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ANS Topical Meeting on

Radiological Accidents — Perspectives and Emergency Planning

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Proceedings

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DISCLAIMER

Foreword

The increasing use of radioactive materials and the increasing public concern about possible accidents involving these materials has led to greater emphasis on preparing for such emergencies. The ANS Topical Meeting on Radiological Accidents—Perspectives and Emergency Planning gave us an opportunity to review our experience with radiological accidents to determine what information from this experience could be applied to improve our preparedness for future accidents.

The meeting covered some of the most important aspects of radiological accidents. We started by inviting several speakers to present papers dealing with radiological accident experience and then solicited other papers on related areas.

Technical response to accidents is of primary interest to many in the nuclear community; most of the papers submitted fell into this area. So many of these papers dealt with the use of computers in response that a session on that topic was arranged.

A very significant impact of most radiological accidents is the cost, especially the cost of cleanup. There were papers on what is known about costs and associated current topics, such as modification and extension of the Price-Anderson Act.

At least as important as the technical response to accidents is how society attempts to deal with them. A session on institutional issues was included to discuss how governments and other organizations respond to and deal with accidents.

Medical effects of accidents are of great concern to the public. Invited papers to review the effects of high doses of radiation as well as very low doses were included in that session.

Although the nuclear industry has an excellent safety record, this fact often does not agree with the public perception of the industry. The final session explored the public response to and perception of radiological emergencies and accidents. This subject will ultimately determine the future use of radioactive materials in this country.

I think you will find that this volume contains an interesting set of papers. The Technical Program Committee did an excellent job of reviewing and selecting the contributions that appear here. Although the Chernobyl accident occurred long after the planning for this meeting was under way, a few papers on Chernobyl were added and many of the other authors managed to incorporate this recent experience. If nothing else, this accident confirmed the timeliness and appropriateness of our topic.

I hope that this review will help us improve our emergency planning, so that we will be even better prepared to deal with any future accidents.

L. Joe Deal
General Chairman

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Introductory Remarks

Joseph Hendrie

It is a pleasure to welcome you to the ANS Topical Meeting on Radiological Accidents—Perspectives and Emergency Planning. It is an auspicious time for such a meeting. Much credit is due to the sponsors and the participating organizations. The sponsors are the Environmental Sciences Division and the Washington Section of the American Nuclear Society, the U.S. Department of Energy, and the U.S. Nuclear Regulatory Commission. The participating organizations are the Federal Emergency Management Agency, the U.S. Environmental Protection Agency, the Health Physics Society, the Conference of Radiation Control Program Directors, the National Emergency Management Association, and the U.S. Department of Transportation. I will speak for all of them in welcoming you here.

When they began the planning for this meeting, the sponsors could hardly have anticipated the greatly increased interest in emergency planning that grew out of the Chernobyl accident. We are very fortunate that the timing was such that we can hear at this meeting the early results from the Vienna meeting at which the Soviets explained Chernobyl. In administering organizations, you get blamed for a lot of things for which you are not responsible, but which happened at a time when you were in charge. That suggests that you ought to get credit for events which occur during your administration for which you had no responsibility at all; therefore, the sponsors of this meeting ought to take full credit for the timing.

Our purpose here is to review what we have learned from past accidents that involve nuclear materials and facilities and to use this knowledge to foster more effective emergency planning for the future. We are going to have sessions on past accidents, medical and health consequences of accidents, lessons learned in the past, current thoughts on radiation dose response, and medical emergency preparedness. Economic considerations in nuclear accidents will be discussed in one session, and we will have sessions on technical problems in emergency planning, institutional problems in emergency planning and emergency response, and, finally, the role of the public in nuclear accident emergency planning and the importance of public perceptions.

Nuclear technology is one of the few technologies in which an appreciation of the possible hazards was present at the beginning and became a serious and ongoing part of the planning and development work. In most technologies, accidents begin happening after the equipment and process have been invented, developed, and placed into service; over time a set of mitigation measures and response measures evolves and becomes embedded in society's practice. In nuclear technology, we started out with an early appreciation of the hazards.

As time went on during the development of nuclear technology, there were criticality accidents, overexposures, and the like. The consequences of these were largely limited to on-site and local areas. Windscale, in the 1950s, was probably the first significant accident that had off-site consequences and affected the public. Other accidents followed; again, the consequences were mostly limited to on-site effects.

The seminal event in this country for reactor accidents and the response to them was, of course, Three Mile Island. My own experience at Three Mile Island strongly emphasizes the importance of emergency planning, especially the importance of aspects of emergency planning that previously had not been given a great deal of attention. The most notable of these aspects was the unexpectedly massive public and media response to the accident, which created a tidal wave of interest and concern that washed across all of us who were trying to understand and deal with the accident, making all of our work much more difficult. Here in this country we have now a very good appreciation of those effects. The emergency planning which has been done since Three Mile Island has, in my view, been an enormous step forward in preparing us should something like that ever happen again and has included measures to deal with that great public interest and concern.

In the aftermath of the Chernobyl accident, there is now more than ever a great interest and concern about radiological accidents and about what should be done in the event of another such accident. I am sure that this meeting will provide us with a chance to examine the past accident record, including the recent events, to find where there might be weak points in our emergency planning structure and to look for places where measures now in use could be improved, made more effective, or made more efficient.

Luncheon Presentations

The Federal Perspective

Samuel W. Speck[a]

I appreciate the opportunity to speak on the nuclear power option in the post-Chernobyl era. I speak as one who until about one week ago had responsibility for commercial nuclear power plant offsite emergency planning at the Federal Emergency Management Agency (FEMA) for some three years. I also speak as a member of the delegation that went to Vienna to meet with the Russians and the other delegations from around the world to discuss what happened at Chernobyl.

Major disasters, whether they be Chernobyl, Bhopal, or the Mexico City earthquake, have a way of significantly impacting the emergency planning that comes after them. When we look back on the Chernobyl incident, I suspect we are going to find it responsible for a number of impacts on the peaceful uses of nuclear energy.

First, the Chernobyl accident made it clear that we will potentially be affected by the use of nuclear power elsewhere in the world, whether or not we continue to make use of the nuclear option in the United States. Russia made it very clear at the Chernobyl conference that she is putting her energy eggs in a nuclear basket. Any increase in electric generating capacity in the Soviet Union during the rest of this century will come from nuclear generating facilities, at least on the European side of the Urals, not from fossil fuel facilities or hydroelectric facilities. There are several reasons for this. One reason is that Russia wants to export her oil and gas to her Eastern European allies and also to Western Europe in exchange for hard currency. So, nuclear power is part of the plan for the overall economic development of the Soviet Union. Elsewhere, by the end of this century, France intends to obtain approximately 75% of her power from the nuclear option, and Japan is moving down the same road.

Another thing that is going to come out of Chernobyl is the internationalization of the management of nuclear power, not the control, but certainly the cooperative internationalization of the management. We plan to participate in a convention on the obligations of all nations for notifying each other when there is a nuclear incident which could have transboundary impacts. I suspect that in the future there will be pressure to develop international referent levels for taking various actions. Given the kind of openness that we saw on many scientific issues at the conference, I anticipate continued progress in the internationalization of the professional nuclear energy community.

The Chernobyl incident caused Russia to open up to some degree. We have not seen that kind of openness in the past, and it is not clear that it is going to continue. Russia may remain closed in many areas, but once there is openness in one particular area, it is much more difficult (and it looks suspicious) to keep other areas closed. I think we saw a little of this when the Russian cruise ship sank recently; Russia was much more open than she had previously been in similar events.

There are a number of other issues raised by the Chernobyl incident. One that is certainly going to have an impact here is the issue of the complexity, the difficulty, and the cost of response and recovery in these incidents. For example, by meteorological manipulation, the Soviets were able to keep rain away from the Chernobyl plant for about one month after the incident. That action protected the surrounding area from the kind of run-off that it might have had otherwise. On the other hand, the lack of rain created more dust problems. Because of the movement of the dust particles in the air when areas were decontaminated, there was a tendency for recontamination. Something that lessened the contamination in one way exacerbated the contamination problem in another. In forests, it takes three to four years for all the radioactive particles to settle to the ground, and thus there are acute problems from fires in such areas. Here the Russians decided that perhaps the best response would be to increase their forest-fire fighting potential. Otherwise, a significant forest fire might well produce a new mini-Chernobyl.

There is a great deal that we found out, but also an enormous amount that we did not find out, in regard to the health impacts of the Chernobyl accident. Milt Levinson relevantly noted that because of the enormous follow-up the Russians are doing with the people who had significant radiation exposure, the health impacts may not be as great as expected. Left unchecked, there might be a modest increase in the rate of deaths from cancer in this population. If you have

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sufficient follow-up, you might ultimately find little or no increase or even a decrease in the overall cancer deaths. The predictions that we have been seeing have focused on the increase in the number of deaths, not the increase in the number of cancers. If these people are checked on an annual basis and cancers from all causes, not just radiation-related cancers, are detected soon enough, then overall cancer-related deaths may turn out to be lower than projected because of this unique medical intervention.

Nevertheless, this event and the offsite elements of it mean that, even if nuclear power is going to be with us globally, its future is perhaps less certain in the United States. Polls taken fairly soon after the accident indicated that more than three out of four Americans opposed building additional nuclear power plants, and more than half favored phasing down or closing those plants that are already in operation.

In this kind of environment, the United States, perhaps more than any other country, is subject to what I would call a "hostage problem." Various elements in our society are able to keep a plant from opening or to close a plant that is open by withholding the requisite support for the opening or for the continued operation of the plant.

Nuclear power was in trouble in the United States long before Chernobyl. It was in trouble, in part, because we developed our nuclear power capability very rapidly. In the process, we did not use the care in some areas in building the plants as well as they should have been built or in managing them as well as they should have been managed. I emphasize in some areas.

Although the release at Three Mile Island has been estimated to be about one-millionth of the release at Chernobyl, the Three Mile Island accident, nevertheless, had a very significant impact in terms of increased fears, in terms of increased costs due to the substantial retrofitting that had to be done, and in terms of the new offsite planning requirements which were mandated by Congress.

Those changes led to delays. Those delays led to increased costs. As people began to look at those increased costs, they began to see that, if you had economic concerns about the cost of nuclear power, it was probably easier to use the safety regulatory process to stop it than it was to use the economic regulatory process. In recent years, we've seen many instances, at the state and local level, of people turning to the safety regulatory structures to try and stop nuclear power for what are essentially economic reasons and, in the process, perverting the safety regulatory process that has been developed in the United States.

This development was possible partly because it came at an unusual time. There had been an oil crisis and threats of severe oil shortages. We miscalculated. We did not appreciate the effect the crisis would have on the traditional linkage between the growth in the gross national product and the growth in use of electrical energy. We did not anticipate the impact of the recession that ensued, and we began to conserve electrical energy in ways we had not conserved in the past. Then we had the decline in oil prices, a glut in oil, and a period in which we had a very modest excess capacity. Thus, the pressure was off. You could play with nuclear power. You could slow down nuclear power without people being worried about brownouts.

In this context, then, I would like to focus on the future of nuclear power from a political perspective, from the perspective of the hostage threat to nuclear power in the United States. I would argue that in the United States we have what is particularly, if not peculiarly, an American problem. After Three Mile Island, Congress and the administration developed a system for requiring offsite planning as a condition of licensing a nuclear power plant. FEMA, just being established at that time, was given the responsibility of advising the Nuclear Regulatory Commission (NRC) by making findings of reasonable assurance that the public health and safety could be protected in the event of an accident. This was to be done by providing guidance, evaluating plans, doing limited training, and evaluating exercises, and then providing findings of reasonable assurance, or the lack thereof, to the NRC. In 1982 another aspect was added. The NRC might also, in the absence of state and local plans, look at utility plans. What it might do with those utility plans is not quite so clear.

When I refer to the hostage problem, I refer particularly to the possibility of a state or a local government refusing to participate in the offsite planning requisite for licensing and, thereby, keeping the plant from operating or closing an operating plant, hence holding a plant hostage. We see a number of hostage problems, and they vary considerably in their nature. There are not too many plants that have not experienced a little of this, when local governments, more than state governments, delay their required participation to see what they can get to enhance their overall emergency management capability. Sometimes their demands include bridges, new fire trucks, and all sorts of things. It is perhaps a little bit of a shakedown, but in most cases it is not carried too far. Then there are situations where the process is simply being slowed until after the next election, because of concern for what the voters might say or how an opponent might use the state or local government's participation in offsite planning. Finally, there is the kind of situation that we have had at Shoreham, and that we may have in Massachusetts in respect to Seabrook, where a government entity wants to stop the plant from ever opening for economic, safety, or political reasons, or a combination of these.

It is easy to overemphasize this problem. We have been incredibly successful, given the tremendous dispersion of power in this country, in bringing some 66 plants and a considerably larger number of units online. We have had successful exercises at least twice at all the operating commercial reactors, and we have completed the CFR 350 requirement for some two-thirds of the plants. We have developed and exercised a Federal Radiological Emergency Response Plan in the last couple of years, a plan that we did not really have until 1984. So we have had considerable success. I think it is easy to overemphasize the problems, the Shorehams of this world, in that context.

What I have defined as a hostage problem up to this point is essentially the refusal of state and local governments to participate in the offsite planning process and thereby to stop a plant from coming online or to cause it to be closed down. The other possibility is one that we have not seen, that the state and local government will simply do a lousy job of offsite planning and exercising in a deliberate effort to show how bad they are, so that FEMA and the NRC cannot make a finding of reasonable assurance under the conditions of such total ineptitude by the state and local governments. When they discover that this approach would probably be a much cheaper way of stopping nuclear power plants than the eight million dollars in legal fees one county has apparently paid, I suspect that we will probably see some try this method. Indeed, some have already indicated to me that they are looking in that direction.

The hostage problem is a particularly, if not peculiarly, American problem for several reasons. In Europe you avoid this problem because the national governments do one of two things. They either mandate that the subnational units (whatever they happen to be) participate in emergency planning, or they simply tell them they are going to assume that the subunit is going to do emergency planning and let it go at that. We, in contrast, require state and local governments to participate if the NRC is going to be able to license, but there is no way that we can mandate that they participate.

There are some other things that make the situation in Europe easier. Europe has far fewer local government units, and their emergency planning zones are usually considerably smaller. Even if there were the same number of local government units, fewer of them would be included in the emergency planning zone.

For all these reasons, the situation is rather different in Europe than in the United States, but things are not all rosy there either. You may have a change in the national government and get an anti-nuclear government, such as the British Labor Party and the German Social Democratic Party, which both have indicated they very well might be.

In the United States, we chose to deal with the problem of regulating the nuclear power industry by an enormous dispersion of power. We disperse power by requiring state and local governments to participate if plants are to be licensed and are to continue to operate. We further disperse power by requiring that every two years a plant must go through what amounts to a relicensing process, because it must exercise its plans and prove that they are still effective. We disperse power at the administrative level. We disperse power within the NRC; I am not going into that except to say that it has made for a somewhat complex, sometimes difficult to understand, and not always terribly well-coordinated process. And, of course, there are nine other agencies involved in emergency planning through the Radiological Assistance Committee process.

Then there is a dispersion of power within Congress. That has created a particularly difficult situation because, in the last couple of years, it has been almost impossible to know where Congress stands or whether the United States even has a nuclear energy policy. FEMA found itself, on one hand, with a joint statement from our appropriations subcommittees in the House and Senate, telling the agency what it expected.

> "In particular, the Committee is concerned about situations where State and local government arbitrarily refused to develop radiological emergency preparedness plans or to participate in the exercise or implementation of such plans. The Committee does not believe that State and local government entities should be permitted to veto the operation of commercial nuclear facilities simply by refusing to participate in the preparation, exercise, and implementation of such plans. In that regard it is the Committee's intention, in its review of such plans, FEMA should presume that Federal, State and local governments will abide by their legal duties to protect public health and safety in an actual emergency. However, where State and local participation in the exercise or implementation of offsite plans is inadequate, the Committee intends for FEMA as a last resort to coordinate the supplemental assistance to Federal agencies that are expected to provide requisite resources within their authorities."

Of course if you go to court as many times as FEMA has and other agencies have, it is in part because there is some question of exactly what your authorities are.

On the other hand, there is the Authorization Committee, chaired by another member of the House from the same party as the one who co-signed the statement I just read to you, indeed from the same state. This chairman has used the hearing process and other means (which I think any person who can tie his shoes and wave "bye-bye" would assume were meant to intimidate) to try to prevent the policy proposed by the Appropriations Committee from being carried out. The Congress simply cannot get its act together because power is so dispersed, not between the chambers, not even between the parties, but within the parties, often between members from the same state.

What are our options, within this context? We can follow the option of compensatory actions, using the state to take compensatory action for locals that will not participate. That is what the State of New York did for Rockland County at Indian Point. That is what New Hampshire has been doing for some of the communities in respect to Seabrook, and that continues to be a viable option. There is some indication that we are seeing locals compensate for state inaction in some other places, perhaps to some degree in Ohio and on some occasions in California.

Now we are getting utility plans. These were first suggested by the House in an 1982-83 NRC authorization bill as something that the NRC might look at, but there is great conflict in Congress over just what this means. Until recently the issue has been hung up on "reasonable assurance," whether FEMA or the NRC could find a reasonable assurance that public health and safety would be protected by the use of the utility plan when the state and locals say they will not participate. There are the problems of whether the utility can carry out its plan if it does not have police powers and whether state and local governments, even if they do participate, can be said to give reasonable assurance if they have not exercised and have not demonstrated any familiarity with the plan.

Recently, the NRC instructed the Atomic Safety and Licensing Board (ASLB) to review the utility plan associated with Long Island Lighting Company and Suffolk County. The legality of this plan is being challenged in the courts. It gave the ASLB two assumptions to follow. One was that the state and local governments would respond in the event of an incident to protect their people, and the second was that they would use the utility plan as the best plan available to them at that time for that response. What will result from that review remains to be seen, but it involves a considerable leap of faith and may take us a considerable leap further in dealing with the hostage problem. It certainly takes us closer to the European approach, but the outcome is unclear. How it would work if state and local governments just decide to do a poor job, instead of completely refraining from participating in emergency planning, is also unclear.

In conclusion, then what are our options? One, I suppose, is to continue to muddle along, satisfied by the fact that most plants have been brought into service and have continued to be in service. A second approach is much greater federal involvement. We have seen a number of options proposed here ranging from the development of some kind of federal brigade or SWAT team which would be ready to go in and provide direction and control, using state and local resources, or perhaps going beyond that and using civil defense authorities. For example, several congressman have suggested developing a federal team that would go in, perhaps in conjunction with nationalization of the national guard, and literally take over. All of these things have a time factor. We know that under the Federal Radiological Emergency Response Plan, valuable as it is, it takes six to eight hours before you can get enough people there to be of much support to state and local governments. So one of the problems with this approach is achieving a timely response with people who really understand the problem and can marshal the local resources.

The third option is to give the state and local governments greater responsibility for emergency preparedness, along the lines of what is done in Europe. Offsite planning could be taken out of the critical path for licensing so that state and local governments could not stop a nuclear power plant from starting or continuing to run by refusing to participate. The political pressure would then no longer be on them to stop the plant, but instead, political pressure would be on them to do as good a job as possible of making certain they were prepared in the event of an incident at that plant. Part of this scheme would include making a determination at the time that the construction permit is issued that an offsite emergency plan for that area is feasible.

Finally, if you simply take the state and locals out of the critical path and say it is up to them to protect their people, what are the federal government's obligations in that regard? It seems to me there are a number. First of all, it would be appropriate to look to the industry for the funding of the federal, the state, and local roles in terms of emergency planning. That is not regarded in all circles as a terribly good idea, but I think much of the industry would be supportive of that if it were done in the appropriate way. Second, continue to improve the guidance that we are giving state and local governments. Frankly, FEMA has had so many responsibilities for trying to put out fires arising from hostage problems during the last couple of years, that the agency has not been able to devote resources to improving the planning guidance or developing the guidance that good offsite emergency planning demands where guidance is not available. The third obligation is training. While we provide limited training at the federal level, this is a place where we could be of increasing assistance to state and local governments and make them feel more secure about meeting their obligations. One thing that we do not do now is provide very much technical assistance in the development of plans and the exercising of those plans. We send the state and locals off on their own and the utility off on its own, and after they come in with their best effort, then we critique it. It would make a lot more sense to be providing more technical assistance in the development of the plan and the exercise process. And finally, in this scenario, we would expect the federal government to continue to evaluate the plans, evaluate the exercises, and publish the results of those evaluations for the public.

It seems to me that the current ad hoc system is not working as well as it should. Congress is placing unreasonable and conflicting burdens on administrative agencies that are simply impossible to reconcile. FEMA's resources have been reduced and diverted to crisis case management at the cost of systems development. The threat of the hostage situation is inevitably going to create an environment where, if any more nuclear power plants are developed, they will probably be sited adjacent to existing plants. This may be done even if these sites are not particularly good places for additional power plants, simply because there would not be the problem of going through a whole new offsite planning process as would be necessary for a new location.

After serving as Associate Director of FEMA with responsibility for offsite emergency planning for three years and since I have been out of that position for a week, I am free to give you an unvarnished version of the situation.

These are some of the thoughts I have as to where we are, how we got here, and where we should be going, in order to preserve the nuclear option.

The Industrial Perspective

James A. Wick

My objective is to share some perceptions about major accident planning and response from the chemical industry perspective. Let's begin with a quick test. I would like everyone in the audience to participate. Last summer, I gave this same test to over 2000 chemical plant managers and their assistants. We were able to tell a few things about them, and I was wondering whether people in the nuclear business are similar or different. Let's see whether we can predict something about you.

Write down the very first thing that pops into your mind. I don't want you to think about these answers. (1) Name a color. (2) Name a flower. (3) Name a piece of furniture. (4) Name an animal in a zoo. Now indicate, by a show of hands, how many answered question No. 1 with "red." About 85 or 90%. How many answered question No. 2 with "rose"? Again 85 or 90%. How many answered question No. 3 with "chair"? About 85%. How many answered question No. 4 with "lion" or "tiger"? The majority.

Most of you, 85 or 90%, answered as I predicted. You are utterly predictable as a group, just as were the chemical plant managers I mentioned. They answered in about the same proportions with the same answers.

How many of you answered all four as I predicted? Only a few. How many answered none as I predicted? Again, maybe a half-dozen. With 85-90% answering as I predicted, only a very few answered all or none of the questions as expected. You are still very much individuals! We have to keep that in mind. As a group, people are predictable when it comes to emergency response and planning, yet we are very much individuals in our lives and in our response to this whole issue.

I want to give you another quick test. Fold your hands for me and see which thumb is on top. There is no incorrect answer. Move your hands apart and quickly bring them back together with the other thumb on top. Hard to do, isn't It doesn't feel right. Fold your arms. See which arm is on top, then move them apart and try to fold them the other way. It's the same phenomenon.

We are very much creatures of habit in everything we do. I would venture to say that your routine in the morning for the first few minutes after getting out of bed is a matter of habit. When you get up, you do the same things every morning in the same sequence, such as brushing your teeth, putting on your pants, or taking a shower. We are creatures of habit, and we have to take that fact into consideration as we go forward in emergency response planning.

For the third little test, I would like to ask you to think back for just a moment to your first formal education, whether it was kindergarten or first grade. Can you tell me your teacher's name? Most of you answered that you can do that. How long has it been since you thought of that person? We have tremendous gray matter potential. In fact, safety programs in both of our industries are very much oriented to imparting information into that gray matter for recall when the time is right. When that time is right, there may be a signal, some sort of stimulus, that tells someone to put on gloves or to look out for a particular hazard or to be sure to wear breathing apparatus in that situation. We want that information recalled at the appropriate time. We should never underestimate the gray matter potential in people. We should continue to work with all people in an attempt to maximize the use of their talents and skills and the potential that has been given to all of us.

I. COMPARING OUR INDUSTRIES WITH PUBLIC PERCEPTIONS

There are many similarities between you in the nuclear industry and us in the chemical industry. Since the accident in Bhopal, we have had to learn a lot. You have blazed a trail, and we thank you for that. Our businesses are indeed alike in that we have the potential to do great harm if we mismanage them. We are also alike in our dedication to managing our technology safely, and since the tragedy in Bhopal, we are also under the tremendous public and governmental scrutiny that has developed due to a lack of confidence and trust in us. This fear of the unknown is commonly called "chemophobia" in the petrochemical industry.

The nuclear industry suffers from some of that same kind of fear. For example, here is a recent UPI clip from my hometown newspaper. Let me read it to you. The date line is Seabrook, New Hampshire.

> "More than a dozen frightened and some 'hysterical' residents reminiscent of the 1938 broadcast classic 'War of the Worlds' called the police during a simulated news broadcast of a meltdown at the Seabrook Power Plant, police said Monday. Seabrook Police Chief, Paul Cronan, said the Sunday night radio broadcast, a political advertisement for an anti-Seabrook Massachusetts Democrat, was irresponsible. The Seabrook police station received 15-20 calls from Massachusetts and New Hampshire residents who heard the 8:30 p.m. broadcast and believed a real nuclear disaster had occurred."

None of us like to see those kinds of reactions.

II. RESPONDING WITH CAER (COMMUNITY AWARENESS AND EMERGENCY RESPONSE)

We are faced with an international issue of trust that can only be solved at the local level. We have taken this lesson to heart in our industry and embarked on major industry-wide initiatives at the local level. These proactive, positive initiatives are targeted to meet real community needs in information and in emergency response and to build public trust through improved relationships and cooperation. We work very closely with and through our trade association, the Chemical Manufacturers Association, and allied trade groups. We have been at it for about a year with renewed vigor, and the preliminary results look positive.

Our industry has discovered several of your resources to help us along the way. We have used consulting firms who have cut their teeth in the nuclear industry on emergency response issues, and they have been able to help us. They understand the differences between our industries, and they have helped us develop guidelines and materials for our local initiatives.

The cornerstone of this effort is our CAER initiative, Community Awareness and Emergency Response. This initiative is a ten-step process of how to go about organizing and implementing a better community emergency response plan at the local level. It doesn't say what the end product should be; it only describes the process. It is liberally based on FEMA-10, the Federal Emergency Management Agency's contingency planning and guidelines document, which is presently being revised. CAER is focusing on improving local emergency response in areas where we have facilities, not just for chemical emergencies, but for the whole family of emergencies, whether they be natural, technological, or nuclear. In fact, we have been successful in piggy-backing on a few of the nuclear emergency response preparedness systems that already exist.

We have 1500 chemical plants today that are active in programs of the CAER type. In my company alone there are 100 U.S. sites, of which 97 are very active in these kinds of programs. About 35 have conducted community-wide, multidisciplinary, multiagency drills within the last year.

Another major initiative has been the National Chemical Referral and Information Center (NCRIC), an enhancement of CHEMTREC. CHEMTREC has a toll-free number for chemical transportation emergencies and provides information and a link to the shipper and/or the manufacturer. The second activity of NCRIC is the Chemical Referral Center, which has a toll-free telephone number for non-emergency information that anyone can use. The third NCRIC activity is CHEMNET, an industrial mutual-aid pact that dispatches the nearest qualified hazardous materials response team to incidents occurring anywhere in the United States. Where we didn't have facilities and teams, we have been able to hire contractors to cover those localities. The last NCRIC activity is improved first-responder training materials.

Many people in my industry have only recently become aware of the tremendous resources of agencies like the Federal Emergency Management Agency, which has a lot of talented people that we have been able to use to help us develop guidelines and procedures and new materials for our plant managers. For example, they have a week-long course on how to conduct a community-wide drill. That course was too long for most of our plant managers to attend. So, using the course as a base, a contractor from the nuclear industry, and people from the chemical industry, we developed a 24-page guide. We intentionally made it short, so a manager could read it in 30 minutes and get an overview of how to organize and conduct a drill in his community.

III. CITIZEN EXPECTATIONS

One of the things that we are learning is what citizens expect of local government. In major emergencies, the area outside the fence is the responsibility of the local government. If we are to make improvements there, we have to make improvements in cooperation with the local government. Before we can do that, we have to understand what demands are being placed on the local government. We concluded that there are about seven things that citizens expect from their local government officials.

1. They must provide advance information, such as alerting signals, response capabilities, and evacuation plans. You have larger emergency planning zones to deal with than we have. We usually think in terms of a mile or less. But, in alerting the community, you are way ahead of us. You have learned some lessons the hard way, and we are learning from you.

One of the things that I tell our plant managers is that if they get involved in a community system of alerting that involves sirens, there are two things to remember: (1) put in a Cadillac, not a cheap siren system; and (2) make sure that the button that activates it is in the hands of the local officials. If you need better communication with them, put in a hotl ine to the local officials.

Here is a letter to the editor that appeared in my local newspaper which demonstrates the first point well.

"On August 22 at 11:15 p.m., my family and I were wakened by several long, loud blasts, which eventually became a continuous scream, from the nuclear warning siren at the Townsend Fire Company. Since the siren is one of the systems designed to warn residents within a 10-mile radius of the Salem/ Hope Creek power facility of malfunctions at the plant, we were thoroughly and justifiably frightened. After turning on the radio and getting no information, we made several phone calls, all of which reassured us that the problem was in the warning system and not the plant."

He goes on to indicate that the siren continued to blow for about two hours, and he lost a night's sleep. That is not an unusual occurence with siren systems around nuclear power-generating facilities, according to the news clips that I read. In fact, they had some problems at the Limerick Station, and a fellow wrote to the local newspaper in West Chester, Pennsylvania. He said, "Look, if these guys can't get this siren system straightened out, what makes them think we have confidence that they can run a reactor."

False alarms produced by warning system malfunctions are a major reason that I have told our plant managers, "If you are putting in a siren system, put in a Cadillac. Don't tell me it can't be done!" There are 25,000 fire departments in the United States, and 24,000 of them are volunteer departments that are alerted through tone-activated siren systems. There are about 1 million volunteer firefighters; if they get up in the middle of the night through inadvertent activation of their siren, they do not tolerate that very long before they make corrections to the system. There are thousands of remotely operated sirens that do not trigger inadvertently very often.

The whole psychology of alerting people and getting them to react in a positive, planned fashion can be a problem. In our business, one of the recommended protective actions may be to stay indoors, shut the windows, and turn off the fans--that is your safe haven. The chemical cloud will pass, the fumes will dissipate, and the area will be safe. These actions are a safer alternative to evacuating, getting in the car, and getting to the intersection at the same time as everyone else. As you well know, that can be a real problem.

Behavioral scientists tell us that people have a natural reaction to fight or flee when subjected to danger such as from a cloud of fumes. They can't fight it so their reaction is to flee. The scientists suggested that if we devote the same amount of energy and effort to training people to evacuate more effectively and efficiently rather than training them to stay indoors, we might have a safer situation. We had never thought about that. We are learning, and we are trying to learn more each day about those kinds of difficult questions that we face at a local level.

2. The second thing that citizens expect local government to do is to assess the magnitude of the emergency quickly and accurately. If you have bad information, you will have two or three unwanted results. For example, let me tell you a story that I heard at Johns Hopkins. They had dispatched a team to Bangladesh after a tsunami went through there several years ago; about 247,000 people were estimated to have been killed in the Delta area of Bangladesh. However, as the early word of the tremendous tragedy spread, the Japanese decided they would respond, and they sent a complete field hospital with doctors, nurses, and staff to Bangladesh. The Germans thought that was a good idea, so they also dispatched one, and the Americans, not to be outdone, sent two. So now Bangladesh had four totally equipped field hospitals. But only the fittest had survived the tsunami. The very young, the very old, the very weak, and females were killed in large numbers. The survivors tended to be physically fit males who had clung to a tree or something for 12 to 14 hours. Their injuries were really abrasions on the inside of the forearms and on the chest. They needed only a little iodine. Bangladesh did not need field hospitals; they were sent due to poor initial assessment.

On that same emergency, incidentally, out of one of the relief planes came a container like the back of a tractor-trailer rig, and in the container was a whole load of teddy bears from a toy manufacturer. Teddy bears are not recognized as a toy in Bangladesh, and the children would not use them. So then they had to divert resources from the emergency to repackaging that tractor-trailer load of teddy bears and shipping them back out of the country.

Quick and accurate assessment of the magnitude of the emergency is critical. Getting the resources that you need and getting them in place fast is what we are talking about. That means having knowledgeable people, good information systems, and good communication systems.

3. The third major expectation of citizens is that we provide emergency mitigation services. These services include fire, police, emergency medical, and shelter.

4. The public expects to be kept informed. We have an important role to play in our industries. Too often the spokesman during an emergency is a university, a health department, or a government official, who may or may not know a thing about your operation. This is the "expert" who is quoted on TV. They are great at speculating in response to questions like "How bad could it be?" Therefore, our industries have a role in providing spokespeople who are knowledgeable, truthful, and accurate, to help keep the public and government officials informed.

5. People expect the local government to rapidly restore services and utilities, even when those services are not their direct responsibility, for example, electrical service.

6. Citizens also expect the local government to provide access to recovery services such as family unification, counseling, insurance claim preparation, and relief assistance.

7. Last, people expect local government to provide information on preventing similar major impacts in the future.

We have a role to play. If we know what is expected of local government, we can identify our role and begin to interact in a positive fashion.

IV. DRILLS

The best way to test where we are and where we ought to go is to conduct drills. You in the nuclear industry are certainly not strangers to multiagency drills. Nuclear power plant drills are conducted to demonstrate performance; the drill must demonstrate performance to get the checks in the blocks in order to continue the license. That is a different kind of drill from the one we want to conduct. We have different objectives for our drill. Ours asks a different question, which is "How can we improve the system?" We are trying to have it achieve those things that the Federal Emergency Management Agency has said that drills ought to do. We've tried to keep our drill scenarios and drill implementation positive and voluntary. The people become stakeholders in building a better system and are better equipped mentally, as well as physically, to deal with future emergencies. We build on what already exists, and we make it work for all types of major emergencies, natural as well as technological.

V. COMMUNICATING RISK

We've also had to learn a lot about communication. Perhaps this is the time for another kind of test. Write a definition of "how fast is fast?" Difficult question, isn't it? We have another version of that question in regard to dumps and environmental concerns. How clean is clean when you clean up a dump site? How safe is safe around your facility?

The nuclear industry is in the lead when it comes to process hazards analysis. In fact, you are very good at sophisticated techniques such as fault tree anaylsis. That is great. We learn from that, but how do you communicate the results from a fault tree analysis to the general public? That has been difficult for us, and I presume it has been difficult for you as well, because you have had a difficult time convincing people that you are the safe industry that you are.

So we need to work on better ways to communicate. We need to learn some of the techniques of the public relations profession. For example, a sign says "Discount for Cash." Isn't that a neat concept? But what would happen if they put up a big sign that said, "Surcharge for Credit"? That's the same thing as a discount for cash, isn't it? But the public relations experts would never put up a sign that said "Surcharge for Credit." We have to learn to communicate in positive ways, rather than negative ways, so we can help our community become more informed and make better decisions.

VI. OUTREACH AND CRISIS COMMUNICATION "MUSTS"

I would like to share with you ten things, which I call common outreach and crisis communication "musts," things that you must do in your community.

1. Plan. Plan for those events and plan again. It is a process, not a product. If you develop a plan, put it in a 3-ring notebook, and call that the end of the process, you are in trouble. The plan has to be a living, working tool. Once I borrowed the emergency response plans for a nuclear power generating station, and I had to use a suitcase to carry them back to Du Pont because there were five three-ring volumes, plus a couple of smaller books. I defy you to go through those volumes at 3:00 a.m. to determine what the appropriate procedures are. The local fire company took those five volumes and extracted about 30 pages that were meaningful to them. Then they had a good working document and a tool they could use. The planning process includes how you communicate with others before and at the time of emergency.

2. Accept Responsibility. Accept the responsibility for a role in crisis communication and crisis management. For instance, one of the things we try very hard to do is to get our plant manager to talk to the local officials. The plant manager is the most credible representative of Du Pont in any community where we have a facility--not a public relations official from Wilmington, not a safety official from Wilmington (the location of our corporate headquarters), not a technical engineer, and not a consultant. The plant manager and his staff are the chief liaison to the community. They have to accept their responsibility for crisis communication and management.

3. Take Charge. Take charge of your facility and establish a communications and command center. If you allow someone else to do this and you allow others to take these responsibilities, you lose control of the situation and you lose your opportunity for input.

4. Never Underestimate the Media. The media are there and they believe strongly, "There is a story here and we're going to get it." You are in the driver's seat; you can provide that story. You need to learn how to provide it in a positive, proactive fashion. When I have gone to accidents, I have helped provide the story, even when we were not the lead agency or the lead company involved. We can help. We can do this in very positive ways that come across well.

For example, holding an onscene press conference is awful. In most media markets, you are going to have at least three TV stations, several newspapers, and a couple of radio stations represented. And then there is you, if you are the spokesman. You go out and give them some details, and suddenly there are 97 million questions being thrown at you. The press conference almost degenerates into a riot. One reason for this, if you think about it and watch your local news, is that Channel 10 will stick their microphone right into your face (don't forget the camera is over their shoulder) and ask you a question. Channel 6 is also taping that answer, but they don't like to show Channel 10's microphone. The next thing you know Channel 6 is doing the same thing, maybe even asking the same question. This goes on and on, and it becomes very difficult to manage.

I can think as fast as any single reporter, but there are five or eight of them out there, and I have trouble keeping up. They can get me in a trap in a hurry. So one of the things we do,once we've given them all the straight facts in press conference fashion, is give them exclusive interviews, one on one. It takes about the same amount of time to give exclusive interviews, and it is a lot more civil. That night, when the clip comes on the 10:00 o'clock news, it shows as an "exclusive" with the Du Pont official. They love the word "exclusive." Incidentally, the first three questions, maybe even the first four or five questions, from every reporter are roughly the same, so you answer the same question over and over again. However, it looks a lot better that night. It looks like you're more in control.

Those are the kinds of lessons that we need to learn in order to communicate with the media. If we communicate effectively with the media, then we communicate effectively with the public. Also, we can't forget our employees. We have an obligation to communicate with them as well. Although we may use a different mechanism, we must communicate with them.

5. Anticipate the Questions and Concerns. What is really on the minds of the people? What are they worried about? Are they fearful for their child's safety or well-being? Are they worried that the industry will fold up and leave town? Those are things of the kind that you can understand only from knowing the community.

6. Grasp the "Real" Issue. You have to grasp what is really bothering the people. For example, you have a man living near a site with groundwater contamination problems. Is he really worried about the groundwater being contaminated? No, he's worried about his daughter who has been drinking that water. So, if you want to move that person from a highly energetic negative position to a negative but low-energy position, perhaps the things to do are (1) provide him with bottled water until the scientists can get in there and examine his well and do all the things that good scientific people do and (2) pay for some medical screening at the nearby hospital for his daughter. It would not cost much, and suddenly the very vocal neighbor is not so vocal, because his real concerns have been dealt with. There is still the concern about groundwater contamination, but now we have more time, and reason and logic can prevail.

7. Be Accurate and Honest. If you try to pull the wool over people's eyes, they will catch you, and your credibility will be lost.

8. Never Speculate, nor Place Blame. Never speculate nor place blame during a crisis communication situation. Speculation is always bad business. Let's talk instead about what is happening and be realistic and honest.

9. Be Able to Articulate Your Positions and Policies Clearly and Simply. You must certainly explain your position clearly and simply as you deal with local officials. You might have the best statement of corporate or departmental policy dealing with this kind of situation, but my guess is that it runs about four lines long in a paragraph that covers the first half of the first page of the manual. That material is pretty difficult to get on the evening news or to communicate effectively to local officials. Communicate in straightforward layman's terms.

10. Practice. In crisis communication and community outreach, you must practice. You must practice to be more proactive.

VII. BENEFITS

But why bother doing this at all? Because we can all benefit.

Our companies will benefit from proactive, positive community relations. We will have more freedom to operate. We won't have as many legislative and regulatory initiatives as we would have otherwise. Our employees will be proud of our companies, our agencies, or our departments as we do the right thing for our community. And certainly, customers will prefer to do business with us because we are reputable, we have integrity, and we are interested in doing the right things.

The community will benefit. If you were introducing risk into the community and the community had no say, then there would probably be some perception of inequitable distribution of that risk; such a perception often results in tremendous backlash and negative reaction to your initiative. When you begin to work in the field of emergency response, planning, and preparedness, the community now has a say in its destiny. The community will also benefit when they receive good information and develop a stake in the venture. Not only do your employees have a stake in what you are trying to accomplish, but community regulators and elected officials will also have a stake in making it work. It becomes a win-win situation. Certainly, the community will win when the tax base has been expanded because the community is recognized as a good place to do business.

And certainly, individuals can benefit: individuals such as you and me. We will grow in our knowledge of other people and in the knowledge of our government, in the elected environment, as well as the regulatory environment. We in the emergency response field have an opportunity to apply our skills to solving the technological problems that are facing our society. Certainly, emergency planning improves our stake in the community in which we live. After all, we live, work, play, and raise our families in these same communities. You bet we care. In fact, today we cannot afford not to be involved. We must be proactive and leaders in the local decision-making process.

Section I

Accident Experience

Chairman: ***Joseph Hendrie***
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Use of Radiological Accident Experience in Establishing Appropriate Perspectives in Emergency Planning

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ABSTRACT Within a nuclear facility, an emergency can range from a situation that only involves the employees of that facility to a series of events that have both onsite and offsite consequences. Analyses of nuclear and non-nuclear emergencies can provide valuable information on the causes of, as well the problems encountered during emergencies. Reports on facility emergencies indicate that up to 90% involve human error. Such events occur more frequently during the night shifts or on weekends. These occurrences may result from the absence of experienced personnel as well as the reduced alertness of onsite personnel. Therefore, this paper emphasizes the human element in a review of accidents that have occurred at nuclear facilities including Windscale, SL-1, the Recuplex criticality, the Wood River Junction criticality, the Browns Ferry fire, Three Mile Island, and Chernobyl. These accidents are described, and their consequences are evaluated. The information obtained from these evaluations may be useful for in- clusion in nuclear plant operating and testing procedures.

INTRODUCTION

An emergency can be defined as a serious situation that develops suddenly and unexpectedly and demands immediate attention. Within a nuclear facility, an emergency can range from a situation involving only facility employees to an event having both onsite and offsite consequences.

The analysis of nuclear emergencies can provide valuable insights on a generic basis, into the causes of such events. In some instances, nuclear emergencies can be initiated through failures of equipment. However, data show that human error is a major source of many industrial accidents, and yet human response after such accidents can also play a major role in bringing accident situations under control.

In the chemical industry, data show that up to 90% of all accidents are induced or further complicated by human error (Joksimovich 1986). In the airline industry, up to 80% of all crashes are thought to involve erroneous human interaction (Lewis 1985a). Similarly, human error appears to have played a major role in essentially all nuclear accidents.

Human error is not confined to the Western nuclear industry; this fact was demonstrated by the recent Chernobyl accident in the Soviet Union. While the exact cause of the accident has not been reliably reported, a number of Soviet plant officials were reportedly removed from their jobs for responding to the accident improperly.

In the preparation of this paper, the authors have examined a range of nuclear accidents from the 1950s to the present that were reported in the literature, and have identified a number of contributing factors which affected human judgment during these events. One common thread found in a large number of accidents is the time of occurrence. The data show that such events, whether severe accidents or operational incidents, appear to occur more frequently during off normal hours (the graveyard shift, weekends, or holidays). Accidents seldom occur during the day shift when the full management team and senior operations personnel are present. Because accidents seldom occur during the day shift, those facility employees most expert in coping with the situation may not be available, and the normal chain of command may be disrupted. At several nuclear power plants, it was also observed that new or less experienced technicians are assigned to the back or graveyard shifts. The lack of experienced human resources and the pressure of an accident situation can be enormous on those individuals who are faced with making important decisions.

An in-depth review of the literature conducted for the U.S. Nuclear Regulatory Commission (NRC) by staff members at the Pacific Northwest Laboratory (Lewis 1985b) determined that human errors generally increase at night and the chance for error is significantly greater if the worker has already been working four or more hours before midnight. The highest error rates were reported to occur between 3 and 6 a.m. The most efficient work is typically performed during the day. When a person is required to work at night and sleep during the day, both work performance and sleep were found to be degraded.

Researchers at Harvard University (Squires 1986) have reported that 30 percent of the operators of heavy equipment fall asleep while driving on the night shift. During a recent annual meeting of the Academy of Behavioral Medicine Research, members of the Harvard Medical School research staff observed that three of the recent major disasters -- Three Mile Island, Bhopal, and Chernobyl -- all happened at night. The concern expressed by behavioral scientists was not only that lowered alertness might cause accidents at nuclear facilities during off-hours, but that the concurrent human ability to detect and correct the problem in a timely manner might also be reduced (Squires 1986). Dr. C. A. Czeisler of the Harvard Medical School, Department of Medicine, has conducted extensive and indepth studies of the effects of rotating shift work schedules (Hearings 1983). Dr. Czeisler observed that workers on night or rotating shifts experience adverse consequences because their circadian rhythm, the physiological functions such as body temperature, hormone secretions, cell division, antibody formation, etc., varies over a 25-hour period.

These physiological functions are regulated by the body's internal clock as well as by indicators from the environment, such as light and temperature. According to Dr. Czeisler, whenever a worker's normal wake/ sleep schedule is interrupted, there is a mismatch between the body's ability and the demands placed upon it in the work place. Stress, gastrointestinal disorders, low morale, high rates of accidents and illness, as well as low productivity, result from this mismatch. Sleep-deprived shift workers often experience involuntary lapses of wakefulness: while they appear to be awake, they may actually be drifting in and out of sleep. Because of these lapses they may be impaired in their ability to respond to warning signals or lights (Czeisler 1980; Rafferty 1983).

Researchers in Sweden (Hearings 1983) have reported that train crewmen fall asleep one out of every six hours during night runs. While asleep, the drivers maintain full pressure on the accelerator, and were found to be unresponsive to warning lights. Czeisler reported in field studies of 1500 workers at a number of industrial facilities, that over 55% of the workers admitted to "nodding off," or falling asleep on the job during any given week. Czeisler's efforts have since focused on identifying effective countermeasures that would mitigate the effects of the interruptions in normal sleep patterns experienced by workers during off-normal hours.

Another common thread to most accidents is that, no matter how well emergency scenarios are developed and emergency planning exercises are conducted, no scenario that fits an emergency situation exists (Vallario and Selby 1981). During an emergency situation, there is little time to extensively consult the emergency operating plan to decide what to do; the major response will have to be based on the quality of the training imparted to emergency response personnel from plan and procedure implementation, as well as thorough periodic drills and exercises.

How an individual will respond to an actual emergency situation is another uncertainty. In Japan, analysis of safety evaluations performed at commercial installations revealed that an appropriate response of operators to an accident could not be expected for at least 10 minutes after they became aware of the emergency situation (Matsuda, Suehiro and Taniguchi 1984). Part of this delay is due to the time required to effectively analyze the situation. Another portion of the delay is likely to result from the reluctance of the operators to acknowledge that they have an emergency. Irrespective of the number of emergency drills and exercises conducted, the real situation may reveal flaws in the emergency plan and in the tempered response. Therefore, everything that even borders on an emergency at a facility should be treated as an emergency.

A REVIEW OF SELECTED NUCLEAR ACCIDENTS

A review of a few of the more familiar nuclear incidents is included to illustrate the sequence of events, the consequences, and the lessons learned. All these accidents began during night shifts, on weekends, or holidays, which suggests that possibly fatigue and/or the absence of more experienced personnel could have been among the causes of these incidents.

Windscale Works of British Nuclear Fuels Accident - 7:25 p.m., Monday, October 7, 1957

The Windscale Works is located on the northwest coast of England. The accident at this facility occurred during a routine maintenance operation in the Head End Treatment Plant. Two air-cooled graphite-moderated uranium reactors were being operated at the site to produce plutonium. At that time, it was known that graphite will store energy as the result of neutron bombardment (the so-called Wigner energy) and that the controlled release of such energy could be used for annealing the graphite structure. Experience showed that achieving a release of all the stored energy in a single step was difficult.

Therefore, a second heating was usually applied.

Unfortunately, at Windscale the second heating was initiated prematurely by the operator because thermocouples used for indicating temperature in the graphite structure showed decreases in graphite temperatures when in fact, the temperature in the graphite structure was rising. This temperature rise was not detected by the operator because of insufficient core instrumentation. The premature second heating caused a uranium fuel cartridge to fail. The fuel cartridge and graphite reacted with air and initiated a fire.

The first indication of the accident was provided by an air sampler located 1/2 mile from the plant. The instrument indicated an increase in beta activity. When the workers suspected a failed fuel cartridge, a remote scanning device was used to locate the cartridge. The scanning device jammed, and the workers donned protective clothes, peered through a plug at the face of the reactor, and found the fuel was red hot. This discovery was the first indication that there was a fire. By that time, the fire had been underway for two days. The intense heat distorted the fuel cartridges and they could not be discharged. However, the cartridges in adjacent channels were successfully removed. By October 10, the spreading fire involved approximately 150 channels in a rectangular cross section.

Although several schemes were attempted, efforts failed to extinguish the fire. Among these efforts was an attempt to cool the graphite by increasing the air flow; this simply caused the fire to burn better. A heat sink was needed to extinguish the fire. Thus, on the fifth day the reactor was flooded with water and the fire was extinguished.

The reactor was extensively damaged. All 35 personnel in the building at the time of the accident inhaled quantities of radioactive materials and were contaminated externally. The release of radioactive materials, primarily iodine and noble gases, into the atmosphere was widespread. Approximately 20,000 Ci of iodine-131 were released through a 405-ft stack. Exposure rates in the surrounding area measured up to 4 mR/h. Radioactive gases such as iodine-131, were transported directly to animal feed and resulted in the contamination of milk. The use of milk was banned within a 30-mile radius of the plant, covering a total area of about 200 square miles. The milk consumption in that area was banned for 25 days; in some of the more heavily contaminated areas, milk consumption was banned for as many as 44 days (Eisenbud 1973; Atomic Energy Office 1957).

Lessons learned from the Windscale accident include:

- A major nuclear accident can have worldwide consequences, both in terms of the airborne releases and in the public impact.
- There was insufficient temperature-indicating instrumentation, which led the operators into making the decision to initiate a second Wigner release, thus initiating the accident.
- The surveillance instrumentation used proved inadequate and failed when an attempt was made to locate and survey the potential damage.
- No written or sufficiently detailed instructions, such as a Pile Operating Manual with special sections on Wigner release, were available.
- An adequate heat sink is required to extinguish a fire.
- The apparent makeshift emergency planning in England was not adequate to handle this emergency situation.
- The countermeasures and public health actions that had been developed for such an emergency were insufficient.

SL-1 U.S. Army Reactor Accident - 9:01 p.m., Monday, January 3, 1961

The Stationary Low Power Reactor (SL-1) accident at the National Reactor Testing Station in Idaho resulted in the first U.S. fatalities from such an event. The SL-1 was an U.S. Army research facility that was constructed as a direct cycle, boiling water reactor with a thermal capacity of 3 MW and an electrical output of 0.8 MW.

The reactor had been used to provide military personnel with operating and maintenance experience, to obtain performance characteristics and fuel burn-up data, and to test components for future improved reactors. The intent was to develop small reactors like the SL-1 for use at Army installations in remote areas such as the Antarctic.

Two weeks before the accident, the reactor had been shut down after being operational for more than two years. Prior to shutdown, the control rods in the reactor core were sticking with increasing frequency. The impedance was thought to be cause by the bowing of boron strips attached to the fuel elements or an accumulation of dirt or corrosion.

On the day of the accident, personnel were inserting cobalt flux measuring wires into the coolant channels between the plates of the fuel assemblies. This task required the removal of some of the control rod drive assemblies and control rods.

The nuclear excursion occurred when 3 military personnel on the night shift attempted to remove the control rods to prepare for the reactor start up. The excursion extensively damaged the reactor core, did minor damage to the reactor building, and fatally injured the 3 people. Since none of the military personnel involved in the accident survived, and because of the rather high exposure rates in the reactor building (approximately 1000 R/h), it was some time before the actual cause of the accident was established.

In May 1961, a board of investigation concluded that the withdrawal of the central control rod resulted in the excursion. Although the building was not designed for tight containment, it was effective in reducing the spread of gaseous fission products to offsite locations. An exposure rate of approximately 50 mR/h was observed about 3/4 mile from the facility (Eisenbud 1973; Vallario and Selby 1981; Cottrell 1962).

The lessons learned from SL-1 accident include:

- The design, in which the withdrawal of a single control rod could cause the reactor to go critical, should be avoided in the future.

- Repeated malfunctioning of control systems should result in remedial actions. The decision to continue operation of the reactor was made at too low a decision level and without review by qualified supervisory personnel.

- Potential accidents need to be anticipated through safety studies and analyses on a continuing formal basis. Source terms should be estimated for a range of accidents to permit the development of plans for realistic emergency preparedness.

- Specialized remote equipment and techniques could be used to determine the condition of the reactor after the accident. This demonstrated the need for the development of a robot that is capable of operating without deteriorating or failing in a relatively high radiation field.

- A remotely operated retrieval system for the nuclear accident dosimeters would be beneficial.

- Shielded medical facilities are required for adequately protecting the medical staff who may be called upon to treat a contaminated accident victim.

- Emergency preparedness instrumentation that is able to assess the high radiation fields which may be present during an accident needs to be developed. The studies performed after the accident indicated that that exposure rates in excess of 1000 R/h could be traversed safely by humans in lifesaving situations.

- Survey instruments must also be designed to operate in a wide range of environments. At the time of the accident the outside temperature was -20°F.

- Contamination control needed to be improved to avoid the spread of radioactive particles. Some of these particles were carried from the plant area on the tires of motor vehicles.

Recuplex Nuclear Criticality - 10:59 a.m., Saturday, April 7, 1962

This was the first accidental nuclear excursion that occurred at any of the production facilities at Hanford. The accident began during a cleanup operation at the plutonium waste recovery facility, known as Recuplex. A concentrated plutonium solution was accidentally pumped from a sump, where it was in thin slab geometry and therefore subcritical, into a 45-liter tank, where it became critical.

The initial pulse consisted of some 10^{17} fissions, with repeated pulses following for 20 minutes. The excursion exposed three personnel in the facility to gamma and neutron radiation doses of 110, 43, and 19 rem, respectively.

The Recuplex incident is mentioned because it occurred only a few months after the SL-1 accident. The SL-1 accident had revealed the importance of retrieving the data from the accident dosimeters in order to estimate the dose received by the exposed workers. In the case of the Recuplex event, a trained senior staff member at the facility performed such retrieval. Without consultation or direction, he ran into the building, following the initial pulse, to retrieve the dosimeter. In so doing, he just barely missed a second much larger pulse consisting of some 10^{18} fissions. Exposure at this level would have been fatal. In this case, a senior staff member who was highly trained in routine and emergency situations did not react properly in an actual emergency. (Vallario and Selby 1981; Zanger 1962; Callihan 1963.)

The lessons learned from the Recuplex accident include:

- Predicting the reactions of personnel, even seasoned "veterans," in an accident situation is difficult if not impossible. If tests could be developed to ascertain such behavior in advance, they would be very useful.

- Robot equipment can be very useful, both in gathering information about an accident and in conducting remedial actions. This lesson was later dramatically confirmed in the longer-range recovery actions at Three Island, Unit 2.

Wood River Junction Criticality - 6:00 p.m., Friday, July 24, 1964

This criticality accident occurred in a fuel recovery plant of United Nuclear Operations in Wood River Junction, Rhode Island. It was the sixth such event to occur in the U.S.

The accident was initiated when a technician erroneously poured 11 liters of concentrated uranyl nitrate into a tank that contained 41 liters of 0.54 M sodium carbonate. Five persons were in the facility at the time of the excursion. The technician saw the typical light blue flash and was knocked to the floor. Although dazed, he ran out of the building to an emergency shack some 500 feet away. He was taken to the Rhode Island hospital, and died 47 hours later. Analyses of hair and blood samples, taken from the technician's body, indicated that he received more than 14,000 rad to the head and 46,000 rad to the pelvic area of his body.

Two supervisory personnel, who arrived on the scene after the first excursion were exposed during a second, lesser excursion that was initiated when they shut down the power to the stirrer that was used to mix the solution. They received approximately 50 rad of gamma radiation and 5 rad of neutron radiation. This portion of the event was similar to the Recuplex incident at Hanford (Auxier 1965).

The lessons learned from the Wood River Junction accident include:

- Better administrative control of fissile materials was needed. Procedural methods and controls had not been updated at the time of the accident.
- Better identification of fissile materials was essential.
- Better training of plant personnel in normal operating procedures was essential. A review by the investigating committee indicated the need for complete written procedures. Except for the plant superintendent and two supervisors, the staff at the facility was not experienced in the kind of operations they performed.

Browns Ferry Nuclear Power Plant Fire - 12:20-p.m, Saturday, March 22, 1975

Although this accident did not involve the release of any radioactive materials, the national media described it as a "disaster and near disaster." The accident was initiated by a team, consisting of an electrician and an engineering aide, who were checking for leaks in the cable spreading room. The team was responsible for locating and plugging holes in the polyurethane sealant surrounding the cables within wall openings. A candle with an open flame was used to indicate the presence of drafts; the candle ignited the polyurethane.

Unfortunately, the fire ignited in an inaccessible location. Fifteen minutes after the fire began, the alarm sounded. The alarm warned people that the CO_2 CARDOX fire extinguishing equipment would be operated. However, because the draft was so strong, this equipment was not effective.

Ultimately, as in the Windscale accident, a heat sink was required to extinguish the fire. Personnel of the Athens Fire Department recommended the use of water to provide the heat sink and extinguish the fire. The plant personnel were concerned, however, that using water would result in shorted electrical circuits, a concern also expressed by the Tennessee Valley Authority Public Safety Officer. Not until 6:00 a.m. on March 23, 1975 did the plant superintendent finally agree to use water. The fire was extinguished about 45 minutes later (Scott 1983; Moeller 1979).

The lessons learned from the Browns Ferry incident include:

- Personnel involved in the work of leak-testing, sealing and inspection need to be trained in fire control and proper fire reporting procedures. They also need to have detailed written, approved procedures for work activities and fire control.
- Fire fighting equipment needs to be fully operational at all times. A metal plate installed for the safety of the personnel prevented the activation of the CO_2 CARDOX system.
- Fire control practices at nuclear facilities at the time emphasized the precaution of not using water on electrical fires. One of the reasons for this emphasis was that the possibility of shorted electrical circuits that could have been instrumental in the loss of safety controls.
- As noted in the Windscale accident, an intense fire requires a heat sink. When the heat sink was provided by the application of water at Browns Ferry, the fire was quickly extinguished.
- Redundant control systems must be sufficiently independent so that they do not interfere with each other during a fire. Control cables, for example, had been run in close proximity in metal conduits. The metal conduits did not provide the intended protection; when the fire melted the insulation of the wires, feedback through indicator lights made safety equipment unavailable.
- Emergency responses needed to be coordinated. The plant superintendent spent too much time in communications with the Central Emergency Center and did not direct the efforts of the plant's emergency response team and the fire fighting teams.

Three Mile Island Unit 2 Nuclear Power Station Accident - Wednesday, 4:00 a.m., March 28, 1979

No other accident until the Chernobyl

event, has had such a profound impact on the public, the nuclear industry, and the regulatory agencies. The accident involved Three Mile Island (TMI) Unit 2, a 906-MW pressurized water reactor, located 9 miles from Harrisburg, Pennsylvania, a city of 70,000.

At the time of the accident, Unit 2 was operating near full capacity. Unit 1 had been shut down for refueling. A condensate pump in the turbine building stopped, which also caused other pumps in the system downstream to shut down. As a result, the reactor scrammed automatically. In accordance with the system's design, emergency feedwater pumps came into operation. However, during an earlier maintenance procedure several valves had been mistakenly left closed, blocking the flow of water. Without this water, the heat and pressure in the primary system began to rise and the power-operated relief valve on top of the pressurizer opened and became stuck. Water poured through the open valve and the pressure fell. After the pressure was reduced, the relief valve was designed to close and restore the integrity of the primary system. In the control room, the relief valve indicator showed it to be closed, when it actually remained open and continued to release water from the primary system. The relief valve remained open 2 1/2 hours; during that time the operators maintained the water in the pressurizer at the normal level, assuming that this was indicative of an acceptable water volume in the primary system. Actually, more coolant was being lost than was being replaced and a small loss-of-coolant accident was underway. Ultimately, water surrounding the core boiled and the top of the reactor core was uncovered. This event resulted in serious damage to the fuel and the associated releases of radioactive fission products to the primary coolant, to the containment, and to the environment.

In the next 11 hours, the operational staff made several attempts to reduce pressure in the primary system. Fifteen hours after the accident began, forced circulation through the core had been established and a relatively stable plant condition was obtained.

Three hours after the accident began, radiation levels in the reactor containment building and the auxiliary building led to the declaration of a site emergency. During the initial phases of the accident, the containment building became contaminated with radioactive water, which was pumped over to the auxiliary building. Abnormal releases were made to the environment through the ventilation system of the auxiliary building.

Dose rates inside the Unit 2 containment ranged up to thousands of rem/h; a measurement of 70 mrem/h was reported at the north gate of the plant. However, dose rates at most ground-level stations around the plant site following the accident were only 1 to 3 mrem/h (Bertini 1980; Okrent and Moeller 1981; NRC 1979).

No injuries were reported. The reactor core was heavily damaged, resulting in extensive cleanup operations.

At least 14 committees reviewed the accident and an ad hoc committee concluded that the offsite collective dose represented minimal risk of additional health effects to the offsite population. The President's Commission pointed to the root problem as being "people-oriented," rather than related to deficiencies in plant design or equipment. The weaknesses identified were not only the shortcomings of individual human beings, but also problems of organizational structure and communications among key individuals and groups. The Commission also found a preoccupation with regulations themselves, rather than with the safety the regulations were supposed to promote. The Commission especially noted the focus on "worst case" accidents to the neglect of less consequential but more probable scenarios (Bertini 1980).

The lessons learned from the TMI accident encompass a wide spectrum of concerns (NRC 1979). The lessons include:

- Training of TMI personnel was deficient and did not prepare the staff to deal with emergency situations. Improved plant personnel training was needed on a periodic basis.

- The TMI control room lacked the application of good ergonomic design. Control rooms should be designed with optimum consideration for human factors.

- There was a need for rigorous emergency preparedness exercises on a periodic basis. After TMI, the NRC required that licensees submit detailed emergency plans and associated implementating procedures, including state and local governmental emergency plans for review and approval. The NRC then determines whether the emergency preparedness capability provided reasonable assurance that the plant personnel could respond promptly and adequately to an emergency and provide measures to protect workers and the public.

- While actual radioactive releases to the environment were negligible, the mental stress to the population may have been the major impact during the TMI accident. The siting of future nuclear power plants at more remote sites would be required.

<u>Chernobyl Nuclear Reactor Accident - 1:23 a.m., Saturday, April 26, 1986</u>

The Chernobyl accident was first noted by a sudden increase in the power of one of four reactors located at the site. While undergoing shutdown, a large emission of steam reacted with the overheated fuel to form the hydrogen that exploded.

When the accident occurred, the reactor was operating at 7% power. The explosion extensively damaged the reactor building; portions of the building collapsed. The reactor was damaged and substantial radioactive releases resulted. Three days later, however, personnel circling the facility in a truck and flying overhead in a helicopter noted that approximately one fourth of the exposed graphite continued to burn.

Based on initial information, temperatures exceeding 9000°F and relatively dry weather were thought to be responsible for transporting the radioactive plume high into the atmosphere and then across the Soviet Union to northeast Europe. Lawrence Livermore National Laboratory scientists suggested, however, that 50% of the radioactive material was released in the hydrogen and steam explosion and the other 50% was dispersed during the graphite fire (Rippon, Blake and Payne 1986).

It was reported that the area around Chernobyl did not immediately receive a high exposure to radiation, probably because of the high plume rise. Soviet emergency plans require evacuation if anticipated doses approach 25 Rem. Evacuation was initiated 36 hours after the accident. It is not known whether the evacuation was delayed because of relatively low radiation levels or for other reasons. It was reported, however, that spectators from the town of Pripyat watched the fire from the plant perimeter on the day of the accident. Although dose rates reported by Soviet sources were relatively low (15 mrem/hr at the zone during the first day of the accident and 0.15 mrem/hr at the zone perimeter), actual rates are not available at this time. Levels observed in Finland, Sweden and the Federal Republic of Germany were 2-40 times that of normal background radiation, mostly due to deposition on the ground. Wide differences in radiation levels in some of the areas appeared to be the result of rainfall. (Wilson 1986).

It is still too early to list all the lessons that were learned from the Chernobyl accident, but some of them are suggested here on the basis of a report by Dr. Richard Wilson, Mallinckrodt Professor of Physics of Harvard University.

- The Chernobyl accident underscores the importance of reactor operator training,
- Emergency planning and the appropriate use of staff during emergencies is essential. Also emergency plans should focus less on how to evacuate, and more on whether and when to evacuate the affected population.
- International cooperation is essential; a more prompt release of data by the Soviets would have been helpful.
- Governments should be careful not to overreact in response to an accident that impacts the international community. During the Chernobyl accident the meterological and radiation protection services of various countries responded promptly and correctly. However, politicians appeared to ignore their council and kept a ban on food and water in areas were it was not needed.

SUMMARY

A review of nuclear facility accidents since the 1950s revealed that human error was found to contribute to a majority of the incidents. The review also indicated that the accidents were caused or complicated by personnel working during off-normal hours.

Contributing factors to the earlier incidents were perhaps deficiencies in the design of facilities or equipment involving this new technology. Later accidents, however, involved facilities and equipment that had become "safer," as a result of the many redundancies built into the operating systems to safeguard the facility and human health. The more recent accidents appear to be the result of errors on the part of humans trying to respond to unusual events involving this increasingly complex technology. Compounding this problem was the fact that many of these accidents may have occurred when the most qualified and best trained personnel were not on duty. In addition, decreased alertness of key personnel during off-normal working hours may have contributed to these events.

Several of the following lessons, learned from these accidents, are being or have been implemented to some degree:

- Nuclear accidents can have worldwide impact on the public, governmental agencies, and the nuclear industry.
- Rotation of operating personnel to work during off-normal hours requires careful planning, taking into consideration the natural body rhythm to maximize performance.
- The response of an individual to an emergency cannot always be predicted. Also, operating personnel require some time to respond to an emergency. Ultimately, the major response will be based on the quality of the training imparted to emergency response personnel.
- Emergency preparedness instrumentation capable of assessing the high radiation fields that may be present is required. This equipment must be operable in a wide

range of environments.

- Remotely operated retrieval and surveillance equipment during emergencies is essential. Control rooms should be designed with optimum consideration for human factors.

- Potential accidents need to be anticipated through safety studies and on continuing formal analysis.

- Regularly scheduled, rigorous emergency preparedness exercises are needed. These exercises must include objective, post-exercise critiques and a willingness on the part of facility management to correct any deficiencies identified.

- Improved plant personnel emergency response training is needed on a periodic basis.

- Emergency responses require well-directed coordination.

- Rigid administrative control of fissile materials is essential.

- Siting of nuclear facilities should be considered with respect to population density.

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**ANS Topical Meeting on Radiological Accidents—
Perspectives and Emergency Planning**

A Chronology of the Chernobyl-4 Accident

G. Donald McPherson

Abstract - This paper presents a chronological description of the events leading to the April 25, 1987, accident at the Chernobyl Unit 4 reactor at Pripyat in the Soviet Union.

I. INTRODUCTION

On August 25-29, 1986, the Union of Soviet Socialist Republics (USSR) presented their report on the April 25, 1986, accident at the Chernobyl Unit 4 reactor to International Atomic Energy Agency Experts' Meeting. The accident chronology included in this paper is based upon information included in the working document of the USSR State Committee on the Utilization of Atomic Energy, which was provided at that meeting, as well as additional information supplied orally by the Soviets at that meeting.

II. REACTOR DESCRIPTION

The Chernobyl reactors are boiling-water-cooled, graphite-moderated, pressure-tube reactors (RBM) 14 m diameter by 9 m high of 1000 MWe capacity. The core is a stack of graphite blocks penetrated by 1680 vertical cooling channels which also contain the fuel bundles. The cooling channels are connected as two separate loops, each one with its own steam separator, four circulating pumps and two feedwater pumps. The latter return water from the condenser to the steam separator. The equipment for each loop is located on one side of the core and is connected to channels only on that side of the core. The pipes leaving the top of the core reactor carry the steam-water mixture to the steam separators. From there the steam goes to the turbines. After passing through a condenser, the steam condensate is returned to the separators. This water is injected in almost jet-pump fashion into a downcomer leading to the circulating pumps, which return the coolant to the reactor.

The only union in the double circuit is that steam from the steam separators on the right side and the left side may be directed to either or both turbines. Thus, one side could feed one turbine, and the other side, the other turbine; or the steam could be mixed to feed both. That is what was being done before the accident.

III. BACKGROUND

The reactor had been running at full power. Personnel at Chernobyl Unit 4 were preparing to shut the reactor down for scheduled maintenance. During the course of the shut-down, the operators planned to perform a test of the capability to use the rotating kinetic energy of the turbo-generator to supply power during the shut-down process.

At the beginning of the test, the reactor power was planned to be 700-1000 MWth (21-33% of full power). Turbogenerator No. 8 was connected to four of the main cooling pumps and two feedwater pumps. The power demand of these pumps would be equivalent to that of the emergency core coolant system. The other four cooling pumps, other two feedwater pumps, and, of course, the other power demands of the reactor building would be carried by the electrical grid. During the test, the operators planned to close the stop valve on the one operating turbine (No. 8) and determine how long the generator could carry the load; i.e., how long it could power the four cooling pumps and two feedwater pumps. At the end of the test, the reactor power would be unchanged. Both turbines on Unit 4 would be stopped, and the grid would be supplying the power to the remaining four cooling pumps and two feedwater pumps, in addition to any other power demands.

Following an earlier test of the same type, the generator exciter block had been modified in an attempt to increase the time during which the run-out power would be available to the emergency equipment. This would allow additional time for the onsite diesel generators to start supplying power to the critical components.

The test was designed by an electrical engineer who was unfamiliar with the operating characteristics of the RBMK reactor. In particular, the test designer was unaware of the positive void reactivity coefficient of the reactor and its increasingly unstable characteristics at lower power. Although the test was to be conducted at 22-31% power (the Soviets state that the exact value was unimportant), the test designer believed that a lower value would be acceptable.

Permission had been requested to conduct the test, but, the Soviets stated that "the plant management did not get around to approving it."

The test occurred on the Saturday before May Day. It was very early in the morning. If the test could not be done while the reactor was being shutdown for this maintenance period, it would be another year before it could be attempted. Given the situation, the operating staff was psychologically unprepared and inattentive to the emergency condition that was arising.

The sequence of events that led to the disaster is detailed in Table I.

IV. AFTER THE ACCIDENT

After early May the situation at the reactor had largely stabilized. The short-lived fission products had decayed, and the radiation exposure rate in the reactor compartments was in the "single R/h" range. Temperatures in the reactor cavity were stable, and in some cases, dropping by 0.5 ^{o}C/day. Ninety-six per cent of the reactor fuel remained in the reactor shaft and adjacent compartments. Any radioactivity still escaping from the reactor was due to wind entrainment of aerosols and did not exceed dozens of Ci/day.

The other three reactor units were contaminated during the accident due to radioactivity being transported through the ventilation system. Sections of the turbine room had high radiation levels since they were contaminated through the destroyed roof of the third block. After decontamination, dose rates in the compartments of Units 1-3 were 2-10 mR/day.

During the accident, large amounts of radioactive material, including graphite and fuel, were discharged over the plant grounds and reactor buildings. Contamination of the ground was not uniform. Several methods of controlling the contamination were used. Polymerizing agents were applied to soil and buildings to reduce air entrainment of particles. As much as 10 cm of soil was removed in some areas. Concrete slabs or mounds of clean soil were placed over the contaminated ground in places. After these measures the total gamma background in the area of the first unit was reduced to 20-30 mR/h. This radiation exposure rate was due principally to releases from Unit 4.

The Soviet long-range plans for mothballing Unit 4 include the following:

- o Building protective walls outside along the perimeter,
- o Erecting concrete and metal dividers between the turbine rooms for Unit 3 and those for Unit 4,
- o Constructing a protective cover over the turbine room,
- o Sealing off "other" compartments as needed, and
- o Providing for ventilation of the damaged unit.

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Table I. Chronology of the accident

Date	Time	Power	Event	Comment/Motivation
April 25	0100	3200 Mwth	Begin power descent to 700-1000 Mwth as required for test. This power level was apparently chosen by test designer; Soviets have stated that the test could have been run at zero power (i.e., immediately following a scram).	
	1305	1600 Mwth	Turbogenerator #7 disconnected. Power for auxiliaries (4 main cooling pumps, 2 electrical feedwater pumps, etc.) was transferred to busbars of turbogenerator #8 (TG #8). Later disengaged scram on 2 turbogenerator trips.	As part of test objective, needed equipment was to be powered from offsite, and TG #8 output was required for test.
	1400	1600 Mwth	Emergency core cooling system (ECCS) disconnected as required by test procedure. This was a violation of regulations.	ECCS was disconnected so that the test could still be rerun.
			Continued power reduction was delayed for 9 h on orders of the power grid dispatcher to meet demand. This allowed a large xenon buildup.	
	2310	1600 Mwth 50% Power	Power reduction resumed.	
	0028	Variable	Switch off local automatic control as permitted at lower power operation; because bulk power set point was preset to a low power level, the reactor power was driven down to below 30 Mwth.	Sharp power reduction results in xenon buildup requiring removal of control rods to maintain criticality.
April 26	0100	200 Mwth variable	Power stabilized at this level. Attempt to increase power to the desired 700-1000 Mwth level was difficult (never accomplished) due to the low amount of operating reactivity margin (ORM.) (Operating reactivity margin is a calculated reactivity margin based on power and control rod distribution and positions, given in terms of number of control rods, where 30 is the minimum allowable. In certain situations this may be lowered to an absolute minimum of 15.)	Xenon poisoning builds up greater than that anticipated for test; as a result, ORM was 6-8 rods, substantially below 15-30 rods required.

Table I. (cont'd)

Date	Time	Power	Event	Comment/Motivation
	0103-0107	200 Mwth variable	At 0103, one additional main cooling pump was placed into service and at 0107 the second main cooling pump was placed into service. This resulted in a total of 8 pumps as the test program directed. The total flow increased to 57,000 m^3/h, above that allowed in the operating regulations. Low power to flow ratio possibly reduced core void to 10% or less. (The Soviets are concerned these pumps may have been operating close to point of cavitation). This resulted in a decrease in steam pressure and water level in steam separators.	Based on the planned test power level, (700-1000), Mwth, 4 main cooling pumps were required to continue operation. Soviets stated that this mode of operation - low core void, very low ORM, slow power measurement response and pumps near cavitation - is very unstable.
	Before 0119:00	200 Mwth variable	In order to keep low steam pressure and low water level from causing reactor shutdown, the operators blocked the emergency protection signals related to these parameters.	By blocking these protection signals, operator is able to proceed towards test.
	0119:10	200 Mwth variable	Operator began manual replenishment of water to steam separator.	Apparently anxious to establish this water level well within the normal range, operator sharply increased the feedwater flow above its nominal value.
	0119:30	200 Mwth	The AR (automatic power regulating) rods moved upward to compensate for reduction in power. Some manual rods removed entirely to allow insertion of AR rods.	The Soviet report implies that this maneuver is probably not permitted.
	0119:40		The feedwater flow to the steam separator had increased by a factor of 3 over the balanced flow for this power level. As this colder water reached the reactor core, there was a sharp drop in the steam fraction of the coolant, and a corresponding power decrease.The feedwater flows into the steam separator at the nozzles of the downcomers leading to the primary coolant pumps. Hence, the core inlet flow temperature responds rapidly to any change in feedwater flow.	

Table I. (cont'd)

Date	Time	Power	Event	Comment/Motivation
	0119:58		Turbine bypass valve (steam dump) closed.	This was done to raise the steam pressure which was too low. However it continued to drop slowly until the start of the accident (0123:40).
	0121:50	200 Mwth variable	Feedwater flow raised to 4 times the balanced flow for this power level.	
			Operator reduced feedwater flow rate sharply to a level of about two-thirds of balanced flow.	This caused increase in inlet temperature and compounded the events one minute later.
	0122:10		Outlet steam quality increases. Therefore automatic rods begin inserting.	Due to approximately 20 sec transient time from steam down to core inlet.
	0122:30	200 Mwth variable	A printout of the actual core flux monitor outputs and the position of all the regulating rods was obtained at this time ("Skala" system). Operator noted that ORM was about 6-8 rods, far below the value where immediate reactor shutdown is required. Nonetheless, no action taken. The flux was "practically arched" in the radial direction and double peaked axially with the higher peak in the top section of the core.	At this highly unusual power shape, the error in estimating the control rod worths was extremely large, hence the error in estimating the ORM was also large.
	0122:45		Feedwater flow stops decreasing at a value 2/3 of balanced flow. Increase in steam quality stops because of pressure increase and feedwater flow stabilization.	
	0123:04	200 Mwth	To begin the test the TG #8 turbine stop-valve was closed and the turbogenerator bypass valve remained closed. The signal for reactor shutdown on closure of both turbo-generator steam valves had been disengaged. This was in violation of the test program and normal operating procedures.	By avoiding reactor shutdown, it would be possible to repeat the test using reactor generated steam. Soviets state that this is the most infelicitous (awkward) moment for this test to be run.

Table I. (cont'd)

Date	Time	Power.	Event	Comment/Motivation
			Flow rate began to fall as the 4 main cooling pumps powered by TG #8 began to run down. Steam pressures also began to increase due to removal of TG steam load and the reduction (by operator action) of the feedwater rate about 1 minute before. These factors combined to increase the coolant void fraction (positive reactivity insertion) and consequently the power.	
	0123:10		Power increase causes pressure increase which causes void collapse. This feeds back to decrease power and automatic rods rise.	
	0123:21		As four pumps coast down, the flow drops, void and power increase. To compensate, automatic power regulating rods begin to lower.	Due to the reactor conditions at the time of the test, the void fraciton is believed to have increased many times more sharply than it would have at normal power. This sharp increase in void fraction rsults in a sharp increase in reactivity.
	0123:31		Reactivity and power increase further because of pump coast down.	
	0123:40	320 Mwth increasing	Unit shift foreman gave order to press emergency scram button.	Soviets stated: "When they were still alive, the operators gave three nations for pressing the AZ5 scram button: 1. the power was increasing 2. the test was proceeding well (and therefore a repeat would be unnecessary) 3. the automatic control rods were moving."

Table I. (cont'd)

Date	Time	Power	Event	Comment/Motivation
	0123:43	Soviet calc. indicates power at 3800 Mwth and rising	The "runaway period came to be much less than 20 seconds." [There are indications that period was 1 second.][a] Over power and short-period alarms come on. The positive void coefficient prompted "deterioration of the situation." Only the Doppler effect partially compensated for the void reactivity increase.	Soviets have stated that void reactivity coefficient 2.0 x 10^{-4}/% steam volume at normal operating conditions was 50% higher at test conditions, but implied it was even higher at this time.
	0123:44		[First power surge terminated by Doppler effect and rod insertion. Water flow continued to decrease (due to run-down of 4 main cooling pumps) and the power continued to increase.]	
	0123:44 (appr.)		Operator heard banging noises and saw rods were stopping before they reached bottom and released servo mechanism to allow rods to fall into core.	The first power surge may have distorted the core such that the control rods could not be bottomed. The banging noises may have been initial ruptures of pressure tubes.
	0123:45		This likely led to fuel fragmentation causing large steam spike which reversed flow to close main reactor pump check valves. Resulting void produces neutron pulse, and second power surge.	
	0123:46		Steam drum pressure exceeds "accident level" and pressure relief valves open. [Pressure rise rate calculated by Soviets was 8-10 atm/sec].	

[a]Items resulting from Soviet mathematical model or that cannot be clearly attributed to measured values are enclosed in square parentheses.

Table I. (cont'd)

Date	Time	Power	Event	Comment/Motivation
	0123:47		Large increase in coolant flow [as channels rupture due to pressure spike].	If initial ruptures occurred as above, this flow increase could be consistent with the upper biological shield being lifted by the pent-up steam in the reactor cylinder and the resultant severing of all pressure tubes and control rod channels.
	0123:48		["Thermal explosion" (steam release) blows top off reactor and destroys reactor hall building].	
	0124 (appr.)		Two explosions - hot fragments and sparks emitted from top of reactor building. Hot fragments caused about 30 fires on roofs, etc. [Mixture of gases containing hydrogen and carbon monoxide capable of thermal explosion if mixed with oxygen, were created in core region]. Thirty fires started in three primary fire sites: (1) turbine room above TG #7; (2) reactor room; and (3) partially destroyed compartments adjacent to reactor room.	[Observation from outside reactor buiding. From this point on, there is no more information from the control room].
	0154		Fire fighting units from Pripyat and Chernobyl arrive. Focused on fighting fire in turbine room to prevent spread to Unit 3. Hand extinguishers and "stationary fire cranes" were used to fight fires in the compartments.	
	0334		Most of fires on turbine room roof were out.	
	0354		Fires on reactor building roof were out.	
	0500		All fires out. Shutdown of Unit 3.	

Table I. (cont'd)

Date	Time	Power	Event	Comment/Motivation
April 27	0113		Shutdown units 1 and 2.	
			Immediately after the accident, an attempt was made to reduce the temperature in the reactor cavity and prevent combustion of the graphite using emergency and auxiliary feedwater pumps. This was unsuccessful. Decision was made to fill the reactor cavity with heat discharging and filtering materials.	
April 28-May 2			Dropped 5000 tons of boron compounds, dolomite, sand, clay, and lead onto damaged reactor. Discharge of radioactivity dropped to several hundred curies/hr.	
			Problem of reducing fuel heatup was solved by pumping nitrogen into space under reactor. Temperatures rose, stopped, and began to drop. As insurance against "extremely improbable" failure of the lower tier of structures, construction of an artificial "heat discharge horizon" was established under the core. Completed by end of June.	

Uranium Yellow Cake Accident—Wichita, Kansas*

Harold R. Borchert

ABSTRACT-A tractor and semi trailer containing Uranium Yellow Cake, had overturned on I-235, Wichita, Kansas on Thursday, March 22, 1979. The truck driver and passenger were transported, with unknown injuries, to the hospital by ambulance. The shipment consisted of 54 drums of Uranium Ore Concentrate Powder. Half of the drums were damaged or had their lids off. Since it was raining at the time of the accident, plastic was used to cover the barrels and spilled material in an attempt to contain the yellow cake. A bulldozer was used to construct a series of dams in the median and the ditch to contain the run-off water from the contaminated area. Adverse and diverse weather conditions hampered the clean up operations over the next several days. The contaminated water and soil were shipped back to the mine for reintroduction into the milling process. The equipment was decontaminated prior to being released from the site. The clean up personnel wore protective clothing and respiratory protection equipment, if necessary. All individuals were surveyed and decontaminated prior to exiting the area.

The Kansas Highway Patrol was notified at 8:51 a.m. on Thursday, March 22, 1979 that a semi tractor and trailer had overturned on I-235 at the North Meridian overpass Wichita, Kansas. A Sedgwick County Fire Department Engine along with a Wichita-Sedgwick County ambulance were dispatched to the scene at 8:58 a.m. Upon arrival at the scene, the responders observed that the vehicle had radioactive placards affixed which indicated that radioactive materials constituted the shipment. Further investigation of the shipment indicated that the cargo consisted of numerous barrels. Upon reviewing the markings on the barrels the responders concluded that the cargo was uranium. The Radiation Specialist, Kansas Department of Health and Environment Wichita Office was notified of the accident at about 9:00 a.m. He telephoned the State Office in Topeka, Kansas at 9:05 a.m. relaying the accident information available to him and then proceeded to the accident site.

The Sedgwick County Assistant Fire Chief, upon learning that the cargo was uranium, requested that the Hazardous Material Response team be notified and activated. The Sedgwick County Civil Preparedness Office was notified at 9:10 a.m. The coordinator and assistant coordinator arrived at the scene about 9:20 a.m. They found that the top of the overturned semi trailer had been ripped open and a number of barrels were scattered about the site. The vehicle which was traveling west and south on I-235 had the right wheels leave the pavement about 213 meters (m) (700 feet) from the site and became mired in the soft unfinished shoulder. The driver had attempted to bring the vehicle back onto the pavement and under control when it overturned about 30.5 m (100 feet) east of the overpass. The overturned semi tractor and trailer was blocking the west bound lanes of I-235 including the median and outer shoulder. The contents consisting of 2.1 X 10^{-1} m^3 (55 gal) drums was spilled during the mishap towards the west and since the overturned vehicle blocked the area, traffic could not proceed past the accident site. This prevented the spread of the material away from the accident scene.

The Civil Preparedness Radiological Officer arrived about 9:25 a.m. and began performing surveys of the area using a Civil Defense CDV-700 GM survey meter. According to the Radiological Officer's survey, the radiation levels were found to be about 2.15 X 10^{-10} C kg^{-1} s^{-1} (3 milliroentgens per hour) (mr/hr) in the immediate area of the spill and about 2.15 X 10^{-9} C kg^{-1} s^{-1} (30 mr/hr) close to the material itself. A Kansas Highway Patrolman

*At the time of the accident the author was a Health Physicist and Chief of the Materials Licensing and Control Section, Bureau of Radiation Control, Division of Environment, Kansas Department of Health and Environment, Topeka, Kansas 66620.

also attempted to obtain radiation readings in the area but was unsuccessful. No other radiation levels were obtained in the area and the previous readings could not be confirmed because it had started raining.

The truck driver and passenger, with unknown and undetermined injuries, were transported by ambulance to the hospital. Assistance was requested from the McConnell Air Force Base Disaster Preparedness Office and they sent an officer with survey equipment to the hospital to perform contamination surveys. The truck driver, passenger, ambulance personnel and ambulance were found to be free from contamination. The truck driver and passenger were treated for their respective injuries. The passenger did not exhibit any signs of injury and was held over night for observation and then released. The truck driver had a couple of cracked ribs and minor cuts and a few bruises. He was hospitalized for about three days and then released.

The Sedgwick County Fire Department and the Hazardous Materials Response Team established a control line about 30.5 m (100 feet) from the site. The traffic was able to be detoured around the site on the eastbound lanes of I-235. Since it was raining the decision was made to cover the material with plastic to lessen the possible dispersal from the site.

The shipping documents were recovered from the cab of the semi tractor and it was found that the contents were indeed radioactive material and listed as Radionuclide U-238 Uranium Ore Concentrate Powder 3,700 mega Bequerels (MBq) (0.10 curies) per drum. The shipment contained 54 drums 19,859 kilograms (kg) (43,782 pounds) gross weight of which the net contents were 18,595 kg (40,996 pounds). The Uranium "Yellow Cake" was being transported by Salt Creek Freightways, Casper, Wyoming from Petrotomics Company, Shirley Basin, Wyoming to the Kerr-McGee Nuclear Corporation, Sequoyah Refining Facility, Gore, Oklahoma for further processing.

The Sedgwick County Civil Preparedness Communications Director notified all three companies about the accident. He also contacted Chem-Trec, the national agency that provides information about hazardous materials. Chem-Trec confirmed that the material was not dangerous to persons in the area. The communications personnel notified the Kansas Division of Emergency Preparedness, the Kansas Adjutant General's Office and the Kansas Department of Health and Environment (KDHE) about the accident.

The KDHE, Bureau of Radiation Control (BRC) Radiation Protection Specialist from the Wichita Office arrived about 9:30 a.m. and assessed the area. He was appointed site commander by the Civil Preparedness Agency because KDHE would ultimately be in charge of the accident area for clean up purposes. The decision was made to cover the area with more plastic because it was raining. Since the accident occurred in a construction area (bridge and highway), a workman from the nearby construction crew was requested to use his bulldozer to construct several bar dikes or dams in the median and ditch to contain the material and water run-off from the contaminated area. More plastic was brought to the scene to cover as much material as possible to lessen the run-off or dispersal from the scene.

There was considerable discussion between the Radiation Protection Specialist at the site and KDHE Topeka Office to fully assess the situation. The rain continued, the dikes and dams had been constructed and the material was covered with plastic, therefore it was recommended that the area be kept secured and static until a Health Physicist from KDHE Topeka Office arrived at the site. The author was directed to proceed to the site, take over command and supervise the activities necessary to clean up the site.

Arriving in the early afternoon the accident scene was assessed and surveys were conducted. The radiation levels found were less than 1.43×10^{-10} C kg^{-1} s^{-1} (2 mr/hr) around the area. Because of the low radiation level from Uranium Yellow Cake, the radiation exposure to personnel did not need to be considered or limited. Since Uranium Yellow Cake is more hazardous chemically than radiologically, dealing with the accident was made easier. The initial assessment of the accident site indicated that of the 54 drums total, 28 drums were undamaged. The remaining 26 drums were damaged to some extent. There were 13 of these drums with their lids off and about 907 kg (2000 pounds) of Uranium Yellow Cake spilled in the trailer, ditch and on the highway. Representatives from the carrier, shipper, a consulting firm specializing in radioactive spills, the insurance carrier and a hazardous materials company arrived over the next several hours. After several discussions and more surveys, a plan of action was formulated to obtain a barrel loader to pick up the drums, load them on another trailer and ship them to Gore, Oklahoma. Control lines were established around the site to prevent the further spread of contamination and limit the access of personnel to the site. Security lines were established about 30.5 m (100 feet) from the control lines. This was done to provide an additional buffer area and was very valuable in limiting personnel access. Discussions were held concerning procedures to be followed and it was decided that there needed to be some flexibility in establishing these as changes would be necessary.

The clean up started with a new semi tractor and trailer being brought to the site and parked outside of the control line. The trailer was lined inside with plastic and the undamaged drums were picked up, inspected and loaded. The remaining damaged drums were also picked up, wrapped in plastic and loaded on the trailer.

Two wreckers were brought to the site to upright the semi tractor and trailer. During the time that the wreckers were positioning themselves and attaching cables to the semi tractor and trailer, there was a wind shift from a north-easterly direction to a west northwesterly direction. This placed the command center and support vehicles downwind of the site. Operations were halted while the command center and support vehicles were moved from the east side of the accident site to the west side placing them upwind from the site. The traffic detoured over the eastbound lanes of I-235, around the site, was stopped for a period of 10 minutes while the semi tractor and trailer were uprighted. There was a cloud of yellow cake dust raised that traveled in a south-easterly direction over the east bound lanes of the highway and the area where the command center and support vehicles had been previously located. This cloud dissipated very quickly and no measurable evidence of contamination was found in the area traversed by the cloud. The east bound lanes of I-235 were reopened and traffic was allowed to proceed around the site.

The clean up progressed rather slowly for the next several days as the crew attempted to recover and pickup as much of the spilled material as possible. This material was placed into barrels for transport back to the Petrotomics Company Mine, Shirley Basin, Wyoming for reprocessing. There were a number of problems associated with the clean up, most of which were beyond anyone's control. The weather was most uncooperative as we experienced snow, drizzle, sleet, rain, strong winds and cold temperatures for several days. During breaks in the weather, surveys were conducted of the area and contaminated areas were outlined with red flags. The surveys were conducted utilizing modified Civil Defense Geiger Mueller (GM) survey instruments. The modification consisted of replacing the side window GM tube with a thin end window GM tube, thus increasing the instruments' sensitivity. Soil and water samples were collected around the perimeter of the marked contaminated area. These samples were sent to the KDHE laboratory for analysis for confirmation of the contaminated area.

Part of the crew, after briefing, started to decontaminate the semi tractor and trailer. The tractor was relatively easy to decontaminate by washing with water. The contaminated water was diverted from the roadway to a diked area in the median. This was done for the ease of removal of the water. The trailer, however, proved to be a much more formidable task in that material had become lodged between the flooring and wooden sides. Several attempts were made to decontaminate the trailer by sweeping, vacuuming and washing without success. The decision was made to remove the interior sections of the trailer, decontaminate them to the extent possible and wrap them in plastic prior to shipment back to Wyoming.

Very tight security was maintained during the entire operation with the insurance company contracting for a private security force. Those individuals necessary for the clean up operations were allowed and authorized in the area. The personnel working in the area where dust or airborne particles posed a problem were required to wear respirators and protective clothing.

When the weather broke, a very thorough and careful survey was conducted of the area to identify and outline the area of contamination. Red flags were used to thoroughly mark and outline the area. Several soil and water samples were collected and analyzed for uranium concentration. Upon receipt of the data, the magnitude of the contamination was more clear and was rather extensive. There were several discussions held over the next few hours in order to outline the direction of the operation. It was determined that it would be necessary to pick up and dispose of a large volume of soil and water. The clean up crew decided that large holding tanks were needed for the water, pumps and tankers would also be needed to pick up and transport the contaminated water. The contaminated soil would need to be moved and handled with a bulldozer, road grader and backhoe. The water could be hauled in transport tankers for disposal and the soil would need to be packaged in drums prior to its transport for disposal. The water and soil would be transported back to the Petrotomics Company mine for reprocessing. The water was pumped from the diked areas by means of a slurry pump into a transport truck. This truck load of about 3.79 m^3 (1,000 gal) was then transferred into the large holding tank. This process continued until all of the water had been picked up from the diked areas. Large transport tankers, about 15.14 m^3 (4,000 gal) each were used to haul the contaminated water back to Wyoming.

The bulldozer was used to push the contaminated soil from the ditches and median into several piles for ease of removal. The contaminated soil was picked up by hand in areas of the ditch and median that were too wet for the use of a bulldozer. Some of the crew put on rubber boots and walked out on boards, placed over the extremely soft areas and with shovels removed the contaminated soil. The shoulders of the ditch and median were bladed several times with a grader to remove the contaminated soil. Each pass with the grader removed about 2-3 cm (1-1.5 in) and this continued until there was no evidence of contamination remaining. In some areas along the shoulder, it was necessary to remove 15-20 cm (6-8 in) of the soil in order to get all the contaminated soil.

The contractor, for this construction area, designed a "nuclear grade barrel filler" from a cement bucket for use in filling the 2.1 X 10^{-1} m^3 (55 gal) drums with the contaminated soil. The contractor had four tubular legs welded to the cement bucket so that each drum could be positioned under it for filling. The backhoe was used to pick up the contaminated soil from the ditch and median dumping it into the cement bucket for loading into the drums. The drums were filled one at a time by this method. They were then wheeled by a hand truck to an area where they were surveyed and decontaminated if necessary. As the drums were identified as clean they were moved to a clean area ready for loading. The drums were loaded two at a time, by means of a barrel loader, onto a semi tractor and trailer rig for transport back to Wyoming. An individual load consisted of about 70 drums each of which weighed 272 kg (600 lbs) and contained about 0.085 m^3 (1/3 yd^3) of dirt.

Upon picking up and removing the contaminated soil and water, the final soil samples of the area indicated a residual concentration of 1.11 mBq-5.18 mBq/gm (0.3-1.4 pCi/g). This compared very favorably to the 2.5 mBq/g (0.67 pCi/g) average for naturally occurring uranium 238 found in the soil in the surrounding area.

The soil and water were all shipped to the mine in Wyoming for reprocessing and recovery of the Uranium Yellow Cake. This was done because they were considered a resource since the concentration of uranium in the water ranged from 592 Bq/m^3 (16 pCi/l) to a high of 2.8 M Bq/m^3 (75,853 pCi/l) and in the soil it ranged up to 20.5 k Bq/m^3 (555 pCi/l).

Decontamination of the heavy equipment namely the backhoe, bulldozer and road grader was formidable in that several successive scrubbings and washings were necessary over several days in order to accomplish the task. The most arduous, however, was decontamination of the steel bridge girders that were on the shoulder of the highway. Yellow cake powder had been spilled on these, since they were very close to the overturned vehicle. The girders were picked up individually, surveyed, decontaminated and then placed on plastic in a clean area. They were also covered with plastic to prevent any possibility of recontamination. The decontamination process was very slow and covered several days before all the equipment was cleaned. Only the road surface remained which was almost as difficult to decontaminate as the girders. Numerous scrubbings and washings were done on the road with very little success. Detergents, liquid soap dilute hydrochloric acid and various other solvents were tried, but were not effective. Finally a hydrolazer (high pressure spray) system was utilized and the decontamination of the road surface was accomplished. All of the equipment was decontaminated to the limits of : a) 5000 disintegrations per minute (D/M)/100 cm^2 fixed alpha, b) a maximum of 15,000 D/M/100 cm^2 fixed alpha and c) 1,000 D/M/100 cm^2 removable alpha prior to being released from the site.

Air sampling was conducted during the operation and no significant air concentrations were observed. The filter cartridges from the face respirators were analyzed of the workers performing decontamination techniques and the concentrations ranged from 2.3 Bq (61.6 pCi)/filter to 5.3 Bq (141.8 pCi)/filter. Urinalysis bioassay was done on 29 individuals associated directly with the clean up operations. The maximum bioassay result was 31 µg Uranium (U)/liter. This level is less than 24% of the recommended action level of 130 ugU/liter for a single uptake on a biweekly sampling program. The personnel dosimetry of those individuals utilizing such did not record any significant exposures above that of background. There were about 24 water samples and 65 soil samples collected and analyzed over the course of the clean up.

In summary, the clean up decontamination operations covered a period of 12 days under some adverse weather conditions. Both the driver and passenger completely recovered from their respective injuries. There were about 700-2.1 X 10^{-1} m^3 (55 gal) drums of soil and seven tanker trucks of 106.12 m^3 (28,000 gal) of water were removed from the site. Several attempts were necessary prior to the successful decontamination of some of the heavy equipment, girders and road surface. The cooperation of all the parties during the entire operation was very good.

The area was decontaminated to the level established by the state. The entire site was officially released from state control at 6:40 p.m., April 2, 1979. The State Highway Department reopened the highway to traffic the next morning.

Aerial Survey Efforts in the Search for Radon-Contaminated Houses in the Reading Prong Area Near Boyertown, Pennsylvania

Raymond A. Hoover and Dennis E. Mateik

ABSTRACT At the request of the Commonwealth of Pennsylvania, the Department of Energy requested EG&G Energy Measurements to fly an aerial radiological survey over a portion of the Reading Prong near Boyertown, Pennsylvania. The survey goal was to help locate regions where buildings contained elevated levels of radon gas. A 250 km2 area was surveyed. A number of sites were located. These sites correlated fairly well with known geologic faults in the area.

In December of 1984, the home of Stanley Watras in Boyertown, Pennsylvania, was found to contain excessive amounts of radon gas. The Commonwealth of Pennsylvania's Department of Environmental Resources (DER) requested assistance from the U.S. Department of Energy (DOE) in locating regions with high probability of containing houses with excessive amounts of radon.

In order to respond to emergencies of this nature, the DOE maintains a Remote Sensing Laboratory (RSL) in Las Vegas, Nevada, and an extension facility in Washington, D.C. The RSL is operated for the DOE by EG&G Energy Measurements, Inc. (EG&G/EM), a contractor of the DOE. One of the major functions of the RSL is to manage an aerial surveillance program called the Aerial Measuring Systems (AMS) (Jobst, 1979). Since its inception in 1958, the AMS has continued a nationwide effort to document baseline radiological conditions surrounding nuclear facilities of interest. These facilities include nuclear power plants, nuclear materials processing plants, and research laboratories employing nuclear materials. At the request of federal or state agencies, and by direction of the DOE, the AMS is deployed for various aerial survey operations. In response to the Commonwealth of Pennsylvania's request, AMS was deployed to conduct an aerial survey of a portion of the Reading Prong near Boyertown, Pennsylvania. The EG&G survey team was tasked with determining which areas in the survey area were most likely to contain houses with excessive amounts of radon, and reporting this information to the local DER officials as rapidly as possible.

II. RADON

Radon is an odorless, colorless, radioactive gas which is produced by the radioactive decay of radium. Radium is part of both the uranium and thorium decay chains, two widely distributed, naturally occurring, radioactive materials. Radon decays by emitting an alpha particle. Long term exposure to elevated concentrations of radon is considered to be a health hazard.

The movement of radon gas from its point of formation in a soil or rock into a building is controlled by many factors. Chief among these are the ability of the radon atom to escape from the interior of the mineral grain of the soil or rock where it was formed, its diffusion and transport through permiable rock and soil (Tanner, 1964) and its ability to penetrate the walls and basement of a building (Fleischer and Turner, 1984). The buildup of radon in a building is further controlled by its ability to diffuse out of the house. Since radon is transported slowly through soils, radon-222 (Rn-222, half-life of 3.8 days) is the isotope which is mostly found in buildings. The other two naturally occurring radon isotopes (Rn-219 and -220) have half-lives which are too short to permit either isotope to accumulate in sufficient amounts to cause any concern.

III. SURVEY SITE DESCRIPTION

Initially, it was decided that an area of approximately 9 square kilometers (km2) (4 square miles) would be surveyed. this area was flown at an altitude of 30 meters (100 ft) and at a line spacing of 45 meters (150 ft). This survey was roughly centered on the Stanley Watras house in both the north-south and east-west directions. At the conclusion of this survey it was decided to increase the size of the survey area to about 250 km2 (100 square miles). At the same time, it was decided that a higher altitude and a wider line spacing would not seriously degrade the data quality and would also shorten the survey time. The new altitude was 60 meters (200 ft) with a line spacing of 90 meters (300 ft). The new survey area was to be roughly centered on the Boyertown Reservoir and was to include all of the townships of Earl and of Colebrookdale, which includes Boyertown.

The Reading Prong is a large geologic formation composed mostly of metamorphised precambrian rocks, whose western edge starts in the vicinity of Reading, Pennsylvania, and stretches several hundred kilometers in a Northeasterly direction. Parts of the Reading Prong are found in Pennsylvania, New Jersey, and New York; Boyertown lies on a small lobe which drops down from the main axis of the formation. In many places the Reading Prong contains rock formations that are enriched in uranium and thorium.

IV. AERIAL MEASUREMENTS

Measurements of the total count rates and the energy spectrum of gamma radiation were made on each of the flight lines at one second intervals. The gamma rays were detected by eight thallium-activated sodium iodide, NaI(Tl), scintillation crystals mounted in external cargo pods on a Messerschmitt-Bolkow-Blohm (MBB) BO-105 helicopter. Each NaI(Tl) crystal was 40 cm x 10 cm x 10 cm in size. Helicopter ground speed was about 36 meters per second. Altitude was monitored by a radar altimeter. (Jobst, 1979; Fritzsche, 1982)

Scintillation pulses from the eight detectors were electronically summed and routed to the Radiation and Environmental Data Acquisition and Recorder (REDAR) system on board the aircraft.

The system's ability to detect an isotope on the ground depends on several factors. These include the system's geometry and the source's strength and distribution. System geometry includes effects due to the detector efficiency, collimation, and the movement of the helicopter. These combine to limit the ability of the system to precisely locate a point source. Source distribution will affect the system sensitivity because the spectra collected are averaged over the detector's field of view.

The REDAR system is composed of several microprocessor-based subsystems. The control subsystem collected and formatted gamma ray spectral data from the detector system. It also collected aircraft positional data and system live time information. Records containing four one-second data points for these parameters were stored on magnetic tape every four seconds. The tape subsystem consisted of a microprocessor and a dual magnetic cartridge digital recorder. Radiological data, along with selected operational parameters, were presented in real time on the CRT's of the on-board display subsystem.

The helicopter position over the survey site was determined by two systems: an ultrahigh frequency ranging system (URS) and the radar altimeter. The URS consisted of two remotely located transponders and an on-board interrogator. The on-board interrogator used the transit time of the UHF pulses from the transponders to obtain the distance from the aircraft to each remote unit. This information was then used to precisely determine the helicopter's position in the survey area. The radar altimeter measured the aircraft altitude above ground level in the same manner. Position and altitude information were processed in real time by the steering microprocessor. These data provided steering indications to the pilot for flying the predetermined flight lines at the desired altitude. Positional data, gamma ray spectral data, atmospheric pressure, and temperature were all recorded on magnetic tape.

The magnetic tapes with the recorded data from the aerial radiological survey were processed after each flight with the Radiation and Environmental Data Analyzer and Computer (REDAC) system. This computerized data analysis system was built into an Airstream motor home which was modified to carry the REDAC.

V. DATA ANALYSIS

Data from the helicopter flights were analyzed while in the field and the results were reported to members of the Pennsylvania DER team who were conducting the ground-based building measurements.

Data analysis for the initial low altitude survey was complicated by a broken radar altimeter. A replacement could not be obtained for several days. Half of the low altitude survey was flown without the radar altimeter. At that altitude there are many visual references for the pilot to key in on and, therefore, it was felt that the pilot could maintain the altitude with a fair degree of accuracy. It was also important that the aerial data be gotten to the DER personnel as rapidly as possible. Another major complicating factor during the survey was the weather. During the course of the survey, it snowed three times and rained heavily once.

The presence of varying amounts of snow and rainwater on the ground makes comparison of the data on a day to day basis difficult. Normally, an exposure rate contour map of an area would have been prepared. The exposure rate values would have been derived from the gross count rates due to terrestrial gamma ray emitters by summing the counts in each channel of each spectrum between 0.040 and 3.0 MeV. These results would then have been converted to exposure rates at one meter above ground by application of a predetermined conversion factor. The presence of snow or rainwater on the ground will change the spectral shape and the count rates due to attenuation. It is possible to correct for the presence of the rainwater or snow (Fritzsche, 1982) but making these corrections are time consuming and were not done. Instead, contours of standard deviations above the background were plotted. For this purpose an average count rate and the standard deviations of the count rate for each day's flight were found. These values were then used to set the contour plot levels. At the end of the survey mission, the levels from each day's flights were normalized to an arbitrary reference value to give continuity to the contour lines.

VI. RESULTS

Figure 1 shows the locations where the count rates are at least two (2) standard deviations above the background. The dashed lines indicate the approximate locations of known faults in the Reading Prong (as obtained from Geologic Map of the Precambrian Rocks and Hardyston Formation of the Boyertown Quadrangle, Berks County, Pennsylvania, Department of Internal Affairs, Atlas 197).

Many of the high count regions can be seen to correspond with the approximate locations of the faults as indicated by the geologic map and with high elevations. A number of high count regions do not seem to correspond to any indicated fault line. This may be due to the presence of unknown faults. The area of high counts which contains the Stanley Watras House is among these areas (Site 1 in Figure 1). A number of houses in this area were found to also contain elevated concentrations of radon gas by the DER ground searchers. The correlation of the high count regions with elevation is probably due to uranium bearing rocks being closer to the surface. The areas of high counts in the far western side of Figure 1 are located outside of what is considered to be the Reading Prong boundaries. Spectra of those areas which do not correspond with fault lines show only the presence of members of the uranium and thorium decay chains.

VII. SUMMARY

A portion of the Reading Prong was surveyed by air in an effort to locate areas which would have higher probabilities of containing radon with excessive radon levels. A number of these areas were located. Most of these areas corrolated with known faults in the Reading Prong. The land around the Stanley Watras house was identified as a site of increased probability of containing excess radon levels. A number of buildings in this area were subsequently identified by the DER ground searchers for further study. No information has been released on the other areas identified as being likely to contain houses having elevated radon levels because of an agreement not to divulge the information which the Commonwealth of Pennsylvania made with the owners of the buildings which were inspected.

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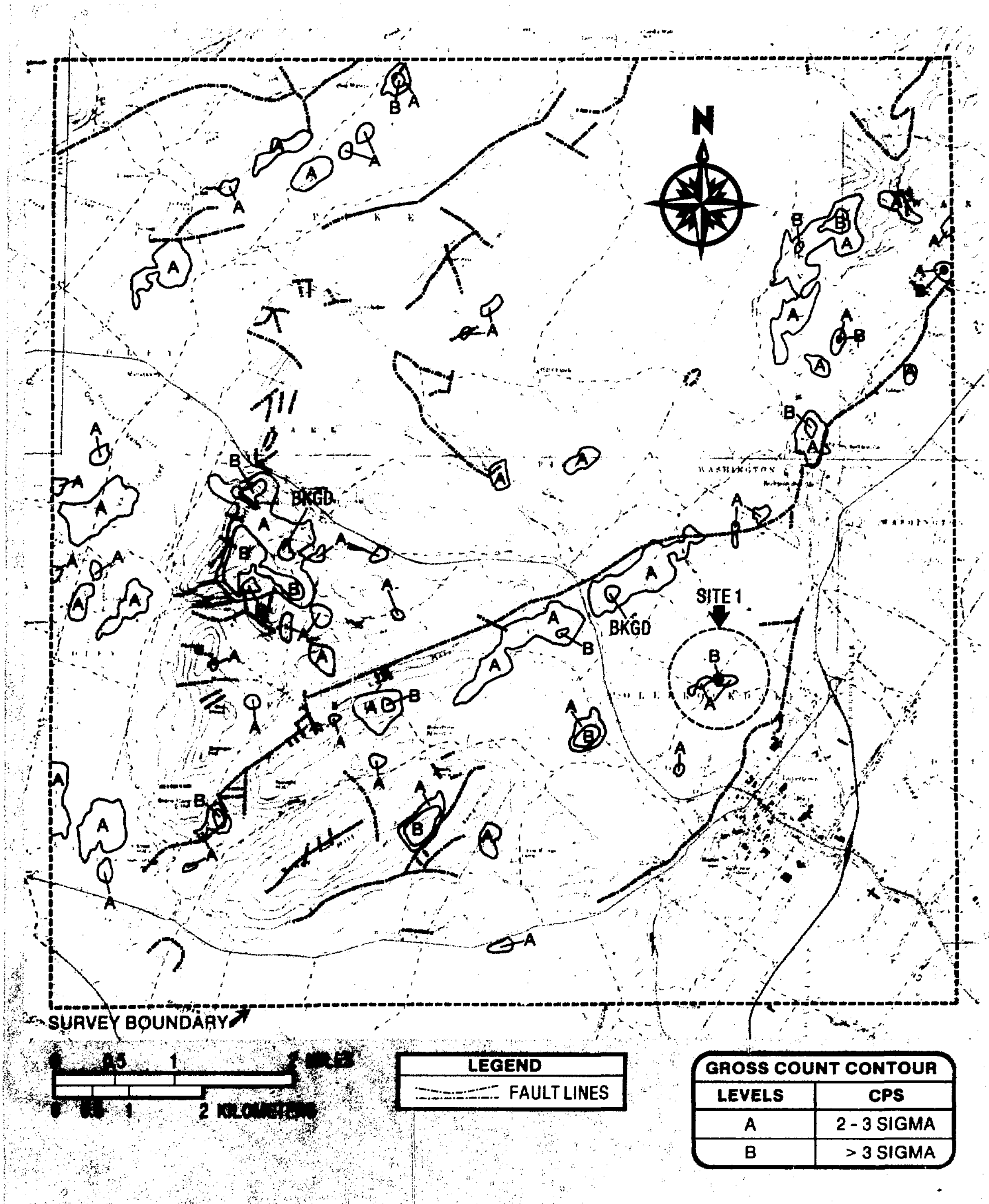

FIGURE 1

MAP OF THE BOYERTOWN PRONG SURVEY AREA. SITE 1 INCLUDES THE LOCATION OF THE WATRAS HOUSE. AREAS WITH NO LABEL ARE BACKGROUND (BKGD). THE SHOWN FAULT LINE LOCATIONS ARE APPROXIMATE.

Emergency Planners, Look Back at TMI-2

Robert L. Long

ABSTRACT. This paper looks back at emergency preparedness and response lessons learned while going through the TMI-2 Accident crisis. It is based on the author's active role as a GPU response team participant and on the results of a two-day workshop conducted by GPU Nuclear to assess the lessons learned, both from the accident and the evolution of emergency preparedness programs following the accident. Generally, the focus is on areas not covered in depth by various regulatory requirements and guidelines.

Emergency preparedness at nuclear power plants changed dramatically following the TMI-2 Accident. NRC and FEMA regulations and guidelines imposed extensive new requirements. And it is generally believed that, as an industry, we are far better prepared to handle nuclear power plant emergencies than was General Public Utilities/Metropolitan Edison in the Spring of 1979.

However at GPU Nuclear, the new GPU company which now operates the TMI and Oyster Creek Nuclear Generating Stations, we believe that a periodic look back at the events of the TMI-2 accident is an effective reminder of the bases for all the new emergency preparedness requirements. We also find that some of the important lessons we learned are not necessarily captured in guidelines and regulations.

For example, radiological support activities both on and off-site identified the difficulties of developing and maintaining high quality, retrievable records. Particularly, during the first several days of the accident, in-plant radiological conditions were changing dramatically. The survey records being kept often failed to note the date and time that measurements were made. Sometimes the description of the location was also too general to really allow other individuals to determine the source of the radiation hazard. Thus, we continue to place strong emphasis in rad tech training on the need to develop and maintain clear and accurate records.

In fact, the subject of records of all kinds was a major concern after the accident as we tried to determine the sequence of events. Thus, we emphasize the need to keep accurate logs, telephone conversation records, data sheets, check lists, etc. such that events can be recreated. Operators and technicians are reminded of the importance of placing accurate time marks on strip charts and noting chart drive speeds. Periodically we have an independent group develop the sequence of events of an emergency preparedness drill scenario using the records kept by the drill participants. This "records" scenario is then compared with the detailed drill scenario data and any problems are identified and addressed.

During the first few days of the TMI-2 accident several workers received exposures in excess of 10CFR20 allowable exposure limits. This has led to extensive attention to and practice of the post-accident radioactive liquid and gaseous sampling techniques. Since noble gas clouds are capable of giving dose rates in excess of 100 rem/hr, and primary water samples can also have very high dose rates, it is also necessary to have available teletectors and other high range portable instruments (up to 1000 rem/hr) to be able to properly perform radiation surveys under accident conditions. Appropriately these activities receive close attention from NRC inspectors and Institute of Nuclear Power Operations (INPO) review teams.

During the TMI-2 accident the ability to handle the plant transient, which lasted for a number of days, and respond to the requirements for emergency plan notifications and continuing demands for updates on plant status and radiological conditions placed unusual demands on the plant staff. One senior manager told a review committee that the plant staff had practiced extensively on handling plant transients at the simulator and on making emergency plan notifications during in-plant drills but they had never really tried to do both at the same time. Thus, our emergency preparedness drills at both the simulator and the plant require the staff to perform both functions and to recognize and use the extensive support capabilities now provided by the emergency response teams.

As we have come to understand the true extent of the damage to the TMI-2 core, we have emphasized the need for continuing awareness and understanding by plant operators, radiological control and other technicians, and management to recognize and accept the real possibility for a core-damaging accident to occur. This effort is enhanced by the use of symptom-based emergency operating procedures and simulator training which focuses attention on maintaining core cooling under a wide variety of transient scenarios.

As is true for many utilities, the majority of company engineers and technologists who provide technical support for the plant, both normally and during an emergency, are located some distance from the plant sites, in our case in Parsippany, N.J., 100 and 150 miles from the Oyster Creek and TMI sites respectively. During the first few days following initiation of the TMI-2 accident, most of these support personnel, plus many others from outside the company, moved to the plant site and set up temporary working arrangements, e.g., in "Trailer City." As discussed in a previous paper (Long, et al 1981), "the working conditions, poor living conditions, the seriousness of the situation, the pressure of the exaggerated public perception of the problems, the evolving organizational problems, separations from and the need to reassure families, and the urgent nature of the work assigned all combined to place the technical support personnel under great stress." Thus, our present emergency plan includes facilities with dedicated communications links - telephones, telecopiers, and computer terminals tied into the plant process computers - which permit and facilitate the accomplishment of the technical support by the staff while allowing them to remain in their home office and home environment. Provisions are made for 24-hour shift coverage of the support activities. Personnel have ready access to all of the needed resources - drawings, technical manuals and references, computer terminals and software packages.

It is generally acknowledged that communications with the news media and the public was very confusing and ineffective during the first several days of the TMI-2 accident. More recently the Chernobyl accident, while much more complicated with respect to accessibility by the press, had numerous examples of misinformation, exaggeration and overreaction to both the real and potential dangers from the radioactive releases. A large full-time communications staff is now engaged at GPUN in developing, fostering, and maintaining effective relationships with news media personnel and the public. After several years effort, we now have agreement for media representatives from the utility, county, state, and NRC to operate from a common, dedicated media center. This permits on-the-spot coordination and verification of information released to the media and the public. We also have trained technical spokesman assigned to the media center to work with the communications personnel at interpreting and making understandable the emergency event information.

Another related activity has been persuasion of the technical personnel, particularly operations and radiation controls, of the need to provide timely and accurate information to the communications staff in the media center. This is a particular challenge when these plant personnel are focusing essentially all of their attention on identifying the causes of the emergency and taking steps to correct and mitigate any adverse consequences. Clearly, both the TMI-2 and Chernobyl accidents emphasize the importance of providing accurate meaningful information to the media and public while the event is ongoing. This need is emphasized and practiced in every emergency drill conducted at TMI-1 and Oyster Creek Nuclear Generating Stations.

As technical support personnel from the GPU companies, as well as companies and government organizations from all over the United States, began to arrive at the TMI-2 site in the first few days following initiation of the accident, major demands were placed on security and training personnel to maintain appropriate controls while enabling rapid access to the site for these hundreds of additional personnel. It is essential to have in-place (or readily available) facilities to check-in personnel, perform rapid security checks, validate previous radiological worker training, provide rapid site familiarization training, and issue appropriate badges, key cards, and dosimetry. The need for an ability to mobilize outside resources was again clearly demonstrated during the Chernobyl accident.

In the summer of 1983, GPU Nuclear conducted a two-day workshop to review and identify the lessons to be learned from the TMI-2 accident and the whole realm of activities evolving in the aftermath of the accident. The results of this workshop, including 95 lessons learned statements and associated questions to evaluate the effectiveness of the responses to each lesson learned statement were reported in Long, 1985. One whole section of the workshop was focused on emergency preparedness. This section and the others associated with operators, radiological controls, maintenance and technical and other support are periodically reviewed to determine whether GPU Nuclear is maintaining the required readiness to respond to any emergency at our nuclear generating stations.

A recent review of these GPUN lessons learned indicates that many of them are reemphasized by the Chernobyl accident (DOE, 1986). While the Russian report is very sketchy with regards to the extent of emergency planning and effectiveness of implementation of any plans, we need to continue to seek information which will aid us in being prepared to handle any emergency which may occur at any nuclear power plant. The viability of nuclear power as a contributor to the world's future energy needs depends on our ability to look back at TMI-2, and now Chernobyl, and learn our lessons well.

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Section II

Technical Aspects

Chairman: ***Harold Lamonds***
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Source Terms Derived from Analyses of Hypothetical Accidents, 1950–1986

William R. Stratton

ABSTRACT - This paper reviews the history of reactor accident source term assumptions. After the Three Mile Island accident, a number of theoretical and experimental studies re-examined possible accident sequences and source terms. Some of these results are summarized in this paper.

Subsequent to the developments during World War II, the Atomic Energy Commission (AEC) was required to site, construct, and operate new production reactors. To obtain advice on how best to solve the policy and technical problems involved, a Reactor Safeguards Committee was created. This committee issued a Summary Report, WASH-3, (AEC, 1950) which advised the AEC on the hazards involved and on siting policies.

The assumptions in WASH-3, relative to escape of fission products (the source term), were uncomplicated; they simply assumed that most of the fission products become airborne subsequent to an accident. This was stated in one place "as practically the total fission product content of the pile," or, in another section, stated as, "assume that ...50% is present in the radioactive cloud." These releases, then, along with various atmospheric diffusion assumptions, served to define hazards as a function of distance from the reactor. Later publications (e.g., Weil, 1955; Parker, 1955; McCullough, 1955) in the early 1950s made comparable assumptions about the release of fission products. In none of these analyses was the release considered mechanistically, taking into account the chemistry of fission products, reactor design, confining buildings, etc.

In 1956, the AEC and the Joint Committee on Atomic Energy recognized the inevitable large-scale application of nuclear energy for the production of electricity, and they undertook a study to gain a more comprehensive understanding of the potential public hazards of nuclear power reactors. This study became The Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants, WASH-740, (AEC, 1957).

This study created what the Nuclear Regulatory Commission has described as hazard states rather than severe accident analyses. Three cases were considered, each arbitrary and with no discussions of sequence of events, chemistry, structural retention, etc. In Case A, the containment is assumed to remain intact (and tight), and hazard to the public derives only from direct radiation from the building. In Case B, the volatile fission products (noble gases, halogens, and 1% of the strontium) are assumed to be released from containment, but all others remain within the protective shielding. Case C is the most severe hazard state, and the release of 50% of the core inventory is postulated. These cases are summarized in Table I.

The Windscale reactor accident on October 9, 1957 (Morowitz, 1981; Clark, 1974; Stewart, 1958) had a significant influence on later studies and on the development of nuclear reactor regulations by the AEC. This United Kingdom reactor was a natural uranium metal, air-cooled, graphite-moderated, plutonium-production reactor. The temperature of the graphite in this reactor was sufficiently low that, as it absorbed neutron and gamma-ray energy, it tended to grow in volume and become distorted. This energy storage effect was first identified by Dr. Eugene Wigner and, hence, is sometimes called "Wigner energy" or "Wigner's disease."

On October 7, 1957, procedures to release the "Wigner energy" were started at Windscale, but the annealing operation did not go smoothly, as had been the case in the eight previous attempts. For a complicated set of reasons, the heating of the graphite was not controlled adequately, and by October 9 (10:00 p.m.), one graphite thermocouple indicated such a high temperature that the pile physicist was required to allow air flow through the core to cool the graphite. The situation continued to deteriorate until, on October 10, increasing amounts of radioactivity were recorded in the 400-ft stack, and the fuel was presumed to be burning. When the fire could not be controlled by injecting carbon dioxide, it was finally quenched on October 12 by flooding the reactor with water.

Because the fuel became very hot and the accident extended over several days, the escape of fission products was significant. The assessment of the amount of radioactivity that escaped to the environment is given in Table II, along with the estimated inventory in the 150 channels in the core that became hot. These data are taken from "Long-Range Travel of the Radioactive Cloud from the Accident at Windscale", (Stewart and Crooks, 1958).

Table I. WASH-740 hazard states[a]

	Case A (Contained)	Case B (Volatile Release)	Case C (Release of 50% of Inventory)
Containment	Tight	Failed	Failed
Release (Source Term)	0.0	100% Xe, Kr, I, Br; 1% Sr	50% Fission Product Inventory
Hazard	Radiation from containment	Airborne activity	Airborne activity

[a]AEC (1957).

Table II. Release during Windscale accident[a]

Isotope	Inventory (150 channels, Ci)	Release (Ci)	Percent of Inventory
^{85}Kr		1,600	
^{89}Sr	275,000	80	0.03
^{90}Sr	11,500	2	0.02
^{97}Zr		460	
^{106}Ru	32,000	80	0.25
^{129m}Te		684	
^{132}Te	161,000	12,000	7.4
^{131}I	162,000	20,000	12.3
^{131m}I		1,764	
^{133}Xe		333,600	
^{137}Cs	12,300	600	4.9
^{134}Ce	218,000	80	0.04

[a]Stewart and Crooks (1958).

The amount of iodine that escaped from the core was at least twice the listed value; between 20,000 and 30,000 Ci are mentioned as being deposited on the filters. The amount of cesium retained by the filter also was comparable to that which escaped to the environment. The estimates of this deposition were 800 to 1000 Ci. Thus, between 11.3% and 13% must have escaped from the core. The chemical form of this iodine is not known, but most was either gaseous or adsorbed on very small particles. This escape of iodine can be explained if the hot, dry, oxidizing environment is recognized.

In 1960, the Advisory Committee on Reactor Safeguards and the AEC staff (ACRS, 1960; Beck, 1960), both commenting on procedures, assumptions, and rules being developed to license nuclear reactors (and other facilities), used the phrase, "source term," for the first time. The definition given was "an arbitrary accident is assumed to occur which results in the release of fission products into the outermost buildings or containment shell. About 100% of the total inventory of noble gases, 50% of the halogens, and 1% of the non-volatile products are assumed to be released (from the fuel). It is then assumed that this mixture leaks out of the outermost barrier at a rate defined by the designed and confirmed leak rate." The recognition that iodine may be the most important element against which to construct defenses is clear.

These severe assumptions, modified only to 25% of the halogens, were incorporated into the AEC document, TID-14844, Calculations of Distance Factors for Power and Test Reactor Site (DiNunno, 1962). In time, this document became embedded in the federal regulations as a part of the rules defining the criteria to which nuclear reactors must be designed and constructed. The success and high degree of acceptance of TID-14844 has led to its use for the past 25 years and its replication worldwide.

The Reactor Safety Study, WASH-1400, (NRC, 1975) was a major step forward in accident evaluation. It clarified the meaning of risk and made quantitative estimates of the probability and consequences of each of the factors involved in an accident. The study showed conclusively that the public risk from even serious core-damage accidents is small; comparisons were made to other risks to which the public is exposed. Even though rigorous analyses were involved, where uncertainties existed, conservative assumptions were made. Important among these were the chemical forms of iodine and cesium. Evidence (thermodynamic and experimental) existed that the preferred chemical forms were cesium iodide and cesium hydroxide, but this evidence was not deemed sufficient to overcome the accepted belief that iodine and cesium would behave as gases or inert aerosols at high temperatures. This neglect of the chemistry of fission products, along with assumptions about containment capability, led to high estimates of the release of some fission products during severe core-damage accidents. Accidents of lesser severity (during which engineered safeguards survived and operated) led to much smaller releases. Essentially all the risk is derived from only the most severe core-damage accidents, even though their probabilities were very small. The full set of accidents was divided into nine release categories for pressurized water reactors (PWRs) and five release categories for boiling water reactors (BWRs). These are summarized in Table III.

The initial criticisms of WASH-1400 were in the probability estimates for the several accidents. More recently, the estimates of release fractions for different accidents have been very closely examined, and while, in general, large reductions are found, the magnitude of the change is being debated.

The accident at Three Mile Island on March 28, 1979, was severe in terms of physical damage to the reactor and trauma to the public, but the release of fission products, except for noble gases, was very small (Kemeny, 1979). The accident was triggered at 4:00 a.m. by failure of

Table III. WASH-1400 accident release categories[a]

Release Cat.	Prob./ Reactor-Yr.	Fraction of Core Inventory Released to the Environment						
		Xe-Kr	I	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
PWR1	9E-7[b]	0.9	0.7	0.4	0.4	0.05	0.4	3E-3
PWR2	8E-6	0.9	0.7	0.7	0.3	0.06	0.02	4E-3
PWR3	4E-6	0.8	0.2	0.2	0.3	0.02	0.03	3E-3
PWR4	5E-7	0.6	0.09	0.04	0.03	5E-3	3E-3	4E-4
PWR5	7E-7	0.3	0.03	9E-3	5E-3	1E-3	6E-4	7E-5
PWR6	6E-6	0.3	8E-4	8E-4	1E-3	9E-5	7E-5	1E-5
PWR7	4E-5	6E-3	2E-5	1E-5	2E-5	1E-6	1E-6	2E-7
PWR8	4E-5	2E-3	1E-4	5E-4	1E-6	1E-8	0	0
PWR9	4E-4	3E-6	1E-7	6E-7	1E-9	1E-11	0	0
BWR1	1E-6	1.0	0.4	0.4	0.7	0.05	0.5	5E-3
BWR2	6E-6	1.0	0.9	0.5	0.3	0.1	0.03	4E-3
BWR3	2E-5	1.0	0.1	0.1	0.3	0.01	0.02	3E-3
BWR4	2E-6	0.6	8E-4	5E-3	4E-3	6E-4	6E-4	1E-4
BWR5	1E-4	5E-4	6E-11	4E-9	8E-12	8E-14	0	0

[a]The data are taken from NRC (1975), App. V, p. V-4.
[b]The common convention is used, i.e., 3×10^{-6} = 3E-6.

the feedwater system which in turn caused the turbine-generator to shut down, the PORV (relief valve on the pressurizer) to open, and the reactor to scram. The drop in system pressure caused the high-pressure injection system to start in 2 minutes, but the operators misinterpreted the information available to them and turned this system off at 4 minutes and 38 seconds. Some of the confusion was caused by closed valves in the auxiliary feedwater system (in violation of technical specifications). These valves were opened at 8 minutes, thus restoring a heat sink for the primary system, but the emergency cooling was not restored to operation for 3 hours and 20 minutes. The main coolant pumps were turned off at 1-1/4 and 1-3/4 hours.

By 6:00 a.m. fission products (probably Xe, Kr) were detected in the containment building, at 6:54 a.m. in the auxiliary building, (causing a station alarm), and by 7:24 a.m. they were detected offsite (causing a general alarm to be declared). Within the reactor vessel, the situation worsened monotonically for about 3 hours (until 7:00 a.m.) when one main coolant pump was activated briefly. Following this injection of water, the condition of the core again worsened steadily until 3 hours and 20 minutes into the accident (7:20 a.m.), when the high-pressure injection system was restarted. We now know that much of the core disintegrated and some fuel melted or liquified a few minutes before this time.

Fission products escaping from the hot fuel into the water (or steam) in the core were carried in the primary system flow. Some remained, some were carried into auxiliary building tanks by way of the let-down system, and some escaped into containment by way of the PORV. Between 2.5×10^6 and 13×10^6 Ci of xenon-133 (inventory 154×10^6) were released from the plant, but only about 15 Ci (inventory 64×10^6) of iodine were detected offsite. No metallic fission products were found outside the containment and auxiliary building (Kemeny, 1979).

The fact that a severe fuel-damage accident occurred in a commercial nuclear power station, the incredibly wide publicity given to the event, and the surprisingly low release of iodine (and no metals) relative to the noble gases, stimulated the technical community to create a research program of unprecedented magnitude. In the United States, this program has been funded by the Nuclear Regulatory Commission through the national laboratories and their contractors, through the Electric Power Research Institute (EPRI), the Industry Degraded Core Rulemaking Program (IDCOR), and through programs funded by single utilities (New York Power Authority (NYPA)/Risk Management Assoc.) and by architect engineering firms [Stone and Webster Engineering Corp.(SWEC)]. Major safety-oriented experiments have been completed at the Argonne, Sandia, and Idaho National Laboratories, and theoretical studies have been done by several organizations.

The theoretical studies have had the objective of reexamining the risk dominant accident sequences found in WASH-1400 and associated with PWR-1,2,3, and BWR-1,2,3. These sequences involve postulated failures of piping, of electric power, and selective failure of the engineered safeguards (containment, containment spray, cooling, emergency core cooling, etc.). Because of redundancy and diversity, the probabilities are low, but a de minimus probability has not yet been defined and accepted. The reexamination of these sequences has been underway for several years.

A sampling of theoretical results from the major U.S. study centers is given below. PWR data are given first for several reactors from various organizations, with BWR results following. The notation insofar as is possible is that of WASH-1400. Some explanatory notes are added to clarify some of the inevitable opaqueness. Generally, the tables are not a complete representation of the organization's efforts, and contain only a representative sampling.

Table IV summarizes a recent EPRI publication relative to the Surry reactor. The S2Cδ sequence did not lead to core uncovering.

Table V contains results that the New York Power Authority and Risk Management Associates calculated for the Indian Point-III plant.

The IDCOR program (IDCOR; 1984, 1986) has reexamined a number of reactors and sequences in detail. Some of their data are shown in Table VI. The drains referred to are those between the two volumes in the ice condenser. The IDCOR program has been underway since 1981 and is drawing to a close this year.

Table IV. Calculated source terms for Surry (PWR)[a]

Fission Product Group	Fraction of Core Inventory Released to Environment During Accident Sequences		
	TMLB'[b]	V[c]	$S_2C\delta$[d]
Xe-Kr	4E-3	1.0	0.0
I-Br	2E-3	6E-2	0.0
Cs-Rb	2E-3	6E-2	0.0
Te-Sb	4E-4	2E-2	0.0
Ba-Sr	2E-4	8E-3	0.0
Ru	5E-7	2E-5	0.0
La	5E-7	2E-5	0.0

[a]Ritzman et al. (1985).
[b]Failure of secondary system steam relief valves, power conversion system, and auxiliary feedwater system, with failure to recover either onsite or offsite electric power within about 1-3 hr following initiating transient which is loss of power.
[c]Dry; low pressure injection system check valve failure. Flooding in Safeguards Building ignored; deposition in building included.
[d]Small loss of coolant accident (equivalent diameter 0.5-2 in) with failure of containment spray injection system and containment failure due to overpressure. Core remained covered, spray operative. Same conclusion in Silberberg et al. (1986), p. 4-57.

Table V. Caculated source terms for Indian Point (PWR)[a]

Fission Product Group	Fraction of Core Inventory Released to Environment During Accident Sequences		
	AB[b]	TMLB'[c]	TMLB'[d]
Kr-Xe			
CsI	3E-6	2E-5	4.2E-6
CsOH	9E-6	1.7E-5	2.2E-6
Te	1.3E-6	6.5E-9	3.3E-9

[a]These studies were completed in 1983-1984 and quoted in ANS (1984). Reexamination with improved codes is planned.
[b]Intermediate to large loss of coolant accident with failure of electric power to engineered safety features.
[c]Failure of secondary system steam relief valves, power conversion system, and auxiliary feedwater system, with failure to recover either onsite or offsite electric power within 1-3 hr following an initiating transient which is loss of offsite power.
[d]Pump seal loss of coolant accident with other characteristics as in footnote c.

VI. Calculated source terms for Zion and Sequoyah (PWRs)[a]

Fission Product Group	Fraction of Inventory Released to Environment During Accident Sequences							
	Zion			Sequoyah				
	TMLB'-δ_ℓ[b]	TMLB'-β[c]	V[d]	TMLB'-δ[e]	V[d]	S_2HF-δ_ℓ[f]	S_2HF-δ_ℓ[g]	S_2HF-β[h]
Xe-Kr	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
I-Br	1.7E-3	1E-2	8E-5	5E-4	1E-4	2E-5	1E-5	1.6E-2
Cs-Rb	1.7E-3	1E-2	8E-5	6E-4	1E-4	7E-5	3E-5	1.6E-2
Te-Sb	2E-5	3E-4	8E-5	3E-5	1E-4	1E-5	2E-5	4-E3
Ba-Sr	1E-5	6E-4	5E-5	1E-5	1E-5	1E-5	1E-5	5E-4
Ru	1E-5	6E-4	1E-5	1E-5	1E-5	1E-5	1E-5	5E-4
La	--	--	--	--	--	--	0	0

[a]Data taken from IDCOR (1986).
[b]Failure of secondary steam relief valves, power conversion system, and auxiliary feed water system, with failure to recover either onsite or offsite electrical power within about 1-3 hr following the initiating transient, which is a loss of offsite AC power. Containment failure due to overpressure.
[c]Same sequence as described in footnote b with containment failure resulting from inadequate isolation of containment openings and penetrations.
[d]Low pressure injection system check valve failure.
[e]Same sequence as in footnote b except containment failure due to overpressure.
[f]Small loss-of-coolant accident (equivalent diameter of 0.5-2 in) with failure of emergency core cooling recirculation system and containment spray recirculation systems. Containment failure due to overpressure. Drains blocked.
[g]Same sequence as described in footnote f with drains open.
[h]Same sequence as described in footnote f with containment failure resulting from inadequate isolation of containment openings and penetrations. Drains open.

Table VII is a small sample of the PWR work completed by the Stone and Webster Engineering Corporation. In addition, a number of parametric studies of the effects of different physical and chemical parameters and their importance in source terms evaluation has been published in the ANS Report of the Special Committee on Source Terms (ANS, 1984).

Table VII. Calculated source terms for Surry (PWR)[a]

Fission Product Group	Fraction of Core Inventory Released to the Environment During Accident Sequences			
	AB-β[b] 1 ft²	AB-δ[c] 1 ft²	TMLB'-β[d] 1 ft²	V[e]
Kr-Xe				
I	6.4E-2	4E-4	1.5E-2	1E-2
Cs	6.3E-2	4E-4	1.3E-2	1E-2
Te	3.9E-2	1.9E-3	6.4E-2	6.5E-3

[a]Data taken from ANS (1984).
[b]Intermediate to large loss of cooling accident with failure of electric power to engineered safety features. Containment failure results from inadequate isolation of containment openings and penetrations.
[c]Same sequence as footnote b, except containment failure at 24 hrs. due to overpressure.
[d]Failure of secondary steam relief valves, power conversion system, and auxiliary feedwater system, with failure to recover either onsite or offsite electric power within about 1-3 hr following an initiating transient which is a loss of offsite AC power. Containment failure results from inadequate isolation of containment openings and penetrations.
[e]Low pressure injection system check valve failure. A decontamination factor of 50 was assumed for piping and water in the safeguards building. Decontamination by building structures was not considered.

Table VIII. Calculated source terms for Surry (PWR)

Fission Product Group	Fraction of Core Inventory Released to the Environment During Accident Sequences						
	S_1G[a,b] S_2G S_3G	AG[c]	TMLB'[d,e]	TMLB'[e,f]	V[d,g]	V[d,h]	S_2C[d,i]
Xe-Kr							
I	0.0	0.57	0.2	4.6E-2	0.3	5E-2	0.0
Cs	0.0	0.57	0.2	3.9E-2	0.3	4E-2	0.0
Te	0.0	0.47	0.1	0.11	6E-2	9E-3	0.0
Sr	0.0	4.7E-3	2E-2		5E-3	1E-3	0.0
Ru	0.0	8E-7	3E-3		2E-7	4E-8	0.0
La	0.0	1.6E-4	1E-4		3E-4	5E-5	0.0

[a]Data from Denning et al. (1986).
[b]Small loss of coolant accidents (equivalent diameters less than about 6 in.) with failure of the containment heat removal system. Containment is tight; no failure predicted.
[c]Intermediate to large loss of coolant accident with failure of containment heat removal system. Steam generator fails due to hot leg break. Containment fails at 50 hr.
[d]Data from Silberberg et al. (1986).
[e]Failure of secondary system steam relief valves, auxiliary feedwater system, and power conversion system with failure to recover either onsite or offsite electric power within about 1-3 hr following an initiating transit which is loss of offsite AC power. Containment fails at 2.5 hr.
[f]Data from BMI (1984).
[g]Low pressure injection system check valve failure (dry).
[h]Low pressure injection system check valve failure (wet).
[i]Small loss of coolant accident (equivalent diameter of 0.5-2 in) with failure of containment spray injection system. Containment is tight; no failure predicted.

Tables VIII and IX show a small fraction of the results obtained by the NRC through its contractors, Battelle Memorial Institute (BMI) and Sandia National Laboratory, for the Surry and Zion reactors. The S_1G, S_2G, S_3G and S_2C sequences for Surry are found to be negligible, but other Surry sequences are significant in their evaluation and, indeed, are found to be comparable to WASH-1400 results.

The recent Risk Management Associates examination of the Fitzgerald BWR-MkI plant, sponsored by EPRI and NYPA, is shown in Table X.

The IDCOR examinations of the Peach Bottom BWR-I and the Grand Gulf BWR-MkIII are presented in Tables XI and XII.

The BWR-MkII has not been studied in the detail that other designs have been examined. However, SWEC has examined the Shoreham BWR-MkII, and some of the results are shown in Table XIII. Additional studies have been completed by BMI and NYPA, but these were not received in time to be included in this report.

Finally, the recent reexamination of the Peach Bottom BWR-MkI by the NRC, BMI, and Sandia is presented in Table XIV.

The dichotomy in source term results from different investigating organizations is apparent from examination of the tables. The suspicion is that the differences derive both from computer code differences and from boundary condition assumptions (hydrogen deflagration, steam explosions, core-concrete interactions, and the postulated direct heating phenomenon). These conjectures, however, remain to be investigated.

Within the past two and one-half years, at least three reviews or summaries of the state of knowledge have appeared. These are:

- The Report of the Special Committee on Source Terms, American Nuclear Society (ANS, 1984).
- Report to the American Physical Society of the Study Group on Radionuclide Release from Severe Accidents at Nuclear Power Plants (APS, 1985, presented to the Nuclear Regulatory Commission, February, 1985).

Table IX. Calculated source terms for Zion (PWR)[a]

Fission Product Group	Fraction of Core Inventory Released to the Environment During Accident Sequences			
	S_2DC[b]	S_2DCF_1[c]	S_2DCF_2[d]	TMLU[e]
Xe-Kr				
I	3E-8	0.22	2.6E-2	5.7E-3
Cs	1.6E-5	0.23	2.7E-2	6E-3
Te	1.9E-3	0.32	3.4E-2	4E-2
Sr	9E-4	3.7E-2	2.5E-3	9E-5
Ru	2.6E-8	1.7E-4	5E-7	4E-7
La	5E-6	1.8E-3	8E-5	4E-7
Ce	1.3E-5	6.4E-4	6E-5	2E-8
Ba	7.3E-4	2.8E-2	2E-3	2E-3

[a]Data from Denning et al. (1986)
[b]Small loss of coolant accident (equivalent diameter of 0.5-2 in) with failure of emergency core cooling injection system and containment spray injection system. Containment failure at 24 hr.
[c]Small loss of coolant accident (equivalent diameter of 0.5-2 in) with failure of emergency core cooling injection system, containment spray injection system, and containment spray recirculation system. Containment failure at 2.2 hr caused by hydrogen burn or direct heating.
[d]Same sequence as in footnote c except containment failure occurs at 14.9 hr due to hydrogen burn or overpressurization.
[e]Transient event with failure of secondary steam relief valves, auxiliary feedwater system, and power conversion system with loss of emergency core cooling system and inoperability of sprays or coolers. Containment failure at 3.2 hr caused by direct heating.

Table X. Calculated source terms for Fitzgerald (BWR)[a]

Fission Product Group	Fraction of Core Inventory Released to the Environment During Accident Sequences			
	AE-γ[b]	TC[c]	TC[d]	TQUV[e]
Kr-Xe				
CsI	5.6E-2	4.9E-5	0.0	1E-5
CsOH	3E-4	4.4E-5	0.0	1E-5
Te	3E-4	1.1E-5	0.0	1E-5
Sr	2E-4	1.0E-4	0.0	1E-7
Ru	1E-7	9.8E-6	0.0	1E-7
La	4.5E-7	2.2E-6	0.0	1E-7

[a]Bieniarz and Deem (1986) and Ritzman et al. (1985).
[b]Rupture of reactor coolant boundary with equivalent diameter greater than 6 in with failure of emergency core cooling injection. Containment failure at 501 min due to overpressure. Release is through reactor building.
[c]Transient event with failure of reactor protection system. Vent or assumed failure of containment occurs at 75 psi, 65 min.
[d]Same sequence as footnote c, with no vent.
[e]Transient event with failure of normal feedwater system to provide core make-up water, plus failure of high pressure coolant injection, reactor core isolation cooling, and low pressure emergency core cooling system to provide core make-up water. Containment fails at 902 min.

- o Nuclear Reactor Accident Source Terms, Report by a Nuclear Energy Agency Group of Experts (NEA, 1986).

The last and most recent of these covered information from Canada, France, the Federal Republic of Germany, Italy, Japan, Sweden, the United Kingdom, and the United States (the many organizations mentioned in this summary). This review report identified a number of (1) main developments since WASH-1400, (2) areas where source term information is sufficient, and (3) uncertainties and areas of disagreement between studies. The most important points of these three categories are listed below:

1. Main developments since WASH-1400
   - A. Source terms were overestimated in the past for most accidents.
   - B. Source terms technology is more complex than the treatment in WASH-1400 indicates, and complete generalization is not possible.
2. Areas where source term information is sufficient:
   - A. In-vessel steam explosions of sufficient magnitude to fail both the RPV and the containment are very unlikely.
   - B. Some containments are stronger than previously thought.
   - C. Experimental release data for the non-volatile fission products are limited; the impact of data is unlikely to be important to the final source terms.
   - D. The most probable chemical compounds for iodine and cesium have been identified.
3. Uncertainties and areas of disagreement:
   - A. The several source term studies had divergent views as to the implications of the present source term predictions.
   - B. The inadvertent omission of possibly important phenomena.
   - C. The boundary and initial conditions related to specifications of the accident sequences and the plant geometry.
   - D. Assumptions relating to application of computer codes.

Table XI. Calculated source terms for Peach Bottom (BWR)[a]

Fission Product Group	Fraction of Core Inventory Released to the Environment During Accident Sequences						
	TW[b]	BC[c]	BC[d]	BC[e]	BC[f]	TQVW[g]	S_1E[h]
Xe-Kr	1.0	1.0	1.0	1.0	1.0	1.0	1.0
I-Br	0.2	0.1	3E-2	3E-2	6E-4	5E-2	4E-2
Cs-Rb	0.2	0.1	3E-2	3E-2	6E-4	5E-2	4E-2
Te-Sb	0.1	0.1	6E-2	4E-3	4E-4	6E-2	6E-2
Ba-Sr	4E-4	4E-4	1E-4	8E-5	4E-6	8E-5	1E-5
Ru	6E-4	1E-3	2E-4	3E-4	1E-5	1E-4	2E-5

[a]Data from IDCOR (1984), p. 10-6.
[b]Transient event with failure to remove residual core heat. No operator action taken although operator actions can preclude any release.
[c]Failure of power to engineered safety features and failure of the reactor protection system. No operator action taken.
[d]Same sequence as footnote c, except operator vents through wetwall when drywall pressure reaches 115 psia.
[e]Same sequence as footnote c, except operator refills condensate storage tank to provide continuous CRD flow.
[f]Same sequence as footnote c, except operator both vents through wetwall and refills the CST.
[g]Transient event with failure of normal feedwater system and the low pressure emergency core cooling system to provide core make-up water and failure to remove residual core heat.
[h]Small pipe break (equivalent diameter of about 2-6 in) with failure of emergency core cooling injection.

E. Details of aerosol modeling in containment.
F. Assumptions relative to steam spikes, steam explosions, hydrogen deflagration, and the postulated direct heating phenomenon.
G. Operator actions can affect the magnitude of the source term significantly. The operator actions that can be accepted for regulatory purposes have not been defined.

In final summary: In view of the enormous effort put forth in the past half dozen years, and now approaching a climax, one must ask the question, how can the community of nuclear reactor safety specialists arrive at a consensus on such an important topic as the magnitude of the predicted source term (relative to public health and safety), following serious core-damage accidents? The question is far too important to remain unresolved.

ADDENDUM (September, 1986)

The accident at the Chernobyl plant in the Soviet Union on April 26, 1986, has been the most serious, by far, in terms of damage to the plant, magnitude of the source term, health consequences

Table XII. Calculated source terms for Grand Gulf (BWR)[a]

Fission Product Group	Fraction of Core Inventory Released to the Environment During Accident Sequences			
	TQW[b]	TC[c]	AE[d]	TQUV[e]
Xe-Kr	1.0	1.0	1.0	1.0
I-Br	3E-4	8E-4	1E-5	7E-5
Cs-Rb	3E-4	8E-4	1E-5	7E-5
Te-Sb	2E-4	8E-4	1E-5	3E-5
Sr-Ba	1E-5	1E-5	1E-5	1E-5
Ru-Mo	1E-5	1E-5	1E-5	1E-5

[a]Data from IDCOR (1984), p. 10-11.
[b]Transient event with failure of normal feedwater system to provide core make-up water and failure to remove residual core heat.
[c]Transient event with failure of the reactor protection system.
[d]Rupture of reactor coolant boundary with an equivalent diameter of greater than 6 in and failure of emergency core cooling injection.
[e]Transient event with failure of normal feedwater systems to provide core make-up water and failure of high pressure coolant injection, reactor core isolation cooling, and low pressure emergency core cooling system to provide core make-up water.

Table XIII. Calculated source terms for Shoreham (BWR)[a]

Fission Product Group	Fraction of Core Inventory Released to the Environment During Accident Sequences			
	AD[b]	AE[c]	TW[d]	TC[e]
Kr-Xe				
I	<0.1	<0.1	<6E-2	<2E-2
Cs	<0.1	<0.1	<6E-2	<2E-2
Te	<0.1	<0.1	<6E-2	<2E-2

[a]Data from Warman (1985).
[b]Rupture of reactor coolant boundary with an equivalent diameter of greater than 6 in and failure of vapor suppression. There is in-leakage to reactor building for 13 hr.
[c]Rupture of reactor coolant boundary with an equivalent diameter of greater than 6 in and failure of emergency core cooling injection.
[d]Transient event with failure to remove residual core heat. Retention or suppression based on an assumed decontamination factor of 100.
[e]Transient event with failure of the reactor protection system. Retention or suppression based on an assumed decontamination factor of 100.

Table XIV. Calculated source terms for Peach Bottom (BWR)[a]

Fission Product Group	Fraction of Core Inventory Released to the Environment During Accident Sequences					
	TC-1[b]	TC-2[c]	TC-3[d]	TB-1[e]	TB-2[f]	V[g]
Xe-Kr	1.0	1.0	1.0	1.0		
I	3E-2	1.3E-2	6E-4	1.2E-2	3.6E-2	0.46
Cs	3.3E-2	1.4E-2	9.6	1.4E-2	4.1E-2	0.44
Te	0.26	8.7E-2	6.5E-3	0.22	0.23	0.26
Sr	0.49	0.3	8.2E-3	0.37	0.46	0.24
Ru	3.5E-7	2E-6	1.3E-7	6E-7	7E-7	1.4E-6
La	1.2E-2	9E-3	3.2E-4	3E-2	3.5E-2	1.3E-2
Ce	2E-2	1.5E-2	4.9E-4	4.8E-2	3.5E-2	2.3E-2
Ba	0.39	0.19	6.2E-3	0.28	0.34	0.2

[a]Data from Denning et al. (1986).
[b]Transient event with failure of reactor protection system. Containment fails at 85 min.
[c]Same sequence as in footnote b except containment failure occurs at 126 min.
[d]Same sequence as in footnote b. Containment is vented at 88 min.
[e]Transient event with failure of power to the reactor protection system. Containment fails at 914 min.
[f]Same sequence as in footnote e except containment fails at 736 min.
[g]Failure of low pressure emergency core cooling system to provide core make-up water.

to station personnel, and integrated radiation dose away from the plant site. The RBMK reactor is graphite-moderated and cooled with boiling water at about 1000 psi. The design is such that a significant positive coefficient of reactivity exists for the removal of water (or steam) from the vertical fuel-coolant flow channels.

The Soviet Report on the Chernobyl Accident (State Committee, 1986) describes the preparations for an experiment. Apparently, these preparations led to a condition of the reactor with little or no boiling in the fuel-coolant channels. A reduction of flow at the same time that the experiment was started led directly to a reactivity induced autocatalytic power excursion. The Soviet estimate of the source term by May 6, 1986, is given in Appendix 4, Table 4.14, page 21 of the DOE translation. They estimate that 20% of the iodine, 13% of the cesium, 5-6% of the barium, and 3-4% of everything else escaped from the plant. Some postulated conditions during the accident are such that it would be reasonable to expect some of the cooling water and fuel to have been ejected vertically at high velocity.

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Radiological Source Term Estimation Methods

P. C. Owczarski, M. Y. Ballinger, J. Mishima, and J. E. Ayer

ABSTRACT Pacific Northwest Laboratory is developing methods for estimating the amount and size distribution of aerosols generated from potential accidents in nuclear fuel cycle facilities. These accidents include fires, explosions, spills, pressurized releases, tornadoes, and criticalities. Experiments have been performed investigating releases from fires, some types of explosions, and spills. Using data from the experiments and information from published literature, models and computer codes are being developed to estimate airborne source terms from all the accidents listed above. These methods will be part of an Accident Analysis Handbook to be published in 1987.

I. INTRODUCTION

Pacific Northwest Laboratory (PNL)[a] is in the final stages of a 7-year project[b] to develop tools for estimating airborne source terms from accidental fires, explosions, spills, tornadoes, and criticalities in nuclear fuel cycle facilities. New experimental data, models, and computer codes have been developed to help estimate source terms. The final product of this project will be an Accident Analysis Handbook to be published in 1987 along with supporting documents.

Two other laboratories are participating in developing project tools. Martin Marietta Corporation at Oak Ridge National Laboratory has developed tools for assessing UF_6 accidents, and Los Alamos National Laboratory has developed codes and experimental data for the transport and deposition of airborne materials in filters and building ventilation systems.

[a]Pacific Northwest Laboratory is operated for the U.S. Department of Energy by Battelle Memorial Institute.

[b]Work supported by the U.S. Nuclear Regulatory Commission under Contract DE-AC06-76RLO 1830, NRC FIN B2481.

This paper briefly describes experimental results and modeling efforts developed in this program.

II. FIRES

PNL has extensively studied accidental fires involving radioactive material. Experimental data were generated from small-scale fires of combustible materials contaminated with depleted uranium dioxide (DUO) or uranyl nitrate hexahydrate (UNH). Models developed from these and earlier release experiments were programmed into a compartment fire code (FIRIN). The following section briefly describes the experiments and code.

A. Data

Experiments were performed in which polymethylmethacrylate (PMMA), polystyrene (PS), polychloroprene (PC), paper, kerosene/tributyl phosphate (TBP), and mixtures of paper/PC and paper/PMMA were burned. Uranium in both the powder (DUO) and liquid (UNH) forms was used as the contaminant. Other parameters that were varied in the experiments included air flow, heat flux, and oxygen concentration. PMMA and PS gave the greatest release (up to 4% of the source contaminant). Most of the release occurred while the material was melting and bubbling, before flaming combustion. The smallest releases (up to 1%) came from burning paper and appeared to be emitted throughout the burn (Halverson and Ballinger, 1984).

A mixture of 30% TBP in kerosene was burned in the presence of aqueous nitric acid solutions with uranyl nitrate dissolved in either phase initially. The weight percent of airborne uranium ranged from 0.2 to 7.1 with the highest releases coming from a burn of an unloaded organic layer over acid with uranium. In all configurations, the mass rate of airborne uranium seemed proportional to the mass rate of smoke made airborne.

The size distribution of smoke and radioactive particles made airborne in the fire experiments was measured using a cascade impactor. Size distributions of smoke particles were all within close range of each other. However, the size distribution of radioactive particles varied widely depending on the fuel type.

B. Models

Dynamic models describing an accidental fire with time-dependent formation of airborne radioactive aerosols have been programmed into the FIRIN code (Chan et al., 1985). This code is basically a compartment fire model. Heat and mass transfer are modeled to provide input to a ventilation code (FIRAC). Compartment pressure, the airborne smoke and radioactive particle characteristics, hot-layer characteristics, aerosol deposition on compartment surfaces, and aerosol flow out of the compartment are calculated as the fire progresses. Aerosol deposition can occur by settling, Brownian diffusion, thermophoresis, and diffusiophoresis. The compartment decontamination factor for fire aerosols can typically range from 1.0 to 10. Filter resistances are calculated as a function of smoke loading. Source terms from heated surfaces, hot overpressurized vessels, boiling liquids, and spills are also calculated in FIRIN.

III. EXPLOSIONS

Explosions have been classified into four types: fast and slow physical and fast and slow chemical explosions. The likelihood of fast physical explosions in fuel cycle facilities appears to be remote (Halverson and Mishima, 1986). Thus, source term calculation methods were not developed for this type of explosion. The TNT-equivalent concept provides a simple solution to the energy release characteristics for fast chemical explosions. Mass release information for metals and liquids is available from the model of Steindler and Seefeldt (1980). For slow chemical explosions, the methods used for fast chemical explosions can be used if applied carefully. A series of pressurized release experiments were performed to provide data on some types of slow physical explosions. Data from these experiments and information in published literature have been used to model the other three types of explosions. Pressurized release data and resulting explosion models are described below.

A. Data

Methods to estimate source terms from explosions were supported by experimental releases of pressurized powders and liquids. Pressure was applied using pressurized air, CO_2 or direct heat. Results of air pressurization experiments were reported by Sutter (1983).

During the air pressurization experiments, 100 g of DUO (1 μm dia) and TiO_2 (1.7 μm dia) powders and 100 cm^3 and 350 cm^3 of uranine and uranyl nitrate solutions were released at pressures ranging from 50 to 500 psig. The largest percent of powder made airborne was 24%. The maximum amount of airborne liquid source was significantly less, about 0.15%.

Experiments in which CO_2 was used as the pressurizing gas for liquid pressurized releases gave a larger source term than comparable air pressurized experiments. Maximum weight percent airborne for CO_2/uranine releases was 0.22. Flashing sprays in which uranine solution was heated before release gave up to 9% airborne, the greatest of any of the liquid pressurized releases.

B. Models

Models are being developed for pressurized releases of powders and liquids. For powders, the model calculates the drag force on a sphere of powder as it moves upward and grows after initial release. The amount airborne is assumed to be proportional to the drag force, and the proportional constant is a function of material properties and controlling parameters. Liquid pressurized releases produce aerosol on release of dissolved gases. The quantity and size distribution of these aerosols also are functions of controlling parameters (pressure, viscosity, surface tension, density, and gas solubility).

For other slow physical explosions, a variety of standard fluid-flow equations are sufficient to calculate energy and mass flow from ruptured vessels. The hoop-stress equation (Halverson and Mishima, 1986) for ruptured pressure vessels provides a good estimate. The isothermal expansion equation provides a conservative estimate of the total energy release in a rupture-type event.

Models for fast chemical explosions are detailed by Halverson and Mishima (1986). Releases from slow chemical explosions can be calculated using the fast chemical explosion models, in some cases, or a universal correlation.

A universal correlation, which can be used for estimating explosion releases, was devised from spill, pressurized release, and explosion data (Halverson and Mishima, 1986). This correlation is the upper boundary of the airborne fraction of material at risk as a function of the energy available for aerosol formation per unit mass of material at risk.

The bounding equation is

$$\log (\text{wt\% airborne}) = -2.6 + \left[18.8 \log\left(\frac{E}{M_o}\right) - \left(\log \frac{E}{M_o}\right)^2 - 67.2\right]^{1/2}$$

E = energy, erg
M_o = source mass, g

From the numerous subsets of data, other best-fit correlations are provided. Figure 1 is a plot of the above equation along with the data subsets.

IV. SPILLS

The lower boundary accidental release event is a free-fall spill of radioactive powder or liquid in static air. Experiments performed at

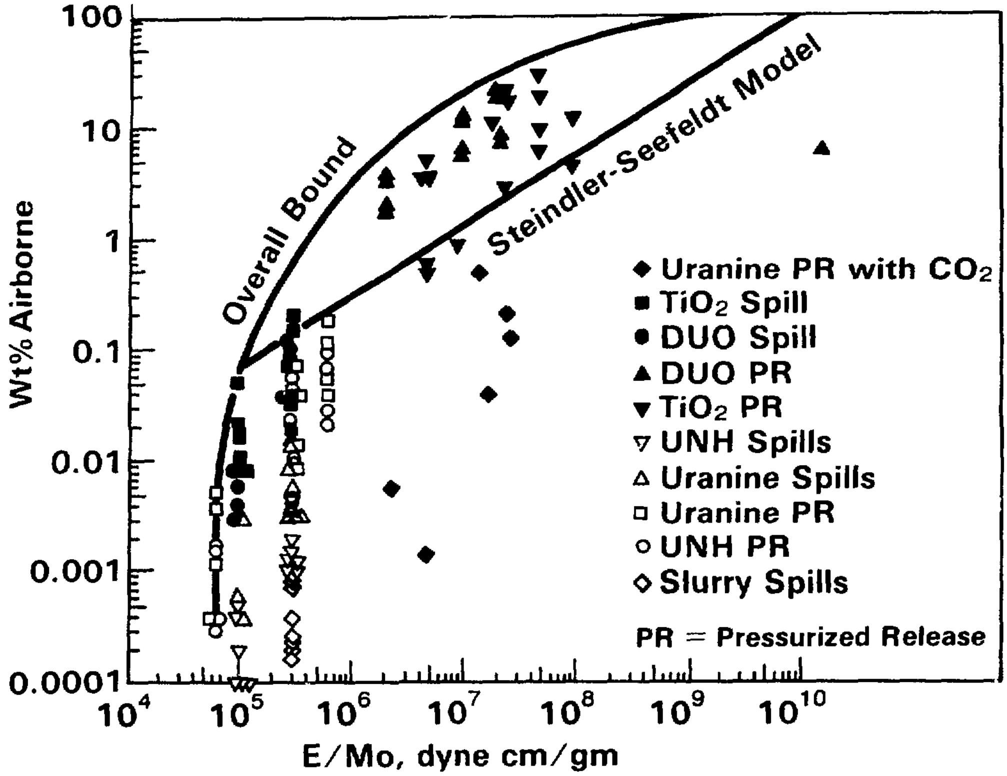

Figure 1. PNL Experiment Results and Curve Fits

PNL measured the mass airborne and particle-size distribution of spill aerosols for various source sizes, source types, and spill heights. These experiments and modeling efforts resulting from them are described in this section.

A. Data

For the first set of experiments, two powder and two liquid sources were used: TiO_2 and DUO and aqueous uranine and UNH (Sutter et al., 1981). For the source powders used (quantities from 25 to 1000 g and fall heights of 1 m and 3 m), the maximum source airborne was 0.12%. The maximum source airborne was an order of magnitude less for the liquids (with source quantities ranging from 125 to 1000 cc at the same fall heights).

Subsequent experiments investigated spills of solutions varying viscosity and surface tension as well as slurries. Weight percent airborne for these experiments was less than half that of previous liquid spills.

The median aerodynamic equivalent diameters for collected airborne particles generated from spills ranged from 1 to 36 μm. All of the spills produced a significant fraction of respirable particles measuring 10 μm dia and less.

B. Models

Models for fractions made airborne in spills are being correlated with controlling parameters. For liquids at low viscosity, the airborne fraction correlates well with a dimensionless combination of liquid density, surface tension, spill height, and liquid viscosity. All liquid spills correlate well with spill height and liquid viscosity.

Powders were dispersed while falling, producing aerosols; in contrast to liquid aerosols formed on impact. A spill model similar to the pressurized release model for powders was developed. The drag force on the volume of powder is calculated as the powder falls. The amount airborne is assumed proportional to the drag force. The Galileo number given below is used to calculate the proportional constant.

$$Ga = \frac{6\ g\ M}{\pi\ \rho_p} \left(\frac{\rho_a}{\mu_a}\right)^2$$

where Ga = Galileo number
g = acceleration of gravity
M = powder mass
ρ_p = powder bulk density
ρ_a = air density
μ_a = air viscosity

These models will be published in a document scheduled for completion later this year.

V. OTHER ACCIDENTS

Methods to estimate the source term for other types of events will be considered. In cases where realistic scenarios cannot be formulated for criticalities, current U.S. Nuclear Regulatory Commission Regulatory Guides (3.33, 3.34, and 3.35) appear adequate. Phenomena (e.g., earthquakes, tornadoes) that may result in severe loss of structural integrity can lead to subdivision of solids on crush/impact. Suspension of powders from homogeneous or heterogenous substrate by high-velocity gases (tornadoes and explosions) are also phenomena of concern.

VI. CONCLUSIONS

Experiments have been performed investigating source term releases from potential accidents in nuclear fuel cycle facilities. Using data from these experiments and information in the literature, methods are being developed to provide radiological source term estimation methods for fires, explosions, spills, tornadoes, and criticalities. Before this study there were few methods available.

Using methods in the Accident Analysis Handbook, a user can estimate the amount airborne and particle-size distribution of aerosols generated by the above accidents. Other parts of the handbook will provide source term calculations for UF_6 accidents (Martin Marietta Corporation at Oak Ridge National Laboratory) and methods for determining the transport and deposition of airborne materials in filters and building ventilation systems (Los Alamos National Laboratory).

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Changing Perspectives on Severe Accident Source Terms

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I. INTRODUCTION

Since the potential hazard to the public from nuclear power plants is almost entirely associated with the possibility of the accidental release of radionuclides from the boundaries of the plant (source terms), it is not at all surprising that source term estimates pervade the regulations that govern power plant operations. Because there has been a substantial quantity of research since the Three Mile Island Unit 2 accident related to the processes that govern severe accident source terms, it is natural at this time to determine if the regulations should be changed, either to relax unnecessarily restrictive regulations or to add further margins of protection, if appropriate. In this regard the U.S. Nuclear Regulatory Commission has committed to reviewing its regulations to evaluate the need for change.[1]

In this paper we describe changes in the understanding of processes that influence source term behavior that have occurred since the regulations were initially formulated.

It is not the purpose of this paper to prejudge how or if the different regulations should be changed. The formulation of regulations involves considerations of policy quite outside technical considerations of source term phenomenology.

II. HISTORICAL OVERVIEW OF SOURCE TERM PERSPECTIVES

In the 1960s the nuclear industry in the United States moved rapidly from the operation of low power demonstration plants to the construction of plants that were many times larger. The licensing agency, the U. S. Atomic Energy Commission at that time, was faced with the challenge of establishing regulatory criteria supported by a very limited base of experimental data related to the behavior of accidents in large nuclear power plants and of plant operating experience. To account for the large uncertainties in this limited data base conservative safety margins were included in the regulations. Most of the regulations and regulatory guides which involve source terms rely on what are referred to as TID-14844[2] release terms. These release terms, which were intended to characterize a severe accident involving some fuel damage but not progressing to vessel failure, assumed 100 percent release of noble gases, 50 percent release of iodine (25 percent from the reactor coolant system) and 1 percent release of all other fission products as solids. This simple prescription, which is independent of accident sequence and plant design features, was convenient for use in regulations. Furthermore, in an era in which the likelihood of severe accidents was generally believed to be very small, the TID-14844 release terms were considered to provide a conservative basis for regulation. These release terms only described the release to the containment, not to the environment. Consistent with the design basis criteria for the plant, the regulatory framework did not recognize the potential for containment failure.

A. Changes in Perspectives Subsequent to WASH-1400

In 1975, the Reactor Safety Study (WASH-1400) was issued.[3] This study led to major new insights into the character of reactor risk. It indicated that severe accidents were substantially more likely than previously believed and that the consequences of a severe accident could vary dramatically, depending on the characteristics of the accident sequence and on the performance of plant features.

The methods of analysis used to predict severe accident source terms in the Reactor Safety Study were quite crude. A minimal data base existed on the release of radionuclides from small samples of fuel (primarily at ORNL) and integral experiments had been performed on the removal of elemental iodine and aerosols from containment atmospheres. The analyses of many complex severe accident processes were performed with very simple models which primarily attempted to conserve mass and energy. Nevertheless, the severe accident source terms developed for WASH-1400 were recognized as being the best available characterization for severe accidents with the available technology. Following WASH-1400, it was clear that some aspects of the regulations, such as the treatment of severe accidents in environmental impact statements and requirements for offsite emergency planning could no longer be considered adequate, and regulations were changed to incorporate WASH-1400 source terms.

B. Changes in Perspectives Subsequent to TMI-2

After the Three Mile Island Unit 2 accident, the importance of developing a better understanding of severe accident processes became widely recognized. What had been considered hypothetical conditions prior to WASH-1400, had at least in part been experienced. The Severe Accident Research Program, that had been initiated by the Nuclear Regulatory Commission after WASH-1400, was accelerated and expanded.[4] Over the past seven years this program, complemented by industrial and international programs, has resulted in a vastly improved understanding of severe accident processes, the ability to model these processes, and a data base for model validation.

As the WASH-1400 methods were examined critically in this time period, there was widely-stated belief that the WASH-1400 source terms were very conservative. Some estimates were that the WASH-1400 values were high by at least one to two orders of magnitude. As a result of this frame of mind, it was generally believed that a review of the regulations could result in significant relaxation of some regulatory criteria, particularly related to emergency offsite planning.

One of the major elements of the NRC's Severe Accident Research Program has been the development of improved methods for source term analysis. Under the direction of the Accident Source Term Program Office, an extensive reassessment was made of the technical bases for source term analysis. A systematic, mechanistic approach to source term analysis was developed. As the initial representation of this approach, a set of state-of-the-art source term codes was assembled, tested, and exposed to extensive peer review. In June of 1986 the "Reassessment of the Technical Bases for Estimating Source Terms", NUREG-0956[5] was published with the recommendation that "The Source Term Code Package is recommended as an integrated analytical tool for NRC evaluation of source terms in regulatory applications <u>provided that uncertainties are considered for each type of application</u>". (Underlining provided by authors.)

C. Current Perspectives

In a previous paper, a comparison was made between results obtained with the suite of source term codes (the predecessor to the Source Term Code Package) with WASH-1400 results.[6] An examination of the models which predict the principal elements of the source term: in-vessel release from fuel, transport in the reactor coolant system, ex-vessel release from fuel, and transport in the containment and secondary buildings, indicated that there is a tendency for the WASH-1400 models to overestimate source terms. In particular, current models predict significant retention of radionuclides in the reactor coolant system for most accident sequences. Retention in the reactor coolant system was not typically given credit in the WASH-1400 analyses. However, the WASH-1400 analyses do not appear to overestimate all sequences. Perhaps more importantly, the uncertainties in source term estimates are quite large and typically encompass WASH-1400 results.

One element of the source term reassessment effort supporting NUREG-0956 was the QUEST[7] program in which quantitative uncertainty estimates were developed for three accident sequences. The QUEST study indicated that the uncertainty in source term estimates is large. The band of uncertainty about the Source Term Code Package results was found to be on the order of a factor of 100 or greater.

The first application study of the Source Term Code Package is the NRC's Risk Reference Document, NUREG-1150,[8] which will describe the risk to the public for six reference nuclear power plants. In this study particular attention is being given to the uncertainties in the description of severe accident phenomena that influence the source term. Based on the results of the QUEST study, it is likely that the uncertainties in source term estimates will propagate into significant uncertainties in the risk estimates for these plants.

In recent years, a much more detailed understanding of severe accident source terms has developed. The methods are much more mechanistic. They are better supported by data. The more detailed examination of severe accident processes has also helped to identify technical issues where a good understanding of source term behavior does not exist. There is a much better characterization of what isn't known about source term behavior, and as a result, the implications of these uncertainties can be better understood.

Thus a perspective on source terms is emerging of broad uncertainty bands rather than point values. The centers of the bands are usually lower than the old WASH-1400 point estimates but the spread in uncertainty is typically broad enough to encompass the WASH-1400 values. Anomalously, a decade ago when the state of understanding of source term phenomena was very limited, the use of point estimates for source terms gave the impression of greater precision than current representations of source terms using uncertainty spreads. It should be recognized that this perspective on source terms is not universally held. The source term estimates obtained in the IDCOR[9] program using the MAAP code tend to be lower than Source Term Code Package results and the sensitivity studies that have been performed by IDCOR[10] (and independently by EPRI)[11] yield narrower uncertainty bands. This further supports the need to attempt to resolve the technical issues that underlie differences in methodologies.

III. THE SIGNIFICANCE OF DIFFERENCES IN SOURCE TERMS

It is difficult to compare source term estimates involving release fractions for seven to nine different elemental groups and to draw inferences about the implications to public health. In this section of the paper, we will use some indices for the relative health consequences for the different elemental groups to develop a measure for comparing release fractions. We will also compare some results of source term analyses with crude criteria for the onset of different types of health effects.

Before continuing, it is imperative to warn the reader about the errors inherent in the simplifications made in these analyses. The contributions to health effects from the different radionuclide groups are neither linear nor independent. Furthermore, the magnitudes of accident consequences, particularly for threshold types of effects, are very sensitive to assumptions in the analysis such as the effectiveness of emergency protective actions. With regard to this simplified treatment of source terms, perhaps the most grievous error is to ignore the influence of duration of release on the measure of consequences. The IDCOR program results[9] indicated quite long periods of release for most of the sequences analyzed (i.e., one to two days). The major part of the release in the Chernobyl[12] accident occurred over a nine day period. The likelihood of early effects that are threshold in nature can be substantially influenced by the duration of the release, since for an extended release an individual will have more time to evacuate, or the wind can shift before the individual is fully exposed.

Despite the limitations of simplified measures of health effects, they can allow the reader to obtain a feeling for the significance of quoted release fractions. In a recent study of the "Relative Importance of Individual Elements to Reactor Accident Consequences Assuming Equal Release Fractions",[13] Alpert developed importance weights for 25 elements with radioisotopes (60) that are potentially significant contributors to offsite consequences. Table 1 reproduces these weights for two contributors to early effects, 24 hour bone marrow dose and lung dose, and for total latent cancers. In the analyses performed with the MACCS[14] code the following important assumptions were made:

- An end-of-cycle core inventory for a 3412 MW PWR
- Equal release fractions of all elements
- Category D stability with a wind speed of 5 meters per second
- Release occurs outside growing season (no direct deposition on crops or pasture)
- A nonbuoyant release of each element at ground level
- A uniform, one-hour release of each element, one hour after shutdown
- Radiation protection factors of 0.75 for cloudshine, 0.45 for groundshine and 1.0 for inhalation
- No emergency protective actions.

In the development of accident source terms, a number of elements with similar chemical behavior are typically analyzed as a group for simplicity. In the Reactor Safety Study[3] seven elemental groups were used. In Table 2 the weighting factors for each of the elements in the Reactor Safety Study groups are summed to give a weighting factor for the group. These weighting factors then provide a simple means to compare the health significance of accident source term estimates among themselves or in reference to health effects benchmarks.

Two benchmarks for comparison are the TMI-2 accident and the Chernobyl accident, both severe accidents but with dramatically different consequences. From the regulatory viewpoint, however, the most significant benchmarks for comparison relate to the thresholds of specific health effects, in particular the thresholds for early fatalities and early injuries.

In the paper, "The Implications of Reduced Source Terms for Ex-Plant Consequence Modeling"[15] Kaiser examined a large number of consequence calculations in order to examine the importance of different levels of source term reduction. Kaiser developed two crude criteria which can be used to evaluate the potential significance of source terms.

> Criterion 1. If the release fraction of volatile fission products is less than 0.1 and the release of the ruthenium and strontium groups is less than 0.05, and the release of the lanthanum group is less than 0.01, the conditional mean number of early fatalities will be very small or zero.

> Criterion 2. If the release fraction of volatile fission products is less than 0.01 and the release of the ruthenium and strontium groups is less than 0.005, and the release of the lanthanum group is less than 0.001, the conditional mean number of early injuries will be very small or zero.

IV. EXAMPLE COMPARISON OF SOURCE TERM RESULTS

In this section we will examine one of the risk dominant sequences from WASH-1400, TMLB'δ, to evaluate the significance of improvements in source term methodology. As points of reference, the TMI 2 accident, Chernobyl accident, Early Fatality Criterion, and Early Injury Criterion are plotted in Figure 1. Note that the TMI 2 accident falls substantially below either of the early effects criteria. The principal contribution to the TMI 2 consequences is the release of approximately 7 percent of the noble gases, not the estimated release of 14 Ci of iodine. The Chernobyl accident is found to exceed the threshold criterion for early fatalities. The lung dose is particularly high because of the large release of the non-volatile elements in the lanthanum group. It should be recognized, however, that although the early fatality criterion was exceeded, no member of the public actually received a lethal dose because of the duration of the release and the emergency protective actions taken.

The WASH-1400 analysis of the TMLB'δ is illustrated in the figure as a point of reference. Case 1 is the Source Term Code Package analysis of the same scenario. In comparing the two results, it must be recognized that point estimate results of the Source Term Code Package cannot be considered best estimate results, since the effect of some important technical issues with high uncertainty is not included in the analysis. Whereas the WASH-1400 source term exceeds the early fatalities criteria by a substantial margin, the STCP analysis for this sequence is essentially equal to the early fatality criterion for bone marrow dose and is less than half the criterion for lung dose. The latent effects health measure for the STCP analysis is approximately one-third of the WASH-1400 value.

If the uncertainties in the analysis are taken into account, however, the perspective changes. One technical issue will be considered as an example of the potential effect of source term uncertainties. In this accident the predicted retention fractions within the reactor coolant system of CsI, CsOH, and Te are 76 percent, 80 percent, and 69 percent, respectively. The fraction of these species that would be released as the result of decay heating following vessel failure is quite uncertain. In Case 2, it is assumed that 50 percent of the originally deposited volatile elements is eventually released from the reactor coolant system and that 50 percent of the material released late in the sequence after containment failure will be released to the environment (based on STCP analyses). In this case, the consequence measure for bone marrow dose would exceed the early fatality criterion.

For the Surry plant, the likelihood of early failure of the containment is believed to be substantially lower than at the time of the Reactor Safety Study. One mechanism that could lead to early failure that was not recognized earlier is direct heating, however. If significant dispersal of the core and direct heating were to occur, the release of radionuclides to the containment would be enhanced. This effect is shown in Case 3. Again the consequences are substantially increased over the base case and the early fatality criteria are exceeded.

However, if the containment were to avoid early failure, the release of fission products to the environment would be reduced. Case 4 shows the effect of delaying containment failure until 12 hours after the start of the accident. Although the consequence measures are reduced, they still exceed the criteria for early injuries. In Case 5, it is assumed that the containment removal processes are completely effective in attenuating all radionuclides but noble gases. This is equivalent to a very late failure case or a filtered vent case. Although the consequence measures are below the early injury criteria, the consequence measures are close to these criteria. A small contribution from the uncertainty in any of the issues affecting source term magnitude could result in exceeding the early injury criteria.

V. CONCLUSIONS

Subsequent to the TMI-2 accident, knowledge of severe accident processes which influence source terms has expanded dramatically. The methods of analysis that have been developed or are under development are more mechanistic and are better supported by experimental data. The uncertainties in source terms are large, however. Considering the magnitude of these uncertainties, it is unlikely that in the near future it can be demonstrated with a high degree of confidence that source terms have been reduced to the degree that early fatalities or early injuries cannot occur in a severe accident. Although it is appropriate at this time to review source term based regulations, it is not clear that substantial relaxation of regulatory criteria will result. Point estimates of source terms should not be used without full consideration of the associated uncertainties.

Furthermore, the magnitude of the source terms is affected not only by the processes of radionuclide release and transport, but by the timing and mode of containment failure and by the performance of containment safety systems such as sprays, suppression pools, icebeds, and air coolers. Thus there is now increased awareness of the potential benefits of accident management to reduce the consequences of severe accidents.

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	Early Exposure		Long-Term Exposure
Element	Normalized 24 Hour Bone Marrow Dose	Normalized Lung Dose	Normalized Total Latent Cancers
Co	0.008	0.01	0.07
Kr	0.1	0.04	0.001
Rb	2E-4	2E-4	1E-5
Sr	0.7	0.9	0.7
Y	0.07	3.5	0.4
Zr	1.0	2.0	0.7
Nb	0.3	0.4	0.2
Mo	0.1	0.7	0.06
Tc	0.03	0.03	0.06
Ru	0.3	3.0	1.0
Rh	0.01	0.08	0.004
Sb	0.06	0.1	0.004
Te	0.8	0.6	0.1
I	1.0	1.0	0.1
Xe	0.01	0.005	1E-4
Cs	0.14	0.09	1.0
Ba	0.5	0.6	0.2
La	1.2	1.6	0.08
Ce	0.2	8.0	2.0
Pr	0.003	0.8	0.08
Nd	0.03	0.3	0.03
Np	1.4	5.0	0.4
Pu	0.003	1.4	3.0
Am	0.001	0.01	0.03
Cm	0.4	5.0	1.1

Table 1. Relative Importance of Individual Elements[13]

		Early Effects		
Group	Elements	24 Hour Bone Marrow	Lung	Latent Cancer
1	Xe, Kr	0.11	0.045	0.0011
2	I, Br	1.0	1.0	0.1
3	Cs, Rb	0.14	0.09	1.0
4	Te, Sb, Se	0.86	0.7	0.10
5	Sr, Ba	1.2	1.5	0.9
6	Ru, Rh, Pd, Mo, Tc	0.44	3.8	1.1
7	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Ce, Pu, Np	4.6	28.0	8.0

Table 2. Group Weights (Normalized)

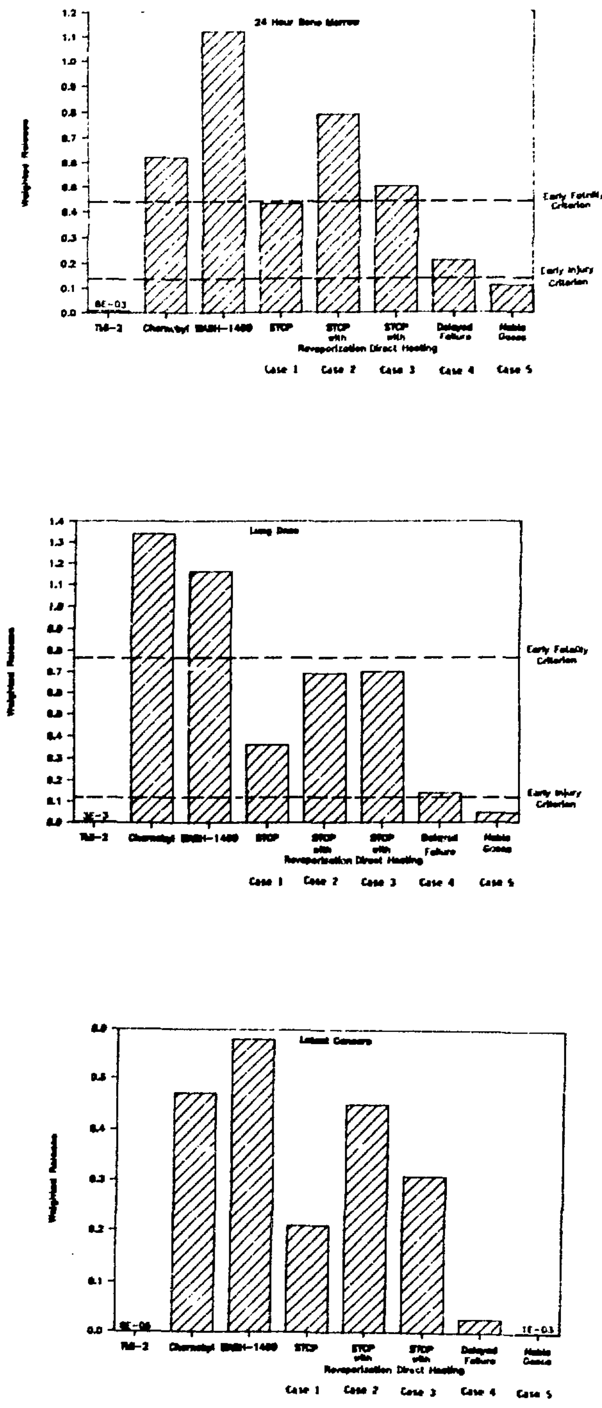

Figure 1. Normalized Consequence Measures for Variations on TMLB'5 Scenario

NRC Perspective on Severe Accident Consequence Assessment

Thomas McKenna and James A. Martin

ABSTRACT - One of the major roles of the Nuclear Regulatory Commission (NRC) during a severe accident (core melt) is to monitor the reactor licensee to assure that they are recommending the appropriate protective actions to offsite officials. Initial protective action recommendations should be made by the plant staff based on predetermined instrument readings (emergency action levels) that indicate the status of plant systems(s) required to protect the public. The determination of the need for additional protective action may require dose assessments/projections. In the past dose assessments have been based on licensee estimates of releases based on the stack monitor reading. This is not adequate because severe accident releases cannot be characterized by the stack monitor. Dose assessments should consider the "probable range" of accident conditions and not just the relationship of stack monitor readings to doses. This paper will summarize the basis for the NRC perspective on consequence assessment and the tools used to monitor licensee consequence assessment and protective action recommendations.

One of the major roles of the NRC during a severe reactor accident (core damage) is to monitor the reactor licensee to assure that the licensee is recommending the appropriate protective actions to offsite officials.

The NRC first monitors and decides if the plant conditions warrant taking action, based on an assessment of core and containment status. Releases resulting in observable early health effects (injuries and deaths) or high individual risks (very high radiation doses) offsite can occur only as a result of a severe core damage coupled with early containment failure (within 24-hours of release from the core). If containment integrity is maintained or if mitigative engineered safety features (e.g., containment sprays) are functioning, the offsite consequences may be limited to the immediate vicinity of the site and early health effects may not be induced even for core melt accidents. However, during a severe nuclear power plant accident early containment failure could not be ruled out or "predicted with confidence." Therefore, severe core damage accidents should be classified as general emergencies, which would warrant initiation of immediate, early protective actions offsite.

Three basic conclusions have been derived from the considerable research on severe accidents concerning the effectiveness of early protective actions for preventing early health effects from a several reactor accident resulting in a major release. These are:

1. To be most effective, protective actions (shelter or evacuation) must be taken before or immediately at the time a major release to the atmosphere occurs.
2. People should immediately evacuate areas near the plant (within a 2-to-5 mile radius) and remain in shelter elsewhere to receive additional information or instructions during the early time frame.
3. Following a major release, the dose from ground contamination in the area beyond which people have seen evacuated based on plant conditions (item 2) may become very important in a few hours (e.g, 4 hr), requiring immediate radiological monitoring to locate hot spots where evacuation would be required.

These basic concepts form the foundation for the NRC's protective action guidance and for the assessments conducted by the NRC during its response.

The specific NRC guidance from NUREG-0654/FEMA REP 1, Rev. 1, Appendix 1, for general emergency protective actions is attached as Figure 1. This figure was sent to licensees via Information Notice 83-28 and is the basic tool used by the NRC response organization when monitoring the recommendations being made by licensees during a severe reactor accident.

Several points concerning Figure 1 should be discussed. Initial scoping of an emergency is to be based on a comparison of the plant instrument readings (i. e., plant conditions) with the predetermined emergency action levels. This comparison will identify the plant conditions referred to in the diagram (e.g., 20% cladding failure, fission products in containment, or containment status). Core damage is defined as release of 20% of the gap activity. This level of core damage was chosen because it is well beyond that expected for any accident if safety systems operate as planned.

The criteria also call for evacuation in what is called a keyhole, that is, in all directions close to the plant and a further distance in the downwind direction (e.g., a 2-mile radius and 5 miles downwind). This ensures that people at greatest risk would evacuate early in the event of a general emergency even if wind direction projections are incorrect.

This guidance indicates that people close to the plant should not evacuate if imminent catastrophic containment failure that could release large fractions of the radioactive material in a puff (<1 hr) is likely. The goal is to shelter people inside their homes as the puff release moves by. This at best would be effective only for puff releases (short duration); for a long-duration release, most of the release and accompanying dose can be avoided by early evacuation. Therefore, effective application requires that the duration and timing of the release be known (predictable), which may not be the case early in a severe accident. In most cases, puff releases would be the result of containment failures that were (1) unpredictable (e.g., bypass accidents or explosions) or (2) the result of overpressurization. Overpressurization failures would most likely be far from imminent (following detection of core damage), allowing considerable time for precautionary evacuation of areas near the plant. The actual time of failure resulting from overpressurization would be unpredictable. Therefore, this part of the criteria is generally applicable only for those situations where the timing of an imminent puff release is certain. An example would be the controlled venting of the containment by the licensee. Consequently, for most core damage accidents (general emergencies) the population close to the plant should be evacuated.

After implementation of protective actions near the plant (based on an assessment of plant conditions) the determination of the need for additional protective action may require dose assessments/projections. Bounding dose calculations also may be very useful in comparing the consequences of various plant response options (e.g., venting the containment versus allowing later containment failure). In the past, dose assessments have been based on licensee stack monitor readings. However, during a severe accident (core damage), releases may not be characterized by the stack monitor. Severe accident releases can bypass the stack monitor or the accident conditions could prevent adequate characterization of release. In addition, since effective protective action requires prompt implementation, an attempt should be made to project the magnitude of a release before it occurs (rather than waiting until it goes out the stack). Therefore, in performing dose assessments, the NRC estimates the possible offsite releases based on the probable range of accident conditions and not just on stack monitor readings. These consequence estimates take into consideration the current and projected status of the core and possible release pathways.

Some simple tools have been developed that could at least provide a bound of possible offsite consequences, based on consideration of plant (core and containment) conditions. Figure 2 is a very simple example.

Figure 2 uses the concept of an event tree to display the potential consequences for public health due to severe accidents. Moving from left to right in the figure, "yes/no" answers to scenarios at the top result in a series of branches, possibly to offsite consequences. For example, if only the radioactive material contained in the fuel pins (gaps) is released with late containment failure, the offsite consequences would be small (branch 7 in Fig. 2). If all the answers are yes, branch 1 indicates extremely severe offsite consequence. Figure 2 illustrates two fundamental public health questions during an emergency response at a light water reactor:

What is the status of the reactor core?
What is the status of the reactor containment?

The answers to these two questions scope the level of threat to the public and the need for offsite emergency response.

Other examples of simple tools used to bound offsite consequences based on plant conditions are the 15 precalculated offsite dose projections for severe accident conditions and for many meteorological conditions presented in NUREG-1062 Dose Calculations for Severe LWR Accidents (USNRC, 1984).

NRC recognizes that dose projections must be used with great caution because of the great uncertainties associated with making such assessments during severe accidents. Table 1 gives an overall estimate of the uncertainties associated with dose assessment for severe reactor accidents. The NRC response staff has estimated that, at best, dose projections made during a severe accident may be within a factor of 10 of the average field monitoring results--at worst, dose projections could be off by several factors of 10. The largest single component of this uncertainty is source term estimation. Unanticipated catastrophic containment failure is an example of an accident for which source term and consequently the offsite dose could be underestimated by a factor of 100,000 or more. Nevertheless, performing dose projections is a useful tool for determining where to conduct early monitoring or where implementation of additional protective measures should be considered, but reliance should be placed on field monitoring results as soon as possible after an actual release.

I would also like to reassure you that the NRC recognizes that its role early in the response to an accident is to monitor. In cooperation with local officials, licensees have developed site-specific criteria for recommending protective actions to the public according to the previously discussed concepts. Licensees are required to report these events to the NRC within 1 hour (versus 15 minutes for offsite officials). It is expected to take an additional hour after notification for the NRC response organizations to be activated and to be prepared to comment on protective action decisions or recommendations. Calling the NRC to confirm a preplanned protective action would only delay protective action implementation.

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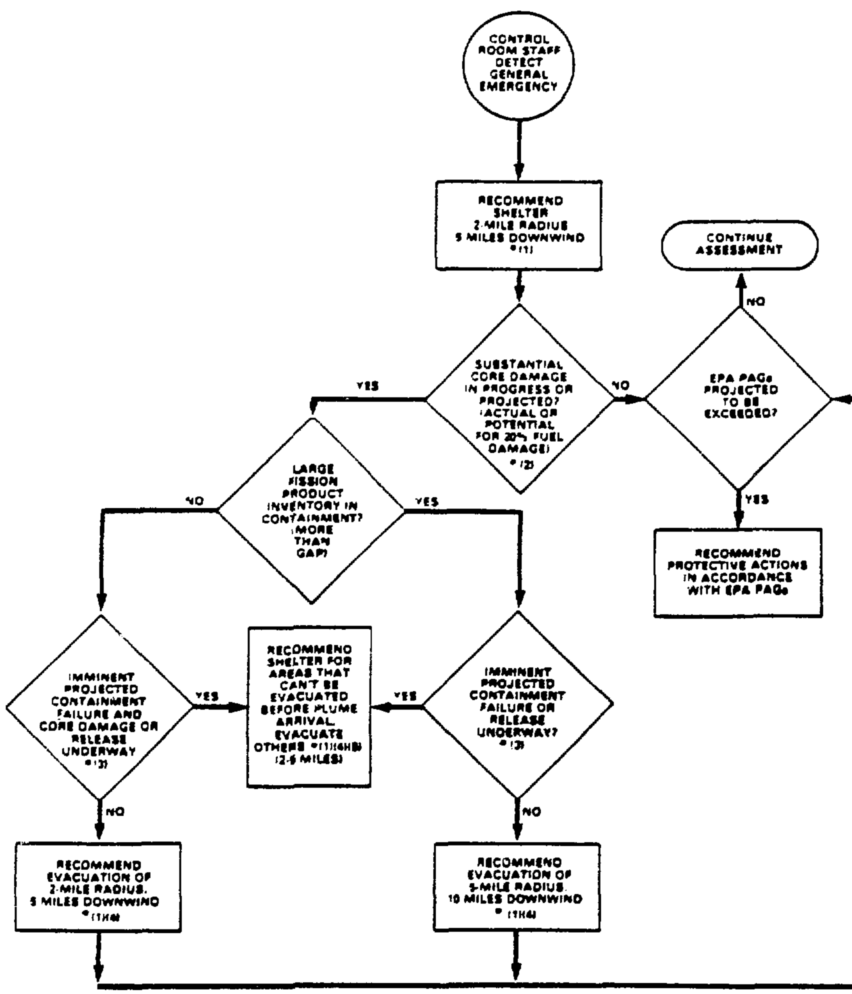
FLOW CHART FOR GENERAL EMERGENCY OFFSITE PROTECTIVE DECISIONS

The following actions will be based on predetermined observable instrumentation and plant status indicators (EALs) contained in the emergency plan and that have been reviewed by offsite officials. However, responsible offsite officials must decide on the feasibility of implementing the protective actions at the time of the accident.



SOURCE: Appendix 1, NUREG-0654-FEMA-REP-1, Rev. 1

*(1) SITUATIONS REQUIRING URGENT ACTION BY OFFSITE OFFICIALS
(Based on Control Room Indicators, No Dose Projections Required)
15-Minute Decisionmaking, Activation of Alerting System and EBS Message

*(2) Actual or projected release of 20% gap from core or loss of physical control of the plant to intruders.

*(3) "Puff" release (rate much greater than designed leak rate).

*(4) For all evacuations, shelter the remainder of the plume EPZ and promptly relocate the population affected by any ground contamination following plume passage.

*(5) Concentrate on evacuation of areas near the plant (e.g. may be time to evacuate 2-mile radius and not the 5-mile radius).

Figure 1. Flow Chart for General Emergency Offsite Protection Actions.

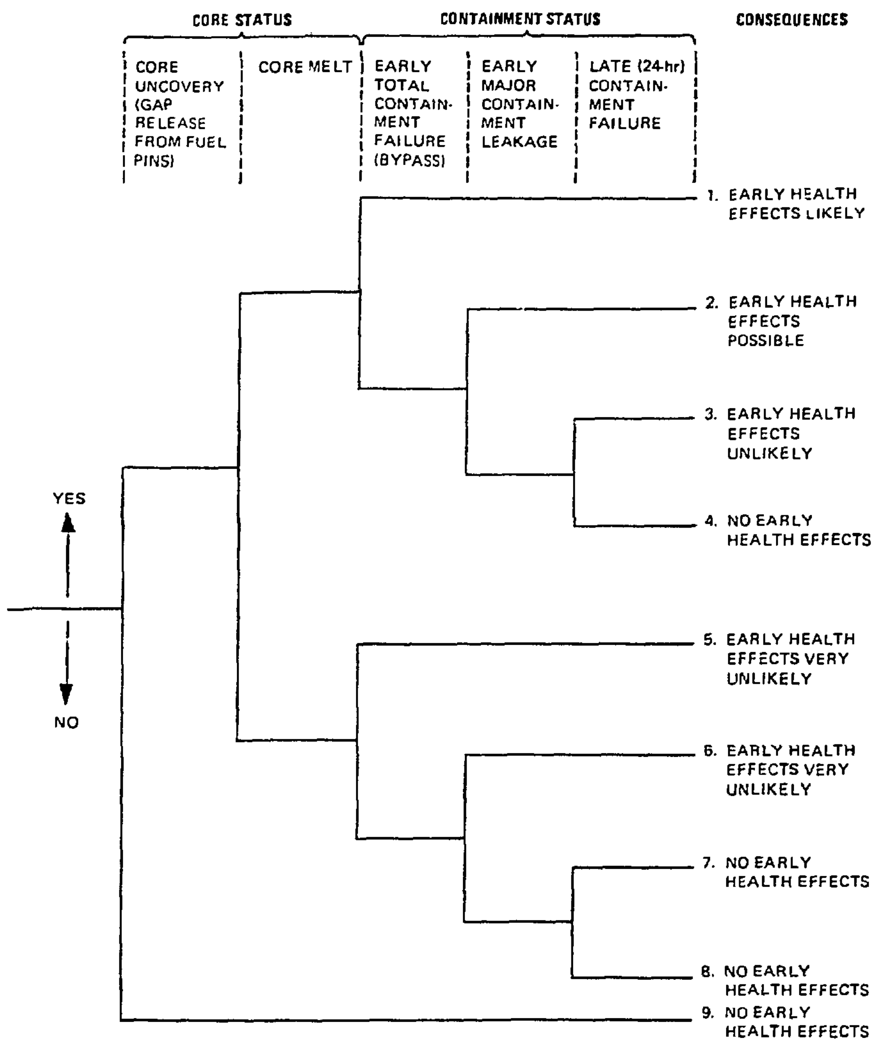


Figure 2.Event Tree For Severe Reactor Accident Consequence Assessment

ESTIMATED RANGE OF UNCERTAINTY BETWEEN PROJECTED AND ACTUAL OFF-SITE DOSE FOR A SEVERE ACCIDENT (CORE MELT)

Element	Uncertainty Factor[a]		
	At Best	Most Likely	Near Worst
Source term (event and sequence)	5	100–1,000	100,000
Dispersion			
Diffusion (concentration)	2	5	10
Transport (direction)	22°	45°	180°
Transport (rate)	1	2	10 (low wind speed)
Dosimetry	3	4	5
Overall (dose and direction)	10, 22°	100–10,000, 45°	100,000, 180°

[a]These estimates are for an averaged dose at a location (e.g., 15–30 min), not for a specific or single monitor reading.

Table 1.Estimated range of uncertainty between early projected dose and actual off-site dose for a severe reactor accident.

A History of Aerial Surveys in Response to Radiological Incidents and Accidents

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ABSTRACT EG&G Energy Measurements Inc., operates the Remote Sensing Laboratory for the U.S. Department of Energy (DOE). The Laboratory plays a key role in the federal response to a radiological incident or accident. It assists the DOE in the establishment of a Federal Radiological Monitoring and Assessment Center (FRMAC). The Remote Sensing Laboratory has played a major role in more than 14 incidents, including lost sources, accidental dispersions, and nuclear reactor incidents.

I. INTRODUCTION

EG&G Energy Measurements, Inc. (EG&G/EM) operates the Remote Sensing Laboratory (RSL) for the United States Department of Energy (USDOE). The Laboratory has major facilities in Las Vegas, Nevada, Washington, D.C., and Santa Barbara, California. It maintains nine twin-engine aircraft, helicopters and fixed-wing, equipped as aerial survey platforms. Remote sensing technologies include: large area radiological mapping, ground-based radiological measurements, high altitude aerial photography, multispectral photography, multispectral aerial scanning, and airborne gas and particulate sampling. The laboratory has acquired and developed a broad variety of remote sensing equipment. Its personnel acquire, analyze, and report data to federal and state agencies.

The RSL is a research facility with state-of-the-art equipment and modern data analysis techniques. It has emerged in response to a unique, clearly-understood need which was apparent more than 40 years ago, viz., the use of radioactive materials demands an aerial measurement system capable of surveying large areas in a short time. Aerial measurements of surface radioactivity were made in the United States as early as 1948, originally to determine the feasibility of airborne prospecting for radioactive ore deposits (Davis and Reinhardt,1957). In the mid-1950s a series of events prompted the U.S. Geological Survey (USGS) and Oak Ridge National Laboratory (ORNL) to develop these systems for entirely different applications. Such events included the United Kingdom Windscale reactor accident, the release of radioactive clouds from nuclear weapon tests in Nevada and the mid-Pacific, and the emergence of commercial power reactors.

In 1959 the U.S. Atomic Energy Commission (AEC) asked EG&G, Inc., to develop a dedicated system for use at the Nevada Test Site. The Aerial Measuring System (AMS), as the program is now known, became operational in November 1960; the first large-area survey was conducted in 1961.

System development and operational capability have grown continuously under the AEC and its successors, the U.S. Energy Research and Development Agency (ERDA) and the USDOE. Aerial radiological surveys are now conducted for a variety of scientific objectives. And because the RSL provides aerial survey platforms, a broad variety of data acquisition and analysis hardware, and a cadre of dedicated remote sensing experts, the Laboratory has become a development center for many remote sensing technologies.

Hundreds of routine aerial surveys have been completed over the entire United States, some over international waters, in the mid-Pacific, and in foreign countries. This report, however, primarily summarizes those Laboratory surveys conducted in response to radiological incidents and accidents. Thirteen surveys will be discussed.

II. LOST COBALT-60 SOURCE

In the summer of 1968 the Laboratory was called by the Radiological Assistance Program Center in Albuquerque, New Mexico. A 330 mCi cobalt-60 source was lost during routine interstate shipment from Salt Lake City, Utah, to Kansas City, Missouri, a distance of 1930 km. At the Kansas City freight terminal the empty lead storage container was found on its side, the lid broken open, the source missing. From Salt Lake City a Laboratory fixed-wing aircraft flew the exact route of the truck at 120 m above the highway. On the second flight day, the crew observed a strong radioactive signal just east of the Missouri River, near St. Joseph, Missouri.

Subsequent passes pin-pointed the location; spectral data confirmed the presence of Co-60.

After landing in St. Joseph, the crew drove to the site and located the source within 10 minutes, using hand-held instruments. The source was a 9-cm-long stainless steel capsule, lying just off the road, within 75 m of the nearest residence. Using standard health physics procedures and an improvised radiation shield, the crew recovered the source and turned it over to the authorities.

III. ATHENA MISSILE

On 5 July 1970, an Athena missile was launched by the U.S. Air Force from Green River, Utah, as part of a routine test program. A system malfunction caused the missile to overshoot the target zone at the White Sands Missile Range in New Mexico. The missile was tracked to low altitude about 650 km south of the United States/Mexico border, near Torreon, Mexico. Because of the rugged terrain and maps with little detail, ground crews failed to find the missile, despite an intensive two-week-long search. Since the missile contained two 470 mCi Co-57 sources as part of its payload, the Laboratory was requested to join the search. After flying a grid pattern for 2-1/2 hours, a fixed-wing aircraft located the source. Positive identification was obtained from gamma spectral data. The following day the aircraft guided ground crews to the impact site, a shallow crater approximately 5 m in diameter.

IV. HAVE SINEW-1

In August 1971 the Laboratory cooperated with the U.S. Air Force and several other agencies in a research project called HAVE SINEW-1. An Air Force Athena rocket carrying six independent payload packages was launched from the Green River Launch Site in Utah. The packages were expected to land, separately, at the White Sands Missile Range. Each package contained either a 50 mCi or a 100 mCi tantalum-182 gamma radiation source as an aid to recovery at White Sands. After the packages were radar-tracked to the Range, the Laboratory conducted an aerial search with a large array of NaI detectors mounted in a fixed-wing aircraft. Each of the payload packages was a simulation of a radioisotope heat source intended for use in instrumented satellites.

Four of the six payloads were recovered intact, with their radiation sources. In addition, two of four 50 mCi sources were recovered, detached from their payloads. The latter apparently disintegrated upon reentry. The two remaining 50 mCi sources, as well as their respective payloads, were never recovered. It has been speculated that these disintegrated prematurely, never reaching the expected impact zone. This early experiment demonstrated effectively that the aerial detection system could be used to search for lost radiation sources.

V. MID-PACIFIC ISLANDS

From 1946 through 1958 the United States conducted a number of nuclear weapons related tests in the mid-Pacific at Enewetak and Bikini Atolls. These are part of the Northern Marshall Islands, 4000 km southwest of Honolulu. Forty-three tests were conducted at Enewetak, 23 at Bikini. In 1972 the AEC conducted a detailed survey of the total radiological environment of Enewetak Atoll (USAEC 1973). More than 4500 samples from the marine, terrestrial, and atmospheric components of the Atoll environment were analyzed with instruments and radiochemical techniques. The Remote Sensing Laboratory provided photographic base maps and a detailed aerial survey of the gamma-radiation levels on all islands of the Atoll. It was found that Sr-90, Cs-137, Co-60, and Pu-239 were the predominant radioactive isotopes still present.

From mid-September through mid-November 1978 an even more extensive aerial survey was conducted in the Northern Marshall Islands (Tipton and Meibaum, 1981). A large array of NaI detectors was mounted on a U.S. Navy helicopter which operated from the flight deck of the USNS Wheeling, a missile tracking ship. A total of eleven atolls and two islands were surveyed. Detailed contour maps were prepared for exposure rate and concentrations of Cs-137 and Co-60.

In addition to the aerial survey work, the Laboratory also provided in-situ gamma measurements during the Enewetak clean-up program from 1977 to 1979. These data were obtained with high-purity germanium detectors mounted on tracked vehicles (Tipton et al., 1981).

VI. BOMARC MISSILE SITE

In November 1973 the Laboratory flew a radiation survey over a deactivated U.S. Air Force Bomarc missile site near McGuire Air Force Base, New Jersey. A fire had occurred in one of the missile bunkers in 1960; fallout from the resulting cloud of smoke and debris distributed weapons grade plutonium around the site. A large array of NaI gamma ray detectors was externally mounted on a military UH-1N helicopter. By flying a flight pattern at 23 m and hovering at 15 m, a Laboratory crew was able to map the dispersion of Am-241, an isotope which accompanies the plutonium and allows it to be mapped from an aerial platform. Surface concentrations as high as 0.81 $\mu g/m^2$ were measured and a distribution map was provided to aid in site clean-up.

VII. COSMOS 954 SATELLITE

Perhaps the most celebrated incident involving the DOE Remote Sensing Laboratory began when U.S. observers noticed a change in orbit of the Russia satellite Cosmos 954 (USDOE, 1978). The satellite, powered by a nuclear reactor, plunged back into the earth's atmosphere on 24 January 1978. It impacted in the Northwest Territories of Canada, scattering radioactive debris over an estimated area of 52,000 km^2. The search and recovery effort, called Operation Morning Light, was a joint program of the Canadian Combined Forces and the USDOE, which coordinated the efforts of many U.S. Federal Agencies and private contractors. Large arrays of detectors, operated by radiation physicists primarily from the United States, were flown at low altitudes in Canadian C-130 aircraft which covered thousands of kilometers each mission. Helicopter teams, equipped with hand-held equipment, investigated hits and recovered debris. The recovery effort was extremely difficult because of major logistics problems in so large a search area, the remoteness of the site, snow, and extreme cold. More than 100 large pieces of radioactive debris were recovered. But it has been concluded that the reactor core disintegrated during reentry, scattering radioactive particulate matter over a broad area. Since it was impossible to locate and recover any more than a small percentage of this debris, even with the best technology presently available, further recovery effort was terminated after several months of intensive work.

VIII. THREE MILE ISLAND

On 28 March 1979 the Three Mile Island (TMI) nuclear reactor was shut down after suffering a major accident which has forever changed the history of nuclear power (Beers et al., 1984). The RSL flew 167 helicopter missions between 28 March and 25 June 1979. Airborne instrumentation was used to identify isotopes, to determine the direction of the plume of radioactive gases emanating from the crippled reactor, and to measure the maximum radiation levels inside the plume. During the first few days the effluent was dominated by Xe-133m and Xe-135. On the first day Kr-88 and its daughter, Rb-88, were also observed. Later the predominant radiation was from Xe-133. The highest exposure rate, 15 mR/h, was measured at 90 m altitude, 0.4 km from the reactor, on 30 March 1979. On the same day an exposure rate of 4.0 mR/h was measured 5 km from the reactor.

The preponderance of radioactive gases, and the absence of radioactive particulate matter in these measurements, suggested that the accident would produce little radioactive fallout downwind of the reactor. This was corroborated by an elaborate sampling program for soil, water, and biota, performed by many Federal and private agencies working cooperatively after the accident.

Confirming evidence was obtained in an aerial survey of an 82-square-kilometer area centered on the nuclear reactor, conducted by an RSL helicopter in October 1982 (Colton, 1983). The highest exposure rates, up to a maximum of 200 μR/h, were inferred from data measured directly over TMI facilities. This radiation was due to Co-58, Co-60, and Cs-137, which was consistent with normal plant operations. For the remainder of the survey area, the inferred radiation exposure rates varied from 6 to 14 μR/h Ground measurements, including soil samples and pressurized ion chamber readings, were in agreement with the corresponding aerial data. With the exception of the activity measured directly over TMI facilities, no evidence was detected of any contamination which might have occurred as a result of past reactor operations or the TMI Unit 2 accident. Ground and aerial results from 1982 agreed with those obtained in a 1976 aerial survey of a considerably larger area (Fritzsche, 1977).

IX. NEUTRON LOGGING SOURCE IN VENEZUELA

In early March 1982 an oil well logging company lost a strong neutron source in Venezuela. It was an americium-beryllium source with a neutron yield of 4×10^7 n/sec; such sources are routinely used for down-hole research in oil drilling operations.

It was assumed, at first, that the source accidentally fell from a storage container broken open during transportation from the field. The transportation route was searched by teams in vehicles and on foot, using conventional hand-held detectors. Villages were searched. A large reward was offered. After six weeks a team from the RSL was called. They brought proportional counters filled with helium-3 to 3 atmospheres absolute pressure. Each of three modules contained eight tubes, 5.1 cm in diameter by 1.8 m long. They were mounted aboard a Venezuelan helicopter for the aerial search.

The helicopter flew the 175-km road between the operations base and the well site, as well as a 0.8-kilometer-wide area on each side of the road and all river crossings. All cities, towns, and villages within several kilometers of the road were flown. The logging site and all drilling rigs in the vicinity were also flown. This work was done at 91 m altitude, with 300-to 460-m line spacing. Seven rivers in the vicinity were flown at 46 m altitude. Calculations indicate that at a survey altitude of 91 m and an aircraft velocity of 35 m/sec, the neutron source would have been readily detectable at lateral displacements as great as 580 m. Despite this sensitivity and despite a well-planned, carefully-executed search, the source was never found. It has been speculated that the source was stolen and quickly removed from the area, either for profit or to embarrass its owners.

X. COBALT-60 DISPERSAL IN MEXICO

In November 1983 one of the most serious accidental dispersions of radioactive material occurred in Ciudad Juarez, Chihuahua, Mexico. A Picker C-3000 Co-60 teletheraphy machine was disassembled for scrap steel. The source capsule, containing an estimated 450 Ci of Co-60, was ruptured. The capsule contained 6000 to 7000 cobalt pellets, each 1 mm x 1 mm in size and having an activity of 40 to 100 mCi. Individual pellets were found at the disassembly site, in the transporting vehicles, along the roads to foundries in Juarez and Chihuahua and in the scrap piles and processing equipment. An estimated 218 Ci of Co-60 was melted into 7000 tons of steel products such as reinforcing bar, metal table bases, and electric motor parts.

When the RSL was deployed on 27 February

1984, nearly 3 months after the initial dispersal of the Co-60 pellets, contaminated steel products had been distributed in northern Mexico, much of the United States and several other countries. Using a helicopter and an array of NaI detectors, Laboratory personnel conducted an aerial survey over El Paso, Texas; Anapra, New Mexico; and the cities of Juarez and Chihuahua in the state of Chihuahua, Mexico. The total area surveyed over these cities was approximately 520 km^2. Both sides of the highway linking Ciudad Juarez and the city of Chihuahua, were also surveyed, a distance of 365 km. Between March 2 and March 24 a total of 700 km^2 was covered; 45 Co-60 anomalies were detected, due either to pellets or to contaminated steel. All of the pellets discovered with hand-held instruments or in the aerial search have been recovered. Most of the contaminated steel products have been returned to Mexico and buried. Unfortunately, because of the nature of the source and its wide dispersion in the environment, complete recovery is neither possible nor certifiable.

XI. READING PRONG SURVEY

In December 1984 the home of Stanley Watras in Boyertown, Pennsylvania, was found to contain excessive levels of radon gas (Hoover and Mateik, 1986). Shortly thereafter the RSL began an aerial survey covering 260 km^2 over the Reading Prong. This is a huge geological formation underlying Boyertown and many other cities in Pennsylvania, New Jersey, and New York. The survey was requested by Pennsylvania's Department of Environmental Resources to help locate regions where buildings might contain elevated levels of radon gas, a naturally radioactive gas produced by the decay of radium contained in the Reading Prong.

The results of this survey were presented in contour maps showing regions which were above the natural background radioactivity in the area. It was observed that elevated activity frequently coincided with known geological faults. It is suspected that other highactivity areas may correspond to previously unknown fault zones or to regions where uranium- and radiumbearing rock are unusually close to the surface. There is some evidence that faults act as conduits, allowing radon to escape to the earth's surface. The results of the survey are being used in further research on the radon problem.

XII. SOUTH HOUSTON

On 6 May 1985 the Laboratory began a 5-week survey of many contaminated sites south of Houston, Texas. During the 1950s and 1960s the Hastings Pharmaceutical Company and several successor companies manufactured technicium and radium sources for medical use, Cs-137 sources for well-logging, and other radiochemical products. Because of poor quality control in manufacturing and careless dumping of trash and product residues, many different sites were contaminated. The Texas Department of Health Resources, acting on anonymous tips, found cesium, cobalt, and radium contamination exceeding 30 to 35 mR/h contact readings in completely uncontrolled areas. After a thorough review of the available data, Health Resources called upon the USDOE for assistance. The RSL flew a grid pattern over South Houston with a helicopter at an altitude of 46 m. A 455 km^2 area was covered with tight line spacing. Industrial areas in four small cities were also surveyed, as well as seven waste dumps. Eleven Cs-137 anomalies were found, five of which had been previously identified. The new sites appeared to be the result of contaminated landfill used at various home construction sites. Several other anomalies were observed: 3 due to Ra-226, 1 to Co-60, and 3 to Ir-192. Thirty-seven other hits were just above the minimum detectable Cs-137 activity of the aerial system. The Texas Department of Health Resources is still investigating and evaluating the results of the aerial survey.

XIII. BESSEMER, ALABAMA

On 2 July 1985 the Laboratory was requested to assist the Alabama Department of Public Health in locating three Cs-137 sources which disappeared when a Uniroyal Tire and Rubber Company production line was scrapped. Two of the sources were 50 mCi, one was 10 mCi; they were used as gauging sources on a tire production assembly line in Opelika, Alabama.

Seven potential sites were identified by Alabama Public Health: steel companies, scrap yards, and a rubber company. In two days all seven were overflown by a helicopter search team at an altitude of 30 m. Because of their small size, all positioning was based on visual navigation and radar altimeter information.

The remains of the three sources were found at the U.S. Pipe and Foundry Co. in Bessemer, Alabama. It has been theorized that the sources were dumped into a hopper along with tons of steel scrap, and smelted to produce steel pipe. The sources appear to have been volatilized during the smelting operation, went up the stack as particles or vapor, which was trapped and collected in a bag house. The debris from the bag house was dumped onto an area measuring 80 m by 18 m which indicated elevated exposure rates. Gamma spectra confirmed the presence of Cs-137. None of the other six sites showed any contamination above background levels.

XIV. SEQUOYAH FUELS URANIUM HEXAFLOURIDE ACCIDENT

On 4 January 1986 a tank containing more than 12,700 kg of uranium hexaflouride exploded at the Sequoyah Fuels Corporation facility in Gore, Oklahoma. The RSL responded quickly to a request from the U.S. Nuclear Regulatory Commission (NRC). Ground-based measurements with a high purity germanium detector were begun within 36 hours of the explosion and continued for a week. The measurements were made principally in the downwind direction from the facility and in public areas. These results were made available to the NRC within hours of collection. An aerial gamma survey was conducted in an 8 km by 8 km area surrounding the facility. Radiation contour lines were overlaid on an aerial photograph of the site. The radiological maps were available on the eighth day following the accident.

Laboratory personnel also obtained high altitude and intermediate altitude photographic imagery, in both the visible and infrared regions of the spectrum. Hundreds of documentary photographs were also taken of the damage and clean-up operations at the facility.

XV. ROUTINE SURVEYS

The purpose of this paper was to present a summary of the radiological accidents and incidents to which the RSL has contributed its unique capabilities. A proper perspective, however, demands wider focus. The routine work of the Laboratory, over more than a quarter century, has made a significant contribution to the definition of our environment. The nature of natural radioactivity, the extremes of its range, and the human contribution to our background are better understood in the United States than anywhere else in the world. Routine Laboratory work has provided the equipment, facilities, and programs for developing truly unique techniques and capabilities.

Routine work for the Laboratory has included:

1. Detailed radiological maps of the Nevada Test Site. Dozens of surveys, conducted with both fixed-wing aircraft and helicopter, have provided contour maps of fallout patterns produced during atmospheric testing (Boyns, 1986).

2. Nuclear Regulatory Commission (NRC) facilities. The NRC has jurisdiction over uranium processing facilities and radiological waste sites which are regularly surveyed by the Laboratory. In addition, commercial nuclear reactors are surveyed. A background survey is conducted before the reactor achieves criticality. Regularly scheduled surveys are made over each operating reactor every few years. To date, 144 surveys have been conducted for the NRC.

3. Research and production laboratories. For the DOE Office of Nuclear Safety the Laboratory surveys the DOE national laboratories, nuclear production facilities, other research facilities and sites of special interest. A total of 57 radiological surveys have been conducted at such sites, not including those specifically discussed above.

4. FUSRAP and UMTRAP sites. For the DOE Office of Nuclear Engineering the Laboratory has surveyed many sites formerly used for the production or disposal of nuclear materials, under the Formerly Utilized Sites Remedial Action Program (FUSRAP), and mines and mill sites, under the Uranium Mill Tailings Remedial Action Program (UMTRAP). A total of 53 such sites have been surveyed.

5. Miscellaneous sites. The Laboratory has also conducted aerial surveys of seven other sites, for the Environmental Protection Agency and the U.S. Geological Survey. At each of these sites there were special concerns related to radiological safety.

6. Ground surveys. The Laboratory has deployed high-purity germanium detectors on ground-based survey platforms for detailed analysis of the isotopic composition and concentrations at contaminated sites. In addition to the work at Enewetak, mentioned above, ground surveys have been completed at 33 additional sites.

In summary, the Remote Sensing Laboratory is a major technical resource of the U.S. Department of Energy. Hence, it plays a key role in the federal response to a radiological incident or accident. It also assists the DOE in the establishment of a Federal Radiological Monitoring and Assessment Center (FRMAC) (Doyle, 1986). Many sophisticated remote sensing systems are deployed to a FRMAC, along with an advance communications system to link the participating local, state, and federal agencies. In anticipation of emergency deployment, Laboratory personnel regularly participate in radiological training exercises; they have played key roles in 16 major exercises since May 1975.

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Status of Aerial Survey Emergency Preparedness and Ground Support Equipment, Calibration, and Sensitivities

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ABSTRACT During the course of EG&G Energy Measurements, Inc. history in aerial surveillance, the scope of response has broadened from routine surveys and accident response with aerial systems, to being prepared to respond to any radiological incident with aerial, ground mobile, and hand-held instrumentation.

The aerial survey system presently consists of four MBB BO-105 helicopters outfitted with gamma pods and specialized navigation systems (MRS or URS) that allow the operator and pilot to fly well-defined survey lines.

Minimum detectable activities (MDA) for various isotopes range from a few tenths of a mCi to 100 mCI for point sources and from 1 to 200 pCi/g for volume sources.

I. AERIAL SYSTEMS

A. Gamma Survey Equipment

A Messerschmitt-Bolkow-Blohm (MBB) BO-105 helicopter is used for low altitude gamma radiation surveys. The aircraft carries a crew of two and a lightweight version of the Radiation and Environmental Data Acquisition and Recorder (REDAR) system. Two pods, each containing either ten 12.7 cm diameter by 5.1 cm thick, or 10.2 cm square by 40.6 cm long sodium iodide, NaI(Tl), detectors, are mounted on the sides of the helicopter.

The preamplifier signal from each detector is calibrated with a Na-22 source. Normalized outputs of each detector is combined in a 10-way summing amplifier for each array. The outputs of each array are matched and combined in a 2-way summing amplifier. Finally, the signal is adjusted in the analog-to-digital converter (ADC) so that calibration peaks appear in pre-selected channels of the multichannel analyzer of the REDAR.

1. The REDAR System. REDAR is a multi-microprocessor, portable data acquisition and real-time analysis system. It has been designed to operate in the severe environments associated with platforms such as helicopters, fixed-wing aircraft, and various ground-based vehicles. The system displays to the operator all required radiation and system information in real time, via a 5-inch CRT display and multiple LED readouts. All pertinent data are recorded on 3M cartridge tapes for post-mission analysis on minicomputer systems.

The system employs five Z-80 microprocessors with AM9511 arithmetic processing chips to perform the data collection, data analysis, data display, position and steering calculations, and data recording which are all under operator control. The system allows access to the main processor bus through both serial and parallel data ports under control of the Control Processor.

The system consists of the following subsystems:

1. Two independent radiation data collection systems
2. A general purpose data I/O system
3. A tape recording/playback system
4. A CRT display system
5. A real-time data analysis system
6. A microwave ranging system with steering calculation and display

The multichannel analyzer collects 1024 channels of gamma ray spectral data (4.0 keV/channel) once every second during the survey operation. The 1024 channels of data are sent to the single channel processor and are compressed into 256 channels with partitions. Table 1 summarizes the spectral data compression performed by REDAR.

The spectrum is divided into the three partitions with the appropriate energy coefficient to make the width of the photopeaks approximately the same in each partition. The resolution of NaI(Tl) crystals varies with energy, permitting the compression of the spectral data without compromising photopeak identification and stripping techniques. In the first partition (channels 0-75), the data is not compressed permitting stripping of low energy photopeaks, such as the 60 keV photopeak from Am-241.

Table 1. REDAR Spectral Data Compression

Eγ(keV)	Channel Input	Energy Coefficient E(keV/channel)	Compressed Channel Output
0-300	0-75	4	0-75
304-1620	76-405	12	76-185
1624-4068	406-1017	36	186-253
4072-4088	1018-1022	N/A	254
>4088-analog	1023	N/A	255
cut off	1024	Unused	256

The spectral compression technique reduces the amount of data storage required by a factor of four.

The 256 channels of spectral data are continuously recorded every second. The REDAR system has two sets of spectral memories. Each memory accumulates four individual spectra. The two memories are operated in a flip-flop mode, every 4 seconds, for continuous data accumulation. While one memory is storing data, the other is being transferred to magnetic tape.

2. Helicopter Positioning. Helicopter position is established using a Del Norte UHF ranging system (URS) and an RT-220 radar altimeter. The URS master unit, mounted in the helicopter, interrogates two remote transponders located outside of the survey area. By measuring the round trip propagation time between the master and the remotes, the unit calculates the distance to each station. These distances are recorded on magnetic tape once each second and in later computer analysis of the data these distances are converted to position coordinates.

Table 2. Minimum Detectable Activity for Several Selected Radioisotopes as a Function of Source Geometries

Isotope	Minimum Detectability Activity*		
	Point Source (mCi)	Surface Source (μCi/m^2)	Volume Source (pCi/g)** α = 10cm
^{241}Am	3.0	0.5	16.0
^{137}Cs	0.5	0.08	1.3
^{134}Cs	0.4	0.06	1.0
^{60}Co	0.3	0.04	0.7
^{238}U	90.0	10.0	200.0

*Assuming a survey altitude of 46 meters

**Conversion factor to pCi/g relate to the average value of a 5 cm deep soil sample.

Under optimum conditions, range accuracy is ± 2 m line of sight to 50 km. The position accuracy under typical survey conditions is ± 5 m.

The radar altimeter similarly measures the time lag for the return of a pulsed signal and converts this to aircraft altitude. For altitudes up to 150 m, the accuracy is ± 0.6 m or $\pm 2\%$, whichever is greater. These data are also recorded on magnetic tape so that any variations in gamma signal strength caused by altitude fluctuations can be compensated.

3. Minimum Detectable Activity. Table 2 indicates the minimum detectable activity for several isotopes as a function of source geometry for the aerial system as employed in the Sandia and ITRI survey of April 1981.(1) These MDAs are typical for most surveys flown with 12.7 cm diameter by 5.1 cm thick crystals.

4. Comparison of Cylindrical and Rectangular Log Detectors. It was mentioned in the description of equipment that the helicopter can be outfitted with either cylindrical or rectangular detectors. In our terminology we refer to these as 5x2 pods or log pods, respectively.

The original thought was to increase the overall area and volume of the detector system to increase sensitivity at the higher energies. This would be particularly useful in the case of looking for lost sources like 137-Cs and 60-Co or other isotopes with higher energies. The usefulness when looking for contamination like 241-Am or other isotopes with gamma rays in the low energy region may not be as pronounced.

In order to evaluate these pods, the effective areas for two sets of two pods (counts in the photopeak divided by uncollided monoenergetic fluence at the listed energies) were determined and are presented in Figure 1.

It can be seen that at 137-Cs energies (662 keV), the difference can approach a factor of two. (Other curves of effective area vs. angle and energy are displayed on the poster.)

One interesting fact is that the overall response of the two systems when looking at distributed sources such as 241-Am show almost no difference in sensitivity. It would

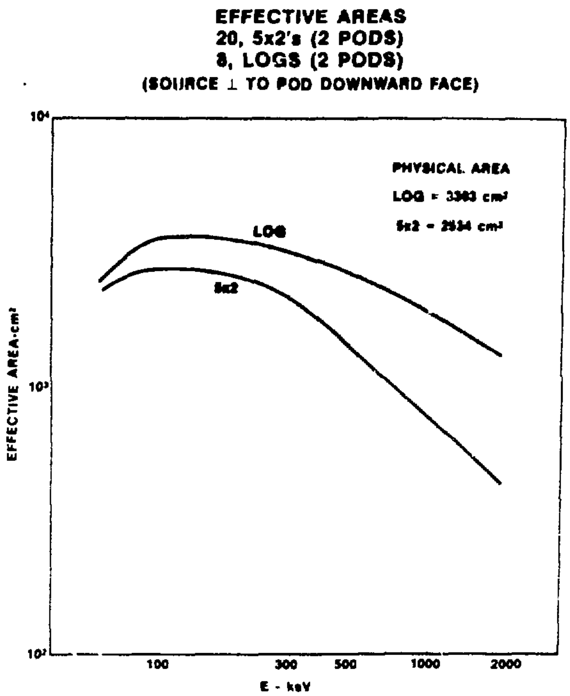

Figure 1. Effective Areas of 5x2 and Log Pods

be more useful operationally to employ the light 5x2 pods because of the weight difference (62 kg) which translates into fuel and allows longer mission times.

B. Neutron Equipment

The MBB BO-105 helicopter can also be outfitted with a neutron pod consisting of four modules of eight Helium-3 proportional counters each for a total of thirty-two tubes. Each tube is 5.1 cm diameter by 1.83 m long and is filled to 2.99 Atm pressure. Each eight tube array consists of 0.64 cm thick polyethylene tubes which are welded to form a rigid system holding the Helium-3 tubes.

The four modules are enclosed in an aluminum pod mounted between the skids on the helicopter. Each module contains its own pre-amplifier and high voltage supply and requires only +12 VDC available from the REDAR.

The principle use of this system would be the recovery of lost neutron sources; e.g., well logging sources commonly used by oil companies.

II. IN SITU GAMMA SURVEY SYSTEM

A. System Components

EG&G/EM has successfully fielded an in situ gamma analysis system employing high purity germanium gamma detectors. This system consisted of the following major components:

1. A vehicle (two types were employed, a four-wheel drive vehicle and a tracked vehicle.)
2. A HPGe detector and collimator shield.
3. A telescopic pneumatic mast capable of locating the detector from 0 to 7.4 meters above ground level (AGL).
4. A 4096 channel pulse height analyzer.
5. A computer, printer, and data storage device.
6. A microwave ranging system (MRS).
7. A 4 kW generator.

1. High Purity Germanium Detector and Collimator Shield. A HPGe detector was used to measure the gamma rays from the radionuclides dispersed in the surface and near-surface soil. The detector was capable of high energy resolution, typically 1 to 2 keV Full-Width-Half-Maximum (FWHM) of detected photopeaks. This high resolution enhanced the ability to identify photopeaks and quantify their emanating isotopes.

A lead and cadmium collimator shield was used to limit the detector field-of-view. The shield forced the detector to view a restricted solid angle and circular area on the ground. The physical angle of the cone was 50° from the vertical; however, the cut-off angle at which gammas could not enter the crystal (detector) was approximately 60° for gammas with energies between 0 and 300 keV.

The detector was calibrated using laboratory point source angular response measurements folded into a sensitivity computation. Several HPGe detectors were used with the in situ system. Each detector of the same design had a similar angular response with the shield mounted in place. There were, however, some variations in the absolute efficiency between the detectors. Conversion factors were generated based on the angular response of one detector and were applied to the other detectors with a relative response factor folded in. The relative response factor was used to correct for differences in the absolute efficiency of each detector. Detector efficiencies were checked periodically and relative response factors were adjusted as necessary.

2. Minimum Detectable Activity (MDA). The minimum detectable soil activity for the in situ gamma system was dependent on the signal count rate and the background count rate. The in situ system used a photopeak (signal) energy window and two background windows on either side of the signal window such that the total width of the background windows was equal to the width of the signal window. The MDA for a bare N-type germanium detector is shown in Figure 2.

The software in the computer of the in situ system evaluated specific photopeaks in real time from the energy spectrum obtained from each measurement. The results were printed out immediately for interpretation by the operator. These results included the isotope name, concentration in the soil, and exposure rate. Table 3 lists the routinely monitored radionuclides and the associated gamma ray energies.

More detail of the in situ system may be found in References 3 and 4.

III. OTHER SYSTEMS

A. Multi Purpose Modular Systems

EG&G/EM is developing other systems that can be used in a variety of manners in

Table 3. Radionuclides and Associated Gamma Ray Energies Routinely Monitored by the In Situ System

Source	Energy (keV)
^{40}K	1460.8
^{54}Mn	834.8
^{60}Co	1173.2 1332.4
^{101}Rh	127.2 197.9 326
^{102}Rh	475.1 697.5 766.8 1046.6
^{106}Ru	621.8
^{125}Sb	427.9 463.4 600.6
^{133}Ba	81 302.7 355.9
^{134}Cs	604.6 795.8 801.8
^{137}Cs	661.6
^{152}Eu	39.5 121.8 244.7 344.2 411.1 444 778.9 964 1408
^{154}Eu	591.8 723.3 873.2 1004.8 1274.5
^{155}Eu	60 86.5 105.3
^{174}Lu	76.5 1241.8
^{232}Th	238.6 583.1 911.1
^{238}U	186.1 351.9 609.3 1120.4
^{239}Pu	129.3
^{241}Am	59.5

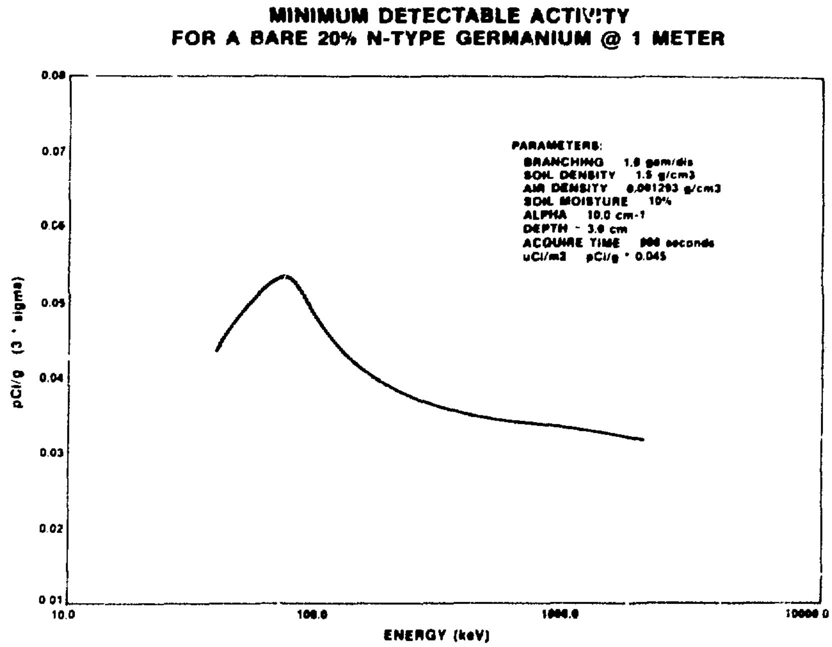

Figure 2. Minimum Detectable Activity (MDA)

non-dedicated platforms for aerial survey or looking for lost sources. These systems consist of discrete modules housing the electronics and detectors.

One example of such a system makes use of an electronics module containing batteries and associated power supplies to provide power to the detector modules. The detector modules are fiber glass containers consisting of the following:

1. Gamma module containing two "log" detectors
2. Neutron module containing eight Helium-3 tubes, 5.1 cm in diameter by 91.4 cm long in the same polyethylene configuration as described in I.B.

All the real time data reduction and system control is done in the electronics unit utilizing a NSC-800 microprocessor.

With the inherent portability of these systems they can be employed in any helicopter with sufficient space or any vehicle (e.g. station wagons or vans). One can then tap 12 VDC power from the platform or use internal batteries to power the system. Data output is in the form of audio or analog to a strip chart.

The flexibility of these systems lends itself greatly to enhanced emergency response.

This work was performed by EG&G/EM for the United States Department of Energy, Office of Nuclear Safety, under Contract Number DE-AC08-83NV10282.

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Aerial Systems Support for Nevada Test Site Weapons Testing

Philip K. Boyns

ABSTRACT EG&G Energy Measurements, Inc. operates two aircraft for the Department of Energy in support of the Nevada Test Site (NTS) activities. A King Air B-200 and a Turbo Beech aircraft are used to perform wind measurements, cloud sampling and cloud tracking operations in support of each test.

I. MISSION

Wind sounding measurements are made with a Doppler radar or a Loran C low frequency navigation system. The aircraft measure winds from approximately 1,000 feet Above Ground Level (AGL) to 12,000 feet Mean Sea Level (MSL) at multiple locations. Meteorological data are supplied to the Test Controller at the NTS before each test.

In case of a venting, the aircraft can collect whole gas and filter samples and provide the Test Controller with visual observation of cloud development and isotopic identification of the consistency of the cloud. The radiologic and navigation equipment allow long-range tracking of airborne radioactivity. The data acquisition system can provide longitude, latitude, altitude, direction, and isotopic content of the cloud on a one second basis. These two aircraft provide both pre- and post-event radiological safety information for nuclear testing at the Nevada Test Site. The aircraft and measuring systems are maintained in a state of readiness to meet routine and accident measurement requirements.

The King Air B-200 (Figure 1) and Turbo Beech (Figure 2) have the same equipment and can perform either cloud sampling or tracking missions.

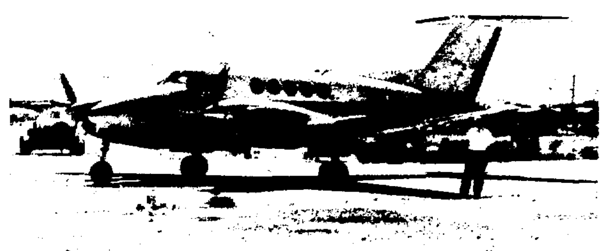


Figure 1. King Air B-200 Aircraft

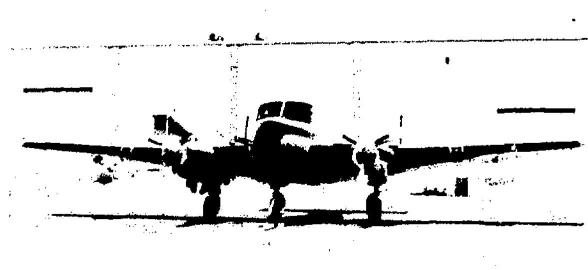


Figure 2. Turbo Beech Aircraft

II. THEORETICAL SENSITIVITY CALCULATIONS FOR TERRESTRIAL AND POINT SOURCES

A computer has been programmed in FORTRAN to compute sensitivity of an airborne detector system. The sensitivity of the detector system is computed for point, surface ($\alpha = \infty$), exponentially and uniformly distributed ($\alpha = 0$) sources. Conversion factors are also computed for concentrations of various depth soil samples versus distributions of the isotopes in the soil. Preselected isotopic distributions in the soil are 0.10, 0.33, 0.50, 1.0, 3.0, 5.0, 7.0 and 10.0 cm. A specific distribution can be entered, if necessary. The conversion factors are given in units of gammas/cm^2-sec, μCi/m^2 and pCi/g for soil samples up to a specified depth are computed in 1.0 cm increments. Model and equations used is shown in Figures 3 and 4.

All conversion factors are calculated for an isotropic, cosine (angle Θ) and an average between isotropic and cosine detecor response functions for all source distributions. A

MODEL

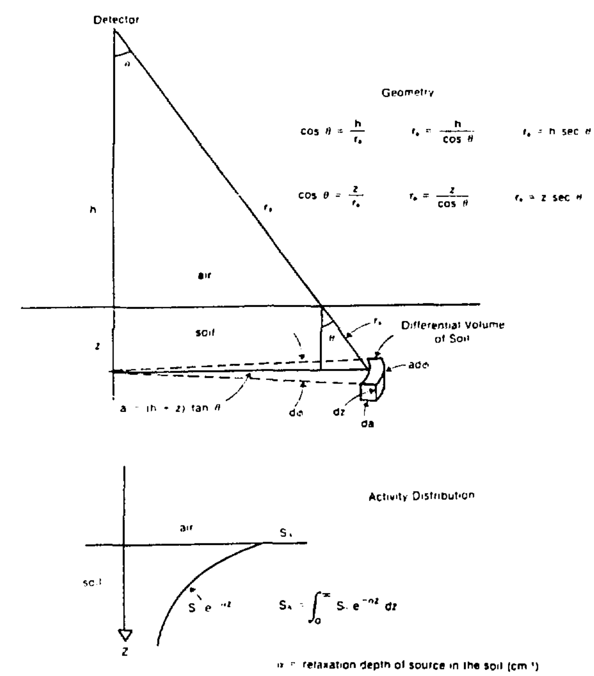


Figure 3. Model Used for Theoretical Calculations for Terrestrial Source Detectability.

$$CR_D = \frac{A S_0 e^{-\alpha z}}{4\pi r^2} e^{-r_a/\lambda_a} e^{-r_s/\lambda_s}$$

A = effective surface area of the detector system

now

$$d\,CR_D = \int_0^{2\pi}\int_0^{\infty}\int_0^{\pi/2} \frac{A S_0 e^{-\alpha z}}{4\pi[(h + z)\sec\theta]^2} e^{-h\sec\theta/\lambda_a} e^{-z\sec\theta/\lambda_s}\, dv$$

Volume

$dv = a\,da\,d\phi\,dz$

$a = (h + z)\tan\theta$

$da = (h + z)\sec^2\theta\, d\theta$

$$\int_0^{2\pi} a\,da\,d\phi = \frac{2\pi a\,da}{4\pi r^2} = \frac{2\pi (h + z)\tan\theta\,(h + z)\sec^2\theta\,d\theta}{4\pi (h + z)^2\sec^2\theta} = \frac{\tan\theta}{2}\,d\theta$$

$$\int_0^{\infty} e^{-\alpha z} e^{-z\sec\theta/\lambda_s}\,dz =$$

$$\int_0^{\infty} e^{-z(\alpha + \sec\theta/\lambda_s)}\,dz = \frac{1}{\alpha + \sec\theta/\lambda_s}$$

Substitution

$$CR_D = \frac{A}{2}\int_0^{\theta} \frac{\tan\theta\, e^{-h\sec\theta/\lambda_a}}{(\alpha + \sec\theta/\lambda_s)}\,d\theta$$

Figure 4. Equations Used to Compute System Sensitivity

special response function may be entered for the detector system. The response may be divided into 20 increments, of any width for angle from 0 to 90°.

The point source conversion factors are calculated for isotropic and cosine response functions with lateral displacements of the source up to two times the altitude. The signal Full-Width-Half-Maximum of the signal is calculated in both time and distance for each lateral displacement for a specific aircraft velocity. (Figures 5 and 6).

Point Source
Effective Bottom Detector Surface Area

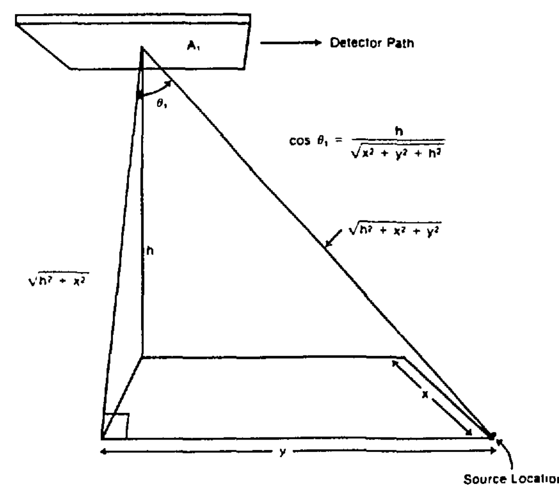


Figure 5. Bottom Surface Area of the Detector System Equation

Point Source
Effective Side Detector Area

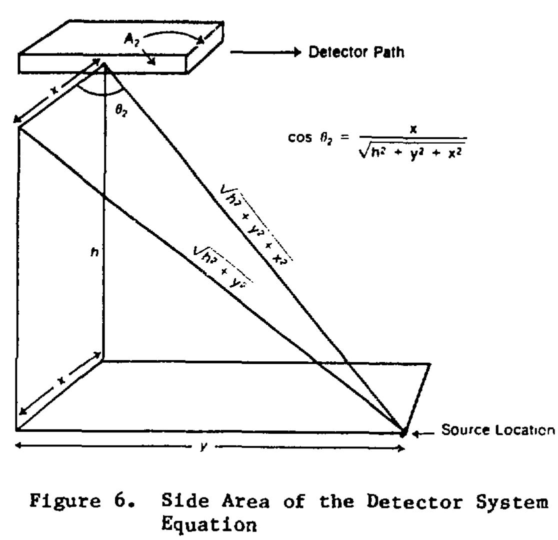


Figure 6. Side Area of the Detector System Equation

Various parameters can be plotted upon request: flux percentage vs. angle (for finite source conversion factors), flux per unit angle, flux per unit area versus angle and the pCi/g versus depth of soil samples.

Conversion factors are shown in Figure 7 for terrestrial sources of Co-60 and Cs-137. Table 1 is the minimum detectable activities for the same isotopes at 91m (300 feet) AGL, normally flown by the aircraft.

Conversion Factors Relating Aerial Photopeak Count Rate Data to Radionuclide Concentration on the Ground for a Variety of Source Distribution Geometries									
	Conversion Factor								
Radionuclide	Point Source on Surface mCi/cps		Uniform Surface Distribution		Exponential Distribution			Uniform Volume Distribution	
	Directly Under Aircraft	At Lateral Displacement of 72 m	μCi/m² cps	Relaxation μR/h cps	Depth (cm)	μCi/m² cps	μR/h cps	pCi/g cps	μR/h cps
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

Figure 7. Conversion Factors for Terrestrial and Point Sources

Table 1. Minimum Detectable Activities

Minimum Detectable Activity for Several Selected Radioisotopes as a Function of Source Geometries[a]			
	Surface Sources		Volume Source $\left(\frac{pCi}{g}\right)^b$ $\alpha = 10$ cm
Isotope	Point Source (mCi)	Distributed Source (μCi/m²) $\alpha = 0$	
^{60}Co	12.6	0.34	4.3
^{137}Cs	24.3	0.77	11.4

[a] Assuming a survey altitude of 91 meters

[b] Conversion factor to pCi/g relates to the average value of a 5-cm deep soil sample

III. THEORETICAL SENSITIVITY CALCULATIONS FOR AIRBORNE SOURCES

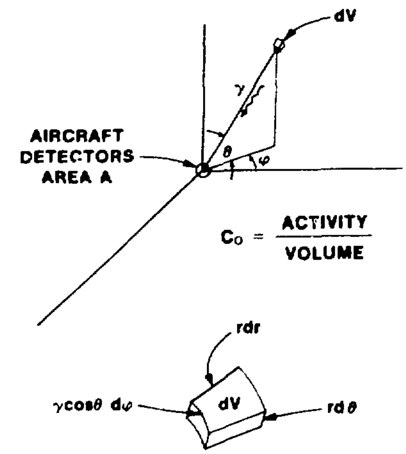
Programs were developed to calculate the sensitivity of the detector system to radioactivity uniformly distributed in a cloud. The calculations assumed the clouds were atleast $3\lambda_a$ in radius for each gamma energy (Figure 8).

Minimum detectable activities and other pertinent data are in Table 2 for airborne clouds of radioactivity for selected isotopes.

IV. EQUIPMENT ON EACH AIRCRAFT

REDAR is a multichannel analyzer that collects 1000 channels of gamma spectral data from 40 keV to 3 MeV. The spectral data are compressed into 256 channels and written on tape with various other parameters, including aircraft altitude above ground level, position data (processed by the LORAN unit into latitude and longitude coordinates), discrete data labels, and outside air pressure. The system

AIRBORNE SOURCES



THE COUNT RATE (CR) AT THE DETECTOR AREA A FROM AN INFINITE CLOUD ($r > 3\lambda_a$) OF UNIFORMLY DISTRIBUTED ACTIVITY (C_0):

$$CR = 2AC_0 \int_0^{\pi/2}\int_0^{\infty}\int_0^{2\pi} \frac{e^{-r/\lambda_a}}{4\pi r^2}\, r^2\cos\theta\, d\theta\, dr\, d\rho$$

$$CR = AC_0 \int_0^{\pi/2}\int_0^{\infty} e^{-r/\lambda_a}\cos\theta\, dr\, d\theta$$

$$CR = AC_0 \int_0^{\infty} e^{-r/\lambda_a}\, dr$$

$$CR = AC_0\lambda_a$$

Figure 8. Model Used for Airborne Sources

Table 2. Minimum Detectable Activities for Airborne Radioisotopes

Minimum Detectable Activity (MDA) for Airborne Radioisotopes

Two 4×4×16 Inch Logs
NaI (Tl)
Surface Area 2500 cm²
ρ(air) = 1.10 g/l

Isotope	E_γ (energy) MeV	I (interaction)	P (photo-fraction)	A_{eff} (area) cm²	μ_a cm²/g	λ_a cm	β (gamma/disc)	Conversion Factor pCi/l/cps	MDA pCi/l
[illegible]Xe	0.081		0.90	2250	0.15	6,060	0.37	[illegible]	[illegible]
[illegible]Xe	0.250	[illegible]	0.95	2375	0.115	7,905	0.91	[illegible]	[illegible]
[illegible]Kr	0.514	0.96	0.74	1795	0.0865	11,560	0.0041	[illegible]	[illegible]
[illegible]Rb	1.86	0.80	0.41	820	0.0460	19,760	0.21	[illegible]	[illegible]
[illegible]	2.40	[illegible]	0.37	712	0.040	22,721	0.35	[illegible]	[illegible]
[illegible]	1.74	0.85	0.49	[illegible]	0.058	16,233	0.37		
	1.28						0.34		
	1.46						[illegible]		
							0.83	[illegible]	[illegible]
[illegible]	[illegible]	0.82	0.44	902	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	0.23		[illegible]	2250	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

also provides the capability for realtime onboard spectral analysis. All of the data are written on tape once every second to provide a record of the mission for extensive post-mission analysis.

The gamma detector package consists of two 4 x 4 x 16 in. NaI(Tl) log detectors and one 1 in. diameter x 2 in. thick NaI(Tl) detector. The detector sensitivities depend on the energy of the penetrating gamma rays and the branching ratio of the parent nuclide; hence, only approximate sensitivities can be given:

Airborne - 2 pCi/liter (5 μR/h)
Ground deposition - 1.0 μCi/m^2 (10 μR/h)
Exposure rate ranges -
Logs: 5 to 100 μR/h
1 x 2: 50 μR/h to 6.0 mR/h

HPI tissue equivalent ion chamber data are recorded by the REDAR system every second. Dose rate is recorded from 1.5 mrad/h to 10 rad/h.

A 660 Victoreen Integrating Exposure Meter provides a total integrated exposure to a maximum reading of 100 R.

A XETEX 305B Digital Exposure Rate Meter provides an auto-ranging display in units of mR/h or R/h.

Two strip chart recorders are used to plot various parameters recorded by the REDAR system, usually the radar altitude and the gross count rates from the log detectors and the 1 x 2 in. crystal.

Portable survey instruments:

a. Eberline E-500B
b. Eberline RO-2A
c. Baird Atomic

High volume charcoal and IPC-1478 paper filters-

Paper filters:	four each 10.16 cm (4 in.) in diameter
Charcoal filter:	22.86 cm (9 in.) x 26.64 cm (10.5 in.) x 3.175 cm (1.25 in.)
Flow Rate:	700 l/min (25 ft.^{3}/min)

The whole gas sample system includes a 5.08 (2 in.) IPC paper filter with a variable flow rate up to 700 l/min (25 ft.^{3}/min). The air filter can be removed and analyzed with a 2 x 2 in. NaI crystal that is shielded with 0.5 in. of lead. The spectral data are displayed and recorded by the REDAR system. Whole gas samples are compressed into bottles for post mission analysis.

LORAN C navigation system provides longitude and latitude for real time displays and are recorded on magnetic tape every second ($\pm$1500 feet).

Auxiliary equipment outputs are also recorded every second:

a. REDAR Altitude (0-2000 ft. AGL)
b. Absolute Air Pressure
c. Outside Air Temperature (OAT)

This work was performed by EG&G/EM for the United States Department of Energy, Office of Nuclear Safety, under Contract Number DE-AC08-83NV10282.

Mobile Robot Response to Actions Associated with the Release of Hazardous Materials

Harvey B. Meieran

ABSTRACT: This paper presents a rational and composite summary of tasks and missions that could be assigned to mobile robots and other teleoperated devices in response to accidental releases of radioactive and other hazardous/toxic materials to the environment. This paper will also discuss specific missions that have been, or could be, assigned to mobile robots operating at the TMI-2 and Chernobyl-4 nuclear power plant sites. Other items and issues that will also be considered are: availability of applicable mobile robot units, expendabilty/durability/decontamination, portability/maneuverability/mobility, communication techniques, power supply, and artificial intelligence/autonomous navigation interactions.

I. INTRODUCTION

Many man-made and natural disasterous accidents precipitate loss-of-life and injuries among persons residing in the vicinity of the incident. Furthermore, releases of contaminants from the incident can also create a long-term (i.e., more than a few days) harsh environment for those who survived the initial effects from the disaster as well as to members of the emergency response teams who are trying to mitigate the consequences of the disaster.

The monitoring, rescue, relief, and cleanup actions associated with the accidental release of radioactive, chemical, biological, and other toxic materials has been historically handled by emergency response team personnel. These incidents also include fires, security, and civil disorder sitations.

Even though individual emergency respose team members can be appropriately attired to protect them from the effects of the hazardous material contaminants, they remain to some degree susceptible to exposure of these materials. Furthermore, the efficiency of their motions and general activities in the affected areas can become compromised and somewhat restricted. On the other hand, the transference of some of these manipulative and monitoring functions to currently available mobile robots and other teleoperated vehicles can eliminate the probability of exposures by the relief worker to the potentially harmful health hazards.

Activities associated with the development and deployment of mobile robots and/or teleoperated devices which may conduct repair, maintenance, decontamination, decommisisioning, and surveillance/inspection missions in nuclear power plants and other nuclear industry facilities are being pursued by a spectrum of domestic and international organizations. These activities are being complemented with parallel efforts for robots in other "hazardous industries", such as explosive ordnance disposal (or bomb disposal, EOD) fire-fighting, and mining.

This paper will discuss the applicability and technologies of currently available vehicles which can assist in responding to incidences associated with the accidental release of hazardous materials. These vehicles have been manufactured/assembled in nine different countries, as noted by Gelhaus and Meieran (1985), and represent a spectrum of locomotion techniques, sophistication of design and operation characteristics, and degree of autonomy and onboard/offboard intelligence. They have been operating in various "hazardous industries", such as nuclear, military and civilian EOD, toxic material handling, general security, civilian civil disorders, fire-fighting, and mining. The commonallity of configurations, applications, and mission assignments for these vehicles enables them to be employed in hazardous situations other than those for which they were originally designed. For example, a robot that was designed for civilian EOD activity has been employed in the nuclear industry. Conversely, a reconfigured mobile robot that was initially designed for the nuclear industry will be employed in the civilian EOD "industry" and can be used to monitor the concentrations of released (to the atmosphere) toxic chemicals.

II. MISSION ASSIGNMENTS

A. General Considerations.

There are many common characteristics associated with disasters which have occured in several non related industries and situations; these industries are, for example, nuclear power, chemical plants, accidents of systems transporting hazardous materials, and volcanic eruptions. The bottom line for these accidents is the presence of a lingering harsh environment created by the accident which could be detrimental to the survivors of the accident as well as to members of the emergency response team.

The common features for these accidents in which there are real or threatened initial or residual deaths and/or injuries can be listed chronologically according to the following ten categories: a) initial human deaths; b) initial human injuries; c) initial livestock deaths; d) creation of a hazardous/harsh environment for the survivors and members of the emergency response/rescue/relief teams; e) attempt to eliminate the main source of the cause of the incident (problem); f) evacuation of the survivors from the scene/locality of the incident; g) restoration of the scene of the accident (if possible) and dead livestock burial; h) delayed deaths and/or injuries to the survivors; i) delayed livestock deaths, intentional slaughter, and burial; and j) residual, long-term decontamination of the local environment (primarily radioactive particles from nuclear power accidents).

It can be seen in Table 1 that there are many common characteristics shared by several well known and exemplified accidents which have recently occurred. The most common and frequent characteristics are: creation of an environment that can be detrimental to the rescue workers (a health hazard); eliminate the main source of the cause of the incident; and evacuation of personnel and survivors of the accident from the affected area (most mandantory and one voluntary), and the long term contamination left as a residue from the accident.

B. TMI 2 Accident.

In the case of the TMI 2 accident, which occurred on March 28, 1979, the contamination was (and still is) attributed to the radioactive particles left within the structure of the reactor containment building. The relatively low quantities of I 131 released to the outside environment decayed to insignificant levels within days after the accident. Up to 50,000 people voluntarily evacuated the areas around the reactor site for a few days. There are currently five operating robots performing a variety of tasks; two robots have been retired from use. None of these robots were installed into TMI 2 prior to 1983, or more than 4 years after the accident occurred.

C. Chernobyl-4 Accident.

In the case of the Chernobyl 4 accident, which occurred on April 26, 1986, the initial two deaths were caused by the fire in and destruction of the reactor building. Several individuals died from accute radiation exposure within days after the accident and the prognosis for others who have been hospitalized is somewhat pessimistic. The environs surrounding the reactor site were heavily contaminated and forced the mandantory evacuation of more than 135,000 persons from a 2830 sq km (1000 sq mile) area around Chernobyl. It was also necessary to destroy and bury a considerable amount of contaminated livestock. The area continues to remain contaminated and local residents are not expected to be able to return to their residence for periods of up to four years.

The operators of the Chernobyl plant have purchased (or leased) at least three mobile robots from Germany the two-tracked MF2 and four-tracked MF3 robots and a 33 Mtonne remote controlled bulldozer. These have been operating at the Chernobyl site within one month after the accident occurred.

C. Lake Nios, Cameroons.

The chemicals released on the night of August 21, 1986 from the bed of the volcanic Lake Nios, Cameroons, instantly killed more than 1500 resi dents and left more than 500 injured. Survivors were later eventually evacuated. Authorities initially expressed some concern over the lingering chemical contaminations which could cause some health problems to the survivors and to members of the rescue teams. There is a stronger and more realistic concern regarding the health hazards being generated by the decay of thousands of head of cattle that were instantly killed by the initial release of chemicals from the volcanic lake.

D. Other Incidents/Accidents.

A large fraction of transportation accidents reported almost daily to FEMA involve trucks and/or trains carrying hazardous and toxic chemicals. When there is a release of these materials, which are frequently accompanied by fires and explosions, the local residents are ordered to evacuate the area. Some recent accidents have forced the evacuation of thousands of people from their residences; they have also been forced to stay away from the residences for periods in excess of several days. The accident at the Bohpal, India Union Carbide plant killed more than 2000 people and injured another 200,000, many of whom were evacuated. These three accidents have also been included in Table 1 as examples of incidents that have similar characteristics to those associated with nuclear power plants.

E. Locations for Mobile Robotic Vehicle Activity.

The locations where mobile robots could be employed in and around these accident sites are: a) indoors (possibly more than one floor or elevation); b) within the physical site boundary for the facility having the accident; and 3) in and around an exclusion area (external to the physical site boundaries) surrounding the affected zone, to include structured and unstructured terrains.

F. Specific Mission Assignments.

It can be seen from the discussion in Section II.A. above and from Table 1 that there are several common categories for which mobile robots could have been used to mitigate the concequences of accidents. First of all, it should be noted that the use of a mobile robot(s) in the specific accidents listed in Table 1 is a speculated consideration. Robots may not necessarily be available to the rescue team at the time they would be most needed. Secondly, suitable mobile robots may not have existed at the time an accident, as in the case for TMI-2.

The primary category which will dictate the need for a mobile robot is d), the development of a harsh environment which can become a hazard to the health of the rescue/relief worker. If there is no hazard, then there is little incentive to employ a mobile robot and little chance that the vehicle could conduct a mission/task that would be more efficient or cost effective than that which could be conducted by a human worker. For this reason it is not anticipated that the utilization of mobile robots could offer a reasonable alternative solution to the employment human rescue workers at the location of an earthquake where hazardous environments, outside of collapsing buildings, would not be expected.

Specific suggested missions associated with each relevent category are listed in Table 2. Some of the specific tasks associated with the various activities are amplified below: security - crowd control (during evacuations) and patrolling deserted streets and environs from which the local population has been evacuated. Core samples are being drilled and taken from the concrete walls of the TMI-2 reactor; these samples will be analyzed for fission product penetration which in turn will assist planning personnel to develop an efficient strategy for decontaminating the reactor's facilities. The carcases from dead and destroyed animals should be buried in an excavated trench as soon as possible (to limit the onset of health hazards among the survivors). The specific activities associated with other missions listed on Table 2 are described by the title of the mission.

III. DESIGN/PERFORMANCE CONSIDERATIONS

A. Locomotion.

1. Mobility/portability.

Specific mobility aspects that would have to be considered are levels of operation in a building (more than one floor), complexity of infrastructure within a building (stairs, location of pipes and other obstacles, width of passageways and doorways, height of walls and ceilings in buildings), type of structuring in an outdoor terrain (texture of surface, placement and height of obstacles, distances to travel).

The robot should be easily transported to the scene of the accident in a small van or a small truck (less than 1 ton) It should also be able to travel under its own power to the final destination. Delays in committing its utilization may cause the rescue workers to be unnecessarilly exposed to hazardous materials.

As it is probable that, under some circumstances, a vehicle may not have the ability to travel to an inaccessable location under its own power, it may be necessary to manually carry that vehicle to the work station. The overall weight of a vehicle should then be restricted to a level that will enable it to be carried by no more than four persons up a flight of stairs.

2. Maneuverability. The *maneuverability* of the vehicle should consider whether or not the robot will be operating in a confined indoor, multiple level indoor, structured outdoor terrain (parking lots, roads) or unstructured outdoor terrain (cross-country, around destroyed facilities).

3. *Locomotion techniques.* Most of the current generation of mobile robots possess either wheeled or track-type locomotion techniques. The third basic technique, legged, has been relegated to the role of a research item at this time. The criteria for the selection of the specific form of locomotion is based upon the requirement for mobility and specific placement of the robot in an outdoor or indoor environment.

B. Cost/Benefit Analyses.

Under normal and non-emergency circumstances, the robot operator may have some reluctance to assign the robot to a mission where there is a finite chance that the vehicle could become lost. Under emergency situations, however, the philosophy of operation demands that the consequences of the hazardous situation be compromised as soon as possible so that there is a minimum loss of life or threat upon the health of the community, and finally to minimize damage to the facilities. The operator thus has the freedom to consider the robot more expendable under emergency situations than under normal situations.

One of the key issues regarding an assured and durable life for the robot is its ability to stay relatively "clean". A device will soon lose its ability to conduct missions if it is not possible to restore the robot to its "mint" condition. The levels of contamination and durability will be dependent upon the design features which will: a) minimize the number and exposure of as-fabricated crevices; b) use suitably hardened components that will not rapidly degrade under high levels of radiation; and c) use materials of construction that will not deteriorate under high levels of radiation nor in corrosive atmospheres; and d) the ability of the vehicle to withstand washdown decontamination actions for either radiological or chemical contaminant agents.

It is desirable that there will be a high degree of assurance that the robot will continue to function uninterrupted during its missions in an emergency situation. That is, the vehicle should have a high availability factor. This assurance can be gained if the robot designer utilizes prudent design and planned maintenance programs (Meieran, 1984). Spare parts and prudent operating scenarios will be able to extend the operating life of the vehicles.

C. Communication Techniques.

There are two basic modes for communication between the vehicle operator and the robot: untethered and tethered. In the former case the communication is usually by radio (RF, UHF, microwave) or by infrared. Other wireless methods have been considered but are not common. The tethered mode is based upon coaxial cable, fiber-optic cable, or twisted pair wires. The former technique permits the robot to be highly mobile and not restricted to travelling distances limited by the length of the cable, along as the power supply package is located on-board the robot. Distances of communication can exceed 2 km (1.2 miles).

The limitations in communication for the teterless robot can be attributed to RF interference and line-of-sight obstacles.

D. Power Supply.

There are two modes of power supply: on-board or off-board systems. In the former case the vehicle can be tetherless (i.e., no umbilical cord from the control station or a stationary power supply system), other than that which may be required for communication between the operator and the vehicle. In the latter case a tether will be required from either a portable power supply system (usually a gasoline engine/generator) or from house power (plug in to the nearest available receptacle). The untethered vehicles will of course be able to have the higher levels of mobility and the greater degrees of freedom. The travel distance for the tethered vehicles will be limited by the length of the umbilical cord and the form of the paths for travel with obsta cles around which the cord could not move.

The untethered robots are generally powered by batteries (or gasoline engines in a few cases) and the tethered robots by house power (110/220/440 VAC). The limitations for use of the battery powered vehicles will be dictated by the time that the robot can operate between battery recharging periods (which usually lasts between 1 and 5 hours). Trickle chargers can extend the period of uninterrupted operation by only a relatively short time. On the other hand, the period of uninterrupted operation for an AC-powered robot is unlimited.

E. Artificial Intelligence/Autonomous Navigation.

No currently available off-the-shelf mobile robot possesses any degree of practical on-board or off-board intelligence which will enable the vehicle to independently transport itself to a workstation (autonomous navigation) and then conduct its mission without having further human instructions. These degrees of sophistication and intelligence are being pursued by several current government/non-government sponsored projects.

F. Status of Current Technology.

In a recently completed survey (Gelhaus and Meieran, 1985), it was noted that there are 69 separate models of mobile robots that are used to conduct a variety of surveillance, inspection, and manipulative missions in a spectrum of hazardous environments. It was also noted by Meieran (1986a and 1986b) that most of these vehicles that were still functioning could be used to conduct missions in and around nuclear facilities during normal circumstances and to a lessor extent they can be utilized as assistive devices during radiological emergencies.

IV. SUMMARY AND CONCLUSIONS

The capabilities and advantages of remote controlled (teleoperated)/robotic mobile vehicles are now coming to the attention of those individuals who are charged with the responsibility to direct efforts to mitigate the consequences of radiological and other hazardous incidents. The direct employment of these technologies can remove the emergency response team member from a harsh environment created by the emergency and thereby eliminate the potential of having the individual exposed to the hazardous materials in that environment.

The optimum features that should be possessed by a currently available, off-the-shelf mobile robot that is directed to respond to radiological and other hazardous environment emergencies are included in the following items: relatively light weight - less than 175 kg (385 lbs) for a vehicle that must work indoors (as well as outdoors) and less than 700 kg (1500 lbs) for a vehicle that is committed to an outdoor terrain having some obstacles; basically able to travel through standard doorways and climb stairs; battery or gasoline fueled power supply; teleoperator controlled; untethered; and an optional manipulator that should be able to lift/transport more than 20 kg (44 lbs). The recent advances in minaturization and the employment of microprocessors and computers have enabled mobile robots to become more reliable and versatile in their missions conducted in the hazardous environment. There are, however, several issues which may limit the ability of these devices to respond with 100 % assurance in these situations; these issues are being addressed to by several national and government-sponsored R & D programs, as for example the Electric Power Research Institute (Palo Alto, CA) and the CESAR facility at the Oak Ridge National Laboratory.

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TABLE 1
SUMMARY OF RECENT INCIDENTS INVOLVING RELEASES OF HAZARDOUS MATERIALS

	Nuclear		Chemical			Natural
			Plant	Transportation		
Chronological Sequence	TMI-2 PA	Cherno-byl-4 USSR	Bohpal India	Miamis-burg OH	San Antonio TX	Lake Nios, Cameroon
a Initial Human Deaths		X	X			X
b Initial Human Injuries		X	X			X
c Livestock Deaths		X	X(?)			X
d Hazardous to Rescue Workers	X	X	X	X	X	X
e Eliminate Main Source of Inci.	X	X	X	X	X	
f Evacuate Survivors	50K	135K	>200K	2K	>1K	.5K
g Bury Dead Livestock		X	X(?)			X
h Delayed Inju-ries + Deaths		X				
i Delayed De-struction & Burial Live.		X				
j Residual Long-Term Contamin.	X	X				X

TABLE 2
SUMMARY OF SPECIFIC MISSION/TASKS TO BE CONDUCTED
BY MOBILE ROBOTS DURING HAZARDOUS INCIDENTS

	Chronological Sequence					
Mission/ Task	Initial Human Deaths Injury	Elimi-nate Main Problem	Evacu-ate Survi-vors	Bury Dead Live-Stock	Delayed Live-Stock Destroy	Residual Long-Term Contam.
Security	X	X	X	X	X	X
Radiation Moitoring	X	X	X	X	X	
Chemical Monitoring	X	X	X	X	X	X
Core Sampling		X				X
Bury Livestock Trench Construct	X	X		X	X	
Destroy Livestock					X	
Sample Aquisition						
Sludge	X	X				X
Air	X	X				X
Smear	X	X				X
Video Surveillance	X	X	X	X	X	X
Fire/Chemical Fight		X				X
Tunneling		X				
Reactor Repair		X				
Reactor Maintenance		X				
Decontaminate						
Scrape/Scabble		X				X
Spray		X				X
Coat/Strip		X				X

Communications Systems for Emergency Deployment Applications

Charles A. Gladden

ABSTRACT The Emergency Response Team (ERT) communications system was developed by the U. S. Department of Energy (DOE) to provide radio and telecommunications service for scientific and management elements located in, and adjacent to, an emergency area. The telephone system consists of six nodes, interconnected via microwave links that support T-1 data links and simultaneous two-way live video. The radio network is a self-contained VHF system arranged around portable and programmable repeaters. The system is comprised of approximately 183 DES voice-private radios and 168 clear text radios. Capability is available in the form of portable International Maritime Satellite (INMARSAT) terminals that allow direct dial access to coast earth stations in the U. S. or other countries.

Communications systems for emergency field deployments have been developed by several organizations such as the National Fire Radio Cache, FBI, White House Communications Agency (WHCA), and military, etc. With rare exception, however, these are developed to provide radio coverage over the immediate area of the emergency and do very little to address the problem of providing voluminous telecommunications within the emergency area or linking the emergency area with the undamaged portion of the international dial network.

During the Cosmos incident (Morning Light) in January 1978, and the Three Mile Island incident in March 1979, it became increasingly obvious that all communications support should be transportable and independent of local communications systems. With this in mind, the Department of Energy's Emergency Response Team (ERT) communications system has been developed around the philosophy that communications in, or adjacent to, the emergency area were non-existent prior to the incident or would be damaged or severely overloaded as a result of the incident. With this established as the basic problem, systems have been developed which provide both radio coverage of the emergency area and telephone service which include the necessary microwave links to assure interface to the outside international dial network. A part of this microwave system is also used to transport live video as well as telecommunications between the several major DOE operations nodes within the emergency area.

One of the more important major elements is the portable telephone system that provides telecommunications between various scientific and management elements located in, and adjacent to, the emergency area. This telephone system consists of six nodes, packaged and assembled in ruggedized packing cases, that serve as operating racks as well as transport containers. These nodes are interconnected via ERT microwave links that support T-1 data links and simultaneous two-way live video. The microwave links are installed in "real time" on any available high points such as buildings, towers, mountains, etc. and allow the majority of command and scientific structures to function at safe distances from nuclear emergencies. In the event trunk circuits are unavailable in the immediate area, up to 24 circuits can be procured at a remote location (up to approximately 50 miles) and transported using existing multiplex and microwave assets. Trunk capacity of the basic telephone system is 48 "loop start" central office trunks, with an additional 48 trunk circuits that can be accommodated in a T-1 format. Station line capacity is 261 lines when all nodes are fully deployed. The six major nodes are the Technical Operations Center, Command Post, Forward Control Point, Working Point, Forward Staging Area, and Main Staging Area.

The radio network is a self-contained VHF system arranged around portable and programmable repeaters. The system is comprised of approximately 200 DES voice private radios and 200 clear text radios. All are portable hand radios and have an R.F. power output rating of approximately six watts.

VHF radio network installation normally begins with arrival of the ERT advance party on scene. The first repeater cell is installed in the vicinity of the advance party command post or in the vicinity of the incident site, if required. Repeater cell coverage is progressively increased by mounting equipment on available high points (mountain tops, towers, etc.) until the area of emergency has thorough R.F. coverage with a minimum of three nets. This coverage area typically has a 25-50 mile radius. Slave repeaters are often used to augment the coverage in shadow areas or in major buildings and complexes with high R.F.attenuation factors. All repeaters are regenerative when operating in the digital mode, thus allowing traffic to be repeated 2-3 times with minimal bit error rate, before arriving at the final destination.

Limited capability has been developed in the form of portable International Maritime Satellite (INMARSAT) terminals that allow direct dial access via the appropriate Pacific or Atlantic satellite, to coast earth stations in the U. S. or other countries. Although this has been used primarily in exercises, its potential value in responding to small to medium-scale accidents/incidents in isolated areas is immense. A prime example of this would be to provide emergency communications in the event of an accident in an isolated area involving weapons damage or radioactive dispersal. This earth terminal has been utilized very successfully in the annual DOE inspection and scientific analysis of the Amchitka project in Alaska. Its small size (approximately 3 standard luggage cases) allows it to be transported on commercial airlines as luggage. Setup time is approximately 15 minutes, and the use of a relatively small 4-foot dish allows the system to remain operational in 40-50 knot winds, if properly sandbagged.

The ERT communications system is serviced, tested, and stored in a western U. S. location. Deployment times vary with equipment requirements; however, the advance party can normally be ready to depart with all equipment in two hours, with a complete deployment taking approximately 8-10 hours.

Future improvements are planned in several areas; however, the most critical is the assembly and fielding of the multichannel satellite system which will provide the ERT complete independence from any existing emergency area communications. Other major improvements planned are to convert the microwave baseband to an all digital operation, convert the VHF radio network to a cellular-satellite operation, and procure 1.5 Mbps live motion video compressors.

In summation, we feel Emergency Response Team emergency communications system meets the objectives of the DOE by providing a portable, lightweight, versatile, high capacity system that can be fielded in a reasonable time and be transported by military air cargo or commercial passenger carrying aircraft. Various incidents and exercises have proven the system can be effective in providing emergency communications in areas where no other systems exist, have been destroyed or overloaded.

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A Mobile Laboratory—Emergency Sample Analyses at the Accident Site

R. H. Wilson

ABSTRACT The Pacific Northwest Laboratory (PNL) provides technical support to the Department of Energy (DOE) with rapid response to the site of a radiological accident for initial assessment of radiological conditions in the surrounding environs. A mobile laboratory manned with a trained crew is capable of collecting, preparing, and analyzing air, liquid, vegetation, and soil samples from the areas surrounding the accident site. Rapid assessment of radiological conditions is important to enable responsible agencies to immediately administer appropriate response and coordinate any necessary longer-term management of the accident site. Sensitive instrumentation and support equipment provide the capability to rapidly assess environmental conditions for uncontrolled areas as stated in DOE Order 5480.1 Chapter XI. A regimen is required to maintain proficiency and to train new crew members. A professional team responding with a well-equipped mobile laboratory capable of providing quick analytical results has a positive impact on public relations. Confidence is also generated among other agencies with an awareness of the technical expertise that is available for assistance and backup in the event of a radiological accident.

INTRODUCTION

The Federal Radiological Monitoring and Assessment Plan (FRMAP), a part of the Federal Radiological Emergency Response Plan (FRERP) establishes a means to request and provide federal radiological assistance and a framework to coordinate radiological monitoring and assessment activities of government agencies in support of federal, state, and local government activities.

The DOE coordinates all federal offsite radiological monitoring and assessment operations during the initial phase of emergency response. The DOE has the responsibility to provide personnel and equipment required to coordinate and perform offsite monitoring and evaluation activities in cooperation with other federal components. The DOE is also designated the cognizant federal agency (CFA) for both onsite and offsite radiological events involving DOE-owned materials. In addition, the DOE is also responsible for maintaining national and regional coordination offices as points of access to federal radiological emergency response.

The DOE operations office in Richland, Washington (DOE-RL) has been designated as the point of access for Region 8 (Washington, Oregon, Alaska). To assist DOE-RL in fulfilling its obligation for response in the event of a Region 8 radiological emergency, PNL is charged with providing appropriate technical support capabilities.

Of primary concern to both PNL and DOE-RL is the rapid and appropriate response to a radiological incident. Following the initial assessment by DOE and PNL, it may be concluded that more sensitive and detailed analyses are required to characterize radiological conditions and define the affected area as quickly as possible. A mobile unit equipped with the proper instrumentation is required to make rapid on-the-spot analyses that identify the radiocontaminants and their concentrations in the environment. With the in-depth measurements provided by a mobile unit, governmental agencies and local authorities can immediately administer appropriate follow-up actions and coordinate any required longer-term management of the accident site.

A mission statement was developed for a mobile unit and from that mission statement the following specifications were set forth to provide the level of analytical capability and support required by DOE-RL:

1. space for the instrumentation and support equipment necessary to make a comprehensive assessment of an accident area in a maximum of three days
2. overnight living accommodations for a maximum of four crew members
3. capability for deployment, operation, and return through the mountainous terrain of remote areas (exclusive of travel on unimproved roads) where supplies may be limited
4. auxiliary power-generating capability sufficient to independently operate laboratory equipment without replenishment for a maximum of three days
5. equipment capable of detecting concentrations of gamma-emitting radioisotopes associated

with major radiological accidents (the instrumentation should be sensitive enough to determine whether or not concentrations in liquid and air samples are above 10 CFR 20 Table II concentration guides and the instruments should be capable of determining comparable concentration levels for vegetation and oil in short counting times, ≤5 minutes)

6. equipment and supplies for the collection, preparation, counting, and analyses of a minimum of 50 environmental samples per day
7. capability to communicate with the Unified Dose Assessment Center (UDAC) facility in Richland when distance and terrain will allow and with crew members working away from the unit
8. dosimeters and read-out instrumentation capable of determining radiation dose in the environment and the dose received by personnel working in the radiation environment
9. mobilization capability within one hour.

MOBILE LABORATORY DESCRIPTION

The mobile laboratory is a 27-foot-long fifth wheel trailer that has been converted into a laboratory for use in the field. A large pickup truck with four-wheel drive and a crew cab is used to pull the mobile laboratory and transport material, equipment, and people.

The basic trailer was built by Conestoga. The Snobird-model trailer interior was redesigned to accommodate counting equipment and still maintain a reasonable level of comfort for the crew. Doors, counter space, and cupboards were rearranged and a heavy support beam was installed underneath the floor above the axle where the heavy lead counting chamber was placed. The dry weight of the redesigned trailer was about 6770 pounds on delivery to PNL. After the equipment and materials were installed, the dry weight of the trailer was 10,000 pounds.

Some of the major items installed in the trailer for the crew's comfort are a 6-ft^3 gas/electric refrigerator, 24,000-BTU ducted furnace, 6-gal hot water heater, 40-gal water tank and pressure water pump, single sink, 40-gal gray water tank, wet-bowl toilet, shower, 27-gal waste tank, air conditioning, storm windows, awning, 6.5-kW-rated generator with 10-gal gas tank supply, two twin beds, and Weathertronics recording wind system. A radio base station was installed with charging units for four portable radios.

COUNTING EQUIPMENT

The detector used in the trailer is a EG&G ORTEC GEM Series HPGe (high-purity germanium coaxial detector system) and a Tracor Northern TN4000 analysis system, keyboard and display, and a printer. A special counting chamber was designed to allow the sample container (450-mL Marinelli beaker) to be raised into counting position and yet have 4 inches of lead shielding on essentially all sides. A precision track with roller bearings was installed in the trailer floor to hold the counting chamber and allow easy movement to its counting position under the detector or anchor it in the center of the trailer for the travel position. Because of its weight (about 800 pounds) positioning and anchoring the counting chamber is very important. If the lead chamber [illegible] loose during travel it would probably go through the wall of the trailer during a turn. A pulley arrangement lowers the detector to the counting position or holds it securely in its travel position.

The detector has been calibrated for four different counting geometries using 450-mL Marinelli beakers containing liquid, soil, vegetation, and 2-in-diameter air filter papers (see Table 1. Counting Efficiencies). Calibration is accomplished using a National Bureau of Standards (NBS) source composed of ^{125}Sb, ^{154}Eu, and ^{155}Eu which supplies several calibration points over a wide range of energies. Other standards such as ^{60}Co, ^{137}Cs, and ^{241}Am are also available. The equipment has been very stable and recalibration is scheduled on a quarterly basis only. A table of calibration factors (Table 2) has been established for the four counting geometries across a broad spectrum of energies so the efficiency for any energy can be determined with reasonable accuracy. Detection levels for gamma-emitting isotopes are well below the concentration limits stated in Table II of 10 CFR 20, Appendix B and in DOE Order 5480.1. These levels are stated for individuals; one-third of these values are considered acceptable for a suitable sample of the population. This sensitivity can be achieved using one-minute counting times.

THE MOBILE LABORATORY TEAM

Fourteen staff members from the PNL Health Physics Department having a variety of expertise in dosimetry and instrumentation make up the mobile laboratory team. These people are on a call list, and in the event of a radiological incident a minimum crew of four team members will be assembled within one hour. During an incident members would be alerted to stand by if the need for the mobile laboratory services had not been determined immediately.

Crew member training is conducted on a quarterly basis. During these training exercises varying aspects of operation such as the preparation for travel, travel to some location, set-up for operations, sample preparation, sample counting, and evaluation of results are practiced. It is not possible to train all members at one time because of other work commitments. A training session usually consists of five or six members and all members participate at least twice a year.

The operational duties of the mobile laboratory crew have been divided into five functions: team leader, communicator/recorder, counter/analyzer operator, sample preparer, and thermoluminescent-dosimeter (TLD) reader operator.

The team leader will be appointed by the PNL Emergency Preparedness Manager or his delegate at the time team members are notified to activate the mobile laboratory. If this individual has not been appointed at the time of notification, the first member arriving to activate the laboratory shall act as team leader to ensure the unit is readied for travel and all members of the crew are assigned to the tasks necessary for the laboratory, pickup truck, and equipment operation.

The communicator/recorder is responsible for maintaining communications with the base station (usually UDAC), crew members, and other teams with portable radios working in the field. A record of all activities will be maintained in a mobile laboratory logbook starting with the preparation for travel and closing with the return to the storage location.

The counter/analyzer operator is responsible for preparing the equipment for operation and checking the supplies for operation in the field.

The person assigned to sample preparations at the accident site also participates in the inventory of equipment to prepare the laboratory for travel.

The TLD-reader operator will have responsibility to ensure that the reader and equipment are operable and adequate supplies are in the laboratory. This operator will also assist in preparation for travel, inventory of equipment, and set-up for operations at the work site.

Because of the versatility and expertise of crew members, work assignments can be changed to relieve crew members or to concentrate effort in some area of the laboratory operation such as sample preparation. This function is time-consuming; when a large number of sample results are required, most of the crew members direct their efforts in this function.

PUBLIC RELATIONS ASPECTS

Wherever the mobile laboratory goes, whether it is on display or involved in an emergency response exercise, it stimulates a great deal of interest among people who work with or tour it. People participating in large emergency response exercises involving federal, state, and local governments, military and civilian agencies are intrigued with the capability, sensitivity, and amount of equipment that is readily portable and available in the field.

In the local communities where radiation is still a mysterious term, people are impressed and somewhat overwhelmed at the display of strange-looking equipment, blinking lights, and strange noises. However, there seems to be some satisfaction in having this peculiar equipment and the "experts" available to help them if an accident did occur near their community.

This response has been particularly noticeable in the state of Oregon during the exercises that have been conducted along the Interstate-84 corridor. This interstate highway is a main route for the thousands of trucks that transport mostly low-level radioactive waste through eastern Oregon to the Hanford Site for disposal. In recent years the public has become much more aware that these shipments are passing through their communities on a daily basis and the people are concerned about potential accidents. PNL provides training in the detection of radiation to local police and fire departments; PNL also presents information on emergency response and demonstrates the capabilities of the mobile laboratory. The training and information relieves a lot of the animosity toward nuclear energy and creates an awareness that an effort is being made to protect people and the environment from the effects of radiation.

ACCIDENT EXPERIENCE

The capability of the mobile laboratory was demonstrated in May of this year during a field training exercise following the Chernobyl accident. The laboratory was moved to a location near the Hanford Site and samples of vegetation and soil were collected for analysis. When the samples were collected, prepared, and counted using the procedural methods established for initial response to a radiological accident, activity in the range of 5 pCi per gram of ^{131}I was detected. A measurable difference in activity was noted among samples collected from the outer and inner foliage of the vegetation sampled. No ^{131}I activity could be detected in soil samples collected in the same location as the vegetation samples.

At the time the Chernobyl "cloud" was predicted to pass over Washington, rainshowers occurred. The mobile laboratory collected and analyzed rainwater samples and found that concentrations of ^{131}I were similar to those levels reported by other laboratories.

The measurement of ^{131}I in the range of 5 pCi per gram was considered very good and demonstrated the capability to meet and greatly exceed the specifications for detection under field conditions.

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FILTER PAPER IN MARINELLI

RAM	keV	Count Eff.%	Eff. +2sd%	Eff. -2sd%	Cal. Fac (cpm/pCi)
Eu Ka xray	42.8	6.28	10.98	1.58	7.17
^{155}Eu	86.6	7.24	7.54	6.94	6.22
^{155}Eu	105.3	8.40	8.81	7.99	5.36
^{154}Eu	123.1	8.25	8.43	8.07	5.46
^{125}Sb	176.4	7.92	8.46	7.38	5.69
^{154}Eu	248.0	5.27	5.63	4.91	8.54
^{125}Sb	427.9	4.66	4.93	4.38	9.67
^{125}Sb	463.4	3.92	4.40	3.43	11.50
^{125}Sb	635.9	3.31	3.64	2.98	13.62
^{154}Eu	723.3	2.47	2.56	2.38	18.23
^{154}Eu	873.2	2.08	2.34	1.81	21.68
^{154}Eu	996.4	1.94	2.09	1.79	23.23
^{154}Eu	1004.8	1.96	2.07	1.85	22.94
^{154}Eu	1274.4	1.73	1.81	1.65	26 01
^{154}Eu	1596.5	1.72	2.06	1.38	26.21

SOIL IN MARINELLI

RAM	keV	Count Eff.%	Eff. +2sd%	Eff. -2sd%	Cal. Fac (cpm/pCi)
^{155}Eu	86.6	2.40	2.69		
^{155}Eu	105.3	2.84	3.19	2.12	18.75
^{154}Eu	123.1	2.91	3.00	2.49	15.87
^{125}Sb	176.4	2.77	3.25	2.82	15.47
^{154}Eu	248.0	2.38	2.70	2.29	16.25
^{125}Sb	427.9	1.77	1.91	2.07	18.90
^{125}Sb	463.4	1.67	1.97	1.64	25.40
^{125}Sb	635.9	1.24	1.52	1.38	26.92
^{154}Eu	723.3	1.10	1.18	0.97	36.23
^{154}Eu	873.2	0.94	1.04	1.02	40.94
^{154}Eu	996.4	0.80	0.92	0.85	47.68
^{154}Eu	1004.8	0.87	0.96	0.68	56.31
^{154}Eu	1274.4	0.77	0.82	0.78	51.76
^{154}Eu	1596.5	0.60	0.83	0.72	58.38
				0.36	75.50

VEGETATION IN MARINELLI

RAM	keV	Count Eff.%	Eff. +2sd%	Eff. -2sd%	Cal. Fac (cpm/pCi)
^{155}Eu	86.6	3.56	3.76	3.36	12.66
^{155}Eu	105.3	3.51	3.77	3.26	12.83
^{154}Eu	123.1	3.92	4.04	3.81	11.48
^{125}Sb	176.4	3.88	4.46	3.31	11.60
^{154}Eu	248.0	2.96	3.26	2.66	15.22
^{125}Sb	427.9	2.25	2.52	1.97	20.04
^{125}Sb	463.4	1.95	2.22	1.68	23.09
^{125}Sb	635.9	1.45	1.75	1.15	30.99
^{154}Eu	723.3	1.32	1.43	1.22	34.00
^{154}Eu	873.2	1.03	1.18	0.88	43.60
^{154}Eu	996.4	0.99	1.26	0.73	45.34
^{154}Eu	1004.8	1.02	1.13	0.92	44.12
^{154}Eu	1274.4	0.84	0.92	0.77	53.42

LIQUID IN MARINELLI

RAM	keV	Count Eff.%	Eff. +2sd%	Eff. -2sd%	Cal. Fac (cpm/pCi)
^{155}Eu	86.6	2.79	3.08	2.49	16.16
^{155}Eu	105.3	2.92	3.11	2.73	15.44
^{154}Eu	123.1	3.14	3.23	3.05	14.36
^{125}Sb	176.4	3.33	4.03	2.63	13.51
^{154}Eu	248.0	2.10	2.40	1.80	21.43
^{125}Sb	427.9	1.93	2.08	1.79	23.33
^{125}Sb	463.4	1.69	1.88	1.50	26.65
^{125}Sb	635.9	1.53	1.77	1.29	29.52
^{154}Eu	723.3	1.11	1.18	1.04	40.75
^{154}Eu	873.2	0.97	1.07	0.88	46.35
^{154}Eu	996.4	0.93	1.02	0.84	48.34
^{154}Eu	1004.8	0.96	1.03	0.90	46.73
^{154}Eu	1274.4	0.82	0.86	0.78	54.96

Table 1. Counting Efficiencies

Energy keV	Veg. Factor	Soil Factor	Airs Factor	Liq. Factor
≤100	12.69	18.77	6.22	16.68
100-150	11.55	15.81	5.39	14.77
151-200	11.55	16.38	5.49	15.02
201-250	13.86	17.66	7.51	19.58
251-300	15.81	19.17	8.58	21.97
301-350	17.33	21.45	9.38	21.97
351-400	18.77	23.10	10.01	23.10
401-450	20.48	25.03	10.48	23.10
451-500	23.10	28.15	11.40	26.50
501-550	25.74	31.07	12.01	27.30
551-600	28.15	33.37	13.06	28.15
601-650	31.07	36.04	13.65	30.03
651-700	33.37	37.54	14.08	32.18
701-750	36.04	40.95	16.09	40.95
751-800	39.17	42.90	18.02	42.90
801-850	39.17	45.05	18.77	45.05
850-900	40.95	47.42	19.58	47.42
901-950	42.90	50.05	21.45	47.42
951-1000	45.05	52.99	22.52	47.42
1000-1100	45.05	56.31	23.71	47.42
1101-1200	50.05	56.31	25.03	50.05
≥1201	52.99	60.06	26.50	52.99

Table 2. Calibration Factors

Developing a Comprehensive and Accountable Data Base After a Radiological Accident

Hollis A. Berry and Zolin G. Burson

ABSTRACT After a radiological accident occurs, it is highly desirable to promptly begin developing a comprehensive and accountable environmental database both for immediate health and safety needs and for long-term documentation. The need to assess and evaluate the impact of the accident as quickly as possible is always very urgent, the technical integrity of the data must also be assured and maintained. Care must therefore be taken to log, collate, and organize the environmental data into a complete and accountable database. The key components of database development are summarized as well as the experience gained in organizing and handling environmental data acquired during:

1. TMI (1978).
2. The St. Lucie Reactor Accident Exercise (through the Federal Radiological Measurement and Assessment Center (FRMAC), March 1984).
3. The Sequoyah Fuels Inc., Uranium Hexafluoride Accident near Gore, Oklahoma (January 1986).
4. Chernobyl Reactor Accident in Russia (April 1986).

I. INTRODUCTION AND BACKGROUND

After a radiological accident of substantial magnitude occurs, physical data obtained can be divided into two categories: (1) data that concerns immediate health and safety needs, and (2) data that satisfies long-term documentation requirements. It is then highly desirable to begin developing a comprehensive and accountable environmental database as soon as possible after the occurrence of a radiological accident. Assessing and evaluating the impact of the accident is always urgent. Yet the technical integrity of the data must also be assured and maintained before any assessment, evaluation, or summary is completed. To adequately assess the accident, the data must be managed and organized into a complete and accountable database.

The Three Mile Island (TMI) reactor accident revealed the fact that environmental monitoring and assessment tasks are extremely important, and become increasingly complex as the magnitude of the accident increases. In responding to the State of Pennsylvania's request for federal technical assistance the United States Department of Energy (DOE) provided ground and aerial monitoring teams, coordinated the monitoring efforts of all teams performing off-site environmental measurements, and created an assessment team of laboratory experts. The assessment team examined the monitoring results to ensure that the correct instruments and samples were utilized, that the instruments were properly calibrated, that common geographic terms and units of measure were used, and that any ambiguous readings were confirmed or denied by additional monitoring.

During the response to the TMI accident, the amount of data being examined soon became overwhelming; it was evident that an environmental database needed to be developed. Some effort was made to put the data in proper perspective but because of the large number of data points being gathered, the lack of qualified personnel and appropriate equipment on site, and the lack of a definitive plan, this task was not adequately addressed.

A definitive plan for handling reactor type accidents has been formalized into DOE's responsibilities in managing the Federal Radiological Monitoring and Assessment Center (FRMAC). Exercises and responses to accidents have further underlined the importance of database development in relation to the monitoring and assessment effort. During the St. Lucie Reactor Accident Exercise in March 1984, preparations were made to develop a comprehensive and accountable database but time was short and the play and data were artificial. Nevertheless, it was evident that more detailed plans and procedures, as well as an adequate portable computer system and more qualified personnel, would be needed in a real accident situation.

A real accident did occur at the Sequoyah Fuels Inc., Uranium Hexafluoride Plant near Gore, Oklahoma in January 1986. Several days after the accident EG&G/EM was asked by DOE to formulate a database for the Nuclear Regulatory Commission (NRC) task force assigned to assess the accident and its impact on the environment.

The accident seemed small and perhaps insignificant, but more than 2000 data points were incorporated into the database in a relatively short time. The need for thoroughness and accountability was again evident.

On April 26, 1986 the Chernobyl Reactor accident occurred in Russia. EG&G/EM assisted DOE in formulating a database of the fallout data received. The importance and usefulness of a database were recognized at the outset, but the effort required to manage incoming data was underestimated and our involvement with the database formulation was delayed until a week after the accident. The amount of data being received at the point of EG&G/EM's involvement was tremendous and the task ahead was staggering. Approximately 10,000 points were entered into the database in a short period of time. Yet, this represented only a small portion of the data available for assessment and evaluation.

The evaluation and entry functions performed on the Chernobyl data were similar to those that would be performed in any FRMAC or emergency response operation. We were able to design the database and its implementation procedures without the extreme pressures inherent in an actual radiological emergency situation.

The Sequoyah Fuels and Chernobyl accidents clearly emphasized the urgency of beginning the database development and assessment processes at the earliest possible time after a major accident so that credible assessments and decisions can be made quickly. We learned that it is unrealistic to design, modify, or formulate from scratch a database system while officials and managers are demanding to know how serious the problem is and what actions should be taken. Instant answers will always be requested of the monitoring and assessment teams in a radiological response. The authorities' ability to provide substantive assessments of the threat to the public health and safety hinges on the ability to rapidly assemble and integrate all available information on radiation measurements, location of surrounding populations, identification of farm animals and crops, surface-fed water supplies, etc. Through the experience gained at Three Mile Island, the St. Lucie exercise, the Sequoyah Fuels accident, and the Chernobyl response, much has been accomplished in identifying major elements of the task of developing a comprehensive and accountable database after a radiological accident.

II. DEVELOPMENT OF EMERGENCY RESPONSE DATABASE CAPABILITIES

The complexities of the assessment tasks in an emergency response are easily overlooked until one examines the many variables of the problem. Short-term and long-term effects, complex food chain and concentrating mechanisms, alpha and beta dose contributions, potential watershed impact on surface drinking water sources, and protective action recommendations for all age groups and for farm animals are just a few of the many results, issues, and questions the assessment team must consider when providing information to the decision-makers. Database development therefore becomes an integral part of data assessment.

The experience gained from previous exercises and operations in formulating databases during an emergency response have identified three major areas of development.

1. Development of plans in cooperation with potential users and/or customers.
2. Development of computer hardware and software capabilities for field and home base support roles.
3. Development of procedural guidelines for receiving and assembling raw data, and verifying its quality and applicability to the response and assessment requirements.

In an emergency response situation, personnel from federal agencies, contractors and state and local organizations will be called upon to assist in the response and also participate in the assessment and monitoring teams. Later, as the incident proceeds out of the emergency stages into the long-term monitoring and recovery phases, a final custodian will be determined. Therefore, it is necessary to coordinate emergency response database development plans with all potential users. Their needs and requirements should be considered in the development of this capability.

The information collection problem in an emergency response is massive and computers have been used extensively to aid in compiling and organizing data. Currently and in the past, within EG&G/EM, emergency response databases have been stored in small, portable tabletop computers. This may not be the best computer for this task, particularly if the database is very large. The right computer system should be compatible with the needs and capabilities of the group assembling the database and doing the assessment as well as potential users and custodians. Other basic requirements are as follows:

1. The system must be easily transportable in ruggedized containers.
2. The basic components must also be fairly rugged to withstand the harsh environments they encounter (baggage handlers, high humidity, etc.).
3. It must support a multiuser/multitask environment.
4. It must be capable of the timely handling of large volumes of data.
5. It must be able to utilize third party hardware and software that are supported and proven.
6. The software must be compatible with other systems.
7. Power requirements must be low to moderate.
8. The equipment must be readily available.
9. It should include the software and peripherals for automated plotting of the data and information transfer.

A larger computer (or an assemblage of the transportable work stations considered above) will be needed at home base. This system can be used during the early stages of the emergency to input and store data from locations not in the critical area and prior to the arrival of the database and assessment teams.

Later, the entire database would reside in this system and be used to produce reports of a more conclusive nature until a final custodian is determined.

The technical integrity of the data, at all stages of development and use, must be assured before the information can be released for action or information purposes. Key aspects of database development can be summarized as follows:

1. All original data sheets should be logged, collated, and cross-referenced for easy referral.
2. The database should be complete, organized, and trackable in every detail.
3. Methods and plans for dealing with qualitative vs. quantitative data need to be developed.
4. The information must be in a form that allows technical integrity of the data to be confirmed as quickly as possible. All data should be evaluated as to quality, validity, and priority of importance to the response prior to its entry into the database. In the early stages of the response, the assessment team should deal only with data that is consistent and appropriate to the incident. All other data can be considered as soon as the immediate threat to the public has diminished. In addition, all entries into the database should be edited and verified.
5. Key data should be retrievable in any category desired. The results need to be summarized, evaluated, and placed in proper perspective quickly and easily and on a continual basis.
6. All the environmental data should be plotted on maps or overlays separately by data type and medium and in appropriate detail.
7. The requirements for distribution of the data will be very great. The need will be continuous for detailed, organized data for assessment purposes and for highly summarized data for management information. The database must be organized so that these requirements can be met easily and quickly.
8. Adequate and qualified staffing should be assigned.

As with any database management program, development evolves and expands as more tasks and requirements are identified. In developing emergency response database capabilities an expansive, dynamic approach must be maintained to remain flexible and responsive to needs and requirements. In addition, the database itself should be dynamic so it can be easily and quickly modified to meet specific requests. The ability to manipulate, massage, and plot the data will also facilitate assessment and decision making in an emergency response situation.

III. IMPLEMENTATION CONSIDERATIONS

In implementing the development of response capabilities in the event of a significant emergency, a number of specifics need to be considered. These include detailed planning, database construction, staffing and deployment.

In planning emergency response database capabilities several activities can aid in this development. These include:

1. The preparation of maps on a variety of scales using a common coordinate system. This (1) aids outside responding organizations in locating sample sites in unfamiliar areas, (2) facilitates plotting the data, (3) assists the assessment team in their evaluation, and (4) allows for easy retrieval of information from common areas.
2. A compilation of baseline environmental information including routine monitoring networks and associated results, population distributions, locations of critical facilities, agricultural areas, potable water supplies, etc. is helpful to the assessment team in determining the impact, if any, resulting from the accident.
3. The standardization of collection and analysis techniques, the formulation of data and information collection forms, the establishing of common and consistent units of measure, and determining the type of information required for a variety of accident scenarios.
4. The development of detailed implementation plans and procedures and the exercise of these plans.
5. The gathering of required supplies including standard office equipment and supplies, a copy machine, an adequate computer, etc. This equipment should be prepared and packaged for deployment.
6. Personnel assignments, training, and crosstraining.

The following information should be included in the computer based database:

1. The reference ID number assigned to each piece of original data received.
2. The time and date the sample or measurement was taken.
3. The identity of the organization that collected the sample and the organization that conducted the analysis.
4. The sample collection ID number and the laboratory analysis ID number.
5. The location at which the sample was collected, based on a common coordinate system.
6. The medium and type of sample in addition to the type of analysis.
7. The results and units of measure.
8. Any significant comments that may affect data assessment.

A considerable amount of the information provided is of a bookkeeping nature but allows the database and assessment team to backtrack and verify incomplete or inconsistent information either with the laboratory that conducted the analysis or the agency that collected the sample. The database format is dependent on the type of accident and response. Preplanned and developed databases should be very generic in format and be able to be modified, as required, during the response.

Adequately staffing the database center at an accident site is essential. In addition to computer operators and clerks, technically qualified people are needed to evaluate the incoming data on a continuous basis. These individuals can also be utilized in the assessment of the data when time and activities allow. For a small response similar to the Sequoyah Fuels Inc., accident four to six people would be sufficient to handle the data. To handle the data from a major reactor accident that would involve the participation of several agencies and organizations as many a two dozen or more people would be required for 24-hour operation. This level of involvement would allow for continuous data entry and evaluation. It would also allow the assessment team to assure the technical integrity of the data.

Finally, in the matter of deployment, database and assessment personnel should be an integral part of the first or advance party responding to a major incident. This will allow smooth data and information flow on a continual basis. In addition, the database and assessment center should be fully staffed as soon as possible.

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A New Method for Presenting Off-Site Radiological Monitoring Data During Emergency Preparedness Exercises

M. P. Moeller, G. F. Martin, E. E. Hickey, and J. D. Jamison

ABSTRACT As scenarios for exercises become more complicated and flexible to challenge emergency response personnel, improved means of presenting data must be developed to meet this need. To provide maximum realism and free play during an exercise, staff at the Pacific Northwest Laboratory (PNL) have recently devised a simple method of presenting realistic radiological field monitoring information for a range of possible releases. The method utilizes only two pieces of paper. The first is a map of the offsite area showing the shape of the plume for the duration of the exercise. The second is a semi-log graph containing curves relating exposure rate and iodine concentration to downwind distance and time. Several techniques are used to maximize the information on the graph.

INTRODUCTION

Current government regulations require that the operator of a commercial nuclear power plant make provisions for the conduct of emergency preparedness exercises. Such provisions include the development of a scenario and the supporting data necessary to drive the exercise. For those in the role of preparing exercise scenarios, data presentation has become an important part of this task. In the last few years, data presentation on inplant reactor parameters has been advanced through the use of simulators which add a new degree of realism and free play. To date, however, the full benefit of the simulators has not been realized due to limitations in the methods used to present radiological data to monitoring teams. Previous methods have not allowed operator response actions on the simulator to determine the timing and magnitude of the radiological release.

In support of a major exercise conducted in the Spring of 1986 at the Department of Energy's Hanford Site, staff at PNL prepared the offsite radiological onitoringm data. A major goal of the exercise was to design a scenario which would provide data for all anticipated participant response actions. Consequently, the release pathway, timing of the release and magnitude of the source term were all to be dependent on the participants' response actions. To meet this objective, a new method for presenting the radiological data was required.

This paper describes the new method for presenting radiological data which was developed for the Hanford Exercise. In order to show the method's effectiveness, a review of the limitations of conventional methods is presented. The new method provides consistent offsite monitoring data for an emergency exercise scenario in which operator actions change the progression of the accident, and hence, the magnitude of the radiological release. In addition, the ease by which the offsite monitoring team controller could pass the data to players was emphasized.

Conventional methods of data presentation do not readily allow for variations in the source term responsive to operator actions. In order to provide data covering a range of potential operator actions, many different sets of tabular data are needed to cover the range of possible radiological releases. Therefore, a tremendous volume of data sheets is required if the scenario is to be flexible. As a result, it is not practical to use conventional methods to provide field team controllers with data for release scenarios corresponding to every potential combination of operator response actions.

The volume of data sheets currently required is determined by the number of dependent variables. Dependent variables include the magnitude of fuel damage, the release pathway, the time of initiation, and the duration of the release. Data that players request include general area radiation exposure rates using both an open and a closed instrument window, gross particulate and iodine air concentrations in counts per minute above background based on the volume of air sampled, radioactive deposition, and contamination levels on the ground in counts per minute per square meter.

METHOD

Instead of computing the required data for all possible combinations of the dependent variables and providing the information to controllers in tabular form, the new method condenses the data into a series of curves presented on a semi-log graph. Data is calculated using a source term of 100 percent fuel damage. Maximum centerline values over the duration of the release are determined for downwind distances. Similarly, the off-centerline distance where radiological data values are equal to background levels is determined for selected downwind distances. These data define the edge of the plume and, hence, its shape. Values for data midway between the centerline and the plume edge are assumed to be two-thirds (66.6%) of the centerline value.

Plotting such data on a semi-log graph is useful in that multiplication and division by a constant on a log scale always results in an increase or decrease on the scale of the same distance. In other words, a new curve which is a multiplicative constant of an existing curve on a log scale will have the same shape and be displaced from the original on the scale. It is this important fact that allows a cursor to be used which adds the desired flexibility. Having plotted the curve for 100 percent fuel damage, the cursor can be used as a measuring stick to determine values corresponding to less failed fuel if the operators take actions that would reduce the level of core damage. While within the plume, values for closed-window readings are assumed to be some fraction less than open-window readings, depending upon the scenario's assumed source term radionuclide mixture. Therefore, the closed-window readings are a constant distance from the open-window curve and can be determined using the cursor.

Data prepared for a scenario can be condensed on a semi-log graph as shown in Figure 1. A map of the exposure area is presented in Figure 2. This map indicates the outer plume edges, the plume centerline, and the midway distance between each plume edge and the centerline. Curves on Figure 1 depict both the general area radiation exposure rate (left-hand abscissa scale) and the gross particulate and iodine air concentrations (right-hand abscissa scale). The air concentration information is given in counts per minute above background per cubic foot of air sampled. Therefore, if an air sampler pumped at 10 cubic feet per minute for five minutes, then the controller would multiply the value obtained on the graph by fifty. Rules of thumb and other relevant information are presented on the border of the graph.

To implement this method of data presentation, controllers must be informed of the event start time (initiation of release) and release category (duration and pathway of release and percent of fuel damage). The release pathway can affect the source term through consideration of filtration systems and release height. Information can easily be communicated by the lead controller through simple, predetermined coded messages to offsite monitoring team controllers. When required to provide instrument readings, a controller should use the following procedure:

- Prior to providing any instrument readings, the field controller should receive the event start time and release category information from the lead controller.

- Knowing the event start time, the field controller will fill in the timeline at the top of the graph. The time of plume arrival at a downwind distance is determined by the wind speed. The timeline, when filled in, should reflect this relationship.

- When a request for an instrument reading is made by a player, the controller will determine the team's current location on the map. It may be useful for the controller to carry a second map without the plume shape indicated so that if there is a question concerning the team's location, the controller can ask a player for assistance without revealing the plume trajectory.

- The controller then determines if the team's location is within the plume. If not, the controller reports background readings. If it is, then the controller determines if the plume has arrived at this downwind location by checking the timeline. If the plume is more than ten minutes upwind of the team's location, the controller reports background readings. If the plume is less than ten minutes upwind of the team's location and still has not arrived, the controller should will report a reading between the background level and the maximum value after the plume has arrived. If the plume has arrived, the controller will report the maximum value for the team's location by proceeding to the next step. If the plume is elevated at this location, report closed-window readings (gamma only) when either open- or closed-window readings have been requested. If the plume has touched down upwind of this location and is at ground level, then open- and closed-window readings should be given as requested.

- To determine the maximum value after the plume has arrived, the controller should identify the team's downwind lateral location relative to the plume centerline, outer plume edges, and the midway distance (midline) between.

- The controller then pinpoints the team's lateral location relative to the three curves on the graph (plume centerline, midline and plume edge) at the team's downwind distance.

- Using the cursor, the controller should adjust the point on the graph based on the measured distances associated with each release category.

- The controller then reports the general area radiation exposure rate (mR/hr) from the left hand abscissa values.

- After the team has taken and measured an air sample, the controller can report the air concentration values using the values from the right hand abscissa multiplied by the volume of air sampled (cubic feet).

Because of the cursor and condensed nature of the data, the method is quite versatile and readily adaptable to different conditions. For example, during the 1986 Hanford Site Exercise, one scenario included a highly-radioactive puff release followed by a continuing, less radioactive release. Data for both of these releases were put on one graph by utilizing two timelines relating the passage of each release over a given location. Another example is to provide for a radical plume direction change merely by utilizing two plume maps indicating the two dominant plume shapes.

Acceptance of the method was noted during the training of controllers for the Hanford Exercise. In less than two hours, controllers understood the method, were able to respond correctly to simulated requests for data, and were able to interpolate between known values presented on the curves to get a value for any location. Many noted correctly that the level of accuracy that they could provide from reading the curves and interpolating is consistent with that available from actual field measurements. Many improved their graphs by including lines that indicated a transformation of a fixed monitoring route roadway into the downwind and lateral coordinate system of the graph.

SUMMARY

As scenarios become more flexible and complex, as through the use of simulators, more versatility is needed to present realistic data to offsite monitoring teams. Conventional methods are not adequate because of the volume of tabular data necessary to correspond to every potential operator response action. The method described in this paper provides a solution that allows flexibility and data in graphic form. Utilization of this new method during a complex exercise indicated that controller performance is enhanced.

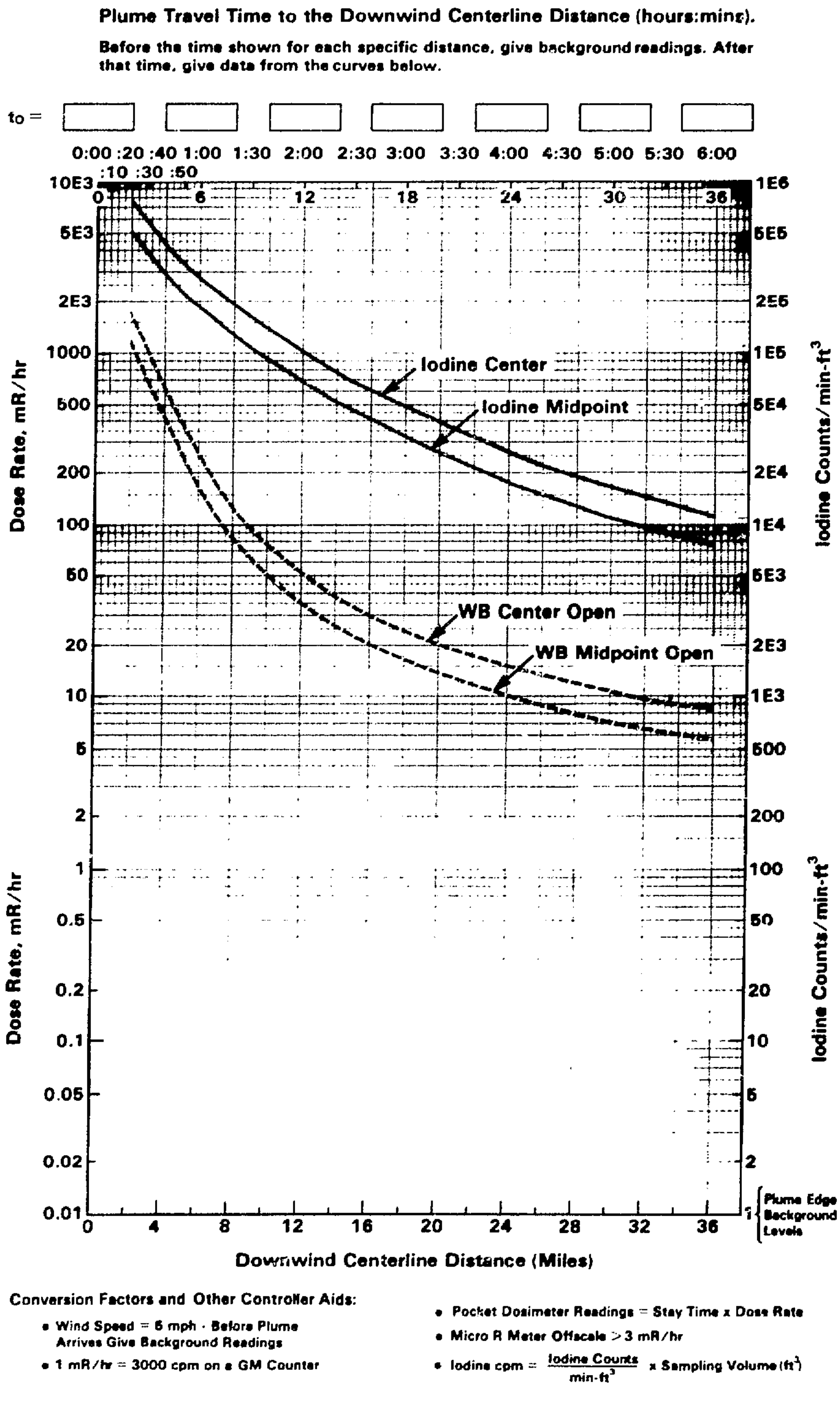

Figure 1. Semi-log graph with radiological data.

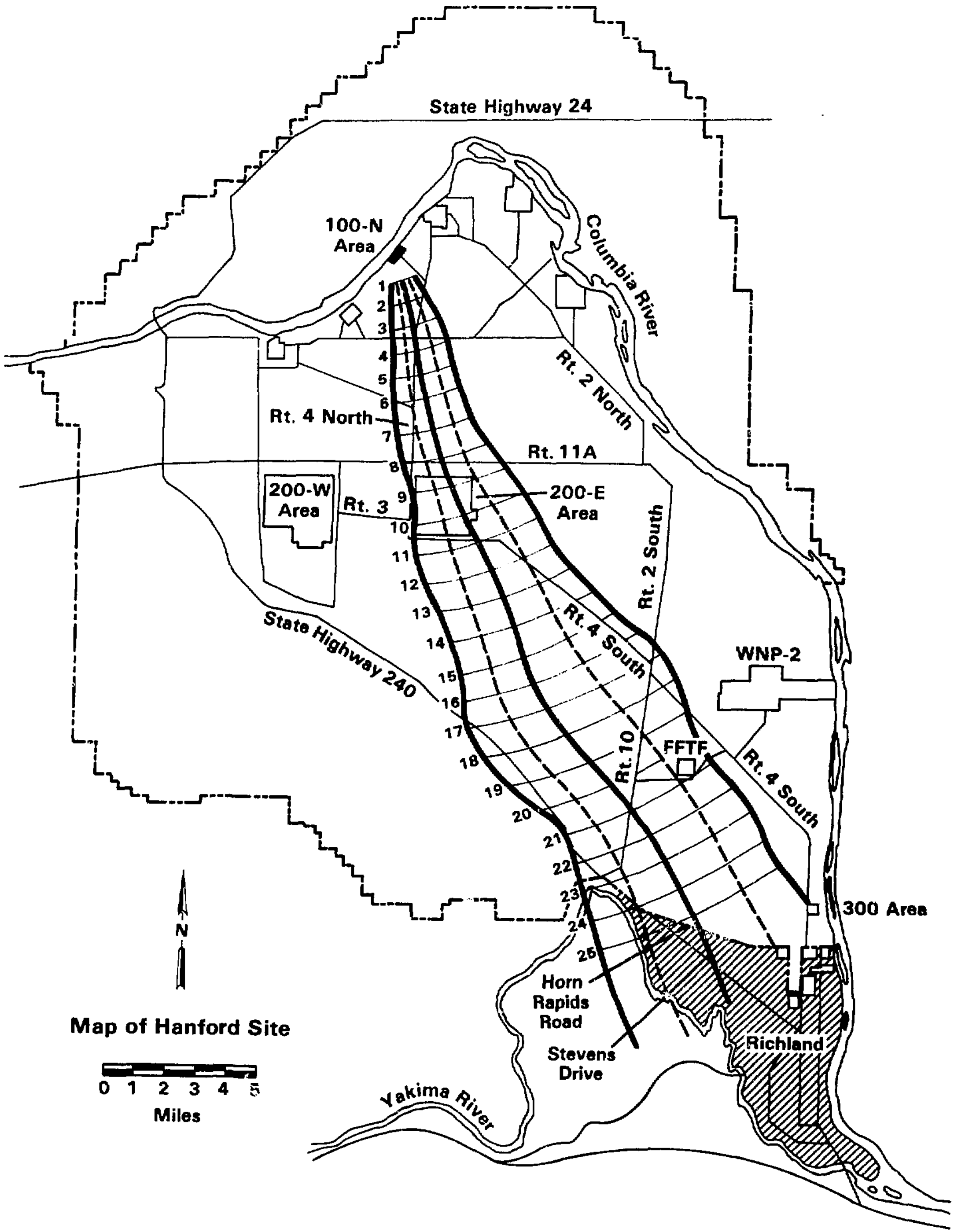

Figure 2. Map of plume exposure area.

Enhancement of the 1985 Browns Ferry Exercise Through the Use of Spiked Samples

James L. McNees

ABSTRACT The use of spiked environmental samples has proven to be a beneficial part of nuclear power plant exercises. Milk, soil, and air sampling cartridges can easily be spiked with realistic concentrations of radionuclide without violating Nuclear Regulatory Commission regulations or policy and with no significant radiation hazard to exercise participants. Analysis and reporting of data from realistic samples significantly improves motivation, attitude, moral, and overall preparedness of the individuals participating in that phase of the exercise.

Since 1979 the State of Alabama has participated in at least one nuclear power plant exercise per year involving the demonstration of the capability to analyze environmental samples. For the first five years, only non-radioactive samples of milk, water, and other items were analyzed in a effort to establish the fact that the State could adequately evaluate real radioactivity at their field laboratory location. Since everyone knew that there was nothing in the samples, those who had participated for several years developed an attitude of flippancy concerning the exercises. Other participants expressed concerns about the level of radiation exposure that they would receive from collecting, transporting, and analyzing environmental samples during a real incident. This problem was heightened when a federal observer suggested to a State participant that we might need to put up a shield to reduce the exposure from milk samples that were awaiting analysis.

In order to fully accomplish exercise objectives of demonstrating our capability of being able to analyze samples and to initiate protective actions on the basis of the U. S. Food and Drug Administrations (FDA) Protective Action Guides, the State of Alabama decided to utilize spiked or radioactive environmental samples for the 1984 nuclear power plant exercise conducted with the Joseph M. Farley plant operated by the Alabama Power Company. Both the State's radiological laboratory and elements from the U. S. Environmental Protection Agency's radiological laboratory from Montgomery participated in the exercise and analyzed the spiked samples at the field location in Dothan, Alabama.

All parties involved acclaimed the use of spiked samples as a positive method of improving the attitude and involvement of laboratory personnel in the exercise. Following the success during the 1984 exercise, it was decided to utilize spiked samples for the November 1985 exercise at Browns Ferry. Starting from the scenario, calculations were made of the amount of iodine-131 that would be anticipated in milk at the nearest dairy to the plant, in air sample cartridges taken at the plume centerline at a point on the boundary of the five mile evacuation sector, and in surface soil samples taken adjacent to the plant following the mock accident. The radioiodine to be used to spike the samples was obtained from a local hospital's radioactive waste storage, where it had been discarded due to insufficient remaining radioactivity for clinical use.

The capsules, which had originally each contained approximately 93 microcuries of iodine-131 had decayed to 0.2 microcuries per capsule by the exercise date of November 13, 1985. Thus three capsules placed in a gallon of milk would produce an approximate concentration of .17 microcuries per liter which fit the scenario, and would adequately demonstrate that the State could rapidly analyze milk against the Food and Drug Administration's emergency protective action guide of .15 microcuries per liter (FDA, 1982). For the air sample cartridge, two capsules placed inside a previously use cartridge would have the same analysis as a cartridge from a sample collected according to the State's field air sampling procedures in a concentration of .69 picocuries per cubic centimeter of air. The spiked soil sample utilized two capsules of iodine-131 placed in cesium contaminated soil that had been previously collected as a sample at a known location of cesium contamination. It contained 0.4 micorcuries of iodine and 0.08 microcuries of cesium-137 in 500 milliliters of soil. The air and soil samples were prepared several days prior to the exercise, while the milk sample was prepared the morning of the exercise. The spiked samples were placed into exercise play at the time they would normally be collected during an actual accident response.

All parties who were expected to handle the spiked samples were informed in advance that some of the samples utilized in the exercise would contain real radioactivity. The radioactivity would be consistent with the scenario, but would be utilized in only nominal quantities. The need to inform all participants in advance of the use of real radioactivity can not be overemphasized.

The U. S. Nuclear Regulatory Commission (NRC) has established exempt quantities of radioactive materials such that any person who receives, possesses, uses, or transfers individual quantities below these amounts is exempt from the requirements for a license and certain regulatory controls (NRC, 1982). For iodine-131 that quantity is 1.0 microcurie. Nuclear Regulatory Commission regulations prohibit the transfer of licensable material in exempt quantities for purposes of commercial distribution without a specific license to do so from the Commission (NRC, 1984). However for non-commercial distribution, as in the case of these spiked samples provided to exercise participants, no license to distribute exempt quantities is required.

Radiation exposure problems in exercises are best controlled by controlling the quantities of materials involved. The exposure rate from each of the 0.2 microcurie iodine capsules utilized on November 13, 1985 was .07 milliroentgen per hour at one inch or 0.5 mircoroentgen per hour at one foot. Placing three capsules in a gallon of milk produced a milk sample containing approximately the same concentration that the Food and Drug Administration stated in their October 22, 1982 Federal Register statement as being permissible for sale to the general public following a nuclear power plant emergency (FDA, 1982). Utilizing a typical hand held survey instrument, no external radiation can be measured from a gallon of milk containing this concentration of radioiodine. Extensive decontamination precautions for a possible spill during transportation or at the field laboratory were not deemed necessary for this concentration of radioactivity. While all samples must contain less than the exempt quantity for the radionuclide involved, lessor amounts can often accomplish the same purpose. Due to the nature of activities that occur during exercises, the quantities of radionuclides utilized in spiked samples should be kept at the minimum necessary to demonstrate laboratory capability and still be realistic for the accident scenario being used.

At the 1984 Farley exercise, during the preexercise briefing, the use of real radioactivity in spiked samples was discussed. Likewise during the preexercise briefing for the 1985 Browns Ferry exercise, the use of real radioactivity in both the environmental sample phase and the medical drill was discuss. Representatives of the various federal agencies including the Nuclear Regulatory Commission were present at these briefings and none of the agencies present expressed any concerns or disapproval of the intended usages of these small amounts of actual radioactive material.

The medical drill, that was a part of the 1985 exercise, utilized technetium 99m and natural thorium as a source of low level contamination. Only inanimate objects were purposefully contaminated. A simulated tornado was to have struck the plant and blown some contaminated objects from the plant. One of the contaminated objects was to have struck and injured a farmer approximately one mile from the plant. The injured farmer and contaminated objects were to be found by field monitoring teams. The magnitude of the potential hazard from this use of real contamination was to be controlled by the limited amount of material to be used and by the six hour half life of technetium 99m. The benefits of the responders having actually seen some real contaminated objects that would at least make their meters respond was thought to outweigh the risks of the use of real radioactivity. The problems and subsequent concerns following this medical drill were a result of the following:

1. The utility's "Scenario Director" was aware of, and had agreed in advance to the use of the technetium 99m, however the utility personnel who participated in the drill had not been informed of the planed us of real radioactivity.
2. An error in calculation had been made concerning the anticipated concentration of the material being used. This resulted in the amount being used being five times the intended amount.
3. Due to a miscommunication concerning the time to begin setting up the scene of the drill, the liquid Technetium did not have time to dry prior to the drill beginning, thus it was easily removable.
4. Utility participants were being informed via their radio system at the plant that the radioactive materials were being simulated and thus ignored warnings to the contrary by the State representative at the scene.
5. The utility's radio system at the joint State-Utility monitoring location was unable to broadcast on the day of the exercise, thus they could hear the participants being incorrectly told that the radioactivity was simulated, but were unable to transmit the correct information.

As a result of the above named problems, six individuals received detectable amounts of technetium contamination on their clothing and/or person. One responder, who did not utilize the protective gloves that were available to him, received a spot of contamination on the back of the middle finger of his right hand of no more than 3 microcuries of technetium 99m. The radiation exposure received by this individual as a result this incident was less than 5 percent of the allowable quarterly limit. Never the less, his finger was needlessly contaminated. The hazard of this amount of external contamination is best illustrated in relation to the fact that thousands of people every day receive an injection into their blood stream of up to 10,000 times as much of this isotope as a part of routine medical diagnostic tests. This is not to imply that the magnitude of these accidental exposures in anyway justify the mistakes that occurred. It does however imply, that the magnitude of the risk inherent with the use of limited quantities of short lived materials should not be sufficient to prevent State and local emergency responders from receiving the types of training experiences necessary to enable them to function in the conditions such as existed adjacent to Chernobyl in the days following the accident.

The events of this medical drill caused a prompt reconsideration of regulatory policy with regards to the use of real radioactivity during exercises. All such usages, including the use of spiked environmental samples were reconsidered by both the State and the Nuclear Regulatory Commission. In interim guidance furnished to the States in March 1986, the Commission staff approved the use of unsealed sources of radioactive material for spiking environmental or food samples for laboratory analysis and the use of sealed check sources and thorium lantern mantels for use in medical drills (NRC, 1986). The Alabama Department of Public Health has incorporated this interim guidance into our program policy.

A group of firefighters would not be considered qualified to be sent to a fire if none of them had ever seen, been to, or helped extinguish a real fire. For this reason fire departments conduct controlled practice burns to train firefighters. Likewise the legions of State and local off site response personnel, who are in theory being trained to mitigate the consequences of a Chernobyl type accident, need to have a training experience that will prepare them for dealing with real radioactivity. There is no better way to provide that experience than through the controlled use of very small amounts of short half life radioactive materials in carefully planned training exercises.

The use of spiked samples has proven to be of positive value to nuclear power plant exercises for both the participants and the federal observers. There is no better way to dispel many of the myths that have been associated with what could be anticipated from such samples. Having actually encountered samples containing levels of radioactivity that could be realistically anticipated to occur, our personnel are much better prepared to handle real samples when the actual event occurs.

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Integrated Emergency Response Program at the Savannah River Plant

Richard W. Benjamin

ABSTRACT. The Savannah River Plant (SRP) has an emergency response system to evaluate rapidly the possible consequences of a release of toxic or radioactive pollutants to the atmosphere or to the onsite streams. This system is based upon a strong research and development technology, and computer simulation. The path, extent, and concentration of a pollutant are predicted by a computer system linked to field instrumentation, and monitored by a mobile laboratory and sample collection teams. Field measurements, using a nontoxic tracer gas, serve as emergency response exercises and provide an extensive data base with which to test atmospheric dispersion codes.

INTRODUCTION

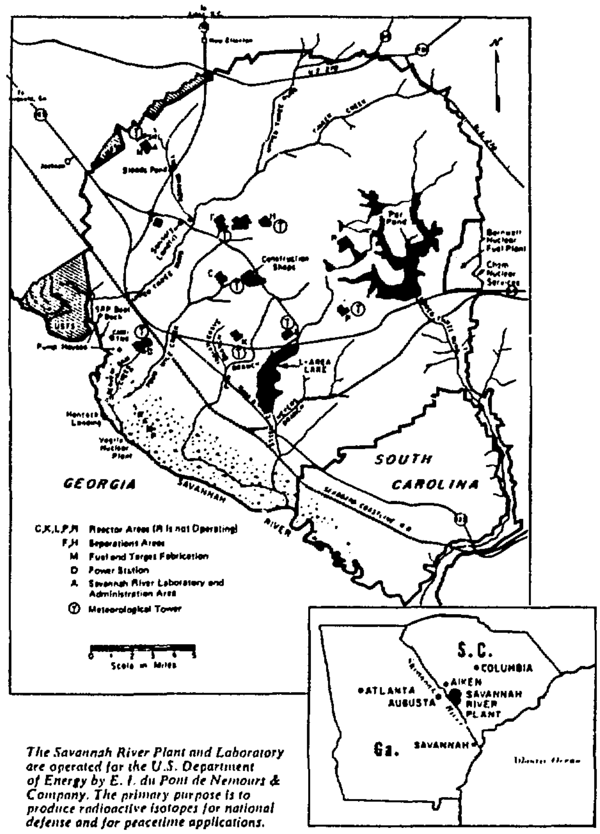
The Savannah River Plant (SRP), a government owned, du Pont operated production facility in South Carolina, produces special nuclear materials for the United States Government and operates virtually the entire nuclear fuel cycle in support of these operations. The wide variety of processes at SRP makes necessary a broadly-based emergency response capability for evaluating and responding to unplanned radioactive and toxic chemical releases to the atmosphere or streams on the site. The emergency response program at SRP includes support in the event of a release and basic research in meteorology, aqueous processes, transport monitoring technology, and computer simulation. The strength of the program lies in the interaction between emergency response and basic and applied research.

The research and development programs include experimental and theoretical dispersion meteorology, dye and thermal studies in streams and lakes, and the development of a unique remote environmental monitoring system. The programs provide the basis and format for an effective emergency response capability, including regular emergency response exercises and effective 24-hour emergency response personnel coverage.

This paper serves as a programmatic overview and introduction to five papers [Addis (1986), Hayes (1986), Hoel (1986), Schubert (1986), and Sigg (1986)] to be presented in the poster session at this conference, which give considerably more detail on particular aspects of the integrated emergency response program.

REVIEW OF SRP FACILITIES AND POTENTIAL RELEASES

SRP is the third largest production site in the DOE complex, comprising some 315 square miles of mostly wooded, gently rolling terrain. The site and its surroundings are shown in Figure 1.



The Savannah River Plant and Laboratory are operated for the U.S. Department of Energy by E. I. du Pont de Nemours & Company. The primary purpose is to produce radioactive isotopes for national defense and for peacetime applications.

FIGURE 1. The Savannah River Plantsite. The Inset Shows SRP's Regional Location

It is bordered on the southwest by the Savannah River, which provides cooling water for the four operating production reactors. The reactors are heavy-water moderated and cooled; they operate at near atmospheric pressure and at a temperature below the boiling point of heavy water. Two large man-made lakes are on site to help cool the thermal effluents of two of the reactors. In addition to the reactors, there are two large chemical reprocessing plants with associated high-level waste tank farms, a tritium processing and handling facility, and a fuel fabrication facility. A high-level waste solidification facility and a naval fuels fabrication plant are under construction. All of these facilities provide the possibility for unplanned releases which would require emergency response procedures.

Radioactive effluents which might have to be monitored in the event of an unplanned release are, e.g., tritium (elemental or oxide forms), gamma ray emitting aerosols, radioiodine, transuranic aerosols, and noble gases. The most likely release from SRP facilities which could travel offsite is tritium, either from the tritium processing facilities or as the result of a reactor moderator spill. Over the years there have been several tritium releases too small to be a health hazard either on or off site. They have, however, served to improve tritium confinement and to develop appropriate tracking and monitoring capabilities. As a result, confinement of tritium has improved dramatically and methods to track and monitor any potential release have been developed and demonstrated. There is also the capability to monitor toxic chemicals, but except for limited quantities of chlorine gas, there are no toxic chemicals on site which could represent a significant hazard.

THE ENVIRONMENTAL TRANSPORT GROUP

The Environmental Transport Group (ETG) has the responsibility for the prompt characterization of atmospheric and aqueous releases from any of the SRP facilities. The primary aim is to provide the proper decision makers with the release location and the predicted dosimetry and exposure information at each stage of an emergency. These decision-makers may be, for example, the shift supervisor at the earliest stages of a release, DOE and du Pont managers, and the monitoring teams seeking to evaluate the environmental effects of the release. Rapid response to a release at SRP is possible because all components of the response system are under the control of one group, the Environmental Transport Group. The system components, personnel and facilities, include meteorology, dosimetry, the dedicated emergency response computers, and the atmospheric monitoring instrumentation.

ETG has a staff of fourteen professionals and eight technicians. Included in this group are five meteorologists, an oceanographer, two health physicists, a nuclear chemist, a computer systems manager, and a small maintenance team to service the widespread meteorological instrumentation on and off site. At least two meteorologists and a supervisor are on call at all times for emergency operations, and additional support personnel are called in as required.

THE EMERGENCY RESPONSE SYSTEM

The emergency response system is a highly automated, real time system which consists of two major components, the WIND system and the TRAC mobile laboratory. The Weather INFormation and Display (WIND) system is used for data acquisition and release consequence calculations for both atmospheric and aquatic releases. The Tracking Radioactive Atmospheric Contaminants (TRAC) mobile laboratory is used for plume tracking and real-time radioactive monitoring in the field. In addition, real-time sample collection teams provide post-release contamination evaluation. These components are described in considerable detail in Addis (1986), Hayes (1986), Hoel (1986), Schubert (1986), and Sigg (1986).

The WIND computer system is used sitewide. Every operating area has a WIND system computer terminal in or near its control room. The system is user-friendly with all of the necessary emergency response codes on a general menu, and the system is used routinely by the site health protection technicians. There is also a WIND system terminal in the Emergency Operations Center (EOC), and an automated alarm system based on stack monitor release limits will be installed in the EOC in the near future.

All data acquisition for the WIND system is achieved through the Remote Environmental Monitoring System (REMS) (Schubert, 1986), which provides the following data:

- Real-time turbulence quality meteorological data are collected from the eight onsite towers and from the WJBF-TV tower just offsite
- Regional meteorological data are obtained from the National Weather Service over the Automated Field Operations Services (AFOS) computer
- Monitors in SRP streams and the Savannah River provide temperature and flow data for aqueous transport
- Real-time source term information and plant boundary concentrations are obtained from the stack and perimeter monitors, respectively

These data are used to provide forecasts of the exposure pathway from an atmospheric release as a function of time, as well as radioactive dose estimates. For aqueous releases, stream and river concentrations as a function of time can be determined.

RESEARCH PROGRAMS

Several basic and applied research and development programs are in progress, and all are aimed at improving, quantifying, and extending the emergency response capabilities of SRP and the national DOE production complex. Research programs also provide the assurance of recruiting and maintaining a highly competent staff of scientists and engineers. These efforts are supported in part by the Office of Health and Environmental Research of the Department of Energy.

Typical of these programs is the Atmospheric Research Program. This program combines:

- Theoretical studies in atmospheric dispersion,
- Experimental field studies aimed at providing a statistically sound data base with which to test atmospheric dispersion models,
- Instrument development to provide the means to measure meteorological parameters, radioactive atmospheric contaminants and tracer gases on a real-time basis

The heart of the Atmospheric Research Program is the Mesoscale Atmospheric Transport Studies (MATS) effort. Its focus is to provide experimental data to test the predictions of the dispersion models. This is done through the use of an experimentally released tracer gas (SF_6), and by tracking the routine small releases from SRP's manufacturing facilities. The data bases from MATS and several other meteorological studies were used as the basis for the Second Model Evaluation Workshop in 1984 [Weber (1985)a, Weber (1985)b, and Kurzeja (1985)]. Radioactive emissions of Kr-85 tritium, and Ar-41 in addition to nonradioactive He-3 are continually used for further dispersion model evaluation and improvement. The development of highly sensitive measurement technology has extended these tracer experiments from on site out to several hundred kilometers.

Studies begun with MATS are being extended in an inter-laboratory program called the STable Atmospheric Boundary Layer Experiment (STABLE). The objective of STABLE is to develop an understanding of turbulence and dispersion in the stable (nocturnal) boundary layer, a condition which is not well-understood and which would be expected to lead to the potentially highest off-site doses in the event of a release. STABLE will include study of a five year data base of turbulence measurements, field experiments, and extensive model evaluation and development.

The Aqueous Research Program of ETG is primarily concerned with aqueous transport processes in the onsite streams and in the Savannah River, and in thermal processes in the onsite lakes and ponds. The major transport concerns are liquid releases, such as tritiated water and hazardous chemicals, and transport of materials adsorbed, such as Cs-137, on stream sediments. A one-dimensional model is used to predict travel times, maximum concentrations, and concentration distributions as a function of time at downstream and river locations for pollutant releases. Stream velocities and dispersion coefficients that were needed in the model were determined from dye studies done in each of the onsite streams. Models are being improved as new information from studies of Cs-137 and uranium transport in onsite streams becomes available. New instrumentation installed in onsite streams will provide real time data on stream/river flow for input to the model. Discharges to onsite streams typically take at least one day to travel offsite and do not require the urgency of an atmospheric release which can cross the SRP boundary in an hour.

CONCLUSIONS

The integrated emergency response program at SRP provides rapid response to an unplanned release. The staff who are active during unplanned releases are the same people who conduct the basic and applied research programs. Many of the requirements for emergency response are the same as those required for the experimental research programs, e.g. operable equipment, good communications, good forecasting, and skilled data taking. Each experiment is conducted as a limited emergency response exercise so that everyone gets plenty of practice. In addition, the participants are knowledgeable about both the research and the emergency response actions, and able to suggest improvements in both areas. The interaction has proven to be very valuable.

ACKNOWLEDGMENT

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Savannah River Plant Emergency Response: Operations and Exercises

Doris D. Hoel

ABSTRACT. The Savannah River Plant is a complete nuclear complex with reactors, fuel fabrication, and fuel reprocessing onsite. Due to the potential for unplanned releases, an emergency response program has been developed to provide information on release consequences. The automated system was developed initially as a real-time site-specific system for the prediction and analysis of atmospheric and aquatic releases. However, it is flexible and has application for incidents at offsite facilities as well. The emergency response system, its components, equipment and capabilities, and the emergency response program, procedures and activities are presented. This includes the special case of joint response with Georgia Power Company's Vogtle Electric Generating Plant.

INTRODUCTION

The Savannah River Plant (SRP) is located in Southwestern South Carolina, 25 miles southeast of Augusta, Georgia, along the Savannah River (Figure 1). It is a government owned, contractor operated facility which produces nuclear materials for defense purposes. The complex includes four operating nuclear materials production reactors, a fuel and target fabrication area, two chemical separations facilities, high-level radioactive waste management operations, a low-level waste burial ground, and (under construction) a new facility to provide fuel for the U.S. Navy's nuclear powered submarines. During SRP's thirty-five years of operation, an emergency preparedness program has been developed, which includes response to nuclear incidents, both on the site and in the southeastern United States.

The following discussion describes emergency response at SRP. It includes components of the system, equipment and capabilities, responsibilities, procedures and activities during an emergency, response exercises, and incidents on site and in the southeastern United States.

DESCRIPTION

A site-specific, automated, real-time emergency response system has been developed at the Savannah River Laboratory (SRL) to predict, measure, and analyze the consequences of any

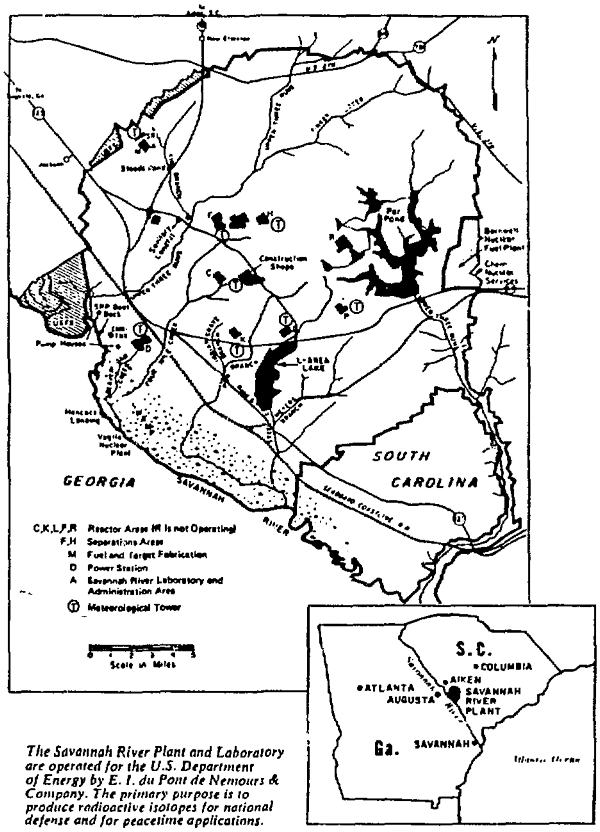

FIGURE 1. Savannah River Plant

unplanned radionuclide release as a result of operations at SRP. There are two major components of the system. One is the Weather INformation and Display (WIND) (Garrett et al., 1983) system for data acquisition and calculation of predicted release consequences. The other is the Tracking Radioactive Atmospheric Contaminants (TRAC) mobile laboratory (Sigg, 1985) capable of near real-time monitoring and plume tracking. In addition, real time sample collection teams are available for post incident radioactive plume tracking. Standard environmental monitoring analytical techniques have also been adapted to provide support for aquatic transport of a radioactive or hazardous waste release.

The WIND System consists of eight meteorology towers for real-time onsite data gathering, an Automated Field Operation and Services (AFOS) National Weather Service (NWS) computer for acquisition of regional meteorological data (Figure 2), two mini-computers, and a network of terminals (Figure 3) for data display and calculation of predicted consequences of either an atmospheric or aquatic release. The WIND System computers are Digital Equipment Corporation VAX 11/780 and VAX 11/750 super mini-computers. Duplicate sets of emergency response codes are kept current on both systems for backup purposes, and data collecting and archiving is automatically switched from one computer to the other in the event of computer malfunction. In addition to meteorological data, stream data from onsite streams and the Savannah River and stack monitoring data from the reactors and separations areas are collected on the VAX computers through the Remote Environmental Monitoring System (REMS). These data provide stream flow and source term information. WIND system data sources are shown in Figure 4. The network of terminals is located throughout the plantsite. Terminals are also in the homes of appropriate personnel for response during off shift hours.

The TRAC (Figure 5) mobile laboratory is designed for plume tracking and near real-time data collection and analysis during an atmospheric release. Its primary purpose is to measure low-level air concentrations of specific radionuclides as they move off the plant site. During an emergency, the TRAC is used to aid in determining dose estimates, to aid in determining adjustments to trajectory calculations, to provide release analysis, and meteorological code verification. Current equipment includes monitors to detect radioactive plumes, gamma and transuranic aerosols, radioiodines, tritium forms and noble gases, a mini-computer for data analyzing and archiving, and an engine/generator for electric power in the field.

FIGURE 2. The AFOS NWS Computer Used for Retrieving Local and Regional Meteorological Data

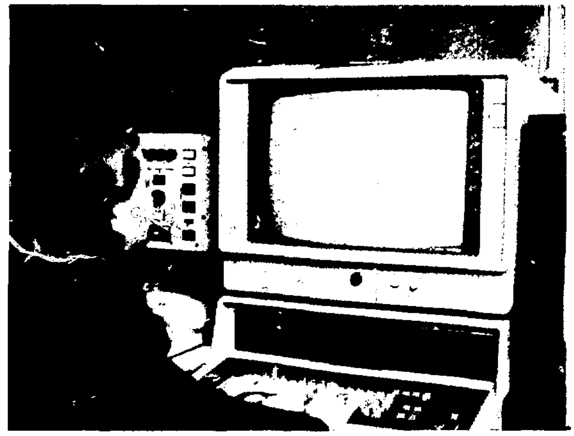

FIGURE 3. WIND System Graphics Terminals for Display of Data and Calculational Results Located Throughout SRP/SRL and in the Homes of Emergency Response Personnel

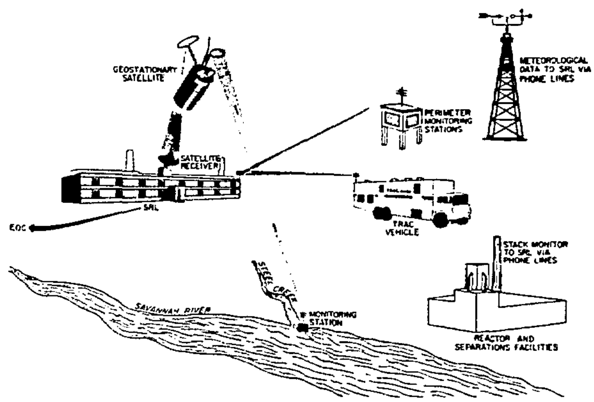

FIGURE 4. WIND Emergency Response System Data Sources

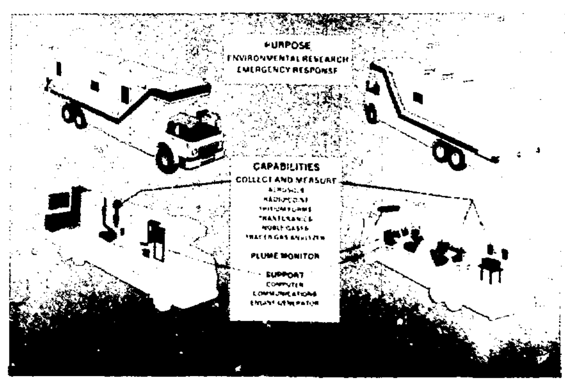

FIGURE 5. Tracking Radioactive Atmospheric Contaminants (TRAC) System

RESULTS

Onsite Incident Response

An emergency situation on the plant site in which SRP's Emergency Operating Center (EOC) is activated requires immediate notification of support and management personnel from SRP, SRL, Department of Energy Savannah River (DOE-SR), and Wackenhut Services, Inc. (WSI). A 24-hour notification procedure has been established in which required emergency response personnel are contacted. All emergency response personnel have been issued pagecoms for contact when away from their office during work hours or away from home during offshift. All pagecoms can be accessed from on or off plant. In addition, three groups of all-call pagecoms have been issued to primary emergency response personnel. The all-call pagecom system is used by EOC communications personnel to simultaneously notify groups of 29 persons of an EOC activation. The all-call system is used in addition to phone notification. Secondary notification for support is made by phone through appropriate departments.

Response time in an emergency depends upon time of notification. The EOC is staffed within minutes during regular shift, in 30-40 minutes during offshift, weekends, and holidays. SRL personnel involved with the WIND System provide immediate response in the Weather Center Analysis Laboratory (WCAL), and in the EOC in about 10 minutes during regular shift; TRAC is operational in approximately 30 minutes; and sampling vans are deployed in 15-30 minutes. During offshift, SRL personnel may respond immediately from home using WIND System terminals. EOC/WCAL response time is 30-40 minutes, TRAC response is about 1.5 hours and sampling van deployment is 45-60 minutes. Increased response time during offshift is the result of travel time to the plant from offsite.

The EOC/WCAL personnel have the responsibility

- to provide weather information, meteorological data, stream data, WIND System trajectory, and dose and deposition calculations
- to make adjustments to calculations for plume rise and mixing depth
- to interact with health protection personnel, area supervision, and technical personnel knowledgeable of the process involved in the incident
- to obtain source term and code input
- to communicate with TRAC and the sampling vans and direct monitoring team activities
- to assist health protection with their monitoring efforts
- to provide data assessment and analysis, and provide data and interpreted results to management and DOE

Figure 6 shows the information flow and interactions during an onsite incident. Finally, after the incident SRL response personnel provide a thorough analysis of all significant releases. The analysis includes trajectory, concentrations, doses, and health effects for both on and offsite populations. The response program continues to improve through research and regular WIND System and plantwide exercises.

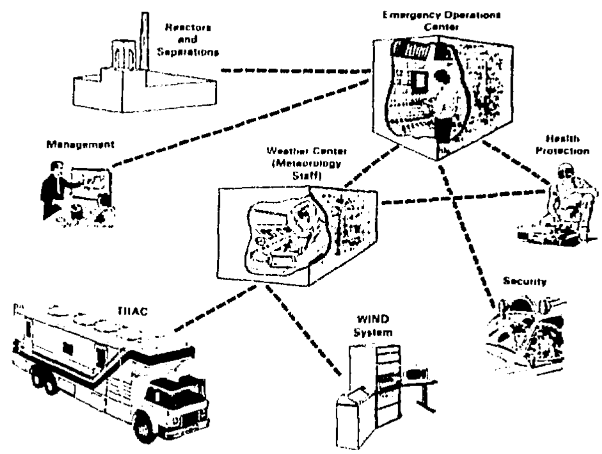

FIGURE 6. Onsite Emergency Response: Information Flow and Interactions

Offsite Incident Response

In the event of a major incident involving nuclear materials in the southeastern U.S., DOE-SR is responsible for organizing and coordinating the radiological monitoring and assessment activities of Federal agencies outside of the site or incident boundary (USDOE, SR-503). The emergency response system at SRL/SRP, although developed as a site specific system, is a major resource for DOE-SR's Region 3 Radiological Assistance Program (RAP).

Under normal circumstances, a request for SRL's assistance would come through DOE-SR's Office of External Affairs. Depending upon the nature of the incident and the amount of assistance required, response would be handled at SRP or may require travel to the incident area. In either case, the responsibilities for onsite response apply. If travel is required, initial WIND System calculations can be made at SRL by personnel who would remain at SRL to support SRP, while designated response personnel are enroute to the incident area. A graphics terminal with a built-in dial-in modem and hard copy unit (as well as a portable backup terminal with print capability) would be transported to the incident area for WIND System calculations. The TRAC vehicle remains in standby mode for either on or offsite incident response. As with onsite emergencies, all calculations, results, analyses, and recommendations concerning offsite impact are reported to DOE. DOE is responsible for approval and forwarding of information to the agency requesting assistance, and the cognizant Federal agency (CFA). Offsite response is shown in Figure 7.

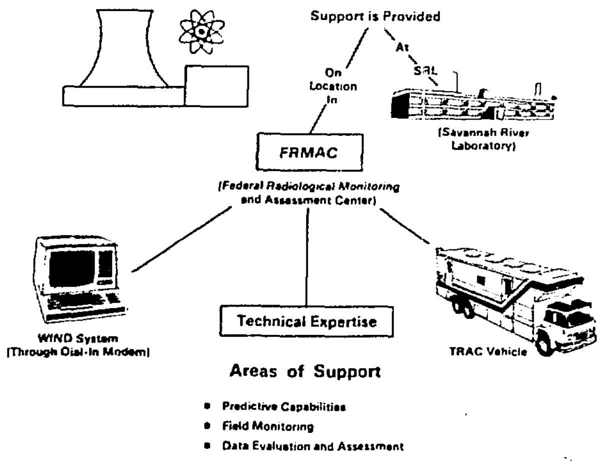

FIGURE 7. Offsite Emergency Response

On March 6-8, 1984, Savannah River participated in the St. Lucie Federal Nuclear Emergency Response Exercise in St. Lucie, Florida. SRL/SRP personnel utilized the WIND System, TRAC vehicle, and sampling vans to provide assistance in the areas of predictive capability, field monitoring, and data evaluation and assessment. This was the first major field exercise which provided a framework for a coordinated Federal response to a major incident.

A special case of offsite response (USDOE, SR 402.1) has also been established with the Georgia Power Company (GPC). GPC's Vogtle Electric Generating Plant is located just across the Savannah River in Georgia (see Figure 1). Due to proximity, DOE-SR and GPC have established a memorandum of agreement to provide for planning and responding to emergencies originating at either facility. SRP/SRL will respond to a Vogtle incident in the same manner as an onsite incident. SRP will be responsible for all monitoring and assessment and protective action on SRP, provide monitoring in South Carolina as requested, provide meteorological data to GPC, and advise GPC and the States of Georgia and South Carolina concerning the incident. Responses are reciprocol for an SRP incident. To facilitate response, direct communication lines between the two facilities and 24-hour-per-day points of contact have been established. Both facilities participate jointly in emergency exercises, the first of which was the Vogtle licensing exercise in April 1986.

Acknowledgment

The information contained in this article was developed during the course of work under Contract No. DE-AC09-76SR00001 with the U.S. Department of Energy.

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A Mobile Laboratory for Near-Real-Time Response to Radioactive Releases

R. A. Sigg

ABSTRACT. A mobile laboratory is improving radiological emergency response and environmental monitoring capabilities at the Savannah River Plant. The laboratory can monitor low concentration levels of radionuclides and can rapidly confirm the location and radionuclide composition of a downwind plume. The analysis system developed for the laboratory includes radionuclide-specific monitors for tritium forms (HT and HTO), gamma-ray emitting aerosols, volatile radioiodine, transuranic aerosols, and noble gases. The monitors can detect radionuclides transported by the atmosphere to offsite population centers at concentrations well below those considered hazardous. An analyzer for a passive chemical tracer gas (sulfur hexafluoride, SF_6) also is onboard. The diverse instruments, their good sensitivity and fast response, and the ability to operate while in transit, make this a unique laboratory.

INTRODUCTION

The diverse nuclear facilities at the Savannah River Plant (SRP) have a very low potential for the release of radionuclides at levels that could create an offsite hazard. However, in keeping with the philosophy that protection of the public and the environment is essential, the Savannah River site has a continuing research and development program to improve its radiological emergency response and environmental monitoring capabilities. A mobile laboratory (Sigg, 1985) (Figure 1), to monitor low concentration levels of radionuclides and to rapidly confirm the location and radionuclide composition of a downwind plume, has been developed as a part of this program.

The large variety of radionuclides produced at the site prompted the development of a number of specialized atmospheric collection and analysis systems for the Tracking Radioactive Atmospheric Contaminants (TRAC) mobile laboratory (Sigg, 1985). These include radionuclide-specific monitors for tritium forms, gamma-ray emitting aerosols, volatile radioiodine, transuranic aerosols, and noble gases. (HT and HTO are used generically to represent the several tritium-containing elemental and oxide chemical forms of hydrogen isotopes.) The monitors can detect radionuclides, transported by the atmosphere to offsite population centers, at concentrations well below those considered hazardous. Their measurements can address legitimate public concerns following any abnormal radionuclide release. An analyzer for a passive chemical tracer gas (sulfur hexafluoride, SF_6) also is onboard. The diverse instruments, along with their good sensitivity and fast time response, make this a unique laboratory. The ability to operate while in transit is an improvement over most mobile facilities. Some of the important functions of TRAC are to

- Provide prompt analyses in the field instead of delayed analyses at site laboratories
- Obtain timely concentration results in the event of an abnormal release in order to
  - assure the plume has been located and profiled before returning from the field
  - provide information almost immediately to the onsite Emergency Operating Center (EOC)
- Test and refine emergency response procedures
- Provide data to evaluate and improve atmospheric transport models by

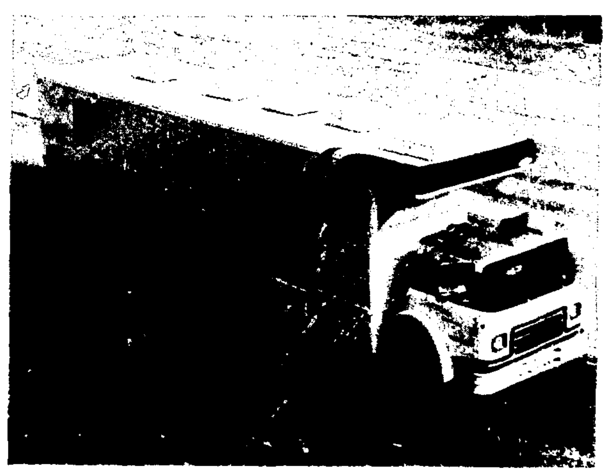

FIGURE 1. Tracking Radioactive Atmospheric Contaminants (TRAC) Laboratory

- monitoring small releases from SRP's normal operations
- measuring releases of SF_6 tracer gas to simulate emergency response operations
- gathering data during emergency response exercises.

PLUME LOCATION

The extensive monitoring capabilities of the Savannah River Plant can be applied most effectively when personnel have downwind plume location information to assist sample collection operations. Three plume locating methods are available to guide sampling operations.

TRAC initially receives forecast plume trajectories and concentrations by radio from the onsite Weather Center Analysis Laboratory (Garrett, 1983).

A direction-sensitive gamma-ray monitor (Figure 2) determines plume locations relative to the vehicle heading. This onboard plume monitor uses twelve large NaI(Tl) spectrometers similar in volume to those used in some aerial surveying applications. The detector array is shielded on the bottom and sides to minimize interferences from naturally occurring radionuclides found in soils and rocks. Successive short (~15 sec) counting intervals yield a time series histogram as the laboratory transects a plume (Figure 3). Low plume concentrations of short lived ^{41}Ar (2 hr) from normal reactor operations have been detected as far as 50 kilometers downwind of an SRP reactor.

A real time analyzer for low concentrations of SF_6 tracer is being tested (Milham, 1986). SF_6, intentionally released from onsite facilities, was detected 50 kilometers downwind (Figure 4) in tests of our real-time plume locating capabilities. When SF_6 and ^{41}Ar are released simultaneously from two different points on the plant, measurements downwind show the plume trajectories are nearly parallel. Location of either plume downwind allows us to deduce the location of the other. In the event of an inadvertent atmospheric radioactivity release, ^{41}Ar plumes could be useful in locating plumes from another facility. The possibility of simultaneously releasing tracer gas at the effluent point is also being explored; in this case, detection of the tracer gas would coincide with the radioactive plume location.

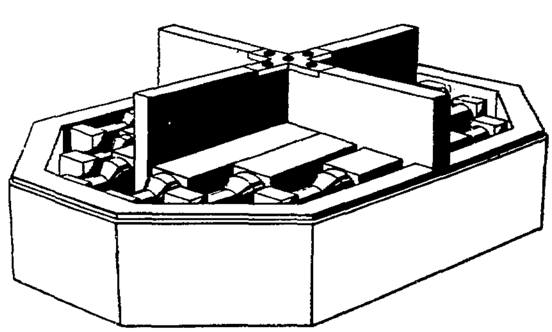

FIGURE 2. Plume Monitor NaI(Tℓ) Detector Array

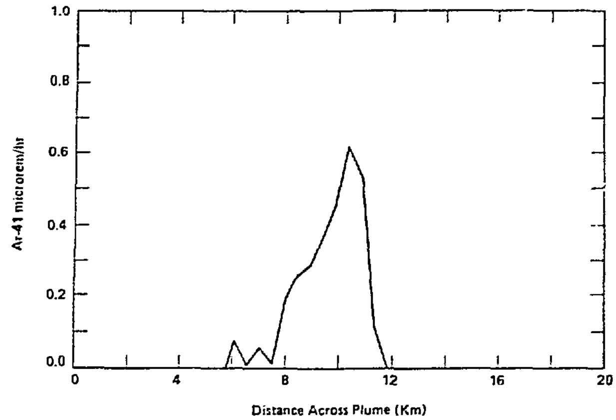

FIGURE 3. Crosswind Transect of an Ar-41 Plume at the SRP Boundary

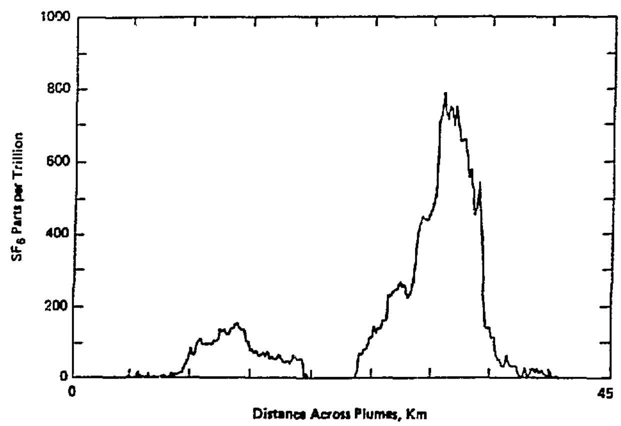

FIGURE 4. Successive Crosswind Transects of an SF_6 Plume at 25 Km Downwind

COLLECTION AND ANALYSIS SYSTEMS

Specialized monitoring systems collect and measure radionuclides from large volumes of air. Concentrations less by a factor of 10^{-6} than the maximum permissible concentration (MPC) limits for the general population in uncontrolled areas can be detected for some nuclides (e.g., ^3H and ^{99m}Tc). Sensitivities are less for monitoring other radionuclides, but all radionuclides of interest can be detected at or below MPC limits in near-real time. These are significant improvements over previous field measurement capabilities.

TRITIUM FORMS

A tritium forms monitor (Figure 5) collects and analyzes tritium either as moisture (HTO) or as elemental gas (HT). Adsorption techniques concentrate individual forms from large volumes of air, and desorbed sample material is analyzed by liquid scintillation spectrometry. The complete process takes about forty minutes. The sensitivity (~20 pCi/m^3) permits detection of any abnormal release, even 40 km downwind. Figure 6 shows separate plumes that were detected onsite for HT and HTO from different facilities.

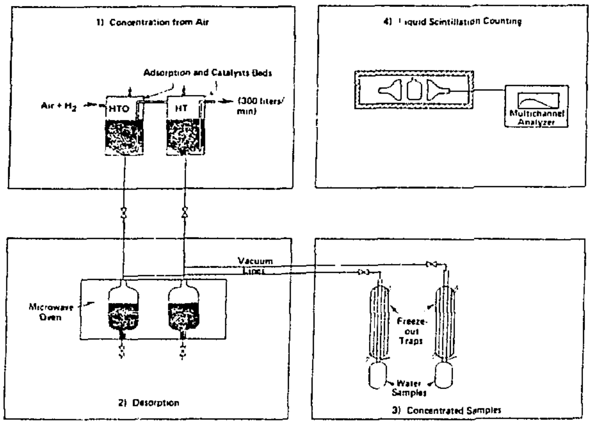

FIGURE 5. Real-Time Tritium Forms Monitor

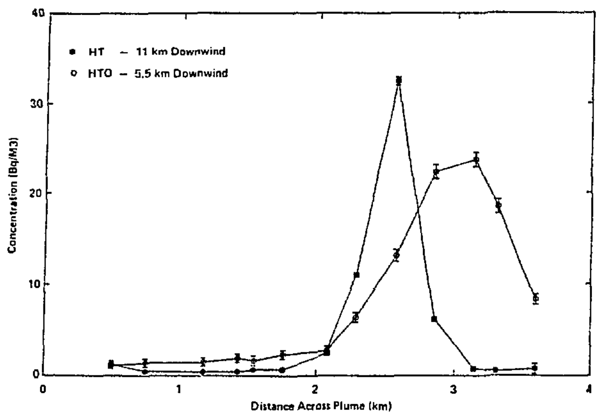

FIGURE 6. Crosswind Transect of HT and HTO Plumes from Two Separate SRP Facilities

GAMMA-RAY AEROSOLS

A monitor collects aerosols from air flowing through a filter at twenty-five cubic meters/minute. In about twenty minutes, post-collection germanium spectrometry (Figure 7) identifies individual gamma-ray emitting radionuclides at concentrations ranging from 1/15 to better than 10^{-5} of MPC (for ^{99m}Tc, ^{132}Te, and ^{141}Ce) without waiting for normal radon daughter interferences to decay. Radioactive debris from the Chernobyl accident was easily detected in the southeastern United States by the monitor (Figure 8).

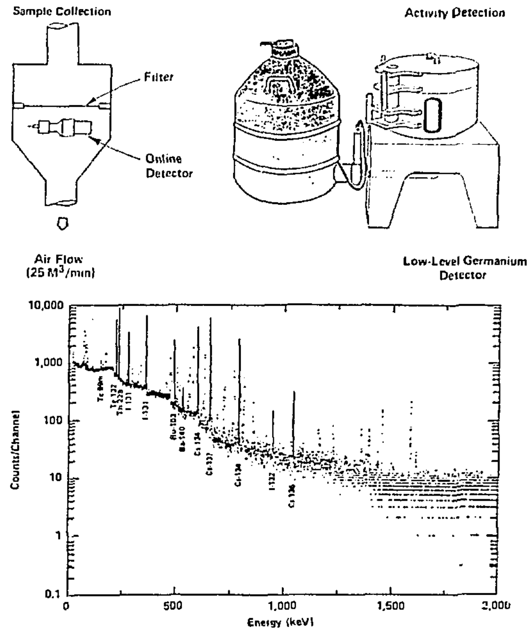

FIGURE 7. Gamma-Ray Aerosol Monitor and Chernobyl Debris Aerosol Spectrum

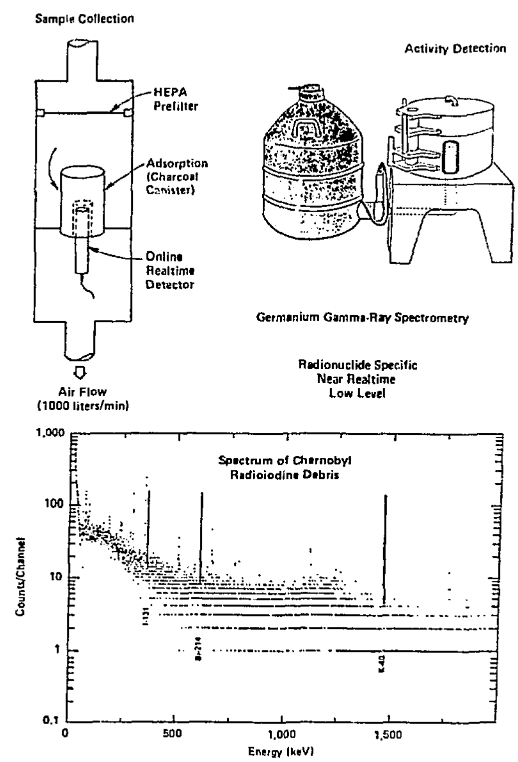

FIGURE 8. Volatile Radioiodine Monitor and a Gamma-Ray Spectrum of Chernobyl Radioiodine Debris

RADIOIODINE

A charcoal canister concentrates radioiodine from an air stream and then gamma-ray spectrometry (Figure 7) evaluates the canister activity. The sensitivity for ^{131}I is 1/3 of MPC when samples are collected for ten minutes and counted for ten minutes. ^{131}I from the Chernobyl accident was also easily detected in the vicinity of the SRP.

TRANSURANICS

A Teflon®-based filter medium collects aerosols from an air stream (Figure 9). The aerosols are deposited at the surface of the media, allowing alpha spectrometry by silicon diodes to distinguish transuranics from radon daughters. Results with detection limits equal to the MPC of ^{239}Pu are produced in an hour.

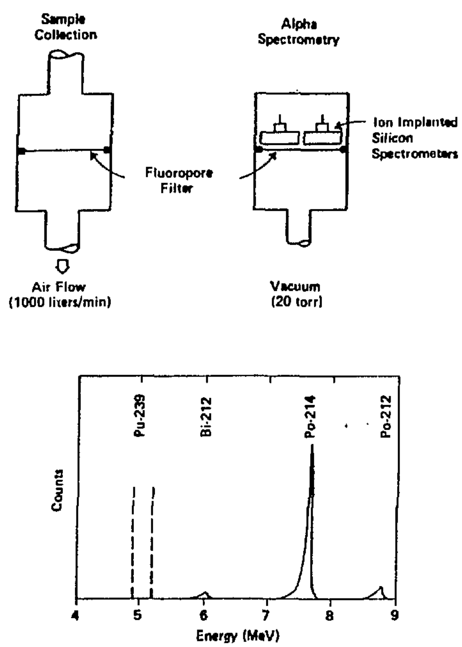

FIGURE 9. Transuranic Aerosol Monitor and a Background Alpha Spectrum

NOBLE GASES

About 6 cubic meters of air are analyzed for noble gas activities. Filtered air is compressed to about 170 atmospheres to concentrate the air in a pressure vessel. A germanium spectrometer in the pressure vessel annulus (Figure 10) analyzes noble gas activities.

APPLICATIONS AND SUPPORT SYSTEMS

The above laboratory systems are supported by a real-time computer and a Loran-C navigation system.

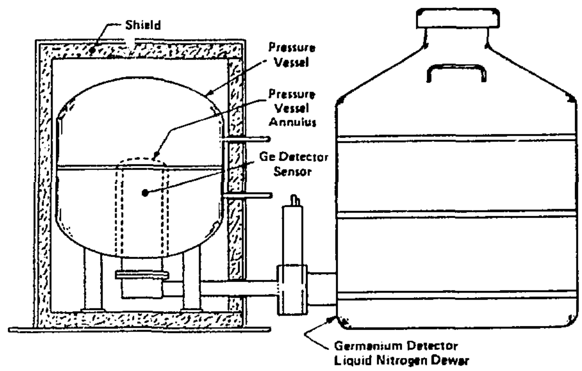

FIGURE 10. Noble Gas Monitor Counting Apparatus

The TRAC mobile laboratory supports onsite environmental research programs through atmospheric transport model validation studies and by monitoring for trace concentrations of radionuclides with the potential for release during normal SRP operations. The laboratory monitored and easily detected radioactive debris from the Chernobyl reactor accident in air samples collected in population centers within 160 km of SRP. The TRAC mobile laboratory also participated in the Federal Emergency Management Agency (FEMA)-sponsored St. Lucie Federal Field Exercise, as well as a joint exercise between SRP and the states of South Carolina and Georgia. TRAC's participation in these exercises and research programs have spinoff benefits that include testing and improving sampling strategies, logistics, communications, and personnel training (Addis, 1986).

ACKNOWLEDGMENT

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Addis, R. P., R. J. Kurzeja, and A. H. Weber, 1986, "Improving Emergency Response Through Field Exercises," This Conference.

Garrett, A. J., M. R. Buckner, and R. A. Mueller, 1983, "The Weather Information and Display Emergency Response System." Nucl. Tech. 60(1):50-59.

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Sigg, R. A., 1985, "A Mobile Laboratory for Near Real-Time Measurements of Very Low-Level Radioactivity." Proceedings of the Fifth DOE Environmental Protection Information Meeting, Albuquerque, NM, CONF-841187, November 5-8, 1984.

Improving Emergency Response Through Field Exercises

R. P. Addis, R. J. Kurzeja, and A. H. Weber

ABSTRACT. Exercises using a tracer gas to simulate the accidental release of gaseous effluent to the atmosphere have been conducted at the Savannah River Plant since 1983. Although the tracer released is a passive gas, the operations are similar to those required for unplanned releases of radionuclides. These exercises, called Mesoscale Atmospheric Transport Studies (MATS), have produced many tangible benefits, including efficient, dependable performance of sampling teams, refinement and documentation of required skills and logistics, and better wind forecasting. The exercises have provided data required to evaluate the performance of dispersion models, and identified areas requiring improvement. For example, the operational Gaussian model was found to accurately predict puff centerline and puff width, but to overpredict the maximum concentration. The relative merits of moving samplers versus stationary samplers under various atmospheric conditions are also discussed.

INTRODUCTION

Fifteen-minute stack releases of the tracer gas, sulfur hexafluoride (SF_6), are used to simulate the unplanned release of radionuclides from facilities at the Savannah River Plant (SRP). Atmospheric dispersion of the tracer gas is detected and measured by both fixed and moving sampling systems.

The goals of these exercises are to:

- Determine atmospheric dispersion model accuracy
- Provide an operational test of the capability to respond during emergencies
- Provide information on crosswind and downwind spread of gaseous effluent

A downwind distance of 30 km was chosen for monitoring because it corresponds to the distance between the release point and the major population centers of Augusta, Georgia, and Aiken, South Carolina. Thirty kilometers also corresponds geographically with an approximately circular series of highways where vehicles can deploy samplers alongside the roads. A schematic map in Figure 1 shows the SRP, the ring of roads encircling the plant, and the configuration for a typical tracer experiment (MATS).

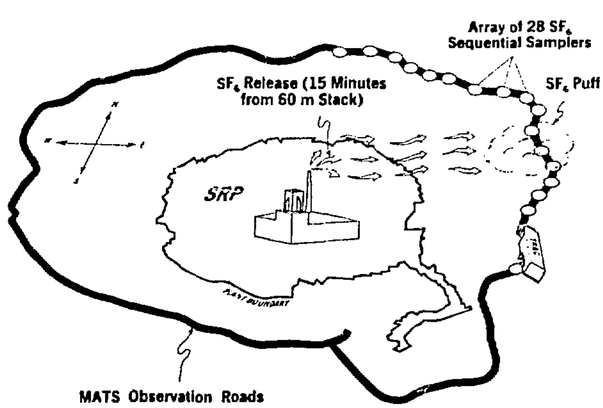

FIGURE 1. A schematic map of the Savannah River Plant, the ring of roads (called the MATS arc) on which air sampling is conducted, and a typical configuration of moving and stationary samplers

DESCRIPTION

During a typical exercise, personnel are advised of the release time and are organized into three sampling teams. Two of the teams load twenty-eight programmable, sequential air samplers into vans. The third team operates a mobile laboratory called Tracking Radioactive Atmospheric Contaminants (TRAC, Sigg, 1985) which contains a continuous operating SF_6 sampling system and other atmospheric monitoring devices. When the three teams leave the site, they receive a puff trajectory forecast. The teams then drive to the intersection of the puff centerline and the sampling arc. While the vans are traveling to the intersection point, meteorologists in the weather center study the surface pressure forecasts and observations, balloon soundings, and measurements from eight instrumented meteorological towers to refine the accuracy of the initial puff impact-zone forecast. These instrumented towers are located near each production facility within the plant boundaries. The spacing of the drop points along the roadside for the sequential samplers is based on turbulence measurements from tower wind instruments. Deployment instructions are then relayed from the weather center to the vans.

As the vans deploy samplers, the TRAC vehicle with its mobile detection capability travels crosswind along the sampling arc to determine the puff's arrival time and actual location. The SF_6 concentration data are stored on computer tapes and discs in real time. Since TRAC is in radio communication with the Weather Center, information can be passed to advise the fixed sampling teams to reprogram or to move samplers to a better location. This procedure can be extended to greater distances downwind as the puff moves along its path. A computer program determines the start time and optimizes the sampling interval for the 10 sequential motor-driven syringes used to collect whole air samples.

After the sequential samplers are retrieved from the roadside and transported back to the laboratory, the SF_6 concentrations are determined by gas chromatography. The TRAC SF_6 concentrations are also transferred to the weather center computer system. Calibration gas standards are available in four concentrations to ensure the accuracy of measurements of both the fixed sequential samplers and the mobile continuous sampler. The concentration data are plotted in time-space cross-sections and compared with models used during emergency response.

RESULTS AND CONCLUSIONS

The exercises have provided the data required to determine the performance of the dispersion forecast models. The operational emergency response Gaussian model, PUFF/PLUME, accurately predicts the width and centerline location of the puff, but overpredicts the centerline maximum concentration on average by 5.4 times. A sequential Gaussian model, 2DPUF, also used in emergency response, more closely predicts the centerline maximum concentration - only overpredicting by 1.5 times.

The logistics of sampler deployment have been improved through heuristically optimizing the spatial distribution of the samplers, the sampling intervals, and perfecting communications and deployment timing between the meteorological forecasters and field crews. The importance of accurate and timely forecasts of transport wind speeds and directions became evident when the field exercises were begun. For example, a transport distance of 30 km results in a mean transport time of about 2 hours for a 15 km/hr wind. Thus, a forecast error of only 10 degrees in direction and 3 km/hr in speed for the transport wind results in a centerline location error of 5 km at the samplers and a timing error of about 30 minutes.

Another practical benefit of the exercises was the testing of sampling strategies. Sampling with a series of stationary sequential samplers is useful in steady or predictable wind regimes. Such a strategy enables the measurement of several cross-sections of the puff as it passes overhead. Results from a typical stationary sampler deployment are shown in Figure 2. However, during variable conditions the stationary sampling strategy often proves to be less than ideal. It is during such conditions that the moving sampler strategy of the TRAC vehicle is most valuable.

The TRAC is able to cross the puff and measure the SF_6 concentrations in real time. Figure 3 shows five successive cross sections of a typical SF_6 release. However, if wind speeds are high, only one transect through the puff may be possible.

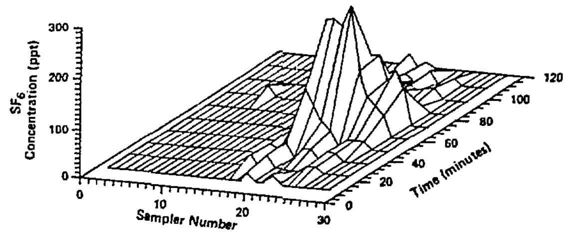

FIGURE 2. Typical results from stationary sequential air samplers showing the SF_6 concentration in time and space as a puff crossed the MATS arc

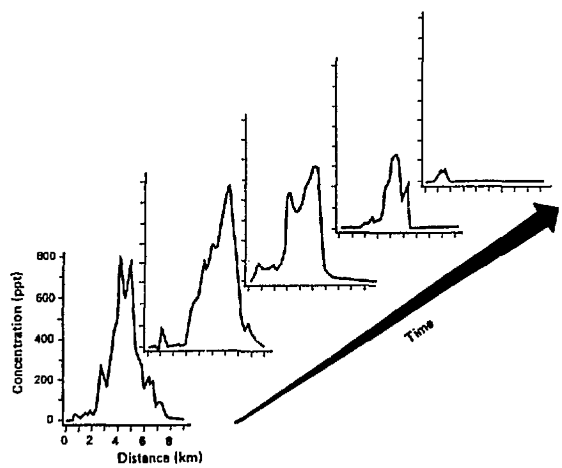

FIGURE 3. Typical cross sections of an SF_6 puff as measured by a continuous SF_6 analyzer onboard TRAC. These are five successive concentration cross-sections for MATS #26, 12/11/85

ACKNOWLEDGMENT

The information contained in this article was developed during the course of work under Contract No. DE-AC09-76SR00001 with the U.S. Department of Energy.

REFERENCES

Sigg, R. A., 1985, "A Mobile Laboratory for Near Real-Time Measurements of Very Low-Level Radioactivity." Proceedings of the 5th DOE Environmental Protection Information Meeting Held in Albuquerque, New Mexico, November 5-8, 1984, CONF-841187, April 1985, pp. 629-636.

Savannah River Plant Remote Environmental Monitoring System

J. F. Schubert

ABSTRACT. The SRP remote environmental monitoring system consists of separations facilities stack monitors, production reactor stack monitors, twelve site perimeter monitors, river and stream monitors, a geostationary operational environmental satellite (GOES) data link, reactor cooling lake thermal monitors, meteorological tower system, Weather INformation and Display (WIND) system computer, and the VANTAGE data base management system. The remote environmental monitoring system when fully implemented will provide automatic monitoring of key stack releases and automatic inclusion of these source terms in the emergency response codes.

INTRODUCTION

The Savannah River Plant (SRP) remote environmental monitoring system is a new concept in emergency response, designed to provide real-time accurate source terms to the emergency response computers in the event of an unplanned release of radioactive or toxic chemicals to the biosphere. Data are collected and archived for immediate use during the emergency response, or for later use in research and development.

The 800 square kilometer Savannah River Plantsite operated by E. I. du Pont de Nemours and Company for the U.S. Department of Energy produces nuclear materials for the U.S. Government. SRP is considered a complete nuclear complex. Major facilities at SRP include:

- Four nuclear production reactors moderated and cooled with heavy water
- Two chemical separations plants for separating and purifying the reactor products
- A Fuel Materials Facility for producing fuel for the U.S. Navy's nuclear powered ships
- A Defense Waste Processing Facility which will process high-level radioactive waste for incorporation in glass for underground storage

Although the facilities of the Savannah River Plant have a very low potential for hazardous releases to the environment, the Environmental Technology Division of the Savannah River Laboratory with the aid of SRP personnel have designed and begun installation of a real-time remote environmental monitoring system. This system places the latest monitoring technology in existing production facilities to provide accurate source terms to a central computer system for effluents which might be released to the biosphere.

Key stack release instruments are presently monitored by control room personnel and the results are transmitted verbally by telephone to the Emergency Operating Center (EOC). These data are then manually entered into the Weather Information and Display (WIND) System computer (Garrett et al., 1981). The WIND System combines the stack release data and local meteorological data and calculates the transport and dispersion of toxic and/or radioactive pollutants. The WIND System also calculates the predicted integrated doses and dose rates for both onsite and offsite populations.

Delays can occur in obtaining the necessary information about the possible dose consequences to the offsite populations because of the large amount of manual data entered and the verbal communications between the EOC and operating areas. To eliminate these delays and the potential for human error, a program is in progress for the installation of electronic stack monitors and the necessary associated electronics to scale and digitize the signals for direct transmission to the WIND System computers.

SYSTEM DESCRIPTION

The SRP remote environmental monitoring system is composed of the following instrumentation: Separations Facilities Stack Monitors, Production Reactor Stack Monitors (Figure 1), Twelve Site Perimeter Monitors (Figure 2), River and Stream Monitors (Figure 3), Reactor Cooling Lake Thermal Monitors, Meteorological Tower System (Figure 4), WIND System Computer, and the VANTAGE Data Base Management System.

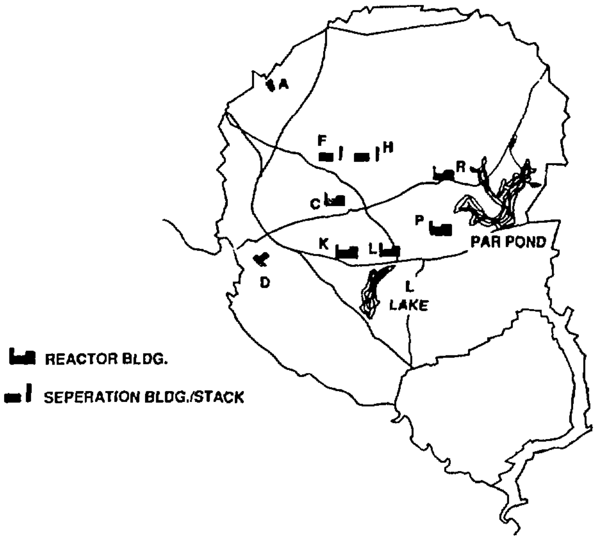

FIGURE 1. Map of the Savannah River Plant Showing Locations of Reactors and Separation Buildings

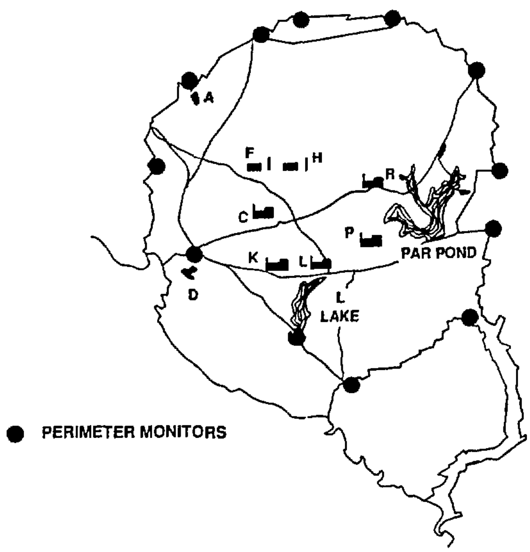

FIGURE 2. Location of the Twelve Perimeter Monitors

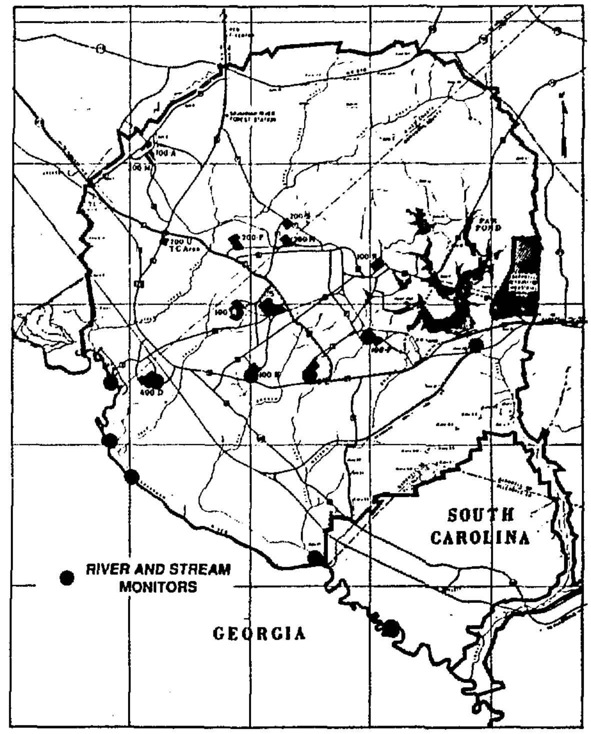

FIGURE 3. Location of the River and Stream Satellite Data Collection Platforms

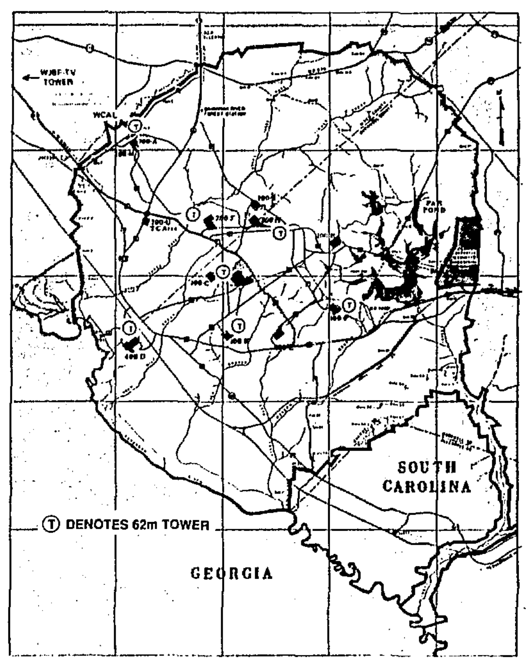

FIGURE 4. Location of the Seven Meteorological Towers, Weather Center Analyses Laboratory and the WJBF-TV Tower

A Geostationary Operational Environmental Satellite (GOES) data link is used to transmit data from remote locations in and around the plantsite (river, streams, and perimeter monitors). Radio-Telemetry data links connect reactor cooling lake temperature monitors with a computer controlled base station. The computer also transmits the received data over telephone lines to the WIND System computer. The noble gas monitor uses fiber optic cables to link the stack monitor to the control room. The fiber optic cable is immune to electro-magnetic interference from motors and relay switches in the local area.

RESULTS

As each element of the system is completed, it is linked to the WIND System computer and the remote environmental monitoring data base. Elements completed to date are: the meteorological tower system; reactor cooling lake temperature monitors; river and stream monitors, and associated GOES satellite up-link and down-link; and a tritium forms stack monitor. The noble gas data links will be available shortly with the installation of data collection and transmission equipment. The real-time monitoring data are combined with the meteorological data using the WIND System VAX 11/780 computer and the VANTAGE Data Base Management System. The VANTAGE Data Base Management System provides seven levels of alarm and alert functions. It also can display up to four variables (in any combination) on an electronic strip chart. VANTAGE archives all the variables in such a fashion that they are immediately available to the WIND System emergency response codes, or are available at a later date for any other purpose.

The remote environmental monitoring system when fully implemented will provide automatic monitoring of key stack releases that are presently monitored by control room personnel. This will eliminate verbal communications and delay in receiving the necessary information needed for the emergency response codes. This system will provide complete real-time accurate source term information for automatic inclusion in the emergency response codes.

ACKNOWLEDGMENT

The information contained in this article was developed during the course of work under Contract No. DE-AC09-76SR00001 with the U.S. Department of Energy.

REFERENCES

Garrett, A. J., M. R. Buckner, and R. A. Mueller, 1983, "The Weather Information and Display Emergency Response System." Nucl. Tech. 60(1):50-69.

Aquatic Emergency Response Model at the Savannah River Plant

David W. Hayes

ABSTRACT. The Savannah River Plant emergency response plans include a stream/river emergency response model to predict travel times, maximum concentrations, and concentration distributions as a function of time at selected dowstream/river locations from each of the major SRP installations. The menu driven model can be operated from any of the terminals that are linked to the real-time computer monitoring system for emergency response.

INTRODUCTION

The Savannah River Plant (SRP) has emergency response plans to be put into effect in the event of a significant release to the environment of toxic or radioactive material. This paper concerns the emergency response to the escape of liquid pollutants to plant streams which flow to the Savannah River and thence to the Atlantic Ocean. More specifically, the paper describes a mathematical model used to predict the immediate consequences of the liquid release.

The computer input information required for the stream/river emergency response model includes the known or estimated quantity, composition, and location where the pollutant entered a plant stream. The model then predicts travel times, maximum concentration, and concentration distributions as a function of time for selected downstream/river locations. The menu driven model can be operated from any of the terminals that are linked to the real-time computer monitoring system for emergency response. These terminals are in the control rooms of all major SRP facilities. The selected downstream locations stored in the model include the public highway (SC Highway 125 that crosses the SRP facility, mouths of streams at the Savannah River, two bridges that cross the Savannah River below SRP, and the water intakes for the Beaufort/Jasper and City of Savannah Water Treatment Plants.

OPERATION OF THE MODEL

The menu-driven aquatic emergency response model contains all of the required stream information in its data base and requires only entry of data pertinent to the release. In the example here, a release from C Reactor to Four Mile Creek will be simulated. The execution of the model requires entry of date, time, type, and units of release (part (a) of Table 1), facility identifier (part (b) of Table 1), the amount of release, and the duration of release (part (c) of Table 1).

TABLE 1

Input Data for a Release
(a) Type, (b) Location,
(c) Amount and Duration

```
    ************************************************
(a) EMERGENCY RESPONSE FOR RELEASES TO PLANT STREAMS
    ************************************************
    INPUT FOR THE TIME AND DATE OF RELEASE:
    NOON IS 0000 PM, MIDNIGHT IS 0000 AM.
    INPUT RELEASE START (DEFAULT; 0756 PM 081786)

    IS RELEASE RADIOACTIVE (Y or N, default is Y):
    ENGLISH OR METRIC UNITS (E or M, default is E): M
    ARE RELEASE UNITS L , KG, CI (Default is CI):

(b)         IF RELEASE LOCATION IS:        PLEASE TYPE

            D. AREA OUTFALL                DD
            HIGHWAY 278 UPPER 3 RUNS       H8
            RD. C (FLOWING STREAMS)        FS
            RD. A (UPPER 3 RUNS)           UA
            H AREA                         HH
            F AREA                         FF
            C AREA OUTFALL                 CC
            P AREA OUTFALL                 PP
            K AREA OUTFALL                 KK
            RD. A (4 MILE CREEK)           A4
            RD. A (STEEL CREEK)            AS
            PAR POND DAM                   PD
            MOUTH OF UPPER FOUR MILE       MU
            MOUTH OF LOWER 3 RUNS          ML
            MOUTH OF STEEL-PENBRANCH       MS
            MOUTH OF 4 MILE CREEK          M4
            MOUTH OF BEAVER DAM            MB
            MOUTH OF UPPER 3 RUNS          MR
            ALTERNATE (MANUAL INPUT)       AL

            ENTER CHOICE HERE - CC

        ************************************************
(c)       SLUG RELEASE OPTION
        ************************************************
       ENTER THE NUMBER OF CI OF MATERIAL RELEASED: 5
       ENTER THE DURATION OF THE RELEASE IN MINUTES: 25
```

The output is designed to show graphically the impact of a release to downriver users and to help in the design of a sample plan to monitor the release. The output consists of a table and several graphs. Table 2 summarizes the input information, in this case for C Reactor at SRP, and gives the results of the model calculation (the maximum concentration and time of arrival at each of the preselected downstream/river locations). The maximum concentration at C-Reactor outfall was 3.3 E-07 Ci/L and this concentration had been reduced to about 1.9 E-07 Ci/L by the time it had arrived at SC Highway 125 230 minutes later due to dispersion processes (Table 2).

TABLE 2

SRL Stream Transport Program

SRL STREAM TRANSPORT PROGRAM 17-AUG-86 7:56 PM DT PAGE 1

DOWNSTREAM CONCENTRATIONS CALCULATED FOR A RELEASE AT C-AREA OUTFALL

RESULTS FOR A SLUG RELEASE OF 5.00 CI, LASTING 25.00 MINUTES

INITIAL CONCENTRATION: 0.3300E-06 CI/L

RELEASE TIME IS 7:56 PM ON 8/17/86
THE REACTOR IS UP
THE RELEASE IS RADIOACTIVE

	MAX CONCENTRATION (CI/L)	ETA (MIN)	TIME	DATE
ROAD A:	9.18812E-06	229.00	11:45 PM	8/17/86
SAVANNAH RIVER:	0.22694E-07	1035.00	1:11 PM	8/18/86
US HIGHWAY 301:	0.10567E-08	2308.00	10:24 AM	8/19/86
STATE HIGHWAY 119:	0.94951E-09	4601.00	12:37 AM	8/21/86
BEAUFORT-JASPER:	0.91551E-09	5490.00	3:26 PM	8/21/86
SAV. WATER PLANT:	0.90096E-09	5899.00	10:15 PM	8/21/86

The travel time to the Savannah River is about 1000 minutes from C Reactor Area and the concentration was reduced by a factor of ten due to dispersion processes. Once flow from Four Mile Creek entered the Savannah River, it was immediately diluted by a factor of about 25. Concentration changes were small after mixing with the Savannah River.

Graphical illustrations are then used to present the tabular results on area maps and to show the time/concentration profiles at downstream/river locations. The first graph (Figure 1) is a map of SRP that shows the location of the release, identifies the public road and mouth of the creek for that release, and shows maximum concentration and estimated time of arrival. The second graph (Figure 2) is a map showing the same type of data as the first graph but for downriver locations. The third graph (Figure 3) is a plot of the concentration as a function of time for each preselected location. The third graph is used to estimate the duration of release and the time the concentration could be above some guideline concentration level. Because the concentration reduction is large after mixing with the Savannah River, a plot showing the log of the concentration is also given so that details of the concentration in the Savannah River can be seen.

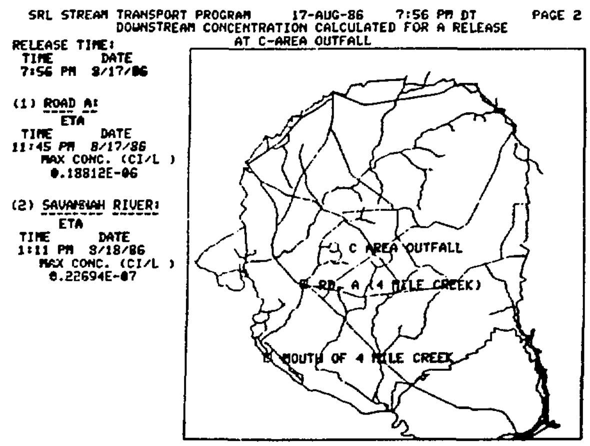

FIGURE 1. Downstream Concentration Calculated for a Release at C-Area Outfall - Onplant

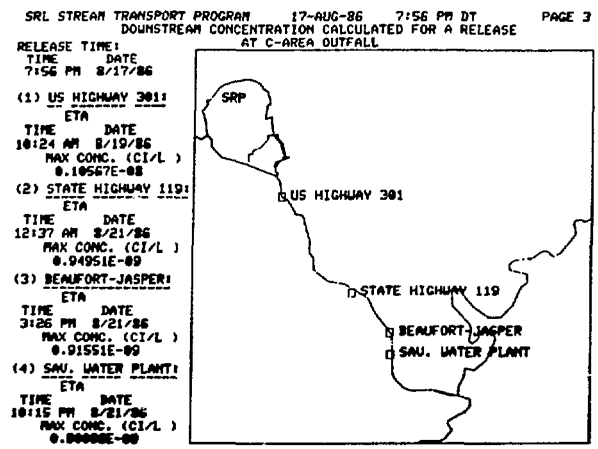

FIGURE 2. Downstream Concentration Calculated for a Release at C-Area Outfall - Offplant

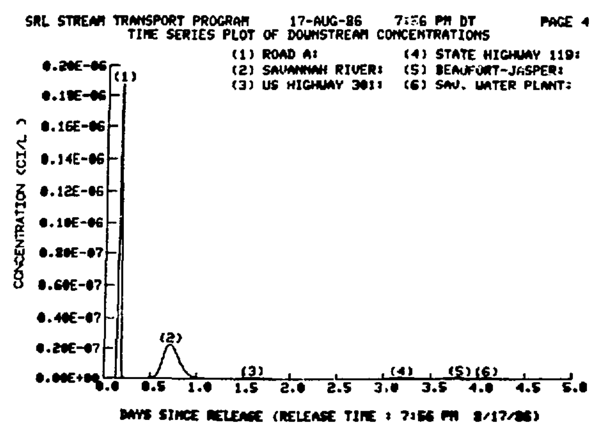

FIGURE 3. Time Series Plot of Downstream Concentrations

MODEL

A one-dimensional model was used to describe pollutant transport in SRP streams and the Savannah River.

$$\frac{\partial C}{\partial t} = D \frac{\partial^2 C}{\partial x^2} - u \frac{\partial C}{\partial x}$$

where C = the average cross sectional concentration
x = coordinate in the direction of flow
u = the mean velocity of flow
D = the longitudinal mixing coefficient
t = time

The model is conservative (i.e., it is assumed that no material loss occurs during transport) and can be used to calculate the results for any release distribution by use of routing. A one-dimensional model is an adequate description of pollutant transport except in the zone of discharge. Mixing in the discharge zone is dominated by convective dispersion which produces a skewing of the pollutant concentration distribution and is not predicted by the above equations. The mixing zone is a small feature (the effective zone is equal to about 50 stream widths) when compared to the length of the stream. These types of models have been used to describe pollutant transport in many streams and rivers (Fischer, 1966; Thackston, Hays, and Krenkel, 1967).

A conservative model from an emergency response view results in higher pollutant concentrations and faster travel times in a stream or river than a nonconservative model where adjustments are made for deposition, sorption, and chemical reactions for pollutants in a stream or river. The conservative model is considerably easier to implement and by use of the pollutant routing technique can handle almost any pollutant release distribution.

EXPERIMENTAL EVALUATION OF STREAM/RIVER TRANSPORT COEFFICIENTS

It was necessary to determine the velocity and dispersion coefficient for each stream and the Savannah River to estimate the travel times and pollutant concentrations at a location. These coefficients were determined using dye tracer studies. A one-dimensional model, TETRAD, was programmed to allow the calculation of stream velocities and dispersion coefficients from measured dye concentration profiles. For each stream, dye was released at the facility and the dye concentration curve was measured as a function of time at downstream locations.

Stream velocities and dispersion coefficients were calculated from the dye curves using TETRAD. TETRAD routed the point-by-point dye concentration from an upstream to a downstream location using an initial set of stream velocity and dispersion coefficients estimated from the statistics of the dye distribution (Godfrey and Frederick, 1963; H. B. Fischer, 1968; Thackston, Hays, and Krenkel, 1967; and Buckner and Hayes, 1975). The stream velocity and dispersion coefficients were then incremented and the process continued until nonlinear least squares estimators were minimized for the coefficients. When the least squares estimators were minimized for the routing, the best estimate for velocity and dispersion was obtained. An example of the dye response curves in Lower Three Runs Creek and the resulting coefficients are shown in Figure 4. Slight deviations of the model from the actual creek dye response can be seen. These coefficients were then used as a data set for the emergency response model. Dispersion coefficients and velocities determined for streams on the Savannah River site ranged from 1 to 25 m/sec and from 0.1 to 3 km/hr.

The model is continually updated with new data and re-evaluated velocity and dispersion coefficients. Real-time flow data telemetered from the streams and river are being incorporated in the model, thus assuring that the appropriate dilution factors are available for each stream at the time of emergency response.

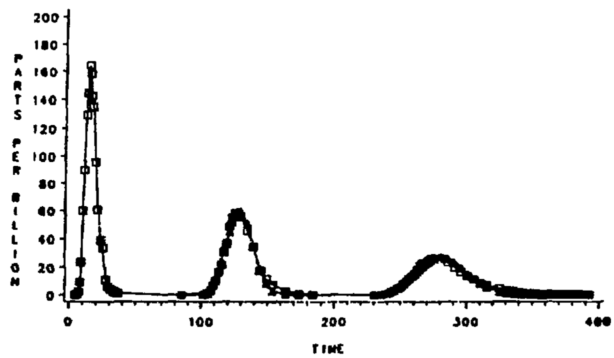

FIGURE 4. Lower Three Runs Stream Dye Tracer Response Curves

SUMMARY

A stream/river emergency model has been developed for use at the Savannah River Plant to predict travel times, maximum concentrations, and concentration distributions as a function of time at selected downstream/river locations from each of the major SRP installations.

ACKNOWLEDGEMENT

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ANS Topical Meeting on Radiological Accidents—
Perspectives and Emergency Planning

Meeting NUREG-0737, II.B.3 Requirements—Backup Analysis of Postaccident Samples

Todd L. Hardt, Mark J. Bradley, and Trudy E. Phillips

ABSTRACT - The Babcock & Wilcox Company has in place a multiparticipant program designed specifically to meet the requirements of NUREG-0737, II.B.3 for offsite analysis of backup, post accident samples. This paper provides a brief review of B&W's experience in analyzing the Three Mile Island (TMI-2) Nuclear Station post accident samples and the subsequent establishment of the Post Accident Sample Analysis Program to meet NUREG-0737 requirements.

Highlighted are the program's "Emergency Protocol" and its contribution to effective emergency planning. The paper also reviews the technical requirements for analyzing and handling post-accident samples. Specifically, the methods for analyzing low-levels of chlorides in a highly borated sample and the problems associated with transporting post-accident samples will be addressed.

INTRODUCTION

NUREG-0737, "Clarification of TMI Action Plan Requirements," II.B.3, "Post-Accident Sampling Capability," Criteria (8) states the following, "... the license shall provide backup sampling through grab samples, and shall demonstrate the capability of analyzing the samples. Established planning for analysis at off-site facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least one sample per day for 7 days following onset of the accident and at least one sample per week until the accident condition no longer exists."

In early 1982, Babcock & Wilcox (B&W) initiated the Post-Accident Sample Analysis Program for backup, offsite analysis of post-accident samples. The program was specifically designed to meet the requirements of NUREG-0737 and benefited from the lessons learned during B&W's handling and analysis of TMI-2's post accident samples. The program consists of three important documents:

Emergency Protocol: A one-page guideline identifying whom to call and the required information for B&W notification of post accident sample shipment.

Project Technical Plan: Describes the state of readiness to be maintained at B&W's Lynchburg Research Center laboratories, response times, and scope of analyses to be performed.

Quality Assurance Plan: Documents compliance with the requirements imposed by 10CFR61, Appendix B.

This paper reviews B&W's experience with analysis of the TMI-2 post-accident samples and the subsequent genesis of the Post-Accident Sample Analysis Program. Discussed in detail are the program's "Emergency Protocol" and backup analysis of post accident samples. Transportation of post accident samples is also discussed.

BACKGROUND

In early April 1979, Babcock & Wilcox was contracted to perform radiochemical analysis on samples taken from the TMI-2 plant following the March 29th accident. The samples were shipped to B&W's Lynchburg Research Center in lead-shielded, 55-gallon drums. The early samples were taken from the plant to the airport by truck and then airlifted to B&W in Lynchburg, Virginia by the National Guard. The later samples were transported by private charter from Harrisburg, Pennsylvania, to Lynchburg, Virginia. Once the sample arrived in Lynchburg, it was transported to the lab by a commercial transport company. Shipment of reactor coolant samples continued on a weekly basis for over 2 1/2 years.

Several valuable lessons were learned during analysis of the initial samples from TMI-2. For example, the fact that a single point of contact did not exist at TMI-2 or B&W

to respond to this kind of emergency created confusion. To address this and other problems experienced following TMI-2 and to meet the requirements of NUREG-0737, B&W's Post-Accident Sample Analysis Program was established in mid-1981 as a multiparticipant program for the Boiling Waste Reactor (BWR) Owners Group. The purpose of the program was to ensure that the capability for backup analysis of post-accident samples was available on an emergency response basis. It was designed specifically to meet the requirements of NUREG-0737, II.B.3 for off-site analysis of post accident samples.

The analysis done on the TMI-2 samples was the minimum required to answer questions of plant condition and stability. Gamma-ray analysis, in particular the identification of cesium and iodine activities, was coupled with strontium analysis to assess the extent of damaged fuel. Boron was measured to verify the shutdown margin. Water chemistry attributes such as pH, conductivity, chlorides and oxygen were used to determine the corrosion potential of the coolant. Analysis of the dissolved gases aided in the estimation of core damage and coolant corrosion potential.

The current program provides the utility emergency planner with an "emergency protocol," which includes a single point of contact protocol for program initiation in the event of an accident. The program provides a 24-hour turnaround confirmatory analysis capability to over one-third of the operating nuclear plants in the United States. The emergency protocol has been exercised many times during plant drills with two utilities testing the complete system of sample shipment, receipt, analysis, and reporting of the results.

EMERGENCY PROTOCOL

Following the TMI-2 accident, samples were transported to the Babcock & Wilcox research center in Lynchburg, Virginia, for analysis. At the time, no mechanism existed for the notification of B&W, or for rapid coordination of resources to analyze the samples. As such, a significant amount of effort was required to gear-up for the initial samples.

In developing the Emergency Protocol (EP) for the Post Accident Sample Analysis Program, the need to make it simple and reliable were considered paramount. Simple meant that the EP could be exercised quickly with little chance of making a mistake. Reliable meant that the EP would accomplish its purpose every time.

The Emergency Protocol is a one-page document. It provides the names and telephone numbers of the emergency contacts; the information required at the time of initiation; and B&W's emergency response activities. For reliability, both a primary and alternate contact are listed, and the work and home phone number for each is provided. The Emergency Protocol has been successfully exercised many times over the past few years as part of plant emergency drills.

BACKUP ANALYSIS OF POST-ACCIDENT SAMPLES

The NRC's "Clarification of TMI Action Plan Requirements," NUREG-0737, specifies the requirements that must be met by licensees of commercial nuclear power reactors for post accident sampling and analysis. Criterion 8 provides the guidelines for using an off-site laboratory for backup sample analyses. All of the post accident analyses have been successfully tested and the required accuracies achieved, as listed in Table 1.

Post accident sample analyses are performed on three sample types -- liquid, gas, and particulate and charcoal filters. Depending on whether the plant is a PWR or BWR, the type of sample to be handled could be an atmospheric gas or liquid or a pressurized liquid. The gas phase of pressurized samples is expanded and flushed from the liquid by means of an argon sparge into an evacuated expansion vial. Gas samples are removed by a syringe for analysis by gas chromatography and gamma-ray spectroscopy. The liquid phase is simply drained from the sample container into a bottle for subsequent analyses.

To perform these analyses within a 24-hour period, the B&W Lynchburg Research Center laboratory is kept in a state of readiness at all times. Reagents are kept fresh and all analytical instruments are maintained in a ready and calibrated state. Therefore, in the event of an accident or a practice drill, a complete set of analyses on a post accident sample (except for strontium 89 and 90) can be performed within 24-hours of receipt of the sample.

The NRC clarification letter for NUREG-0737 requires that laboratory procedures for the analysis of post-accident reactor

TABLE 1. Post-Accident Sample Analyses

Analysis	Range	Accuracy	Method
Gamma Activity	50-2000 keV	20%	Spectroscopy
Sr-89,-90	0-1000 uCi/cc	25%	Radiochemical
Boron	0.1-100 ppm	5%	I.C.
	100-10,000 ppm	10%	Titration
Chloride	>500 ppb	10%	I.C.
	<500 ppb	50 ppb	I.C.
Conductivity	1-10,000 uohms	10%	Cond. Bridge
H, O, Kr	> 0.5%	5%	GC
	< 0.5%	20%	GC
pH	1-13	0.3 units	Electrode

coolant samples be tested with a simulated post accident sample. The B&W procedures have been successfully tested for the analysis of chloride and boron in the prescribed test matrix. Both analyses used the method of ion chromatography and were shown to meet or exceed the specifications of the guidelines.

TRANSPORTATION OF POST-ACCIDENT SAMPLES

An undiluted liquid, post accident grab sample drawn for off-site analysis will typically be 10 to 30 mls and have an activity as high as 1 curie/ml. The sample may be pressurized or unpressurized and be in anything from a glass vial to a shielded, steel bomb. The transportation of these highly radioactive liquid and gaseous samples is difficult.

There are two methods for shipping post-accident samples -- Type A containers and Type B containers.

- Type A - Containers are designed to retain the integrity of containment and shielding under normal transport conditions.
- Type B - Containers are designed to retain the integrity of containment and shielding under normal transport conditions and hypothetical accident test conditions.

Type B packaging is more restrictive and can thus carry higher activity samples. The primary regulations that govern the fabrication, testing, licensing and use of Type A and B containers are:

- 10CFR71 - Nuclear Regulatory Commission regulations on transportation of radioactive material.
- 49CFR173 - Department of Transportation regulations on transportation of hazardous materials.

A typical post accident sample will require a Type B container for shipment immediately following an accident. However, after a short decay period (16 hrs to 7 days), these samples can be shipped as Type A. Thus, the two options for transporting a post accident sample are:

1. Type B Shipment: The post accident sample is drawn and then shipped immediately via truck or rail to the off-site laboratory for analysis.
2. Type A Shipment: The post accident sample is drawn and then stored for deacy. When the sample reaches Type A, levels, they are transported by air to the offsite laboratory.

The decision to use a Type A or Type B cask is a function of many factors. Two key factors are the technical adequacy of the transport method selected and the 'up-front' cost. The up-front cost is a function of the cost of the container, and the lead pigs needed to hold samples while the containers are in use. Typically, a Type A cask will cost much less than a Type B. As a minimum, two casks should be available. This is a cost of $150,000 to $300,000 for two Type B's versus $20,000 to $40,000 for two Type A's.

For ALARA purposes, a post accident sample should be drawn, stored, and transported in the same lead pig. A Type B container can take up to nine days for a round trip between the off-site laboratory and the plant. Thus, even with two Type B casks, up to seven lead pigs will be needed to handle the first week's samples. The number of lead pigs needed the Type A transportation scenario will depend on the roundtrip flight time (one day or less) and the sample decay time. For a BWR, 10-ml, depressurized sample, the decay time to Type A is approximately 16 hours. For a PWR, 15-ml, pressurized sample, the decay time to Type A is approximately seven days. Thus, use of a Type A cask will require no more, and probably fewer lead pigs for drawing and storing samples than for a Type B.

In general, using a Type A cask for transport or post accident samples will significantly reduce up-front costs. Keep in mind, following the TMI-2 accident, the initial samples were flown to Babcock & Wilcox's Lynchburg Research Center by Air National Guard and the subsequent samples were flown in Type A containers using a private carrier service.

EXPERIENCE

During the four years that this program has been in effect, the entire system has been exercised one time. During this drill, unexpected problems arose concerning the sample shipment and the actual sample bomb used to hold the sample. A power station notified B&W through the emergency protocol that a post-accident practice sample was being shipped to Lynchburg by air for analysis. The sample had not been received by noon the following day as expected and a tracer located it at the airport in Richmond. Due to its weight, the sample could not be transported on small plane, and this, subsequently arrived by truck. This drill pointed out the need for each utility to define the most efficient way to transport the sample to its backup laboratory. A follow up sample was shipped a month later by truck with no delays.

An additional problem surfaced when the sample bomb was removed from the shipping cask. We found that the sample container was too large to fit onto the B&W sampling rig. Several hours were spent replacing fittings on the container and constructing the tubing to be used so that the smallest possible volume would be introduced to the expansion side of the system. This pointed out the need for either a standardized sampling bomb or each utility to inform its backup lab of the correct dimensions of the sample container.

Despite the problems encountered, we were able to complete all analyses and report results to the utility within the prescribed 24 hours. The single point of contact emergency protocol worked very smoothly during this drill as it had in other, less-extensive drills.

SUMMARY

The Babcock & Wilcox Post-Accident Sample Analysis Program successfully fulfills the requirements of NUREG-0737, II.B.3 for off-site analysis of post-accident samples. Key to that success is the emergency protocol for program initiation and the ability of the laboratory to analyze post-accident samples to the required precision and accuracies.

When initiating similar programs for meeting regulatory requirements, important factors to consider are:

1. The emergency protocol needs to be both simple and reliable.

2. Use vendors and multiparticipant programs to meet regulatory requirements. The ideal situation is to be able to have the vendor addressing regulatory questions on the program rather than each participant on an individual basis.

An Evaluation of Emergency Systems for the Monitoring, Sampling, and Analysis of Reactor Coolant, Containment Atmosphere, and Airborne Effluents

Andrew P. Hull, Wayne H. Knox, and John R. White

ABSTRACT

A postimplementation review has been made in NRC Region I of the postaccident sampling systems (PASS), the gaseous effluent monitors, and provisions for sampling effluent particulates and radioiodines, which were required by the NRC subsequent to the TMI-2 accident (NUREG-0737). Prefabricated PASSs were predominant. Problems included inadequate purge times, separation of dissolved gases, excessive dilution, and the accuracy of analytical techniques in the presence of interferences. Microprocessor-controlled high-range gas monitors with integral provisions for sampling particulates and radioiodines in high concentrations were widely used. Calibration information was generally insufficient for the unambiguous conversion of monitor reading to release rates of a varying postaccident mixture of radiogases. The referenced sampling guidance (ANSI-N 13.1-1969) was inappropriate for the long sampling lines customarily used. Generic research is needed to establish the behavior of particulates and radioiodines in these lines.

I. INTRODUCTION

The accident at Unit-2 of the Three Mile Island Nuclear Power Station (TMI) on March 28, 1979 disclosed numerous deficiencies in the installed system for the collection and analysis of primary coolant and containment atmosphere samples under post-accident conditions. This system was typical of those then installed at U.S. Nuclear Power Stations. The accident also disclosed limitations in the capability of its gaseous monitors and the adequacy of sampling systems to deal with the concentrations of airborne effluents that might be anticipated under post-accident conditions. They were also typical of those being employed at the time of the accident.

Subsequently, the U.S. Nuclear Regulatory Commission (NRC) drew up a number of short-term recommendations which were based on the lessons learned from the accident and which were described in the report NUREG-0578.[1] They included measures for the improvement of post-accident sampling capability and for the expanded range of radiation monitors. These recommendations were developed into specific tasks in report NUREG-0660[2] and finalized for implementation in a clarification, NUREG-0737[3]. Its specific requirement for Post Accident Sampling Capability are set forth in Item II.B.3.

Those for High-Range Noble Gas Effluent Monitors are set forth in Item II.F.1, Attachment 1, and those for the sampling and analysis or Measurement of High-Range Radioiodine and Particulate Effluents in Gaseous Streams are set forth in Item II.F.1, Attachment 2.

An implementation deadline of January 1, 1982 was specified. It was also indicated in NUREG-0737 that these systems would be subject to a post-implementation review. Responsibility for their post-implementation review was assigned by the NRC Office of Inspection and Enforcement to the NRC's regional offices. In mid-1983, Region I contracted with the Safety and Environmental Protection Division of Brookhaven National Laboratory for technical assistance in their performance. Each has required the identification and documentation of the licensee's commitments, clarifications, schedules and orders. A subsequent on-site inspection has included the physical verification and validation of the installation and operability of equipment, as well as the verification of the adequacy of the licensee's procedures and of the qualification and training of licensee's personnel.

Starting in late 1983, on-site reviews have been completed at the rate of about one per month for the twenty operating licensee sites in Region I, which currently contain a total of twenty-five operating reactors. They are located in four of the New England states: New York, New Jersey, Pennsylvania and Maryland.

II. APPROACH

Before the on-site reviews commenced, the individual elements that should be included in the overall review effort were considered in a Management Oversight and Readiness Tree (MORT). Following this, a specific set of instructions and/or questions related to each review component was prepared. These included such subcategories as design, monitoring system, procedures, structures, hardware and support services, readout and recording, personnel and training.

III. FINDINGS

A. POST ACCIDENT SAMPLING SYSTEM (II.B.3)

As indicated in NUREG-0578, the purpose behind the requirement for the improved post-accident sampling capability was the prompt provision of information for the assessment and control of the course of an accident. In particular, the II.B.3 required chemical and radiological analyses are intended to provide information for the assessment of core damage and coolant characteristics. The required analyses of containment atmosphere are intended to establish the presence and concentration of hydrogen, as well as to provide information for core damage assessment.

The principles of core damage assessment are based on the grouping of fission products according to their volatility. Thus, the fraction of each group that would be released would depend on the temperature and fuel fragmentation during an accident. An extended discussion of this subject was presented in the Rogevin Report [4], from which the grouping shown in Table I is excerpted. An inventory of the major fission products in a reference 3651 Mw(t) BWR which has been in operation for three years, arranged according to release groups, is shown in Table II.

In very broad terms, an accident in which only noble gases were released would be indicative of fuel cladding failure, one in which the volatile nuclides of I and Cs were present would be indicative of high fuel temperatures and one in which the non-volatiles were also present. In a more complex situation, the core damage would be determined by a set of simultaneous equations which take into account the ratios of the release groups.

Table I

Categories of Fission Product Releases in Order of Decreasing Volatility

I	Noble Gases (Kr, Xe)
II	Halogens (I, Br)
III	Alkali metals (Cs, Rb)
IV	Tellurium (Te)
V	Alkaline earths (Sr, Ba)
VI	Noble metals (Ru, Rh, Pd, Mo, Tc)
VII	Rare earths and actinides
VIII	Refractory oxides of Zr and Nb

The licensees of the twenty operating power reactors in NRC Region I have installed a variety of systems to meet the requirements of NUREG-0737, Item II.B.3. As shown in a summary in Table III, they range from relatively simple licensee designed systems which are intended solely to obtain samples of reactor coolant and of the containment atmosphere for subsequent laboratory analysis, to elaborate vendor or architect engineer designed systems which are intended to perform all of the required analyses in line, with the laboratory only as a back-up.

Table II

Inventory of Major Fission Products in a Reference Plant Operated at 3651 MWt for Three Years

	Group-Rogevin Report	Isotope*	Half-Life	Inventory** 10^6 Ci
Noble gases	I	Kr-85m	4.48h	24.6
		Kr-85	10.72y	1.1
		Kr-87	76.3m	47.1
		Kr-88	2.84h	66.8
		Xe-133	5.25d	202.0
		Xe-135	9.11h	26.1
Halogens	II	I-131	8.04d	96.0
		I-132	2.3h	140
		I-133	20.8h	201
		I-134	52.6m	221
		I-135	6.61h	189
Alkali Metals	III	Cs-134	2.06y	19.6
		Cs-137	30.17y	12.1
		Cs-138	32.2m	178.0
Tellurium Group	IV	Te-132	78.2h	138
Alkaline Earths	V	Sr-91	9.5h	115
		Sr-92	2.71h	123
		Ba-140	12.8d	173
Noble Metals	VI	Mo-99	66.02h	183
		Ru-103	39.4d	155
Rare Earths	VII	Y-92	3.54h	124
		La-140	40.2h	184
		Ce-141	32.5d	161
		Ce-144	284.3d	129
Refractories	VIII	Zr-95	64.0d	161
		Zr-97	16.9h	166

* *Only the representative isotopes which have relatively large inventory and considered to be easy to measure are listed here.*

** At the time of reactor shutdown.

None of the reviewed PASS Systems were adjudged perfect in every respect. However, of the seventeen which could be fully tested at the time of the review, all met the basic requirements of Item II.B.3. Of the three that could not, one was inoperative, one had an improperly installed valve which made it impossible to obtain a sample of the containment atmosphere and one could not conduct a test of the containment atmosphere sampling system due to its Technical Specifications, which did not permit the opening of the valves which maintain containment isolation during operation.

The representativeness of the PASS samples reactor coolant and the licensee's radiological analytical capability were tested by making a comparison of the results of their analysis with those from the plant's normal sample sink. The accuracy of the licensee's chemical analytical capability was tested by the use of standards of known content. With one exception, the systems which were designed to perform all or most of the required analyses in-line demonstrated the greatest operational readiness and ability to provide prompt data. This was especially evident for the one system that was used to analyze routine samples. The exception required a long startup, so that the required sampling and analysis with it could not be completed within the stipulated three hours.

Table III

Summary of Installed Post-Accident Sampling Systems

Sample Collection (no on-line analysis capability)

Design	No.	Remarks
Licensee	5	
G.E.	7	

Sample collection (limited on-line analysis capability)

Design	No.	Remarks
General Dynamics	3	In-line-pH, Cond.
Quadrex	1	In-line-pH, B, Cl
Sentry	1	In-line-pH, Cond., DO, Dis H
	1	In-line-pH, Cond, DO, Dis H, Cl
Stone & Webster	1	In-line-pH, B, Cl

Full in-line analysis capability (including isotopic)

Design	No.	Remarks
Sentry	1	
Combustion Eng.	1	

A schematic of the coolant sampling portion of a relatively simple PASS is shown in Figure 1 and that for containment atmosphere sampling is shown in Figure 2. The principal features of the associated control panel are shown in Figure 3. The panel also contains a mimic diagram of the system, with pilot lights to indicate that the intended steps (i.e. value close up opening) have occurred. It is obvious that even this relatively simple system is in fact quite complex, so the detailed and lengthy procedures are required to guide the PASS operator through the sequence of steps necessary to obtain the desired samples.

The principal deficiencies that were identified during the review of the PASS systems are summarized in Table IV. It should be noted that in many instances the findings of inadequacy of surveillance was made on the basis of the lack of a suitable schedule and/or the excessive time interval required to get a system fully back on line after a fault had been identified by the licensee. Although most purge times seemed adequate, in many instances the licensee had not conclusively established this by making calculations of the volume of the line(s) to be purged. Midway in the review, the Office of Inspection and Enforcement indicated that the containment atmosphere sample was primarily for fission gas measurements, so that sample line losses should not considered a significant factor unless the licensee intended to use measurements of airborne radioiodines in the containment atmosphere in the assessment of core damage.[10] This in the view of the authors'

Table IV

Principal Deficiencies Identified in Review of Post-Accident Sampling Systems

Frequency	Deficiency
12	Inadequate surveillance and maintenance
11	Assurance from study that purge times adequate
10	Assurance of representativeness of radioiodines in containment air sample
10	Adequacy of documentation that shielding in sample room and/or of sample during transport sufficient to enable operation within GDC-19 criteria.
7	Procedures did not call for proper pressure and/or temperature corrections
7	Procedures inadequate or in need of revision to conform to actual operation of PASS
5	Excessive dilution of sample prior to screening for actual level of activity
5	Liquid in stripped gas
4	Flow not assured in the absence of a flow meter
4	Insufficient or no backup for one or more in-line analyses
3	Inadequate test of all features of system by licensee prior to on-site review
3	Inadequate training or insufficient number of trained personnel to assure ability to operate system during post-accident conditions.
3	Inadequate assurance that sample could be obtained when reactor depressurized (no pump in PASS).
3	Needle bent during attempt to perforate septum of sample collection vial (GE design system)
3	Improper interpretation of flow produced by critical orifice (or of pressure required to maintain design flow)
3	Unsuitable cask/shield vial for sample transport
3	Volume delivered by ball valve (for dilution) not established by actual measurement (GE design system)
3	Chemical analysis procedure not adequately tested for possible interferences

Figure 1

PASS LIQUID SAMPLING SIMPLIFIED P&ID

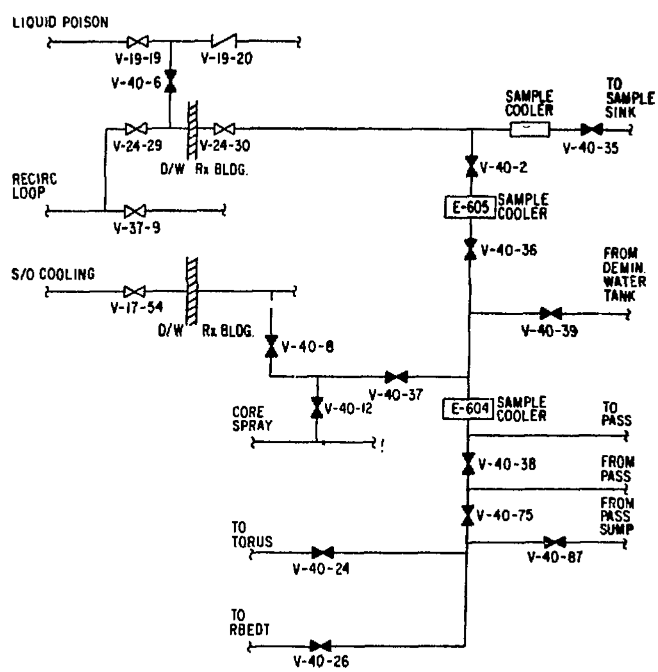


Figure 2

CONTAINMENT ATMOSPHERE PASS SAMPLING DIAGRAM
SIMPLIFIED P & ID

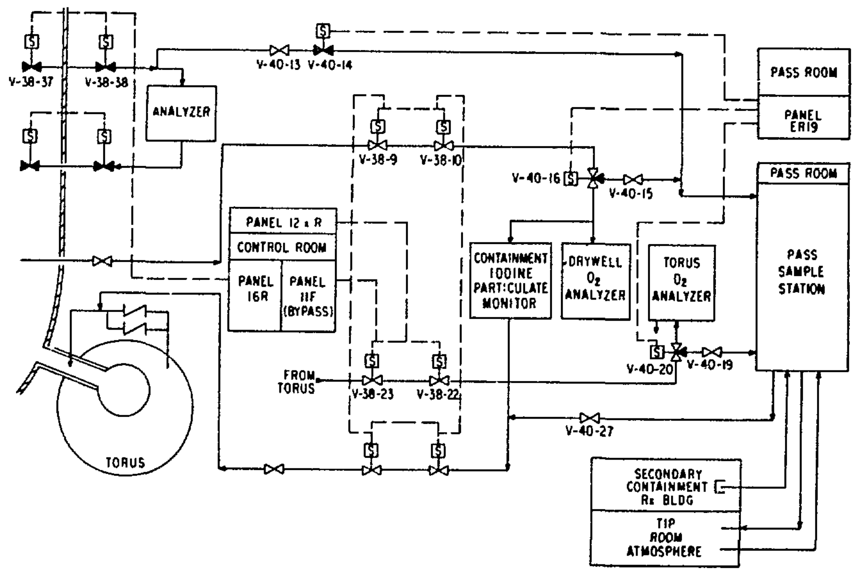
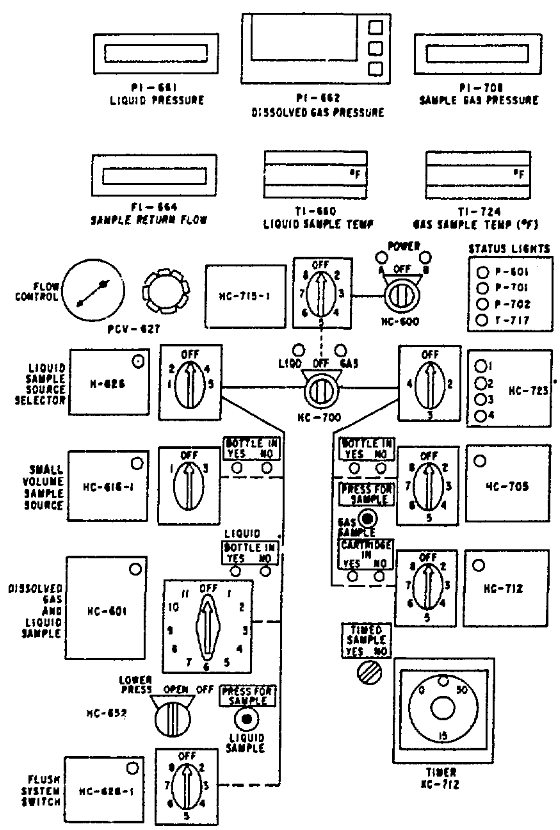


Figure 3

GE PASS CONTROL PANEL



downplays the value of this measurement for the establishment of the potential source term if the containment should leak or fail outright.

In most cases, the shielding provided for PASS systems and sample transport appeared adequate, but many licensees had not conducted a formal study to establish that the GDC-19 criteria (5 rem whole body, 5 rem extremity dose) could be met.

The balance of the listed deficiencies and the measures necessary to address them should be self-explanatory.

B. HIGH-RANGE NOBLE GAS MONITORS

A summary of the installed high-range noble gas monitors, according to their location (on-line or off-line), type of detector, and vendor is shown in Table V. It is evident that the Region I licensee have chosen a variety of approaches to comply with the requirements of Item II.F.1-1. The typical Boiling Water Reactor (BWR) contained either one monitored release point under accident conditions, the unit vent, or a second monitored release point for the standby gas treatment system. The Pressurized Water Reactors (PWR) were more variable, with from one monitored unit vent and a main steam line monitor to three monitored vents and a steam relief monitor.

Two licensees installed on-line monitors, using ion chambers which were located in or immediately adjacent to stacks or ducts, while seventeen installed off-line monitors. Of the latter, six installed "gas only" high-range monitors as additions to their pre-existing low-range monitors. A schematic of such a monitor

TABLE V

SUMMARY OF INSTALLED MID- AND HIGH-RANGE NOBLE GAS MONITORS

No.	Range	Detector	Vendor	Model	Operating Mode	Data Processor	Background Subtraction
On-Line							
2	Mid/High	Ion Chamber	(1) GA (1) Victoreen	RD-2A 847	Continuous	No	No
Off-Line							
Gas Only							
1	Mid/High	Plastic	NMC	GA-270	High Alarm	No	No
1	Mid High	GM Ion Chamber	Victoreen Victoreen	 847	Continuous	No	No
3	Mid/High	Ion Chamber	Victoreen	847	Continuous	No	No
1	Mid/High	Ion Chamber	Reuter-Stokes	C4-2510-101	High alarm	No	No
Integrated Gas Monitors and Particulate-Iodine Samplers							
5	Mid High	Cd-Te Cd-Te	GA	WRGM	High Alarm	Yes	No
3	Mid High	GM GM	Eberline	SPING-4	Continuous	Yes	Yes
2	Mid High	GM GM	Kaman	KGM-HRH	High Alarm	Yes	No
1	Mid/High	Ge-Li	SAI	RAGEMS	Continuous	Yes	NA
1	Mid High	GM GM	Eberline	AXM-1	High Alarm	Yes	Yes

which utilizes an ion chamber, is shown in Figure 4. Twelve licensees installed commercially available integrated monitors with modules for both monitoring and sampling. A view of a typical one (the Kaman KDGM-HR) is shown in Figure 5.

These installations have also incorporated a variety of approaches to the problem of achieving the required full-range sensitivity. Typically, three overlapping-range detectors have been provided, as shown in Figure 6 (for the Genral Atomics WRGM). In order to achieve the upper limit of 10^5 uCi/cm^3 (^{133}Xe equivalent), most of these monitors are designed so that their high-range detectors view a limited volume of gas, as compared to that viewed by their mid- or low-range detectors. An example, for the enhanced high-range detector of the Kaman HRH, is shown in Figure 7.

Although Item II.F.1-1 was not specific on the calibration of noble gas monitors up to the required upper range, the NRC has provided some guidance. It recognized the problem of the availability of suitable noble gases, i.e. ^{133}Xe in sufficient concentrations and of their utilization by licensees if they were available. Therefore, the Staff recommended that a one-time "type" calibration in the laboratory over the intended range be performed and that the transfer procedure of ANSI N323-1978 be utilized in conjunction with solid sources at appropriate energies for on-site calibrations.[11]

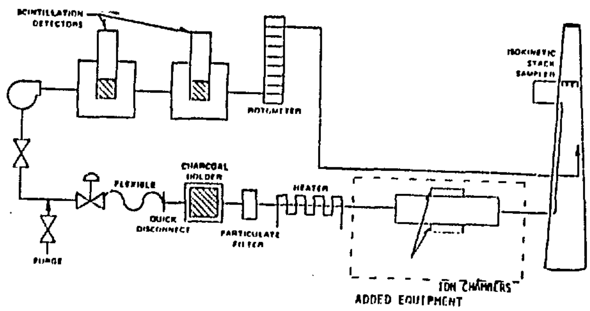


Figure 4. High-range effluent process radiation monitors

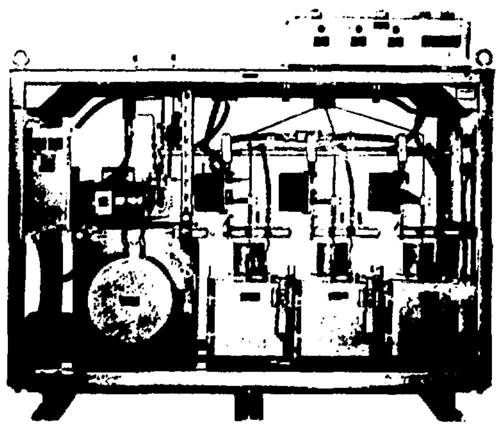


Figure 5. Kaman HRH high-range noble gas monitor and sampler

As suggested by Table VI, most of the vendors appear to have performed only a "one point" primary calibration, utilizing ^{133}Xe and ^{85}Kr. They have then performed a number of transfer calibrations with solid sources with a range of activities and energies, to establish the energy response and/or range capability of a given detector.

A summary of the sampling arrangements which have been provided to achieve compliance with Item II.F.1-2 and which have been reviewed to date is shown in Table VII. Again, a variety of approaches is evident. Some licensees (including the five who have utilized "gas only" monitors to comply with Item II.F.1-1) installed independent sampling facilities. One licensee wrote emergency sampling procedures with incorporated pre-existing unshielded collector for routine sampling. Five added additional shielded particulate and iodine sample positions which were connected to an existing low-range sample line, while one added a pre-fabricated multiple sample-position module.

Eleven licensees have installed integrated monitor/samplers which contain micro-processor modules that provide for the automatic or remote collection of a sample at one of three individual sample positions, as is also shown in Figure 8. Another licensee located its integrated unit in what would become a high-radiation field during post-accident conditions, so elected to create another more remote sampling station. These integrated monitor samplers typically provide for a much reduced flow of a few hundred cm^3/min, as compared to the 1-2 cfm flow that is typically provided for low- and mid-range sampling. The intent is to thereby limit the total amount of activity that would be collected at concentrations which approach the upper design criterion of 100 uCi/cm^3 for the stipulated 30-minute sampling period.

IV. LESSONS LEARNED

A. HIGH-RANGE NOBLE GAS MONITORS

Oversimplifications in the conversion of the direct indications of the installed gas monitor, typically in cpm or mR/hr, to effluent concentrations and/or rates of release were among the principal shortcomings encountered in the reviews.

The guidance in NUREG-0737, II.F.1-1 states "Design range values may be expressed in ^{133}Xe equivalent values for monitors employing gamma radiation detectors" (as most do). This concept has not been generally understood or employed by vendors or by the reviewed licensees. In some instances, they have employed uninterpreted actual calibration data for ^{133}Xe or ^{85}Kr to establish detector response, without a recognition of their limitations. The former emits low energy photons, with a mean energy of 0.045 MeV per disintegration. Thus, they may be significantly absorbed in the housing or walls of a detector. In contrast, ^{85}Kr is principally a beta emitter, with accompanying bremstrahlung gamma radiations and a 0.51 MeV photon with a yield of only 0.4%. This is apparent from Figure 8, which illustrates the direct response with distance of Eberline's high-range detector to each of these nuclides. When corrected respectively for absorption and bremstrahlung, the true energy response of this detector is about midway between the two curves, so using one point from either could lead to a factor of two error.

Beyond this, these uninterpreted calibration data were in some instances also employed to calculate release rates (in uCi/sec), without regard to the variable energy response characteristic of the detector. This characteristic may be close to linear with energy, as shown in Figure 9, for the Kaman KDGM-HR, or may be quite non-linear as shown in Figure 10, for the General Atomics WRGM. Beyond the inherent response of the detector itself, its energy response may also be dependent on the geometry in which it is installed and the type and thickness of the intervening duct or pipe walls which may absorb radiations before they reach the detector.

All of the reviewed licensees have installed monitors which in principle met the upper range criterion of 10^5 uCi/cm^3. However, only two had calibrated the installed high-range monitors on-site with radiogases in concentrations approaching 10^5 uCi/cm^3. The vendor calibration information supplied by Kaman, as shown in Figure 11, suggested that a test with actual radiogases approaching these concentrations had been performed with ^{133}Xe. However, on the basis of field testing which employed ^{85}Kr it was found by another investigator that this monitor could not meet the specified upper range.[12] It is our understanding that since these tests, the Kaman high-range detector has been modified so that it can do so. A similar fall-off which appeared to be due to a large dead-time at high concentrations was reported by a consultant to a Region I licensee in a field calibration of the high-range detector (SA-9) of the Eberline SPING.[13]

TABLE VI

CONCENTRATIONS FOR VENDOR CALIBRATIONS OF II F.1-1 HIGH RANGE MONITORS

	^{133}Xe Concentrations uCi/cm^3	^{85}Kr Concentrations uCi/cm^3
Eberline		
Mid-Range SPING NGD-1 (SA-13)	0.13	0.47
High-Range SPING AXM-1(SA-14)	0.26	1.47
SA-15, SA-9	1.75	9.98
General Atomics		
Mid/High Range-WRCM	0.65	11.1*
Kaman		
High-Range-HRH	$5x10^4$	$1.5x10^5$*

* Based on calibration data supplied by vendor, as inferred for NBS Reference Data.

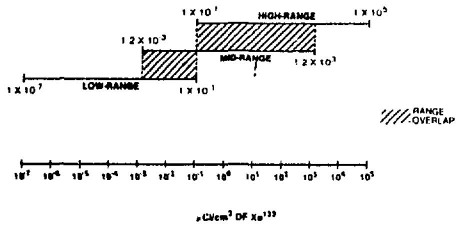


Figure 6. Ranges of General Atomics wide-range gas monitor.

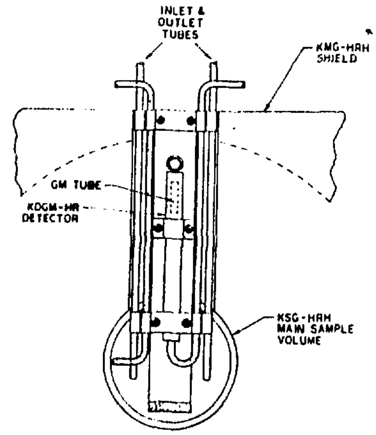


Figure 7. KMG-HRH-Enhanced high-range geometry.

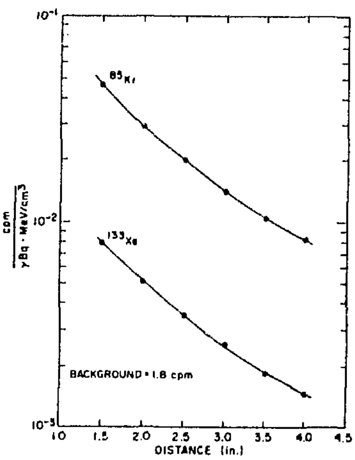


Figure 8. Response of Eberline SA-9 high-range detector to ^{85}Kr and ^{133}Xe.

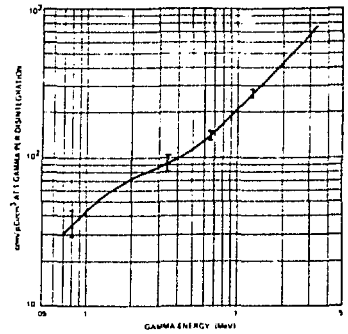


Figure 9. KDGM-HR enhanced detector in KSG-HRH sampler, enhanced high-range position energy dependence characteristic.

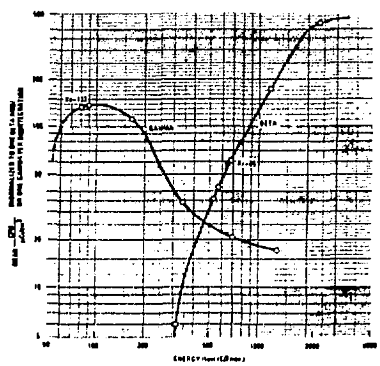


Figure 10. General Atomics wide-range gas monitor RD-72 high-range detector energy response curve.

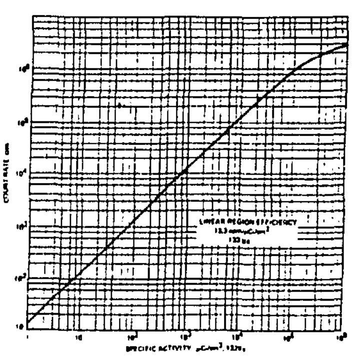


Figure 11. HDGM-HR Enhanced detector in KSG-HRH enhanced high-range position efficiency to Xenon-133.

TABLE VII

SAMPLING AND ANALYSIS OF PLANT EFFLUENTS, II.F.1-2

No.	Range	Vendor	Model	Sample Positions	Shielded	Filter Selection	Remarks
Independent Utility Design							
5	-	-	-	1	Yes		(In each instance)
Vendor Design							
1	-	NRC In .	MAP-5	3	Yes	Local/remote control	Timed sample
1	-	Kaman	HRH	1	Yes		
Integrated Units							
4	Mid/High	GA	WRGM	3	Yes	Local/remote control	Timed sample
3	Mid/High	Eberline	SPING-4	1	No	Fixed	
2	Mid/High	Kaman	KGM-HRH	3	Yes	Automatic (GM Monitor)	Automatically timed sample
3	All	SAI	RAGEMS	1*	Yes	Automatic	Note 1
1	Mid/High	Eberline	AXM-1	1	Yes	Fixed (GM Monitor)	

Note: One licensee has installed this system, but does not utilize its Ge-Li detection feature.

Some licensees have recognized the variable energy response of high-range monitors by the provision of corrections in their software for making off-site dose assessments. However, this does not provide guidance for a reactor operator or supervisor who may have to make manual calculations of effluent release rates before skilled post-accident dose assessors are likely to be available.

As indicated in Table VI, three licensees selected the Eberline SPING-4 as a high-range monitor for effluent noble gasses. During the reviews, it was ascertained that the micro-processor of this monitor is not radiation hardened, thus making it doubtful that it would operate reliably in high-radiation fields. However, in one case the monitor was supplemented by the Eberline SA-10 and SA-9 mid- or high-range detectors, for which the sensitive components are remotely located. When the SPING-4 component of this unit senses high radiation fields, it is isolated from the sample stream, thus increasing its reliability of function throughout an accident sequence.

In several instances, licensees with installed micro-processor controlled high-range gas monitors were found to have a limited number of plant personnel with sufficient training to be able to retrieve data beyond that routinely displayed. The review also revealed that several of these monitors had experienced frequent and/or extended down time of their automatic features, due to the failure of their flow sensors which appear to be sensitive to entrained dust particles and which therefore call for frequent preventative maintenance.

Except for those with installed integrated units which function automatically, provision and/or procedures had not been incorporated by many licensees for the isolation and/or purging of their low-level gas monitors, should their range be exceeded. Thus their recovery and availability would be doubtful following an accident as effluent concentrations declined to within the low-range region.

C. SAMPLING AND ANALYSIS OF PLANT EFFLUENTS

The principal deficiency encountered in the review of arrangements for the sampling of radioiodines and particulates was the inability of licensees to document that their sampling systems could collect representative samples. This is particularly so for those with long sampling lines, in which considerable deposition losses of elemental radioiodines could occur even when installed in accordance with the design guidance of ANSI N13.1-1969.

The transmission of elemental iodines through long sampling lines has been measured under controlled conditions in the laboratory by Unrein et al.[14] Their studies suggest that it depends upon the relative rates of deposition and resuspension from their walls. Transmission factors greater than 50% were found for 1" sampling lines at flow rates of 2-3 cfm, for injection periods of several hours. However, these studies did not indicate how long it took to reach equilibrium between deposition and resuspension after an initial injection. Only a small fraction (<1%) of the injected elemental iodine was transmitted through the 1/4" sampling line with a 0.06 cfm flow rate as utilized in the General Atomics WRGM, which is shown schematically in Figure 12.

The NRC's proposed guidance suggests that the closest approximation to representativeness may be achieved at equilibrium, when deposition and reentrainment or re-suspension are equal. This could be expected to occur most rapidly in a continuously operated system, rather than one in which flow is initiated only upon the occurrence of higher-range concentrations. The Kaman and the Eberline AXM-1 monitors approximate this in that, upon an indication of abnormal gas concentrations, they isokinetically obtain a small local side-stream flow (of a few hundred cm^3/min) from the low-range monitoring/sampling line, in which a much greater flow (1-2 cfm) is maintained.

From the reviews, it was apparent that most architect/engineers and licensees have been aware of the need for the heat tracing of sampling lines when they are exposed to "outdoor" conditions. However, it was also apparent that many of them have not recognized a similar need for the heat tracing of long indoor horizontal sampling lines in which condensation could occur, especially under the high moisture loads of some accident sequences. In a few reviews condensation was found in the sampling medium of sampling positions.

Although II.F.1-2 calls for continuous sampling, the procedures of five licensees called only for the analysis of a grab sample to be collected post-accident over a short period of time (to limit the amount collected to the capability of their laboratory Ge-Li analysis systems), with no indication in their procedures of how they would evaluate the preceding sample to establish the total amount released from the onset of accident conditions.

In several instances, which included the three SPING-4s, the three SAI RAGEMS and one licensee devised installation, the filter assembly for the collection of particulates and iodines was either unshielded or inadequately shielded. None had conducted an analysis to assure that with such an arrangement, the samples could be collected, retained and transported within the GDC-19 dose limits (5 rem whole body and 75 rem to the extremities). It should be noted that by two successive 1/200 dilutions, the RAGEMS should collect only relatively low activity samples under all accident conditions.

All of the licensees had Ag-Zeolite collection media available for sampling under accident conditions. Almost all of the installations provided for isokinetic sampling at normal stack flow rates, but only a few could maintain it if large deviations from these flows were to occur under accident conditions. Of those that could not, none had developed correction factors, as called for in Item II.F.1-2.

Only a few licensees had developed adequate procedures for the analysis of "hot" samples, in which the collected activity might considerably exceed the upper limit which could be analyzed by their GeLi counting and analysis systems. Although several had established procedures for counting samples with greater than normal activity in a geometry distant from the detector, only a few would be able to cope with samples approaching the 85-170 Ci of radioiodines which would be collected at a concentration of 100 uCi/cm^3 at normal flow rates of 1-2 cfm for the stipulated 30-minute sampling period.

VI. COMMENTS AND RECOMMENDATIONS

Except for the GE designed PASS, which was basically the same except for variations in sample line arrangements and the associated valves at individual facilities, a wide variety in PASS systems were encountered in the reviews which have been conducted over the past two years. Many required frequent and considerable attention to keep them fully operational and all required frequent retraining to maintain operator proficiency with their controls and their detailed operating procedures.

The example of the one in-line system that is also used for routine sampling suggests that the readiness and availability of the other systems could be enhanced if they were too were also periodically used for routine sampling, in between the infrequent occasions they are utilized for exercises or mandatory retraining.

A wide variety of approaches to the monitoring of noble gases and the sampling of particulates and radioiodines in high concentrations have also been encountered in the reviews.

If the monitoring requirements were solely those for the noble gases, ion chambers would seem the most straightforward detectors, in view of their simplicity, wide range capability, and linear energy response characteristics. However, they are relatively insensitive and therefore require a large volume of contained gas which is difficult to shield from extraneous radiations. An example of one such installation is shown in Figure 13. The 0.1" -thick steel wall in which the detector was housed would have a large absorption for low energy photons, such as those from ^{133}Xe, compared to a much smaller absorption of the higher energy photons from shorter-lived noble gases.

The integrated monitoring/sampling devices which incorporate microprocessor data handling and control accomplish the full range requirements of Item II.F-1.1 by routing the flow to more then one detector, each of which is designed to be sensitive to portions of the full range requirement. This permits the isolation of the low-range detector during periods of high concentrations. It also facilitates the routing of flow to a selected shielded filter assembly at the same time. Their ability to store and to provide a history of release rates over time makes them attractive for both routine and accident monitoring. Additionally, the use of a monitor for every-day purposes adds to their reliability for accident monitoring. If not so utilized, they require regular surveillance and maintenance to assure their availability.

Much of the confusion over the use of the ^{133}Xe equivalent concept in the calibration of high-range noble gas monitors could be elminated by the adoption of the "Ci-Mev" concept as described by Mourad.[15] A simplified version of the same concept, which utilizes the average noble gas energy as a function of time post-shut down as shown in Figure 14 in dose calculations, was described by Lahti at the 1985 Annual Meeting of the Health Physics Society.[16]

To minimize the ambient post-accident radiation fields, most of the post-accident monitors and/or samplers are located at considerable distances from the points of effluent release, thus necessitating long sampling lines (typically 1" x 100-250'). This creates a dilemma between the desirability of maintaining a high flow rate in the sample line so as to minimize deposition losses and the desirability of minimizing the amount of collected radioactivity on

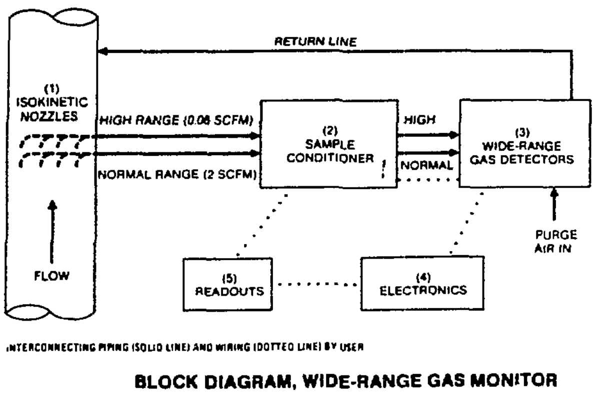


Figure 12. Block diagram, wide-range gas monitor.

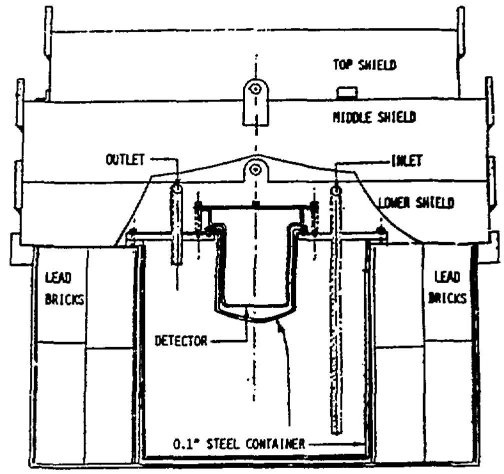


Figure 13. High-range noble gas monitor.

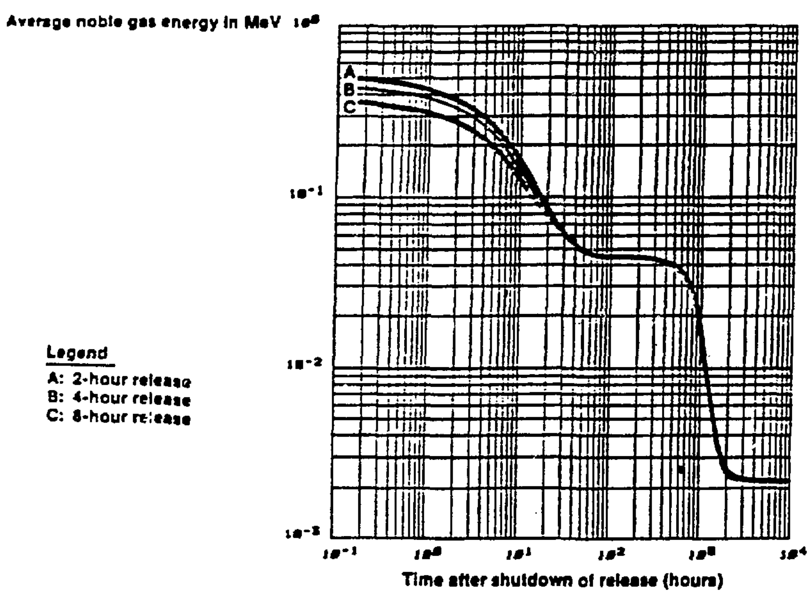


Figure 14. Average Noble Gas Energies of Total Releases.

the sampler. It is solved in some monitors, by the provision of a second stage of isokinetic sampling with a probe situated within the high-flow line close to the sampling head, but with a much small flow (a few hundred cm^3/min) through the high-concentration sampler. This seems desirable on the grounds of both convenience in handling and analysis and of ALARA considerations.

ACKNOWLEDGEMENTS

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Appreciation is also appropriate for the typists, Marie Cooney and Cheryll Christie who prepared the text of many revisions of the reports and of this manuscript.

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A State-of-the-Art Approach to Emergency Preparedness—Remote Monitoring of Nuclear Power Plants

James A. Blackburn and Michael C. Parker

ABSTRACT Immediately following the nuclear accident at Three Mile Island, the State of Illinois began to design a state-of-the-art emergency preparedness system for the thirteen nuclear power reactors within its borders. This system incorporates an on-line reactor parameter data communication link, an on-line automated isotopic gaseous effluent monitoring system, and gross gamma monitors installed around each site. Liquid effluent monitors will soon be installed also.

The sensitivities and capabilities of this remote monitoring system have been clearly demonstrated both during abnormal events at the reactor sites and during emergency preparedness exercises. These experiences readily illustrate the system's ability rapidly to provide comprehensive, technical data to the Department's staff should an accident occur at an Illinois reactor site.

I. INTRODUCTION

Immediately following the nuclear accident at Three Mile Island in 1979, the State of Illinois began to design a state-of-the-art emergency preparedness system for the 13 nuclear power reactors being operated or constructed within its borders. In 1980, the Illinois Department of Nuclear Safety (IDNS) was formed, initially by Executive Order of Governor James R. Thompson, then by confirming legislation. An immediate challenge was to provide a mechanism whereby governmental agencies would receive more timely and accurate information regarding both the radioactive composition and the magnitude of any accidental release of radioactivity to the environment.

The initial goals of the Remote Monitoring System (RMS), in the event of accidents, were: to analyze as accurately as possible the discrete radioactive components being released from the reactor site; to assess the magnitude of their radiological impact on the populace; and to transmit the results of the analyses to the Departmental decisionmakers as rapidly as possible. It was quickly realized that predicting radiation doses to members of the public was almost impossible without additional knowledge of reactor conditions. Since 1984, the RMS has been significantly enhanced and expanded by incorporating on-line information regarding the status of essential safety systems at the plant. Crucial factors in theRMS design were to provide reliability in the acquisition and transmission of data and to minimize the amount of time required of utility staff for collection and transmission to IDNS.

The current objectives of the RMS are threefold: early warning of nuclear reactor events having a potential off-site impact; fast risk analysis of reactor systems; and rapid identification, quantification, and verification of a radioactive release to the environment. Each of these objectives plays an essential role in assuring the ability to recommend prompt off-site protective actions.

II. SYSTEM DESIGN

The Illinois Department of Nuclear Safety's Remote Monitoring System incorporates three major components: gross gamma detectors radially positioned around each nuclear power station; on-line automated, isotopic gaseous effluent monitors, which sample from major engineered release points; and an on-line reactor parameter data communication link to each facility's on-site computer. In addition, on-line liquid effluent monitors, which will be located at each plant's liquid discharge points, are scheduled for installation at two sites within the next year. All RMS components are connected, through dedicated data communication links, to the IDNS Radiological Emergency Assessment Center (REAC) located in Springfield, Illinois. There, a technical staff comprised of nuclear engineers, health physicists, and other nuclear safety specialists reviews the data and performs analyses of plant conditions. This REAC staff is divided into two analytical groups: one concerned with the status of reactor safety systems; the other with environmental assessment.

Confirmation of a gaseous release to the environment is accomplished by a network of up to sixteen pressurized ion chambers installed radially around each plant at a distance of approximately two miles. These monitors have a dynamic range of 2.6 x 10^{-10} to 2.6 x 10^{-3} X units per hour (10^{-6} to 10^{1} Roentgens per hour). The siting of these instruments involved a balancing of many factors, including maximizing the probability for plume detection, while minimizing plume transit time to the detector and the potential for error, either due to observing building shine from the containment or sky shine from an elevated plume. Using typical Illinois atmospheric conditions, the off-site system will detect a plume with a centerline reading of two milliRoentgens per hour approximately 90 percent of the time. This criterion for detection requires one monitor to report values double the levels associated with natural background. Non-detection is limited to extremely stable conditions with wind directions bisecting the distance between adjacent detectors. Since these atmospheric conditions can be easily described, the Department is planning to develop software to recognize such conditions rapidly and display an appropriate warning for the analytical staff.

Quantification of a release is accomplished through radiological analyses of data supplied by the IDNS gaseous effluent monitor. This monitoring system is complicated, incorporating high purity germanium detectors and gamma spectroscope to identify and quantify components of the isotopic source term, i.e., those radioactive contaminants released into the environment, whether particulates, volatile iodines, or noble gases. This instrumentation has a dynamic range of 3.7 x 10^{-9} to 3.7 x 10^{2} Bq/cc (10^{-13} to 10^{-2} uCi/cc) for the particulate and iodine stations, and of 3.7 x 10^{-5} x 3.7 x 10^{8} Bq/cc (10^{-9} to 10^{4} uCi/cc) for noble gases. The radiological data gathered by this gaseous effluent monitoring system is continually received and updated by a combination of PDP 11/34 and 11/70 computers. Software has been developed to compute atmospheric dispersion and postulate environmental exposures from a release, based upon current meteorological conditions and effluent radioactivity levels.

Early warning and risk analysis of reactor events are accomplished utilizing the IDNS Data Link (DDL). DDL receives approximately 1200 to 1500 key reactor and engineered safety system parameters every two to four minutes directly from each reactor's on-site computer. Early warning of abnormal reactor conditions will soon be provided through software which will monitor each plant's engineered safety system configuration, identify the presence of key abnormal events, and indicate proper operability of engineered safety systems. Additionally, DNS is researching the use of expert system technology to develop a decision-aide to diagnose reactor system status during abnormal events and to apprise the REAC team of possible sequences leading to a release of radioactivity.

The analysis of the liquid effluent streams may use only gross gamma monitoring rather than complete isotopic analysis. Such a reduction in complexity and cost is being considered due to the longer lead time before the onset of exposure to the general public by the water pathway and the relative ease with which such exposure can be reduced or averted.

III. CURRENT STATUS

At the present time, the gamma monitors are installed and operating around all reactor sites, providing baseline data for those facilities still under construction. A gaseous effluent monitoring system has been operational at LaSalle since 1982. Additional systems are currently under construction for installation later this year at Zion and Dresden. The contract has been signed for fabrication and installation of systems at Quad Cities, Byron, Braidwood, and Clinton during the next four years. The DNS data link is installed and operational for all reactors except Braidwood. Braidwood will be incorporated into the DDL network prior to initial loading of fuel. As previously mentioned, the liquid effluent monitoring systems are currently undergoing initial review as a pilot project for the Zion and Dresden facilities. Upon satisfactory completion of that evaluation, additional systems will be procured and installed at all Illinois reactor facilities.

IV. DDL SOFTWARE DEVELOPMENT

Although the DDL analysis software is only in the first of three developmental stages, it has already become a major facet of the Departmental approach to Emergency Preparedness and Response. The initial phase in DDL software development is to establish a data base with access to both historical and current data. At the present time, a four-day data base is available for instant access. The basic capability is to display data for up to seven signals chosen at random from among the 1200-1500 points available from each reactor. These data may be either sequential two-minute measurements or samples chosen at specified intervals. The user is also allowed to track real-time trends of the chosen signals. Under this mode, the oldest data, approximately forty minutes old, automatically scrolls off the terminal screen as the current information is received. An alternate format allows real-time display of the current status of up to twenty different signals, although the trending capability is sacrificed. This latter format is truly user selectable, having the flexibility to group points associated with a given subject, a given reactor, or selected at random from the system at large.

The second phase of software development for the DDL system will be to design an "early warning" alarm into the system. This feature will consist of software programs which will constantly monitor various data packages for indications of abnormal conditions, such as loss of off-site power. Although not necessarily an indication of accident conditions, such degradation is important to assess the ability of the facility to respond adequately in the event of an accident. The Departmental response to such an alarm will be to alert the reactor analysis staff to ensure its awareness of the situation.

The third and final phase of DDL software development is presently envisioned as an expert system to aid the reactor analyst in his ability to determine the current status of reactor safety systems, project in a faster-than-real-time environment the possible consequences of the abnormal events and advise on protective measures off-site. It is not the intent or the desire of the Department to direct utility remedial actions based on this data. DNS will use this information only to the extent that likely consequences of an event can be analyzed in support of off-site protective actions.

V. TECHNICAL DIFFICULTIES

The development and operation of such an extensive remote monitoring system has had its share of technical difficulties and challenges. The off-site monitors, for example, require the use of a 300 volt battery which is both relatively expensive and short-lived. In addition, impending failure is characterized by erroneous indications of increasing radiation levels, sometimes as high as ten to fifteen times normal levels. Additional problems encountered with the off-site detectors include loss of electrical power, failure of the telecommunications link to Springfield, vandalism, and susceptibility of the electronic components to power surges caused by lightning.

Despite these difficulties, over the past year (July 1985 through June 1986), the off-site component of the system, comprised of 96 detectors with their modems, approximately 2130 kilometers (1325 miles) of telecommunication lines, two computer systems with attendant hardware peripherals and required software, achieved a system-wide operability factor (availability of data for display in Springfield) of 93.1 percent.

To increase the reliability of the off-site monitors even further, the Department is developing a 300 volt power supply to replace the dry cell battery, researching the ability to use solar collectors for back-up power, and evaluating radio communications to a central collection point rather than leased lines to a telecommunications bridge. In addition, the Department has already installed electrical surge protectors on the power supplies to reduce susceptibility to lightning.

The complexity of the isotopic gaseous effluent monitoring system creates complicated problems. A major cause of system failure is failure of off-the-shelf components and a lack of environmental support systems, rather than problems with the custom-designed components. For example, the system has failed several times following failure of the air conditioning system for the Technical Support Center (TSC), the location for the system's on-site process computer. Although a TSC is required by NRC regulations, apparently the operability of its support systems is not. Sometimes, substantial effort is required to obtain repair of this essential support system by the utility.

A recent failure of the effluent monitoring system was the result of loss of electricity to the TSC. Although supplied with back-up generating capability, the circuit breaker feeding the system had been manually thrown due to a "noisy transformer" which was annoying a worker in the area.

To eliminate these types of operational problems, the Department is requiring the construction of a dedicated building on-site. This building will house the entire gaseous effluent monitoring system, including its process computer and dedicated support equipment such as back-up power supplies, heating and air conditioning, and instrument air.

Not all of the problems with the effluent monitoring system are the result of variable in-plant environmental conditions. A substantial amount of difficulty was finally traced to microphonics in the coaxial cables connecting the gross count rate meter to the multichannel analyzer. Since this gross indicator is designed to signal the system to rapidly increasing radiation levels within the sample, the presence of these spurious signals caused the system to decrease the analysis times and consequently the system's sensitivity. Although the system never specifically failed due to this problem, the ability to detect radioactive contaminants being released into the environment was substantially impaired.

In spite of these difficulties, the gaseous effluent monitoring system has performed well, particularly for a prototypical system. Over the past year (July 1985 through June 1986) the system's three stations, particulates, iodines, and noble gases, were operational 86.7, 83.3, and 89.8 percent, respectively.

In contrast to the approach used for the off-site and the gaseous effluent monitoring systems, the DDL incorporates no State-owned sensors or instrumentation, relying exclusively on utility installed and maintained instrumentation. The Departmental components to the system are limited to a pair of statistical multiplexors, dedicated telecommunication links, and the REAC computer system. To minimize the cost and time required for system installation, this component is currently limited to a subset of the signals presently residing on the utility's computers at the various reactor sites. Such an approach minimizes the direct responsibility for proper operation and reliability, although at a sacrifice of control and assurance that the system will be functional when needed. The utility's computers, for example, may be removed from service for preventive maintenance with little, if any, advance notice to the Department or Departmental input. In addition, utility personnel have, on occasion, unilaterally altered the contents of the transmitted data stream, adversely impacting the Department's data analysis capability.

VI. OPERATIONAL RESULTS

The primary intent of the RMS is to provide the capability of independently assessing the possible off-site impact of an accident at a nuclear power reactor. Such a capability is a major support in recommending specific protective actions for the general public. Since monitoring effluent streams during normal operations is of secondary importance, only a limited amount of data analysis has been performed to date. Gaseous effluents from the reactors have remained below the minimum detectable levels of the off-site monitoring system. Environmental radiation levels nominally average between seven and ten microRoentgens per hour due to natural background. During times of precipitation, however, the ambient levels are substantially elevated, often reaching as high as fifteen to twenty microRoentgens per hour.

The data generated by the gaseous effluent monitoring system has routinely indicated the presence of radioactivity in each of the three stations, when the reactor is at power. Typical fission products including isotopes of Cobalt, Manganese, Sodium, Iodine, Krypton and Xenon, for example, were identified by the system during May, 1986. Research is being done to match the ongoing results obtained from the gaseous effluent monitoring system and the daily grab samples gathered by the utility, as a portion of its Appendix I requirements. Although not yet available for publication, these comparisons appear to correspond remarkably well, considering the variation in methodology used for obtaining and analyzing samples.

Late in January, 1986, a small leak developed in one or more of a reactor's fuel rods. The DNS Gaseous Effluent Monitoring System readily identified both increased levels of radioactivity and additional nuclides within the effluent stream. The noble gas station, for example, reported effluent concentrations averaging 2.7×10^{-1} Bq/cc (7.4×10^{-6} uCi/cc) for the 64 analyses performed between January 23 and February 1. This compares with an average of 3.4×10^{-3} Bq/cc (9.4×10^{-8} uCi/cc) for the 96 analyses obtained between January 1 and January 15. By combining these data with DDL parameters, such as off-gas radiation levels and the readings from the utility's effluent radiation monitors, the Department was able to verify the presence of this leaking fuel.

The DDL was directly utilized by the Department during the feedwater transient event at the LaSalle station on June 1, 1986. The system did not document the water level as being less than the low water level scram setpoint, due to the transient's short duration when compared to the two-minute sampling frequency. However, the system clearly showed the initial downward trend of the water level and other affected parameters, both during the initial event and the subsequent manual shutdown of the reactor. During this event, the utility telephoned the REAC at hourly intervals, indicating the power levels and the reactor status. By monitoring DDL, technical staff within the REAC were able independently to verify the status of individual control rod banks, reactor power levels, reactor water levels, etc.

VII. EMERGENCY USE OF THE RMS

Although the RMS monitors the activities of the Illinois reactor facilities during normal operations, its purpose is to provide data during emergency conditions at a nuclear power reactor. During a major accident, the REAC staff would initially analyze the various DDL parameters for trends and to diagnose the problem. The reactor analysis staff would concentrate on the status of the reactor's engineered safety systems in an effort to predict key trends and their consequences, while the environmental analysis staff would focus on containment radiation levels, area radiation monitors, and effluent concentrations. In anticipation of a possible radioactive release into the environment, the environmental analysis staff would also calculate dose reduction factors due to atmospheric dispersion, using the meteorological parameters available from the DDL. If the situation warrants, calculations would be performed to determine the protective actions which would give the least dose to the general public living in the immediate environs surrounding the facility.

If a release occurs, its presence would be verified by both the DDL and the gaseous effluent monitoring system, and the postulated impact on the general public calculated. Using the isotopic source term provided by the gaseous effluent monitoring system, the environmental analyst would also be able to predict the presence of released radionuclides which would contribute to ground contamination. Verification of the wind direction and the anticipated off-site radiation levels would be available from both the environmental monitors and the Department's field teams dispatched to the area. Following the termination of the release, the isotopic source term, in conjunction with contamination readings obtained from the mobile field teams, would be used to determine and document the integrated dose to the off-site population.

Although the Remote Monitoring System does not solve all problems associated with assessment of reactor accidents and their impact upon the general public, the Illinois Department of Nuclear Safety believes that this approach is a technological advancement in providing accurate and timely information to decision makers during a radiological accident at a nuclear power reactor.

Section III

Computer Applications

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Computer-Assisted Emergency Preparedness and Response

James W. Morentz

ABSTRACT. No tool available to emergency planners and response personnel offers more actual and potential management capability improvement that the microcomputer. The low cost, fast access speed, data format flexibility, data manipulation versatility, and color graphic display clarity make the microcomputer -- when applied appropriately -- an extraordinary resource for the emergency manager to draw upon. A computer program called the Emergency Information System® is in use around several nuclear powerplants. The system helps describe and clarify situations at a nuclear facility for knowledgeable observers. It is used for communication with surrounding jurisdictions, and is even linked into state emergency operations centers where the broadest emergency decisions are made. The EIS uses computer-generated maps to display evacuation routes and merges the graphic displays with several databases of value to emergency planners: special populations, resources, traffic control, alert lists, plans, and others. The microcomputer, operating alone or in a network of up to 20 computers, provides real-time decision support to emergency response operations staff.

While a nuclear powerplant stays in one location, an emergency sets in motion a swirl of events that traverse geography, demography, and political jurisdictions. The spatial movement of an emergency is, we believe, the single most complex aspect of its effective management.

No tool available to emergency planners and response personnel offers more actual and potential management capability improvement than the microcomputer. The low cost, fast access speed, data format flexibility, data manipulation versatility and color graphic display clarity make the microcomputer -- when applied appropriately -- an extraordinary resource for the emergency manager to draw upon.

This paper describes actual experience and potential applications of a computer software package for nuclear emergency planning. The computer program, called the Emergency Information System®, has been developed by the author's firm for use on-site at a nuclear facility, in communication with surrounding jurisdictions, and even linked into state emergency operations centers where the broadest emergency decisions are made.

I. NUCLEAR FACILITY EIS

At a nuclear facility, the Emergency Information System becomes an important tool of the emergency planner. The combination of databases which can be related geographically give the on-site planner superior capabilities in emergency response analysis.

Let us take three examples, warning sirens, special populations and public-technical information on accidents, all of which are responsibilities shared by the local government and the facility.

The special capability of the EIS is to link geographical data with text data. In the case of warning sirens, the following scenario illustrates the power of this tool.

Siren sound propagation distances are entered on the geo-relational database of the EIS. When sirens are ordered to sound, the typing of the word "siren" brings to the screen a map of the evacuation zone and circles (or other shapes) showing the siren sound propagation. If a siren is found to fail (via a call-back system or report from the field), the user selects the failed siren from a list. Pressing the [M]ap key returns the evacuation zone to the screen and the failed siren's entire propagation zone is blinking. The user then moves the screen cursor into the blinking area representing the failed siren sound and presses a single key. Appearing almost instantly is the plan for replacing the warning siren with mobile sirens, loud speakers, or door-to-door contact. Exact street names, responsible agencies, contact persons and phone numbers, and the number of people to be warned all appear. In seconds, the emergency manager can confer with the responsible authorities and set in motion alternative warning methods.

Now, for another example, let us look at special population needs. The EIS is equipped with a geo-relational special emergency needs database. Contained on the database is information on the name of the facility, address, phone, contact name, number of people at the dependent care facility, notes on evacuation plans, the agency responsible for carrying out the evacuation, the lead time necessary for an evacuation to take place, the shelter location to which people will be moved, and other important information.

Of premier utility to the emergency manager is the fact that all of this information is also referenced by geographic location. Thus, with the press of a few keys, the user has displayed on the screen's map the location of each special care facility. Pressing another key, a text listing is displayed showing the name of all facilities, the number of people at each, and the total number of people to be evacuated from the care facilities.

With the press of two keys, the user can identify how many busses, for example, with wheelchair lifts are necessary to evacuate a specific nursing home. Then, with the press of a single key, the user can return to the map and have that specific nursing home appear, blinking at its proper location on the map.

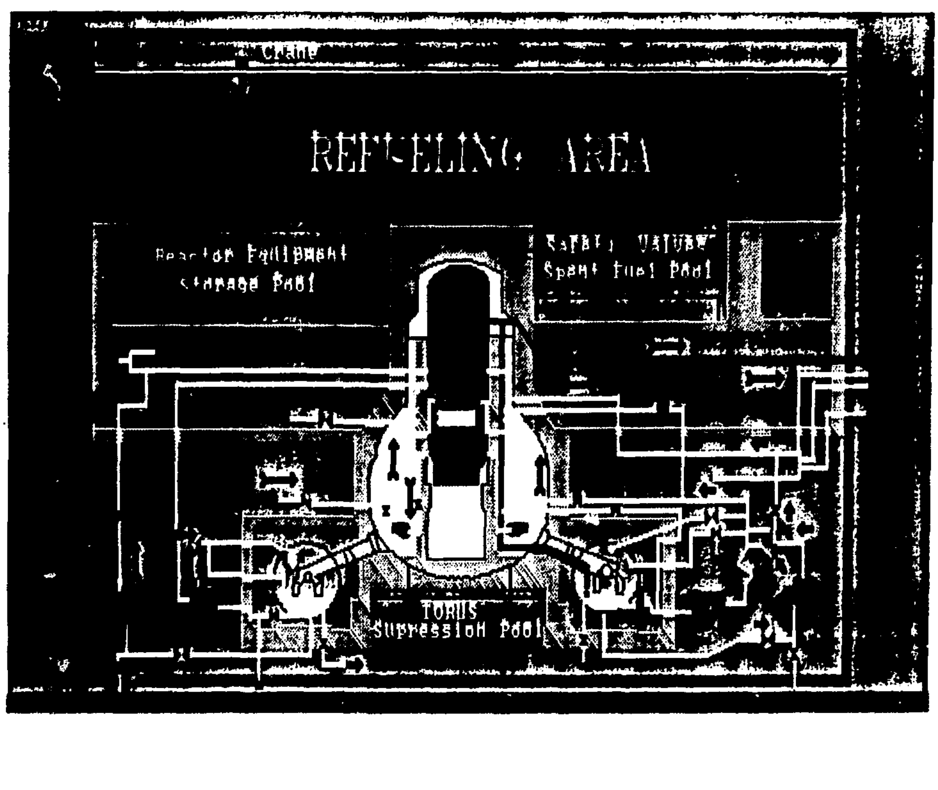
As a final example of on-site applications of the Emergency Information System, we refer to providing public-technical information about the Three Mile Island boiling water reactor. The versatile graphic capabilities of the EIS have enabled us to "zoom" right inside the plant to display a graphic depiction of the steam cycle of the plant. An initial graphic encompasses the entire steam cycle. Moving to any of the three main parts of the cycle and pressing a single key "zooms" the user to a detailed graphic of that portion of the steam cycle, as shown below.

Then, using the geo-relational database capability of the EIS, the user can request text information about any of the component parts of the reactor.

For example, in its development of the Three Mile Island EIS, the Pennsylvania Emergency Management Agency focused on public-technical information; that is, generally available public information of a technical nature that could be used to inform knowledgeable media and aid decision-making. The focus of the text developed about Three Mile Island was on the impact of a reported failure of the component parts of the reactor.

Thus, if a report comes in that a particular pump has failed, public information personnel at the plant, or emergency personnel responsible for contacting governments, have at their fingertips a sophisticated graphic and text display to use to provide appropriate public-technical information. The speedy data display process is as follows:

First, the user "zooms" in on Three Mile Island and the steam cycle. Second, the user types the word "pump." Instantly, circles appear around all the pumps designated as vital to the functioning of the facility. Third, the user can move the cursor directly to the pump of choice if its location is known, or switch to a summary text screen and select the right pump from a list of pump names. If the list approach is selected, the user can go directly to detailed text descriptions or return to the steam cycle graphic where the selected pump will be blinking to note its location. Fourth, from the graphic or the summary text display, a series of full text displays are retrieved. These describe the function of the specific pump selected, the implications of its failure, and



the potential for expanded emergency conditions. The full text description may be a single screen or many screens, each of which are displayed less than a second after the user presses a single key.

The value of this graphic and text capability for briefing the public news media or government executives with explicit technical information accompanied by excellent graphics should not be underestimated. If one thing has been made clear by past incidents, it is that effective communication is essential. The EIS as an executive briefing tool is unexcelled. Six different types of graphic pointers are available on the system (cross-hair, checkmark, arrow, hand with pointing finger, hand held in stop position, and user-sized outline box). The user can select the most appropriate pointing graphic at the press of a key, changing it instantly to match the presentation's need. Thus clarity of presentation in graphic and text becomes one of the improved products of the use of the Emergency Information System.

II. OFF-SITE AUTOMATED PREPAREDNESS AND RESPONSE

The Emergency Information System is currently used in large and small municipalities and counties in this country and abroad. It has been purchased by nuclear facilities and other corporations for the use of local governments. As a result, the EIS can serve as an important information link between powerplants and the off-site emergency responders in local and county government. The EIS has been used in radiological emergency planning drills and exercises. It is upon that exercise experience that this and the following section are based.

Off-site emergency preparedness is aided tremendously by the Emergency Information System in planning, evacuation route monitoring, and resource management.

The EIS Automated Emergency Plan allows speedy retrieval of standard operating procedure checklists for all types of emergencies. For radiological emergencies, the Automated Plan allows the user to quickly arrive at specific lists of assignments for all personnel. These can be used in advance of an emergency for printing out plans for everyone to review and for training new personnel. During an emergency, top management can use the plans as a checklist of performance to monitor whether specific benchmarks have been achieved. Following an emergency, plan evaluation and improvement are aided by the computer. Debriefing meetings can be held about plan utility, in which a user calls up each part of the plan in sequence. As recommendations are made, they can be typed directly onto the computer for all in attendance to verify. Almost instantly, then, a new or revised plan section is ready for training, exercising, and use.

Evacuation route monitoring is readily accomplished with the EIS. At the local level, three elements of evacuation are crucial: route planning, alternate route identification, and traffic control point staffing.

Using the EIS, a local government can design an evacuation route system that takes into account all prior facility planning. The government can then enter routes as part of the EIS (as shown on county-level map on the following page), providing street names, number of people to be evacuated, and other important information. The evacuation routes can be coded by sector and can each identify a lead time or clearance time and make note of the destination of evacuees. When the user requests evacuation routes to be displayed, they are drawn on the screen as red lines, and a text listing of all routes can be generated.

Alternate routes can also be entered onto the system so that if a main route is blocked the user can quickly display alternate routes (as dotted or dashed lines of red, blue, or yellow) and obtain a list of those route names. If a main route is closed, it and its alternate routes can be set to blink on the map as an alert to all personnel to the deviation from plans.

Resource management is another off-site application of the EIS to radiological emergency planning and response. The EIS can be used to locate dosimeters, monitor their deployment around the Emergency Planning Zone (EPZ), and record their recall after an exercise or emergency. In the evacuation stage, the EIS will identify each Traffic Control Point, its staffing numbers, responsible agency, and lead time for staffing. (See EPZ-level evacuation map on next page.)

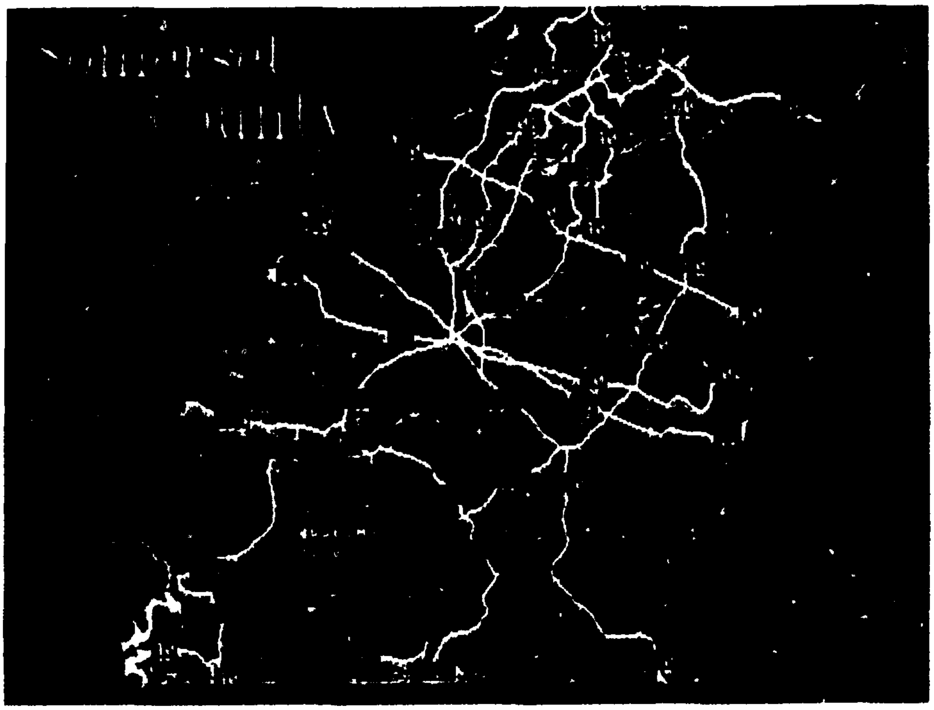
With these brief examples it should be clear that the advantages of microcomputer decision support tools for radiological emergency planning and response are great. The advantages are even greater when the computer is viewed as part of a communications network linking powerplants, local governments, and the state emergency office. In the next section we will discuss the aspects of the EIS that bring the state into the network.

III. STATE EMERGENCY OFFICE COMPUTER APPLICATIONS

The state emergency operations center (EOC) has a key mandated role in radiological emergencies. The EIS supports that role not only through better information filtered up through local governments by means of the automated Event Log, but also by giving the state EOC a firm grasp on the SPATIAL DIMENSIONS of the emergency.

Far removed as it normally is, the state EOC nevertheless is usually responsible for tremendously important decisions regarding a radiological incident. In order to make the proper decisions, the state EOC must be able to provide an accurate picture to the Governor of the options and consequences of decisions.

There is no better tool for achieving this end than computer graphics. And, we suggest, the geo-relational data management system that is the backbone of the Emergency Information System brings to emergency decision-making all the best attributes of this new tool.



County-Level Map of Evacuation Routes

EPZ-Level Evacuation Map

At the state EOC, the EIS can portray an accurate picture of the geographic aspects of the emergency through a series of "zoom" maps that fully depict the EPZ from 50 or more miles out right down to the crucial intersections. Each map is separately digitized, showing all the information that an EMERGENCY MANAGEMENT DECISION-MAKER needs. These are, above all else, decision-makers' maps, not geographers' maps.

Most importantly, the EIS does not stop with maps. It builds into the decision-support system the geographic representation of data. Thus, when considering an evacuation, the state-level decision-makers can "play" with "What if?" options and graphically see their outcome.

"What if we decide to evacuate in one hour?", the user might ask. The EIS would show all reception centers that can be opened in one hour. Or in two hours, if desired.

What if a plume spread northwest and we had to move all people immediately out of its path to the east and southwest? What evacuation routes are available in this area (and the user draws a rectangle on the screen)? Shown immediately on the screen are the evacuation routes. A text listing of the routes is also available at the press of a single key.

Now, what about Access Control Points? If we evacuate that zone, where will our ACPs be placed? Again, ACPs appear as dots on the map. Those selected for staffing in this specific evacuation can be set to blink by the user.

Now, the Governor might ask, what if we don't do a northwest sector evacuation but do an entire 10-mile zone evacuation? Show me everything I've just seen for the whole EPZ. And in a few seconds the interrogation of all the constituent geo-relational databases of the Emergency Information System is complete and the Governor has the options.

IV. SUMMARY

Microcomputers have opened to local and state emergency managers capabilities for decision analysis that were possessed only by the military in the past. As our technological society has become more complex, at least one technological advance is now available to help government and industry leaders sort out the complexity and make the best possible decisions when it comes to public safety.

FEMA'S Integrated Emergency Management Information System (IEMIS)

Robert T. Jaske and Wayne Meitzler

ABSTRACT - FEMA is implementing a computerized system for use in optimizing planning, and for supporting exercises of these plans. Called the Integrated Emergency Management Information System (IEMIS), it consists of a base geographic information system upon which analytical models are superimposed in order to load data and report results analytically. At present, it supports FEMA's work in off-site preparedness around nuclear power stations, but is being developed to deal with a full range of natural and technological accident hazards for which emergency evacuation or population movement is required.

I. INTRODUCTION

At the present time, IEMIS is operational at FEMA Headquarters, the 10 FEMA Regions, the National Emergency Training Center, twelve States and a number of local jurisdictions.

The IEMIS computer equipment is divided into three major components: the operational field workstations, the FEMA mini-computer and data base, and networked user nodes.

Presently, a field workstation supports each Regional Office. In the future, each Region would become a node with its own mini-computer acting as a local hub for State and local use. The field workstation consists of a medium resolution color graphic terminal to make system requests and display results, a letter quality printer to print alphanumeric reports, a color hardcopy device to provide hardcopy of the information on the color display, and communications equipment interconnecting the field terminal and the FEMA mini-computer.

For the users located near the FEMA mini-computer, high resolution workstations are available to display results, and film recorders to make hardcopy of the displayed information. The high resolution terminals are also supplemented by large screen projection, 2x3 meters.

Both the field workstations and the high resolution workstations connect to the IEMIS computer which performs the modeling calculation and data storage. The system is built around commercially available hardware. The computer is a Digital Equipment Corporation VAX 11/750 with four megabytes of the internal memory and two gigabytes of disk storage, magnetic tape units, and a line printer. At present, over 50 field workstations are operational in the IEMIS environment. Included are IBM compatible PCs which can be adapted to serve as IEMIS terminals. Any VT 100 compatible terminal may connect to IEMIS, but graphic functions require special features.

As currently operating, IEMIS includes the following elements:

a. Computer utilities, word processing, (VMS/TDMS/Datatrieve/INGRES);

b. Emergency Information Coordinating Center elements-message functions, notification, weather abstracting, team briefing;

c. The National Map and resources data base as a geographic information system;

d. Program data bases, such as Radiological Emergency Planning Program;

e. Exercise Evaluation and Simulation Facility; and

f. Colorgraphic chart and graph preparation.

Other compo[illegible]nts will be added as models an program data bases are integrated into the system. Under development at this time are: hurricane evacuation planning and exercising, transportation accident evaluation, management of accidents involving toxic and hazardous chemicals, dam failure, conflagrations and earthquake.

FEMA is working with State and local governments in the development of data bases and in developing the application of IEMIS to these program areas. In order to stimulate interest, FEMA makes developmental copies of IEMIS in executable format to State and local governments, associated universities and contractors.

II. NATIONAL MAP AND GEOGRAPHIC INFORMATION SYSTEM

Presently, IEMIS contains a digital cartographic data base derived from the 1:2,000,000 scale, sectional maps of the National Atlas of the United States. This color graphic version is a unique adaptation of the original work by the United States Geological Survey, and is arranged within IEMIS for unlimited pan and zoom for the entire United States. All displays may be made at one of six standard map projections, and the data structure will handle global inputs including International Date Line transactions.

As implemented in IEMIS, the mapping system includes political boundaries, transportation networks (roads and railroads), hydrographic data features (streams and water bodies) elevations, and names of geographic and populated places. These data were entered into the computer in a format that records topologic relations with related features on the source map. This format allows graphic applications, such as drawing streams and roads for automatic map plotting, as well as analytical applications, such as population and area calculations, and checking data for consistency and accuracy. The format also permits display of maps at any scale and introduction of data from maps of any scale.

An important feature of the attribute coding is that individual features within each category are ranked from the most significant to the least significant. This scheme allows the user to select a minimal amount of data and selectively increase this amount to the level of detail needed to support the theme and scale of the desired map. The data can also be grouped together in various ways to produce logical sets of information.

As implemented, the system permits the cutting out of precise sections, performing unlimited editing, additions, color changing, and reinsertion of edited sections. It is, thus designed to be continuously maintained without service interruption. Each user may create unique maps filed in a personal file for use in support of individual projects or in specialized instruction, or alternatively return edited map segments to the master data file which may exist in several locations. These functions can be performed off-line on PCs for economy in developing local data bases.

The user can obtain the location of any point in latitude/longitude, linear distances between any points, and the perimeter of any polygon. The user can also add or delete text, color or shade any polygon, and find the population or resources data in any streams for evacuation models. The system also includes the geographic place names listed in the National Atlas. FEMA is also planning to upgrade the National Map system substantially over the next several years into a generalized geographic information system.

The USGS 1:100,000 scale maps presently being prepared for the 1990 census (by 1988), will be integrated into the system as the quadrangles are digitized. Digital elevations and resources data in topologically structured layers may now be entered into the data base.

As part of the ongoing map applications, FEMA maintains a Memorandum of Understanding with the USGS whereby the parties agree to the 1:2,000,000 scale map to 1:100,000 scale. As (NMSS) segments are produced by USGS they will blend into the FEMA system. The two agencies will share a common, computerized status management sub-program maintained on IEMIS and available to all users.

III. PRINCIPAL IEMIS COMPONENTS

While the long range goal of IEMIS is to support the entire range of emergencies for which planning, exercising and response is thought feasible, as a practical matter, developments progress in an incremental fashion. At this time, the following program areas are addressed, with a comment as to the status of development.

Table 1

Program	Status Comment
Radiological Emergency Preparedness	Supports all regulatory objectives.
Hurricane Evacuation Planning	Model in advanced state, preliminary planning possible.
Hazardous Materials	Diffused gas modeling and evacuation support.
Crisis Relocation Planning	Preliminary work now possible. More data needed in system.
Dam Break	Base maps and elevations encoded, coding in progress.

As a means of describing basic IEMIS capability, a short discussion of the major models involved in the programs above follows. The model suite is available to all the program areas either singly or in multiple combination. For example, the MESORAD model will interact with the I-DYNEV evacuation model and allow studies of concentrations of highly diffused radioisotopes which could occur as a result of power accidents. Since as an interim step, it reports concentrations of duffused (not subject to gravity effects) gases, it can also estimate concentrations of highly toxic, diffused chemicals over the simple terrain (no deep valleys or shore breeze effects) where two dimensional puff models are applicable.

MESORAD computes the dose arising from the release of radioisotopes listed in the joint NRC/FEMA Preparedness Guide NUREG-0654/FEMA-REP-1 Rev. 1[1]. MESORAD consists basically of the diffusion code MESOI[2] and a simplified radiation dose code described by Scherpelz et al, 1985[3]. Outputs from the diffusion code are considered geometrically and dose calculated using a semi-infinite cloud assumption over a grid of 1000 cells whose size can be varied from a minimum of 0.4 mi^2 (1.04 km^2) to a maximum of about 10 mi^2 (26.8 km^2). This rather coarse representation provides a relatively macroscopic estimate. This almost matches the approximate size of neighborhoods in evacuation centroids of the evacuation model and allows integration of the dose and transportation models by applications of a shelter factor to each population centroid. This model format is computationally efficient and allows quick estimation of dose by successive interaction in support of plans reviews, exercises and drills. In its existing form it is not intended for response activities.

The techniques utilized in MESORAD contribute to advancing the state-of-the-science for real-time dose assessments with the development of a new method to treat the finite cloud approximation. The method, entitled the Discrete-Point Approximation, is used in the estimation of external doses when the dimensions of the cloud are small compared to the mean-free-paths of gamma radiation. This method, used in place of a point-kernel integration, provides a two decade savings in computational time at the cost of typically a one percent (but up to 10 percent under some conditions) difference in estimate. This represents an acceptable tolerance in light of the decades uncertainty in source term and, to a lesser degree, atmospheric dispersion.

A revised version of MESORAD is now operational on the system. This revised version may, after a suitable period of training and indoctrination, become a first order reference model for some response activities.

IV. EVACUATION - I-DYNEV, INTERACTIVE DYNAMIC EVACUATION

I-DYNEV is the principal model framework which ties a number of sub-models into an integrated subsystem. The main source of the subsystem is the U.S. Federal Highway Admininstration model named TRAFLO described by Lieberman[4] before the January 1982 meeting of the Transportation Research Board. TRAFLO was developed to operate at three levels: Level I treating individual vehicles, Level II treating a macroscopic grouping of vehicles in clusters, and Level III a gross representation in terms of traffic parameters. I-DYNEV uses the Level II process, treating groups of vehicles by means of movement specific histograms.

TRAFLO is actually a system of models composed of:

a) A macroscopic urban network model called NETFLO;

b) A macroscopic freeway model called FREFLO; and

c) A traffic assignment model.

The traffic assignment model used originally in TRAFLO is the equilibrium model TRAFFIC developed at the "Universite de Montreal", which has been extensively validated. It is planned, however, to also interface other assignment models with TRAFLO.

The physical traffic environment, which must be specified as input data by the user in order to exercise the I-DYNEV system, consists of the following features:

a) Topology of the roadway system;

b) Geometrics of each roadway component;

c) Channelization of traffic on each roadway;

d) Motorist behavior which, in aggregate, determines the operational performance of vehicles in the system;

e) Circulation pattern of traffic on the roadway system;

f) Specification of the traffic control devices and their operational characteristics;

g) Traffic volumes entering and leaving the roadway; and

h) Traffic composition.

To provide an efficient framework for defining these specifications, the physical environment is represented as a network comprised of uni-directional links and nodes. The links of the network generally represent urban streets or freeway sections. The nodes of the network generally represent urban intersections or points where a geometric property changes (e.g., a lane drop, change in grade or a major mid-block traffic generator)[5].

Over a period of several years, I-DYNEV will be supplanted by a more generic form which will be capable of simulating evacuations ranging from point source (typically hazardous materials sources) to regional events (typically hurricanes).

V. REGIONAL EVACUATION (REGEVAC)

I-DYNEV is being expanded to deal with regional problems which require a higher degree of origin/destination, control and accountability. When complete, the regional evacuation system will allow the user to specify which destination nodes are associated with each origin node. This is being done to allow specific assignment of population groups by neighborhood or district to specific relocation centers. While this was done in the trip distribution phase of I-DYNEV, this distribution was tentative and was to some extent overruled by the trip assignment and ultimate loading pattern for the simulation. A more specific direction and control strategy is thought necessary in large cases where the impact zone of the hazard is variable (e.g. hurricanes) and requires neighborhood assignment to shelter areas when the onset rate of the developing incident exceeds the dispersal rate of evacuees to individual destinations. Data from hurricane evacuations show significant increases in persons using public shelters when the hurricane suddenly changes course and threatens areas previously thought relatively secure.

REGEVAC will contain 3 levels of modeling (2 for I-DYNEV) of which two[6] are currently available for preliminary work. It allows a preliminary development of trip distribution based on the available highway network and control tactics which provides a first approximation. Then based on the known population dynamics, a trip assignment can be executed which readies the input factors for the final or simulation model which determines the actual traffic performance. Because of the size of the problem, the simulation model will operate on a segmented network, but except for data entry and modification purposes is transparent to the user.

Features of REGEVAC not possible with the existing I-DYNEV include the ability to create networks directly in the iteritive graphic mode, to modify existing links in previously defined cases, and to significantly relate the denial of links to a system as progressive flooding from an advancing flood wave. These features will significantly reduce manual coding burden, decrease errors previously created by incompletely coded network changes and allow flood level input from other models such as the National Weather SLOSH models, the Corps of Engineers HEC-1 and others. In addition, through testing with pilot projects involving real time flow measurements of both water and traffic, FEMA plans to work with local jurisdictions in the applications of REGEVAC as a means of building public confidence.

REGEVAC is designed to operate in conjunction with a computerized geographic information system in which resources elements with locational significance can be identified, incorporated into traffic flow planning, and ultimately used to guide recovery operations. Such attributes as demography, highway information and shelters and public facilities can be attached to visible map vectors and incorporated directly into modes which simulate planning options.

The mapping data of choice at this time is a combination of 1:2,000,000 National Atlas Data and the USGS/Census 1:100,000 scale data for streets, roads and streams now being digitized for the 1990 Census. In addition, FEMA is supplementing these data with facilities' data from its extensive files for ultimate use for a number of major disaster problems involving transportation, energy and fuel distribution, population protection and national resources management.

VI. Hazardous Materials

Currently, FEMA is deploying advance elements of IEMIS related to hazardous and toxic materials accidents. The U.S. Coast Guard Hazardous Assessment Computer System (HACS) is operational as a study sub-model suite, and a heat radiation envelop model is being deployed as a training aid. As a multi-purpose element applicable to many hazards, a siren alerting, sound propagation model is being integrated into the color graphic, menu driven format. Releases of vapors and aerosols following assumptions governing diffused gases can be modeled with the concentration sub-model of MESORAD. Several excellent heavy gas models are being evaluated by the Department of Energy, and one will be selected for inclusion in IEMIS.

As the data within the national IEMIS GIS is developed, the universal capability to support accidents from fixed and mobile sources will develop. This capability is also applicable to the modeling of radioactive exposure from accidents to mobile facilities.

VII. NEAR TERM DEVELOPMENTS

Because of the technical achievements possible with the system, IEMIS is being expanded to serve as the basis for a general distributed data processing system combining the elements of a geographic information system and a suite of action oriented simulation models which can be selectively brought to bear on disaster management problems.

VIII. FUTURE DEVELOPMENT

While the future is by no means assured, IEMIS provides the basis for a truly national system using a fully distributed data base, common data exchange standards, and full access by all components of civil government.

° In conjunction with a pilot project in the Tulsa, Oklahoma area, FEMA will develop software which will operate a nationally accepted code for simulating dam failure and incoporate it into the system. Using a combination of data picks from the simulation, produce an evacuation plan for the management of the threatened area. In addition, FEMA will support development of means to integrate flood insurance maps.

° FEMA is also arranging to integrate a sample of a metropolitan data base created by a gas utility company with the operational simulation capability described in this paper. In this formal demonstration, the software will be used to show how utility lines and services can be integrated with transportation models and hazardous impact models in order to create a metropolitan planning instrument. In addition, because of its high quality, the data base will be based to illustrate other uses for digital data of this type such as vehicle management, real estate development and cadastral applications. Subway planning and operation could also be integrated.

By the completion of the 1990 Census in the United States, a national geographic information system will be operational to support a multitude of administrative and commercial activities. Through data exchange standardization, all parties with interests in public resources should be able to participate.

On the equipment side, we see the development of computing equipment continuing unabated through 1990 such that the ordinary personal computer will be a very powerful machine, probably a 32 bit address, virtual core with at least 10×10^6 bytes of random access memory. Such machines will be extensively networked using common file protocols and will exchange information at unprecedented rates over both land lines and radio frequencies.

Thus, the emergency management community will become a full partner to the burgeoning use of electronic media and will be able to avoid the necessity of building immense data bases for statistically infrequent events by using everyday geographical and cadastral data bases in a special way.

FEMA fully intends to maintain the leadership posture evident in IEMIS, and within the achieveable time limits of technological adaptation, assist in creation of what could become an international system.

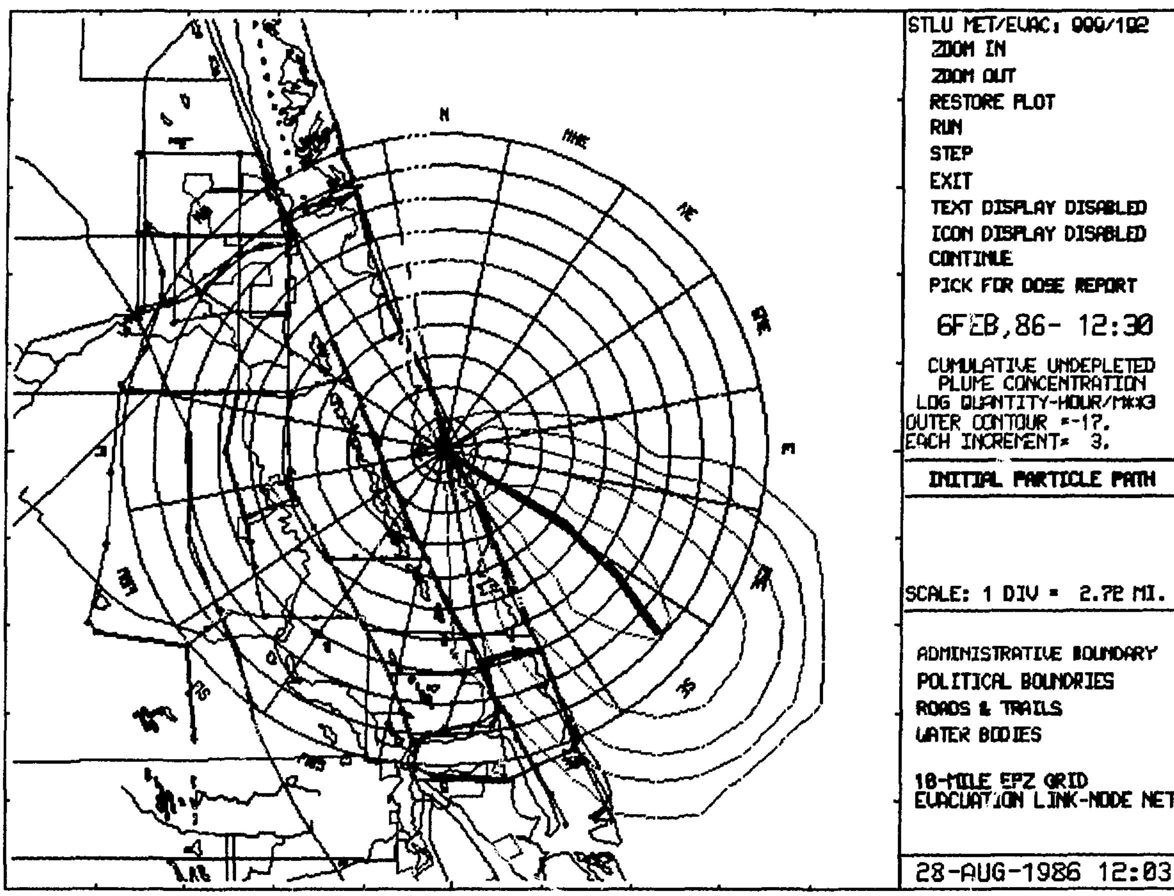


Figure 1 - Section of Map for Fort Pierce, Florida area with hypothetical release from St. Lucie Power Station.

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The MAPSS Nuclear Emergency Management System

Howard A. Price, Jr., Larry D. Sadler, and Robert W. Johnson, Jr.

ABSTRACT: With the MAPSS NEMS in place, a distinctly higher level of confidence is imparted to the emergency personnel managing a nuclear plant accident. The graphics are unique and are the same throughout all levels of response. Standard Operating Procedures are available instantly to all personnel, and system security provides partitioning of critical levels of command as well as documentation of all actions taken through the emergency. Maintenance is reduced to a very simple level and the entire system operates at meaningful life saving speed.

INTRODUCTION

The MAPSS Corporation is a small high-tech company with experience in the design and implementation of computer-based systems. MAPSS personnel have designed and implemented systems for several governmental agencies, including the Department of Defense and the Department of Energy. They range from small business systems, running on personal computers, to nuclear materials tracking systems involving large scale mainframes. MAPSS personnel have considerable experience in the design, manufacture, and application of specialized nuclear instrumentation. MAPSS has developed a system, called MAPSS=MAGIC which is flexible enough to be the core of the MAPSS Nuclear Emergency Management System (NEMS). The MAPSS=MAGIC system is the culmination of more than 7 years of design and development. MAPSS personnel have worked in conjunction with the nuclear industry for over 20 years. Some of MAPSS consultants have made significant contributions to the nuclear industry.

DEFINITION OF THE PROBLEM

The MAPSS Corporation has learned that, in spite of the intensive effort which has been made since the Three Mile Island (TMI) accident in Pennsylvania, and the subsequent establishment of the Federal Emergency Management Agency (FEMA), no satisfactory command, control, and communication information system has been established at the state or federal level.

The experiences at Three Mile Island (TMI) and more recently at Chernobyl, USSR, dramatically illustrate the need for effective control and communications at every operating nuclear installation. TMI and Chernobyl have demonstrated what can and will happen if an adequate nuclear emergency management system is not in place. If TMI and the USSR had had a responsive operating nuclear emergency management system, much of the panic and confusion that resulted would have been avoided. Yet today, more than seven years after TMI, no computer based state-of-the-art system is in use and no two manual systems now in place are alike.

Existing warning systems, not only in the United States but throughout the world, are in the great majority a piecemeal and awkward combination of telecopiers, telephones and manual procedures. Even with the stand alone PC based systems now on the market, considerable time is still wasted in passing complex details via the telephone with the recipient copying data on paper by hand. The telecopying of material between operations centers often suffers from network jamming because of telecopier overload. In addition, telecopiers are unreliable because of frequent equipment failures at critical times; telecopier performance has dropped lower than 50% in some exercises. In short, the existing communication systems have been stressed beyond their capabilities in exercises and most likely will not work during an actual emergency. It is obvious that an improved method of handling the critical information is necessary for adequate and timely decision making.

In order to establish a common ground on which to address what an emergency management system should do, the following list of the minimum requirements have been identified. Requirements may be different for particular applications, but an emergency management system should:

1. monitor situations and direct the activities of emergency personnel, resources, and equipment;

2. notify the affected areas in the event of an accident;

3. inform and direct the media in a responsive manner;

4. prepare for, coordinate, and conduct evacuation as events dictate;

5. conduct training of emergency personnel; and

6. coordinate and standardize operating procedures for emergency situations.

The MAPSS Corporation has developed a comprehensive Nuclear Emergency Management System (NEMS) which addresses these requirements from a "systems" point of view rather than the fragmented approach of the existing systems. The MAPSS NEMS is designed to be used primarily by personnel with little or no computer background. Information transmission, which now can take minutes, hours or longer, can be accomplished in seconds.

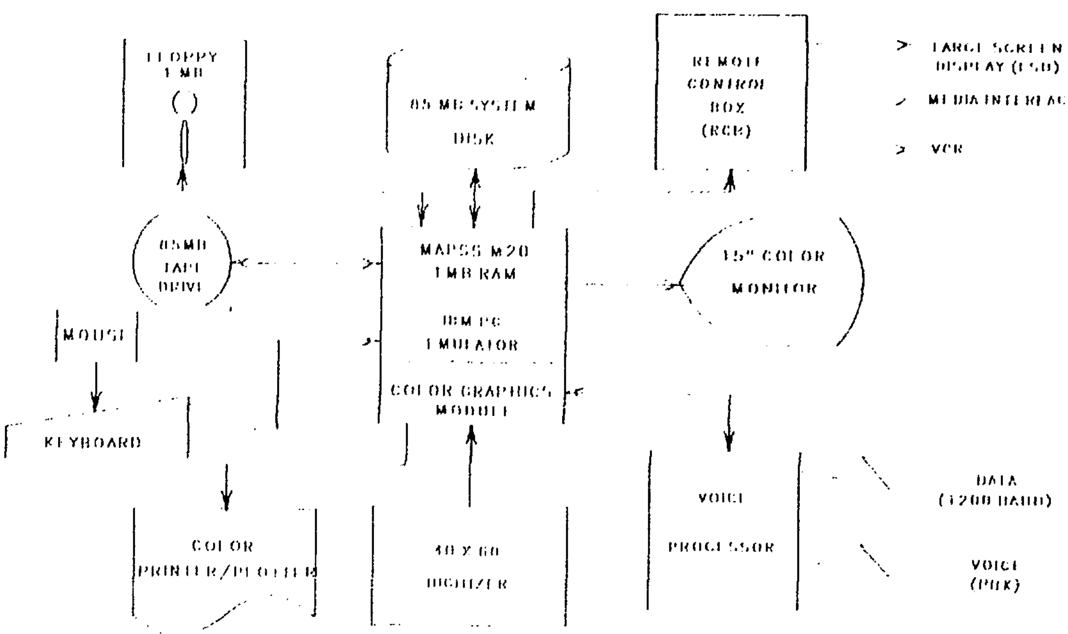
The MAPSS NEMS permits the establishment of a command, control and communications network which will allow the timely flow of information among participating parties. These may be any one of several levels of government: from federal to state to local government. The system will communicate with any of the previously mentioned agencies in any combination or sequence.

The MAPSS NEMS is a microcomputer-based system designed to aid in the command, control and coordination of emergency response teams in the event of an accident at a nuclear installation. Since the system is built around a microcomputer, it is portable in that if an Emergency Operations Center (EOC) has to be relocated, the system may be relocated with the personnel. The system would require a telephone line and 110V AC to become operational. The MAPSS NEMS uses a distributed processing approach which incorporates:

o a high-speed local area network;

o remote networking;

o voice and data transmission capabilities;



o multi-tasking real-time operating system;

o high resolution color graphics displays;

o up to 340M bytes of hard disk;

o color printer/plotters;

o graphic digitizers;

o video output drivers; and

o IBM PC compatiblity.

The software which drives the NEMS is a combination of a menu and function key driven system. The MAPSS NEMS takes full advantage of the real-time multi-tasking operating system in that most of the communications are done in background. The software is written in FORTRAN 77 and assembly language. The system responds to input from external computer links. The operator may initiate various functions, either through menus or the use of function keys. However, as defined by system requirements, certain functions are available only at predetermined times and are accessable only by authorized users. An example might be that the operator can run the plume model only when the situation has reached the Site Area Emergency level. The entire system is password protected: Passwords are used for two purposes, first, to restrict access to the system to authorized users only, and second, to restrict access to critical information or software routines. This feature insures that only authorized personnel have the capability to calculate evacuation recommendations, run plume models, or review the event log. The entire software system is implemented in a modular fashion; which permits the addition of new features and/or enhancements to the system without having to recompile the entire software system. For instance, the plume model to be used may not be decided on until the system design phase.

SYSTEM OVERVIEW

The managing utility has responsibility for informing the appropriate EOCs of an emergency situation and any subsequent escalation or closeout. The MAPSS NEMS monitors the managing utilities' Emergency Command Center (ECC) via a computerized link and will automatically notify the EOCs of any emergency situation. On command from the operator at the EOC, the Standard Operating Procedures (SOP) for the particular situation will be initiated. When the EOC has been informed of a change in conditions, it will inform the appropriate state agencies. The EOC has the additional responsibilty of keeping the news media aware of the situation and coordinating news releases for the public. The MAPSS NEMS will permit authorized users at the utility, state, and local level to review and compose guidance messages and media releases. All news releases can be viewed simltaneously at the appropriate sites; and once agreed on by all parties, the release is printed and distributed to the media. If the situation requires, the MAPSS NEMS can feed video directly to the news media via standard video (RGB) outputs. The image on the system's monitor would be the same one that would be released to the media. This allows important messages, evacuation routes and sectors, etc. to be viewed by the public. Information from the various radiation monitoring teams can be entered into the system and new contamination patterns can be calculated. This process is accomplished by using a contouring technique which will combine both real data from the field and the projected data from a plume model. In addition to monitoring the ECC's computer system and supplying data to the outlying operations centers, the MAPSS NEMS automatically maintains and updates an accurate record (Event Log) of all incoming and outgoing messages, as well as the acknowledgements to those messages. This Event Log is maintained at all times, not just when an emergency is in progress.

At the city/county level during an emergency the operators have the ability to display the location of the shelters, the status of those shelters, the evacuation routes, the pickup points, paired schools, and roadblock status. If during the operation of the system, a new incoming message is received, the operator will be notified by an audible tone and a flashing one-line message in the upper left hand corner of the screen. This feature insures that all messages are received and are available for recall. The city/county can also put its own data in their system that is pertenient locally such as city/county SOPs, phone list of key personnel, radio frequencies and call signs, equipment status, other city/county officals phone numbers, or.. whatever else is necessary for the city/county to manage a emergency. The city/county is supplied with data directly from the managing agency and coordinates its activities with the EOC.

If the communication link is lost between EOCs, each EOC has the ability to run in a stand alone fashion. It will update its Event Log with the exact time communications were lost and what actions were taken and by whom during the loss. When communications have been reestablished, the Event Log on the controlling master computer will be updated to reflect the actions taken at the EOC which had been "lost".

EXAMPLES OF THE SYSTEM

All of the examples used are from the State of Tennessee. Although Tennessee's reporting network may be somewhat different, the philosphy and data flow of the systems are very similar. The following examples illustrate the graphics capabilities of the system and demonstrate its flexibility.

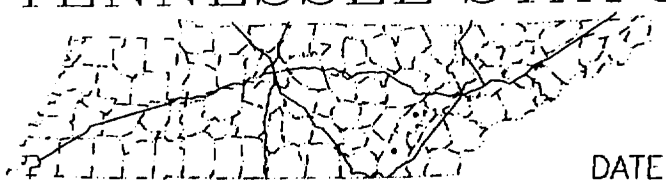
NORMAL MODE

The following is an example of what the screen would display during NORMAL condition. Notice that the date/time group will be updated as will the condition of the entire notification network. The system is displaying the name of the duty officer at the Emergency Operations Center (EOC). It also displays the location and status of all the outlying centers. In addition, the system displays the current condition of each of the nuclear power plants and the operating status of each individual unit. If the color of the site or the plant is green, the condition is normal; if the color is blue, the site or plant is offline; if the color is violet, the site or the plant is not responding to the computer's polling. The system will display the status map of the affected area on a large screen display (LSD) and it will be updated at regular intervals reflecting situation changes.

UNUSUAL EVENT MODE

This is an example of the screen when an Unsual Event occurs. The status of the plant has changed to yellow, the plant site as shown on the state map is now yellow, and the unit status at the plant is also yellow. At this time the system will start to update at a much faster rate. The system normally updates once every ten minutes. Once an Unusual Event has occurred, the update rate is increased to once every five minutes. Since update time varies with each system, the times given here are merely examples. As the situation continues to degrade, the update time will be reduced as needed until the system becomes fully dedicated to a particular event (full dedication would not take place until a General Emergency was declared). The notification network continues to check each outlying site

TENNESSEE STATUS MAP



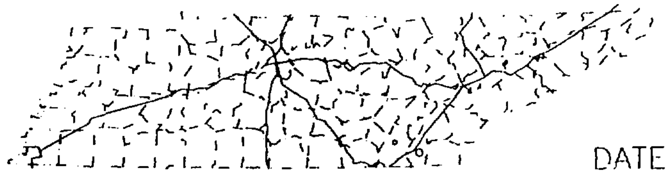
DATE: 23JAN86
TIME: 13:21 CST

STATE DUTY OFFICER: ANDY EDLY
NUCLEAR POWER PLANT STATUS
SEQUOYAH PLANT CONDITION: NORMAL
UNIT 1 100%
UNIT 2 OFFLINE MAINTENANCE
WATTS BAR PLANT CONDITION: NORMAL
UNIT 1 100%
UNIT 2 100%

STATE NOTIFICATION NETWORK:

SEOC	HAM EOC (SQN)	McMIN EOC (WBN)
	BRAD EOC (SQN)	MEIGS EOC (WBN)
CECC	FCC/RMCC (SQN)	RHEA EOC (WBN)
	JIC (SQN)	FCC/RMCC (WBN)
		JIC (WBN)

TENNESSEE STATUS MAP



DATE: 23JAN86
TIME: 13:30 CST

STATE DUTY OFFICER: ANDY EDLY
NUCLEAR POWER PLANT STATUS
SEQUOYAH PLANT CONDITION: NORMAL
UNIT 1 100%
UNIT 2 OFFLINE MAINTENANCE
WATTS BAR PLANT CONDITION: UNUSUAL EVENT
UNIT 1 100%
UNIT 2 100%
PRESS <GO>

STATE NOTIFICATION NETWORK:

SEOC	HAM EOC (SQN)	McMIN EOC (WBN)
	BRAD EOC (SQN)	MEIGS EOC (WBN)
CECC	FCC/RMCC (SQN)	RHEA EOC (WBN)
	JIC (SQN)	FCC/RMCC (WBN)
		JIC (WBN)

and report the status of each even if the site is not involved in this particular event.

The next picture is the Standard Operating Procedure that would come up on the operator's display when he/she presses the <GO> key on the keyboard. Some of the sites have responded and a couple have not. Those sites which have not responded would be displayed in violet. At this time the operator would verbally contact the agencies or persons not responding to the alert. As the respondents react, their status will return to the normal color code which is white. At the same time, the event log is updated each time someone is notified and it will also record who acknowledged the message. However, if the operator must verbally contact any party, the operator must log manually the time of notification. To notify a selection, the operator simply places the screen's cursor on the selection and presses the <GO> key. In this example Rad Health and the Highway Patrol have not been notified. It is obvious that the system continues to monitor the entire notification network (SNN).

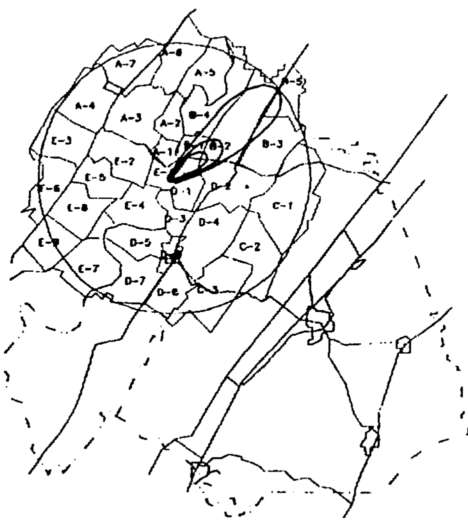
EVACUATION RECOMMENDATIONS

This picture shows evacuation recommendations based on the predicted plume from the plume program. The MAPSS NEMS will calculate which evacuation sectors within the 10 mile EPZ will be covered by the projected plume and make evacuation recommendations based on those projections. The plume program to be used would be the one selected by the utility and integrated into the MAPSS NEMS. The system will give evacuation recommendations based on the time intervals selected during the plume calculation process. In this example the plume has been calculated for 0 to 4 hours, 4 to 8 hours, and greater than 8 hours. The system has identified those sectors which would be affected by the plume and has listed them to the right of the picture and color coded them to correspond with the color coding of the plume. It should be noted that MAPSS does not write plume models; MAPSS will integrate the plume model that the utility selects.

CONCLUSION

Even though the MAPSS NEMS is designed primarily for nuclear emergency management applications, the system can be utilized to manage other emergenices such as natural diasters, hazardous materials spills, and other catastrophic

EVACUATION RECOMMENDATIONS



EVACUATE THE FOLLOWING SECTORS:

0 4 HOURS

A1, B1, B2, D1, D2, E1

4-8 HOURS

A1, B1, B2, D1, D2, E1

8+ HOURS

A1, B1, B2, B3, B4, B5
D1, D2, E1

events that would require real-time accurate and responsive emergency management.

Although the MAPSS NEMS is designed primarily as a nuclear emergency management system, it is versatile enough to be utilized in day to day operations. The word processor, spread sheet, and graphics, which are standard in our system, can be used for reports, graphs, equipment lists, maintenance and training schedules.

MAPSS has developed the most comprehensive and responsive system available. In order to keep pace with the increasing demands and complexity of emergency management, MAPSS continues to enhance the system. Some of the future enhancements may include:

o interface with FEMA's Integrated Emergency Management Information System (IEMIS); and

o use FM radios and other technologies as a backup to transfer data between sites in the event of phone line failure;

o a version of the NEMS which will run on the IBM PC and compatibles.

The MAPSS Corporation is committed to a systems approach to the emergency management challenge. MAPSS will continue to be at the forefront of emergency management systems by offering a comprehensive cost effective turnkey system which includes:

o <u>delivery</u> and <u>installation</u> of the hardware and software;

o <u>training</u> the operators and providing the technical assistance;

o <u>expanding</u> the system as requirements dictate;

o <u>upgrading</u> the system as enhancements come online.

A Comparison of Computerized Dose Projection Models and Their Impact on Protective Action Decision Making

Susan M. Reilly

ABSTRACT. The increased use of computers at nuclear power plants provides more accurate and timely assessment of conditions and consequences during emergency situations. Dose projection has been one of the prime beneficiaries of this new technology, and the resulting calculations have a direct impact on protective action decision-making. In this paper, a quantitative comparison of three dose projection models illustrates that projected doses can vary from factors of 2-3 up to factors of 100 or more for similar input conditions. The impact on protective action recommendations is great, resulting in recommendations that vary from "no action" to "evacuation."

I. OBJECTIVES

This project has two objectives: first, to quantitatively compare three different dose projection models; and second, to compare the protective action recommendations that might be made based on each set of dose projection results. The models which have been selected for this comparison are: (1) the U.S. Nuclear Regulatory Commission's IRDAM model; (2) a straight-line Gaussian (Class "A") model currently in use at a U.S. utility; and (3) a time-dependent (Class "B") model, also available at a U.S. utility.

The IRDAM model runs on a personal computer, while both utility models run on VAX 11/780 mainframes. Each of these models are described in the following section of this report.

These particular models were chosen to represent a cross-section of those dose projection methods currently available in the United States. IRDAM provides a particularly applicable point of reference since it is the model used by the NRC site dose assessment team when they respond to emergencies at operating plants. Since the NRC would be involved in discussions of protective action recommendations, it is useful to know how the NRC dose projections might differ from those of the utility; and therefore, how protective action decision-making could be affected.

II. DESCRIPTION OF MODELS

A. Model 1: IRDAM

The Interactive Rapid Dose Assessment Model (IRDAM) is the model currently used by the U.S. NRC site dose assessment team during emergencies at nuclear power plants. Version 5.0 (August, 1985) was used in this analysis.

IRDAM provides a straight-line Gaussian projection based on user-supplied input data. The IRDAM program is used for all U.S. utilities, and is designed to reference a given file for site-specific data upon the user entering the site name. (Such site-specific data might be: type of plant (BWR/PWR), number of units on site, etc.) Other input, such as current meteorological and release information, is provided by the user. The output of the IRDAM program is in tabular form, with both whole body and thyroid dose rates provided at several pre-determined or user-supplied distances.

B. Model 2: Utility "A"

Model 2 is a straight-line Gaussian (Class "A") model which produces transport and diffusion estimates in the plume exposure Emergency Planning Zone (EPZ). The model is designed to run on real-time data: the computer is hard-wired directly into the plant's meteorological tower and effluent monitoring system. It is possible, however, to override the automatic mode and manually input data.

Model 2 is structured around an accident "menu" which permits the user to provide as much or as little information as is available. The more specific the inputs, the more accurate the output, but the computer has a set of default values which are used when no user inputs are provided (e.g., accident type, time after shutdown, isotopic mix). Dose projection results are provided either in tabular or in color graphic form. Graphics include a site map and a multicolor plume plot.

C. Model 3: Utility "B"

Model 3 is a time-dependent (Class "B") model which can represent actual spatial and temporal variations of plume distribution and can provide estimates of relative deposition within the ingestion EPZ. No "centerline" dose values are calculated; values are integrated within individual segments of the plume.

Aside from the dose projection calculations themselves, Model 3 and Model 2 are very similar. Like Model 2, Model 3 also runs on real-time data, and is menu-driven by the user. Data are available in tabular or graphic form.

Both Model 2 and Model 3 have several characteristics which are common to each other but which differentiate them from IRDAM (see Table 1). Key differences are:

- o Isotopic mix is a function of accident type rather than being based on a standard iodine: noble gas ratio.
- o Decay time between reactor shutdown and start of release is incorporated on an isotope-by-isotope basis.
- o Downwind deposition and depletion of the plume is considered.

III. VARIABLES IN THE ANALYSIS

Each of the three models was run for the following sets of input conditions:

Wind speeds:	4, 10 miles per hour
Stability Classes:	A through G
Filtered Release:	Loss of Coolant Accident
Unfiltered Release:	Steam Line Break
Time Between Shutdown and Start of Release:	0 and 2 hours
Wind Direction:	Constant during release
Release Duration:	2 hours
Gamma Dose Type:	Semi-infinite

IV. RESULTS OF DOSE PROJECTION COMPARISONS

Table 3 summarizes the ratios of projected dose rates from IRDAM to those of Model A and Model B for the input conditions indicated on the table. Dose projections vary from factors of 1-3 up to factors of several orders of magnitude. Sources of discrepancy can be divided into three categories:

1) Dispersion

Figures 1 and 2 illustrate that X/Q variations between models account for up to a factor of 3, with IRDAM being consistently conservative.

2) Source Terms

Models "A" and "B" use a decay-corrected source term of 15 specific isotopes (13 noble gas and 2 iodine). Dose rates are calculated for each isotope, then summed to obtain the total dose rate. For the IRDAM runs, noble gas to iodine fractions were provided that were identical to those being used by Models A and B (Table 2); however, since specific isotopes were not indicated, IRDAM results depend on the average noble gas and iodine dose factors being assumed in the IRDAM program.

TABLE 1. MODEL SUMMARY MATRIX

Characteristic	I	A	B
Straight Line Gaussian ('A' model)	X	X	
Time Dependent (Class 'B' model)			X
Whole Body Gamma Dose Calcs	X	X	X
Adult Thyroid Dose Calcs		(1)	(1)
Child Thyroid Dose Calcs	X	(1)	(1)
Elevated Releases	(2)		
Ground Level Releases	X	X	X
R. G. 1.109 Dose Factors	X	X	X
Time Between Reactor Shutdown and Release Considered	(3)	(3)	(3)
Gross I:NG Ratio Incorporated	X		
I:NG Ratio Determined by Isotope		X	X
Downwind Decay of Plume	X	X	X
Deposition Considered			X
Lake Effect Capability			X
Finite Gamma Dose			X
Semi-Infinite Gamma Dose	X	X	

I = IRDAM, A = Utility "A" model, B = Utility "B" model.

NOTES:

('X' indicates model has that characteristc)

(1) User chooses adult or child.
(2) IRDAM calculates ground level releases as elevated, with a 10-meter release height.
(3) All three methods consider time after shutdown, but in different ways.

TABLE 2. PERCENT IODINE IN EFFLUENT

Accident Type	% I of Total I + NG 0 (hours after shutdown)	2	8
Filtered (Loss of Coolant)	2.1	4.5	5.8
Unfiltered (Steam Line Break)	80.2	96.6	98.0

LOSS OF COOLANT ACCIDENT

Stability		IRDAM: A Hours After Shutdown		IRDAM: B Hours After Shutdown	
		0	2	0	2
B	WB	1.2	1.2	2.0	1.3
	CT	0.78	1.1	0.78	1.1
D	WB	0.46	0.95	2.0	1.2
	CT	0.58	0.85	0.95	1.4
F	WB	0.36	0.36	2.4	2.0
	CT	0.79	1.1	1.6	3.5

STEAM LINE BREAK

Stability		IRDAM: A Hours After Shutdown		IRDAM: B Hours After Shutdown	
		0	2	0	2
B	WB	0.16	0.028	0.041	0.0066
	CT	0.97	1.6	0.95	1.6
D	WB	0.063	0.031	0.029	0.0061
	CT	0.70	1.2	1.2	2.0
F	WB	0.048	0.0091	0.028	0.0050
	CT	0.94	1.6	1.9	5.2

TABLE 3. Summary of Site Boundary (0.7 miles) Dose Projection Comparisons. Values are the ratios of IRDAM results to those of Models A and B, as indicated. (WB = Whole body, CT = Child Thyroid).

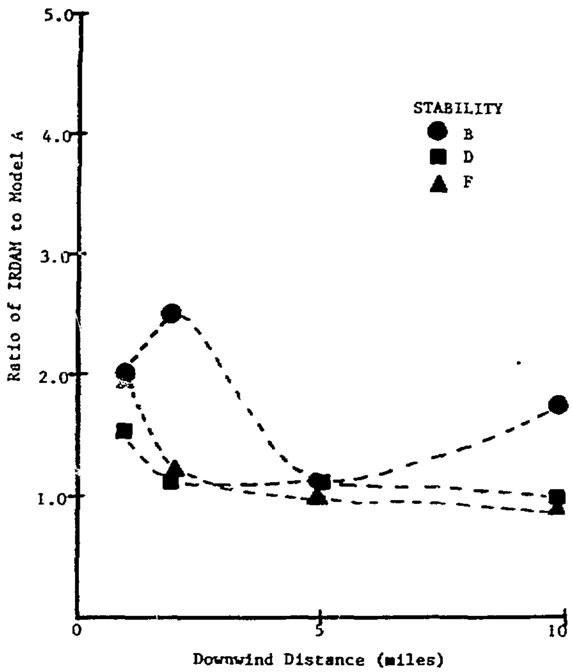


Figure 1. Ratio of X/Q values between IRDAM and Model A for various stability classes. Ratios showed less than 3% variation between 4 mph and 10 mph.

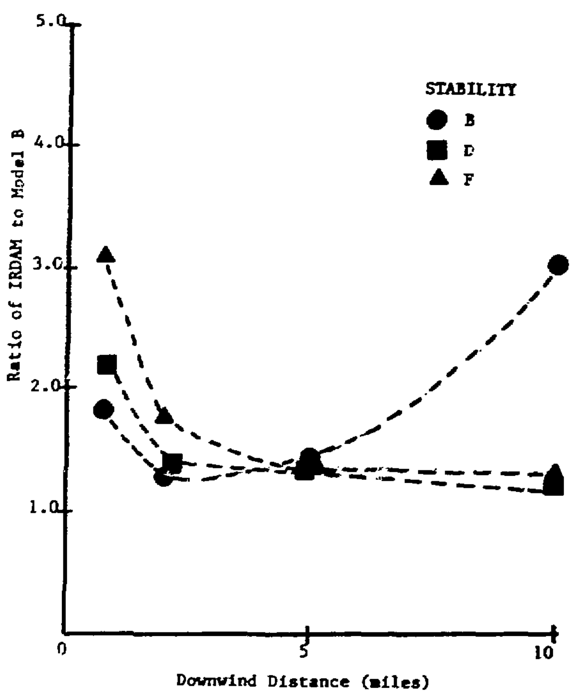


Figure 2. Ratio of X/Q values between IRDAM and Model B for various stability classes. Ratios showed less than 3% variation between 4 mph and 10 mph.

3) Other Factors

a. The major discrepancy between models is seen for the case of whole body dose resulting from a steam line break. This discrepancy results from the fact that Models A and B include the iodine isotopes in their calculation of whole body dose, while IRDAM does not. In the case of a filtered release, this omission is not significant; however, in a release that consists of greater than 80% iodine, the whole body dose resulting from iodine is substantial. It is important to realize, however, that in a plume that is 80% iodine, Protective Action Recommendations (PARs) will be driven by thyroid dose, not by whole body dose.

b. Finite vs semi-infinite dose models, deposition, and depletion of the plume all contribute to differences between results of IRDAM and the other models. For the cases examined in this paper, these factors contribute to less than a factor of two difference in results (as evidenced by comparing dose ratios at the site boundary to those at ten miles).

V. IMPACT ON PROTECTIVE ACTION RECOMMENDATIONS

The basic guidance for protective action decision-making is found in EPA-520/1-75-001 "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents." Protective actions are based on total projected dose (i.e., dose rate multiplied by the expected duration of the release), and when this dose exceeds certain triggerpoints, actions to protect the public may be required. These Protective Action Guides (PAG's) and their corresponding actions are as follows:

Total Dose (Rem)		Action Considered
Whole Body	Child Thyroid	
dose $<$ 1	dose $<$ 5	none
1 $\leq$ dose $<$ 5	5 $\leq$ dose $<$ 25	shelter
5 $\leq$ dose	25 $\leq$ dose	evacuate

(Note: non-metric units are used herein as both the dose models and the PAG's themselves are expressed in these units. For reference, 1 Rem = 0.01 sievert, 1 Ci = 3.7E10 becquerel, and 1 mile = 1609 meters.)

Figures 3 and 4 illustrate the effects that different model results have on protective action decision-making. In Figure 3, results were compared for the case where there is good agreement (less than 20% difference) between models: IRDAM vs. Model A for the case of a LOCA, 0 hours after shutdown, B stability, 4 mile per hour wind, 1000 Ci/sec release rate, 2 hour duration. In the case of the whole body dose, both models exceed the PAG limit for shelter at the site boundary, but require no action at other downwind distances. One would expect a similar protective action recommendation (PAR) to be made in both cases; e.g., shelter to 2 miles, no action beyond that. In the case of the thyroid dose projection, however, Model A exceeds the PAG limit for shelter at 5 miles while IRDAM does not. This could result in a PAR based on IRDAM to evacuate to 2 miles, shelter to 5 miles, no action beyond that; while Model A indicates evacuation to 2 miles, shelter from 2 miles to 10 miles, and no action beyond 10 miles.

Now examine the impact on PARs when models produce signficantly different projected doses. Figure 4 illustrates the integrated whole body and thyroid doses for the case of IRDAM vs. Models A and B, steam line break, 2 hours after shutdown, F stability, 4 mph wind, 660 Ci/sec release rate for whole body calculation, 1.6 Ci/sec for thyroid calculation, 2 hour release duration. In the case of the whole body dose, IRDAM predicts that no action is required - while Models A and B indicate that the evacuation PAG is exceeded beyond 5 miles and the shelter PAG is exceeded beyond 10 miles. In the case of the thyroid dose, IRDAM predicts that the evacuation PAG is exceeded beyond the site boundary (0.7 miles), while the shelter PAG is exceeded beyond 2 miles. In contrast, Model B results show that the evacuation PAG is never exceeded, and that the shelter PAG is exceeded between the site boundary distance and 2 miles, while no action is required beyond that distance. Model A results are similar to IRDAM's

V. SUMMARY

This paper illustrates the dramatic effect that the use of different dose projection models can have on protective action decision-making. Even though models may appear similar on the surface, subleties such as time-dependent dose factors contribute to widely-varying dose projections - and consequently to widely-varying protective action recommendations. To avoid confusion during emergencies, dose assessment personnel at both the utility and governmental agencies should have a thorough understanding of the capabilities and limitations of their particular models.

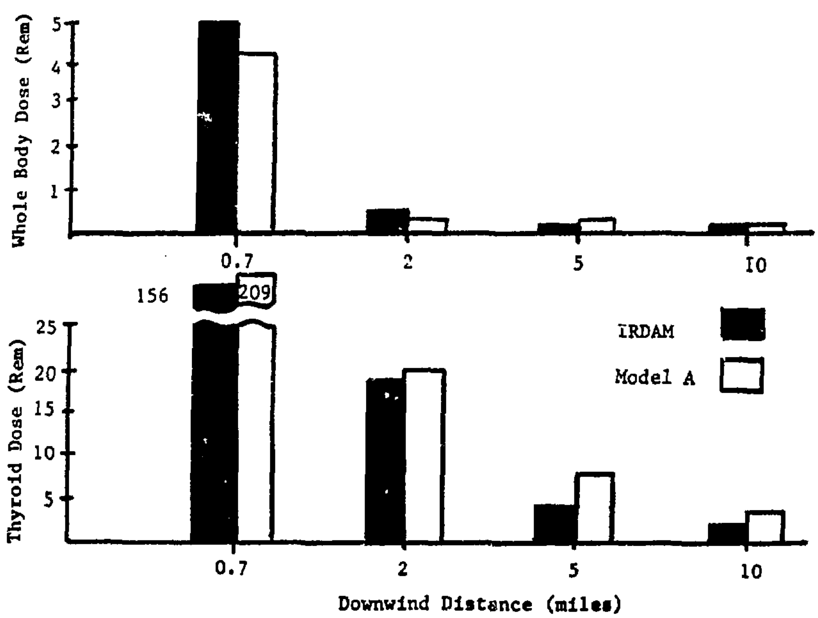


Figure 3. Comparison of Projected Doses Relative to PAG Limits. Case 1: LOCA, 0 hours after shutdown, B stability. Case 1 results represent agreement of projected doses to within 20%, yet there is a difference in which PAG triggerpoints are met at each distance.

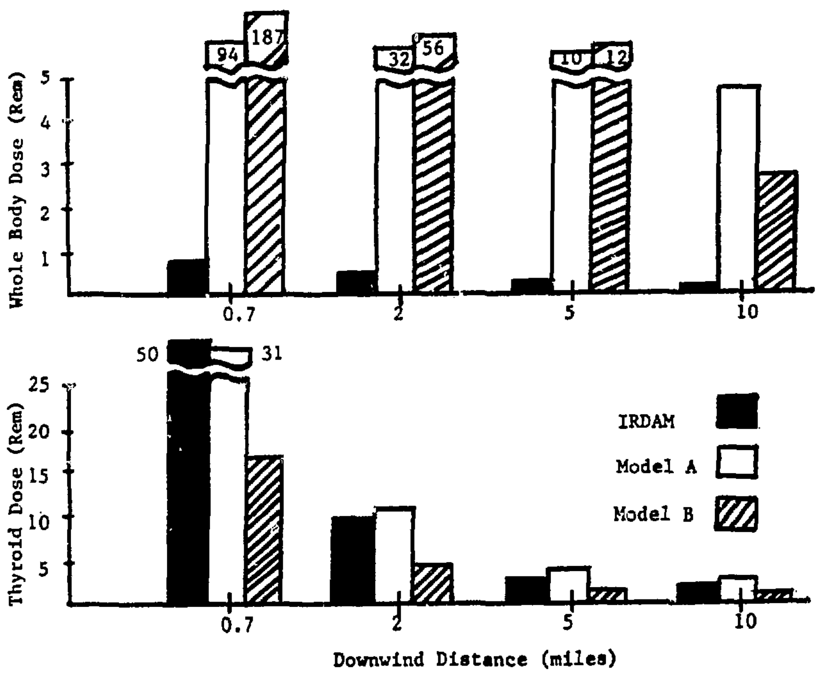


Figure 4. Comparison of Projected Doses Relative to PAG Limits. Case 2: steam line break, 2 hours after shutdown, F stability. The marked differences in projected doses between these models result in widely varying protective action recommendations.

An Importance Ranking of Various Aspects of Off-Site Radiological Emergency Preparedness

John W. Hockert and Thomas F. Carter

ABSTRACT

Under contract to the Edison Electric Institute, IEAL developed a method to assess the relative importance of various aspects of offsite radiological emergency preparedness. The basic approach involved structuring the 35 objectives that the Federal Emergency Management Agency expects offsite emergency planners to demonstrate during nuclear power plant emergency preparedness exercises into a hierarchy based upon the emergency response capabilities they support. The analytical hierarchy process (AHP) was employed to derive the quantitative relative importance of each of the 35 objectives based upon its contribution to the overall capability of offsite agencies to assist in protecting public health and safety in the event of an emergency at a nuclear power plant. The judgments of a cross-section of state and local emergency planners, federal regulators, and intervenors were solicited to rank the 35 objectives.

I. INTRODUCTION

At present, the primary mechanism employed by federal evaluators to determine the adequacy of offsite emergency preparedness is observation of the full-scale exercise conducted at least once every two years at each nuclear power reactor. Although the Federal Emergency Management Agency (FEMA) has continued to refine the methodology used during these evaluations since taking over this responsibility in 1979, it is generally recognized that additional improvements can be made in the evaluation methodology. In particular, a useful adjunct to the current evaluation methodology would be a systematic method to recognize and incorporate into the exercise evaluation process the considerable differences in the real importance among the NUREG-0654 planning standards and evaluation criteria. Without such a method, there is the possibility that utility and state and local government attention and resources will be directed toward relatively insignificant exercise deficiencies at the expense of real emergency preparedness problems. In addition, this method would also serve to foster a consistent understanding among the FEMA regions and evaluators as well as state and local government agencies and utility decision makers as to exactly what constitutes adequate emergency preparedness. Furthermore, such a method could also be used to best allocate governmental and utility radiological emergency preparedness resources and minimize compliance costs to rate-payers.

This paper describes a means to achieve the first part of this objective: that is, to systematically assess the relative importance of the various aspects of offsite radiological emergency preparedness. The methodology presented builds upon the modular exercise evaluation concept developed by FEMA to create a hierarchy describing required offsite emergency preparedness capabilities. Based upon this hierarchy, proven analytical methods for quantification of expert judgment were employed to develop a structured interview format to be used with experts in the field of radiological emergency preparedness. Based upon their qualitative judgements, expressed as responses to a carefully selected set of questions, the methodology provided a means to calculate a quantitative measure of the overall importance of each of the 35 standard objectives that FEMA expects offsite emergency planners to periodically demonstrate during exercises. The methodology also provided quantitative measures of the consistency of the qualitative judgments leading to the relative importance measures.

II. METHODOLOGY

The evaluation of the level of offsite agency preparedness to respond to nuclear power plant emergencies is a difficult and complicated undertaking. Proper emergency response may well depend upon the actions of from dozens to, conceivably, hundreds of individuals. The actions of these individuals throughout the emergency are interrelated in a complex manner that depends upon both the specific nature of the emergency and the consequences of previous emergency response actions. In evaluating such a complex system, it is necessary to foster and to maintain an awareness of the importance of the proper functioning of each component of emergency preparedness to the overall capability of offsite agencies to protect the public health and safety. Otherwise, there is a distinct possibility that evaluators will concentrate their attention on activities that they are expert in or that they find interesting, independent of the importance of such activities.

In developing their modular approach for evaluating radiological emergency preparedness exercises (FEMA, 1983), FEMA has made great strides in structuring the evaluation process. This approach identifies a set of 35 standardized objectives to be demonstrated during radiological emergency preparedness exercises. These objectives have been drawn from those elements of NUREG-0654/FEMA-REP-1, Rev. 1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, that are both observable during exercises and pertinent to state or local governments. The next logical step in this structuring process is the identification of the relative importance of each of these 35 objectives.

The identification of the relative importance of each of the 35 objectives is also made difficult by the complexity of their interrelationship with one another. This process is made even more difficult because the requisite offsite emergency response capabilities differ considerably for different types of nuclear plant emergencies. Therefore, the capabilities required to achieve certain objectives may be quite important in responding to extremely unlikely accidents and yet be unimportant for the great majority of nuclear plant emergencies. In making a determination of overall relative importance, such capabilities must be compared with other capabilities of moderate importance for virtually all nuclear plant emergencies. Thus, the method employed to determine the relative importance of the 35 objectives must facilitate such complex comparisons. Furthermore, in order to be useful, the method must be reasonably simple to use and understandable. These considerations strongly suggest the use of a methodology that makes maximum use of the considered judgment of individuals experienced and knowledgeable in radiological emergency preparedness.

An extensive body of literature exists describing proven methods to consistently account for and effectively use expert judgment in importance ranking. (Freeling, 1982, Saaty, 1977, 1978 a,b,) Thomas L. Saaty, formerly of the University of Pennsylvania's Wharton School and now with Pennsylvania State University, has published several journal articles that address a structured method for developing quantitative measures of the relative importance of various objectives supporting the same goal. The first stage of this method is to identify the goals that each objective supports. This leads to the construction of a hierarchy.

The use of such a hierarchy has several advantages. First, the hierarchy provides a meaningful integration of related objectives. This not only focuses the evaluator's attention on those aspects of the related objectives that contribute most directly to their goal, but it also relieves the evaluator of the difficult problem of directly comparing objectives that support disparate goals. This increases both the effectiveness and the efficiency of the process. Second, the use of a hierarchy structures the evaluation so that issues related to system performance occur at lower levels, while policy issues dominate at the higher levels. Thus, information about the relative importance of higher level goals can be effectively obtained from policy makers, while information about the relative importance of individual objectives in achieving those goals can be obtained from individuals more familiar with system details. Third, the use of hierarchies increases the reliability and flexibility of the process. The achievement of the overall goal is divided among the levels of the hierarchy in such a manner that each achieves a portion of the goal and the totality meets the overall goal. Therefore, changes to the manner in which one portion of the goal is met (i.e., changes to one part of the hierarchy) do not affect the analysis for those parts of the hierarchy supporting other portions of the overall goal.

The hierarchy constructed to support the evaluation of the relative importance of FEMA's set of 35 standardized objectives to be demonstrated during radiological emergency preparedness exercises was derived by identifying the three capabilities necessary to protect public health and safety:

1. *The capability to obtain information necessary to determine actions to be taken to protect public health and safety;*
2. *The capability to maintain the command and control necessary to support effective decision-making and incident management; and*
3. *The capability to implement appropriate protective actions when the decision is made to do so.*

These three capabilities were then analyzed into subordinate capabilities until, eventually, all 35 of the objectives were linked with the capabilities that they support. The detailed hierarchy and a listing of all 35 objectives are included in the project report (IEAL, 1985).

The next stage of the method involved the estimation of the knowledgeable individual's underlying estimate of the importance of each goal or objective through use of ratio judgments, which are a form of pairwise comparisons. This was accomplished by having the individual answer a series of questions regarding the relative importance of each pair of objectives. The questions are derived directly from the hierarchy, and the possible answers are taken from a set of preselected responses that have evolved from Saaty's research. The set of questions used to rank the importance of the 35 objectives and the possible answers to these questions are presented in the project report (IEAL, 1985). The method for quantifying these answers and using them to derive an individual quantitative importance weighting for each of the 35 objectives is also discussed in the referenced report (IEAL, 1985).

Expert judgments of the relative importance of objectives supporting each higher level goal and the goals themselves were solicited by sending detailed questionnaires to FEMA and Nuclear Regulatory Commission (NRC) headquarters and regional staff, Regional Assistance Committee (RAC) members, other federal, state, and local emergency planners, and members of public interest groups. Thirty-five responses were received from individuals representing a reasonable cross-section of these organizations. Rankings were derived based upon each individual's responses, and an overall composite ranking was developed, based upon each individual's responses within his or her areas of expertise. In addition, the responses were analyzed for internal consistency and to determine the degree to which the responses were correlated with the respondent's organizational affiliation.

III. RESULTS

The individual respondent's detailed rankings of the 35 exercise objectives were found to vary. This is not surprising in light of the complexity of emergency preparedness and the extent to which the relative importance of specific capabilities during a given emergency depends upon the detailed accident scenario and site conditions. The observed variation did not appear to be significantly related to the respondent's organizational affiliation, geographical location, or area of expertise. This variation indicates that the specific relative importance weights provided by the AHP should be regarded as somewhat uncertain.

However, in reviewing the overall response, a pattern did emerge. While the detailed rankings varied, there was general consensus on a group of seven exercise objectives that were considered very important by a large majority of respondents and a group of ten exercise objectives that were considered relatively unimportant by a large majority of respondents. With one or two exceptions, the uncertainties in the individual rankings were such that objectives could be ambiguously assigned to one of these two groups or to a third group of objectives of moderate importance.

The seven objectives that were considered very important by the large majority of respondents were, in approximate order of decreasing importance:

Ability to project dosage to the public via plume exposure (Objective 10);
Ability to communicate with all appropriate locations, organizations, and field personnel (Objective 5);
Ability to make decisions and coordinate activities (Objective 3);
Ability to mobilize staff and activate facilities promptly (Objective 1);
Ability to project dosage to the public via ingestion exposure (Objective 11);
Ability to monitor and control emergency worker exposure (Objective 20); and
Ability to evacuate onsite personnel (Objective 23)

The ten objectives considered relatively unimportant by the large majority of respondents were, in approximate order of decreasing importance:

Ability to administer potassium iodide to the general public (Objective 22*)
Ability to brief the media in a clear, accurate, and timely manner (Objective 24);
Ability to establish and operate rumor control (Objective 26);
Adequacy of facilities for mass care of evacuees (Objective 28);
Ability to deal with impediments to evacuation, such as inclement weather (Objective 16);
Ability to coordinate information releases (Objective 25);
Adequacy of procedures for registration and radiological monitoring of evacuees (Objective 27);
Ability to control access to evacuated areas (Objective 17);
Ability to estimate total population exposure (Objective 34); and
Ability to determine and implement

* FEMA's original Objective 22, the ability to administer potassium iodide, was separated, for this analysis, into the ability to administer potassium iodide to emergency workers and the ability to administer it to the general public.

recovery and reentry measures (Objective 35)

The remaining objectives were considered to be of moderate importance by the majority of respondents. The variation in the relative importance accorded the objectives in this group by the various respondents was greater than the variation seen in either of the other two groups.

IV. CONCLUSION

The results of this analysis have potentially far reaching implications for the conduct and evaluation of offsite radiological emergency response exercises. For example, it is reasonable to expect that those objectives considered very important would be exercised most frequently, evaluated most thoroughly, and, if not demonstrated satisfactorily, would be the basis of a Category A deficiency or negative finding. Likewise, the relatively unimportant objectives would be exercised infrequently, accorded little evaluation effort, and, if not demonstrated satisfactorily, would rarely result in anything more severe than a citation as a correctable weakness. The moderately important objectives would, on average, receive moderate emphasis or be emphasized within a reduced emergency planning zone, or in specific cases, be classed as relatively important or unimportant depending upon the detailed local conditions or exercise scenario considerations.

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A Goal-Oriented Functional Tree Structure for Nuclear Power Plant Emergency Preparedness

Richard V. Calabrese and Marvin L. Roush

ABSTRACT Guidelines for development and implementation of emergency response plans do not provide the planner/implementer with an adequate overview of the functions to be achieved or a measure of their relative importance. To provide a framework in which this importance can be recognized, understood and quantified, a logical goal oriented tree structure has been developed which integrates and gives a clear visual representation of the functions required to meet the emergency preparedness objective. The tree considers a spectrum of both high and low probability events which may require mitigation both onsite and offsite.

The ultimate objective: to Minimize the Ill Effects of a Nuclear Power Plant Incident is satisfied by functions concerned with prevention and mitigation of human injury and property damage. A complete and detailed structure which specifies the subfunctions and success paths which satisfy these functions has been developed. Institutional activities such as planning, training, procedure development, monitoring and decision making do not enter the tree directly. Instead, the logic structure defines the extent to which these activities must be considered and the information systems and decision models required for successful implementation of the plan.

The top structure of the tree is presented and a few branches are considered in detail. The impact of institutional activities, information systems, etc. is discussed. Tree quantification is considered.

I. INTRODUCTION

Numerous documents provide guidance toward development and implementation of nuclear power plant emergency response plans. Most are concerned with the adequacy of personnel, supporting facilities and institutional activities performed in support of the plan. These activities include contingency planning, task organization, procedure development and specification, monitoring and dose projection, and practice drills to aid in plan maintenance and improvement. Unfortunately, such documents do not provide the planner/implementer with a good basis for understanding the functions to be achieved and a clear definition of their overall importance toward attaining the ultimate objective. To provide a framework within which this "importance" can be recognized, understood and quantified, we have developed a logical tree structure which integrates and gives a clear visual representation of the functions required to meet the emergency preparedness objective. Rather than focus solely upon low probability, high consequence events with radiological impact on the public, we consider the spectrum of events which may require mitigation both onsite and offsite.

The goal oriented tree presented herein is an outgrowth of the logic structures defined as part of the Top-Down Integrated Approach to Safe, Reliable, Economic Nuclear Power (e.g., Combustion Engineering, 1982; Technology for Energy Corp., 1982). The approach and results represent refinements of a detailed functional classification performed for the Dept. of Energy (DOE) by these authors (University Research Foundation, 1983) as part of the DOE Information Systems Project. The structure provides an easily understood overview of all required functions and their interrelationships, and a logical framework for quantitative evaluation. It is anticipated that the model can be a useful training tool to allow an individual to view specific parts of a plan in proper context. It might also be useful to provide a contextual overview of regulations to allow a proper perspective of the need for and the contribution of a single regulation to the overall objective, as well as the impact of the regulation upon achievement of other functions.

We have attempted to structure this goal tree according to the following simple guidelines: For each box, one can look "down" in the tree to see how the functional objective is accomplished. One can look "up" in the tree to see why the functional objective is required. Near the top of the tree, each function is specified in a very general way. Subfunctions or components of a higher level goal are then developed at lower levels. The lowest levels represent one or more success paths by which specific objectives can be accomplished. Effort has been made to address issues in a generic way that is applicable to any plant and to account for all possible contingencies, except sabotage. Therefore, at lower levels not all subfunctions/success paths will be relevant at a specific plant.

II. OVERVIEW OF TREE STRUCTURE

The top structure, shown in Figure 1, states the ultimate goal of emergency preparedness; that is, to Minimize the Ill Effects of a Nuclear Power Plant Incident that could potentially threaten the health, safety and economic well being of both the public and plant personnel. The objectives to be accomplished are prevention and mitigation of human injury and property damage. The functions which address these objectives can be classified as either preventative (pre-injury) or mitigative (post-injury). Mitigative functions are included since any realistic criteria for success must accommodate some casualties and/or property damage, if only due to the initiating event.

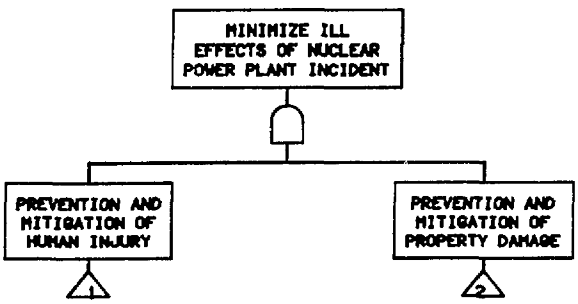


Figure 1 - Top Tree Structure

Portions of the tree that are presented herein are each assigned a number. This number appears inside a triangle located to the left of the top box of that branch. If the lowest boxes of that branch contain a triangle beneath them, then the number inside that triangle specifies the tree number on which it is continued.

The functions that satisfy the ultimate goal are given in Trees 1 and 2. These are:

a) Prevention of Human Injury. Plant personnel and the public must be protected from conventional injury and from exposure to radiation and toxic chemicals.

b) Prevention of Psychological Stress. The psychological impact on the public from an incident of any degree is to be minimized.

c) Human Casualty Control (Mitigation). People injured as a result of conventional accidents and/or exposure to radiation or toxic chemicals must be identified, treated and otherwise cared for both onsite and offsite.

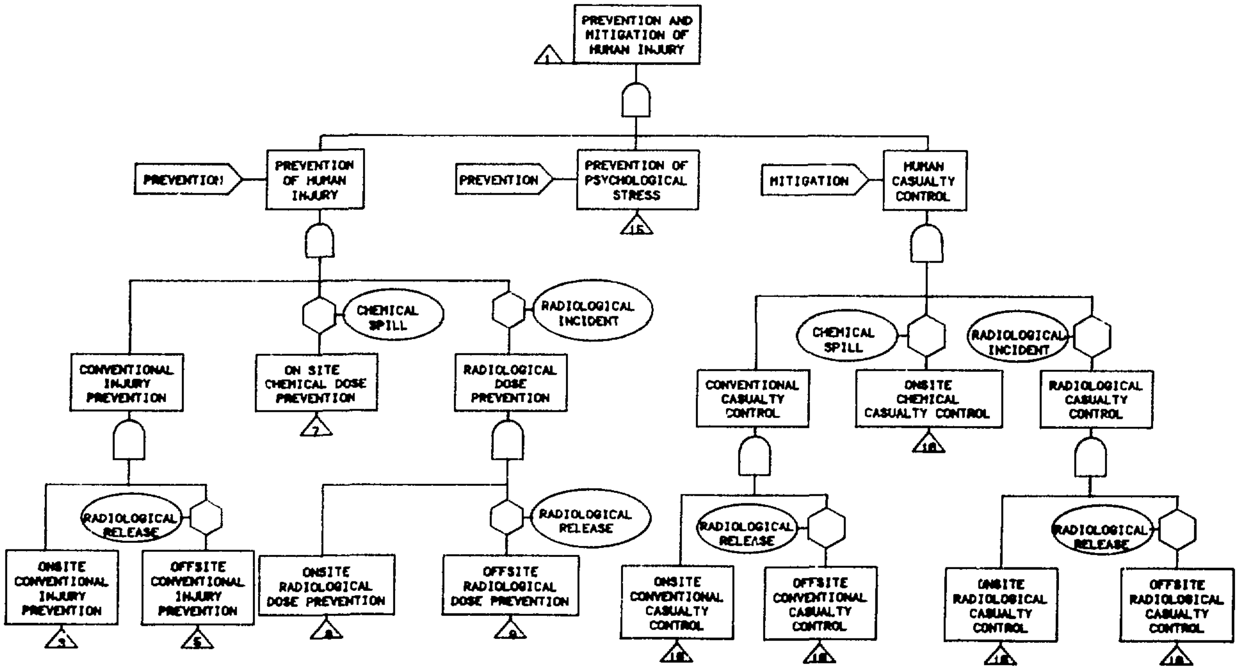
d) Prevention of Property Damage. Damage to plant and public property is to be prevented during an incident.

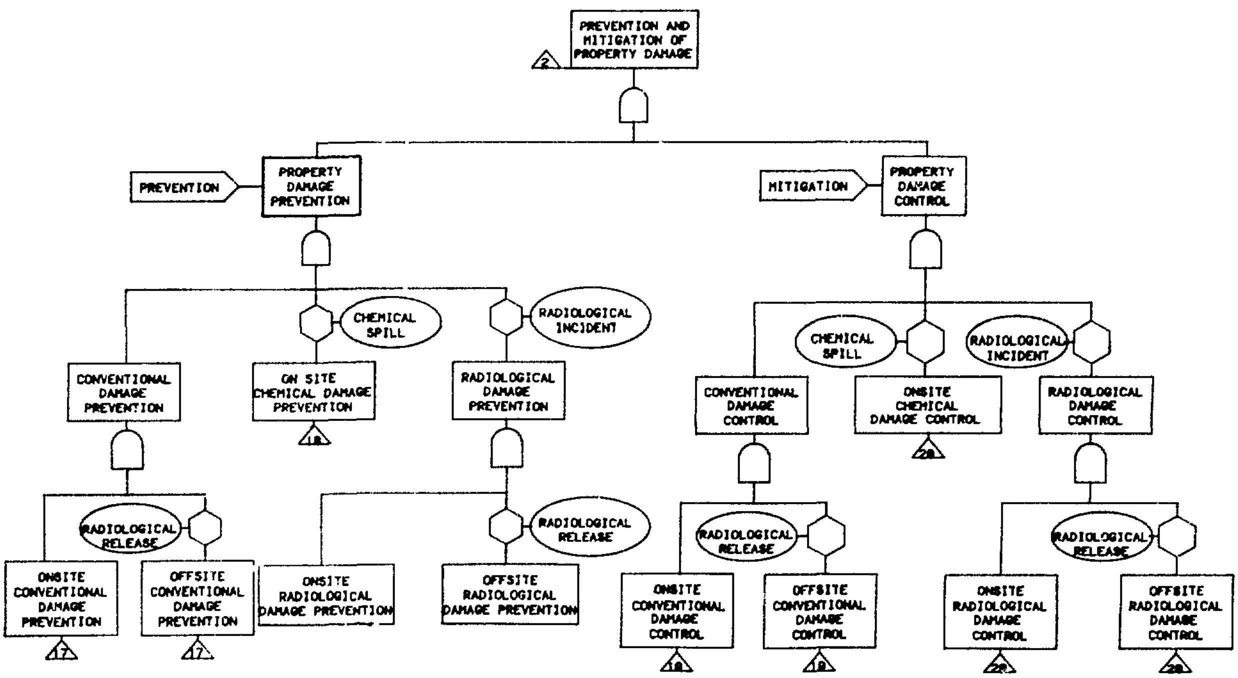
e) Property Damage Control (Mitigation). Plant and public property suffering conventional damage and/or contamination by radioactivity or toxic (or corrosive) chemicals must be identified, secured, and possibly restored to its pre-incident condition.

Except for Prevention of Psychological Stress, each function is divided into its component parts. These are conventional, radiological and chemical. The conventional and radiological subfunctions are further divided into their onsite and offsite components. Only onsite measures are considered for the case of a chemical spill. The utility is not responsible for the impact of offsite spills on the public. Furthermore, it is unlikely that an onsite spill will impact the public. However, said impact could be included if warranted. The need to consider chemical spills becomes apparent when one considers that release of toxic vapors from the failure of an onsite ammonium hydroxide storage tank or chlorine cylinder can have a higher probability of impacting control room personnel than does a radiological release to the environment. Licensing issues for the proposed Midland Nuclear Power Plant included detailed consideration of the impact of toxic and corrosive chemicals released from nearby chemical plants.

Several conditionals appear in Trees 1 and 2. They serve to identify the conditions for which each subfunction should be considered. Onsite conventional injuries and property damage must be considered for all incidents. The same is true for psychological stress. Offsite functions must only be considered when the potential exists for a radiological release to the environment. The tree is structured to accommodate radiological incidents confined to plant structures and to consider the impact of offsite chemical spills on plant personnel and property.

The functions defined above are accomplished by a complex series of human actions/interactions. Numerous activities must be performed by designated plant personnel and civilians, often under less than ideal conditions. Many require cooperation among plant officials and civilian authorities. Their success is dependent upon proper planning, accurate dissemination of information and an ability to make rational decisions in a timely manner. However, such considerations do not directly enter into the functional classification. Planning, which includes the identification, organization and designation of tasks and the development of implementation procedures to insure that they are accomplished in a timely fashion, is an institutional activity. Each functional activity is best accomplished when a well designed and tested plan exists. Information can only be disseminated in a timely fashion when a well designed communications network is in place. Again, development of such a network and procedures for its use are institutional activities. The functional classification serves to identify





the requisite institutional activities. Each institutional activity could be classified according to a tree structure which parallels the functional tree, once its goals have been properly defined.

The acquisition, analysis and interpretation of data to insure that accurate information is disseminated and that proper and timely decisions are made are activities which aid in the implementation of the plan. Like institutional activities, decision models and information systems can also be classified according to a parallel tree structure. Furthermore the functional classification serves to identify information needs and the decisions which must be made. Separate consideration of institutional activities and Information System/Decision Models also serves to facilitate quantification of the goal oriented tree.

While a detailed functional classification has been carried out, it is too extensive to be presented here. Instead, the major elements of the tree will be presented followed by a discussion of selected individual success paths. As previously noted, the tree super structure presented here represents both a refinement of and a supplement to the tree developed for the DOE Information Systems Project (University Research Foundation, 1983). The work was undertaken to provide a more logical and easily understood representation by emphasizing common features among the various subfunctions and success paths. The lower levels of the tree remain the same and are given in complete detail in the aforementioned report. In this sense, the structures presented here serve as an introduction to and a useful aid in interpreting the more detailed functional classification.

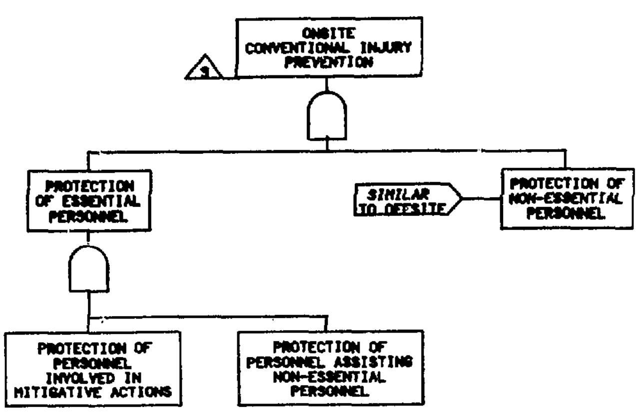
In presenting the logic structures, the impact of institutional activities, data acquisition and decision making will be considered. Emphasis will be given to the function Offsite Radiological Dose Prevention which appears under Prevention of Human Injury. This is satisfied by subfunctions for source mitigation, evacuation, sheltering, etc.

III. CONVENTIONAL/CHEMICAL INJURY PREVENTION

A. Conventional Injury Prevention.

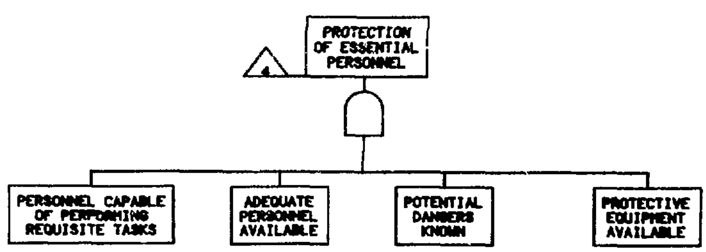
This function is concerned with protecting emergency response workers and the public from conventional injury during the course of their actions.

1. Onsite Conventional Injury Prevention. Tree 3 specifies the subfunctions required to meet this goal. Non-essential personnel must be protected or escorted to the site boundary. The required actions are similar to those taken to protect the public. Two types of essential personnel are identified, simply because the tasks they must perform are quite different. At the next tree level (not shown), specific tasks are identified. These include responding to equipment failures, fires, seismic events, etc., or providing transportation/equipment/supplies to other response teams. For most of these, conventional injuries are prevented by a common type of success path illustrated in Tree 4. Individuals must be capable of performing the task; that is, they must understand all aspects of the task. In order to identify adequate personnel, the magnitude of the task and personnel limitations must be understood. Team members must be aware of potential dangers and the requisite protective equipment (task specific) must be available. This tree is applied in turn to each task with lower tree levels specifying the details of these requirements.



2. Offsite Conventional Injury Prevention. This goal includes protecting the public and emergency response teams assisting them (Tree 5). Protection of essential personnel is satisfied by subfunctions similar to those for the protection of those assisting onsite non-essential personnel. The public is protected by satisfying three general subfunctions. Access to dangerous areas must be restricted. This subfunction is activated only if property damage is caused by initiating events such as earthquakes, tornadoes, etc., or by public panic. Damaged property must be identified and secured and the potential danger must be publicized. Rioting and looting must be controlled by avoiding panic and providing proper deterrents. This is insured by adequate response capability, competent teams and adequate security. Transportation hazards must be minimized on all appropriate types of carriers. A common success path is shown in Tree 6. Traffic congestion and panic tendencies must be minimized. Vehicles must be loaded and operated safely. The needs of special population groups such as the ill and elderly must be accommodated.

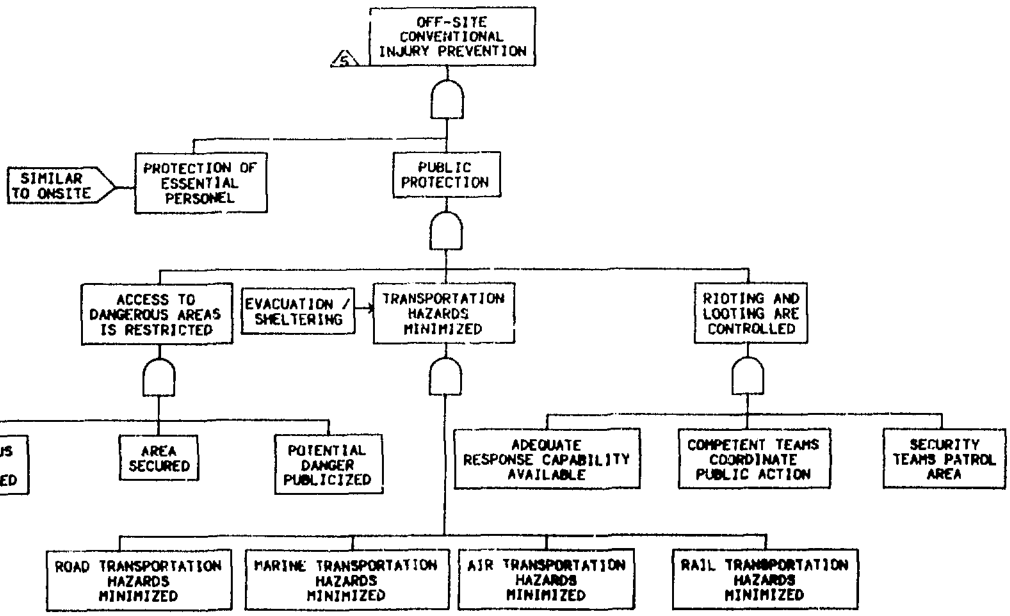
Lower levels of the Offsite Conventional Injury Protection tree identify specific items/population groups that must be considered and

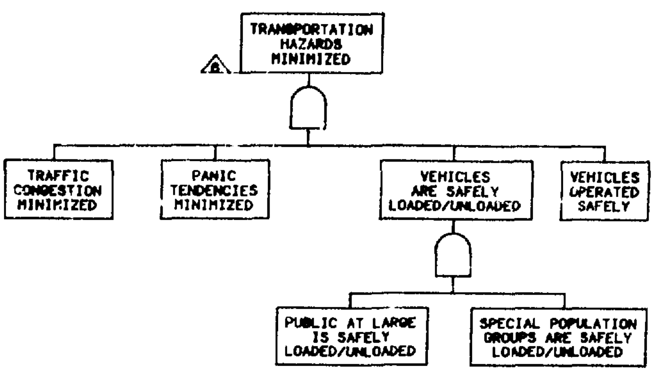


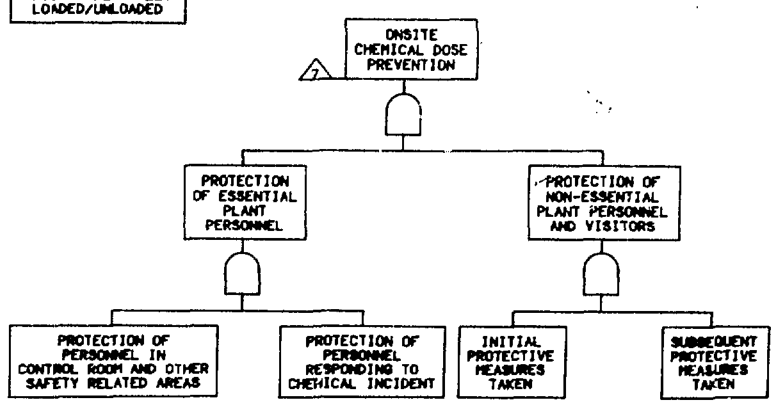
detail the specific tasks and facilities (including communications) that are required. This is common to the success paths for all major functions.

B. Onsite Chemical Dose Prevention.

Tree 7 reveals that the top tree structure for this function is similar to that for Onsite Conventional Injury Prevention. Since workers must be protected from overexposure the general subfunctions are quite similar to those for Onsite Radiological Dose Prevention to be discussed subsequently. Of course, specific tasks and equipment requirements will differ.







IV. RADIOLOGICAL DOSE PREVENTION

The onsite component of this function is defined to include monitoring teams and those assisting the public.

A. Onsite Radiological Dose Prevention.

Like the other onsite "prevention" functions, the radiological function involves protection of both essential and non-essential personnel (Tree 8). Essential personnel are divided into those involved in emergency response outside plant buildings and those who remain inside the plant. Those outside plant buildings include monitoring teams and those assisting non-essential personnel. Their safety is insured by adequate knowledge of environmental hazards, and use of protective clothing/equipment and/or shift rotation to avoid overexposure. Personnel within the plant must be protected from radiation sources within the plant and from radioisotopes released to the atmosphere. The former corresponds to the protection of workers entering high radiation areas. This subfunction has not been developed since it is more the concern of health physicists than emergency planners. The latter subfunction is satisfied by success paths which insure the habitability of the control room, emergency response center and other safety related areas.

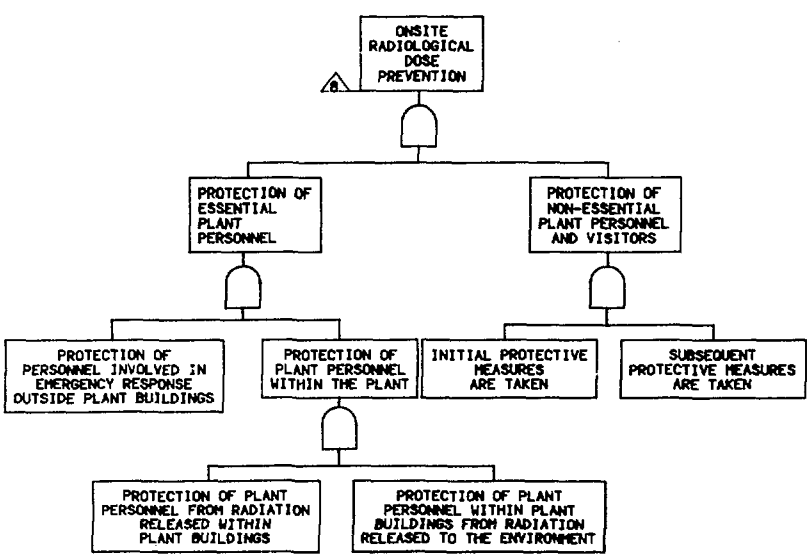
Initial and subsequent measures are taken to insure the safety of non-essential personnel. Initial measures include identifying, notifying, accounting for and sheltering these individuals. Subsequent measures include sheltering or evacuation and possibly the use of blocking agents.

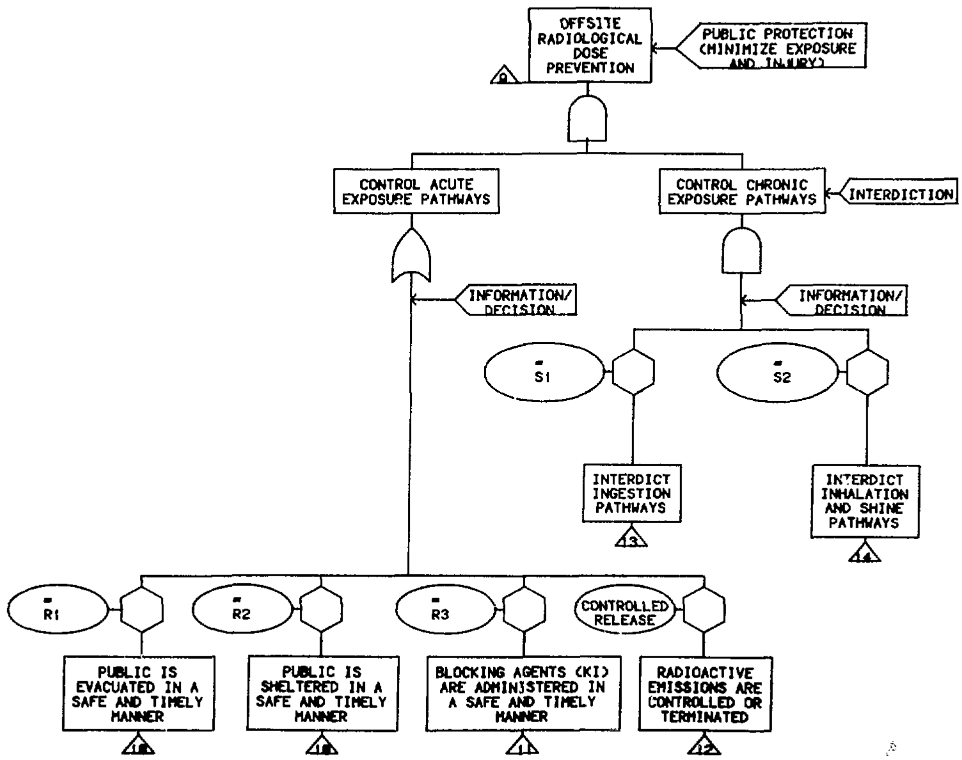
B. Offsite Radiological Dose Prevention.

To protect the public, two subfunctions are required as shown in Tree 9. Both acute and chronic exposure pathways must be controlled. Acute pathways are those to which the public is exposed during and immediately after a release of radioactivity to the environment. These include inhalation, cloud shine, and short-term ground shine. Chronic exposure pathways are longer term pathways of concern during the period of days (or possibly years) after the incident. These include longer term ground shine, inhalation of resuspended material, and ingestion of contaminated foodstocks.

1. Control Acute Exposure Pathways. Methods currently accepted for controlling public exposure to radioactivity are evacuation and sheltering. However, the use of blocking agents such as potassium iodide (KI) is included for completeness. Each of these methods may be used alone or in conjunction with one or both of the others. For instance, it may not be practical or desirable to evacuate the entire population. During a severe incident it is possible that the people residing in areas to which the radioactive plume is carried by the winds will be evacuated and those in surrounding areas will be sheltered. Blocking agents may be distributed to evacuees and those sheltered in areas adjacent to those under evacuation. Evacuees may be directed to engage in mobile sheltering techniques such as breathing through wet towels while en route to their cars and keeping their car windows and air intake vents closed to avoid the influx of contaminated air. Furthermore, it may be recommended that only particularly susceptible members of the population, such as pregnant women, be evacuated and all others be sheltered or given blocking agents. It may only be possible to evacuate the mobile population requiring sheltering for immobile persons, such as bedridden individuals and the like. It may be desirable to evacuate the general public but not institutionalized individuals.

The decision as to which of the aforementioned methods should be employed, and where, when, and to whom is a complex one requiring plant, monitoring and weather data. As pre-





viously mentioned, the decision process with its requisite information is not included as part of the functional classification presented here. The output of the decision process is represented by conditionals placed on the subfunctions "Evacuation," "Sheltering" and "Blocking Agents" displayed as R1, R2 and R3, respectively, in Tree 9.

In principle, the area surrounding the plant can be divided into several sectors, the size and number depending upon population density, road network, meteorology, terrain features, etc. The decision to evacuate, shelter, use KI or to use a combination of these can be made for each sector and population group and for various time periods. The conditionals therefore represent a matrix of decisions. The matrix coefficients prioritize and define the requisite subfunctions. For instance, a coefficient of 0 can be used to define the associated subfunction as unnecessary for the sector/population group/time period it represents, a 1 can define it as a primary success path, a 2 as an alternate success path, etc. Depending upon the situation, two or more of the conditionals may have the same coefficient for a common element. For example, sheltering with ingestion of KI may be the primary success path. Again, the coefficients of the matrices are the output of the decision process.

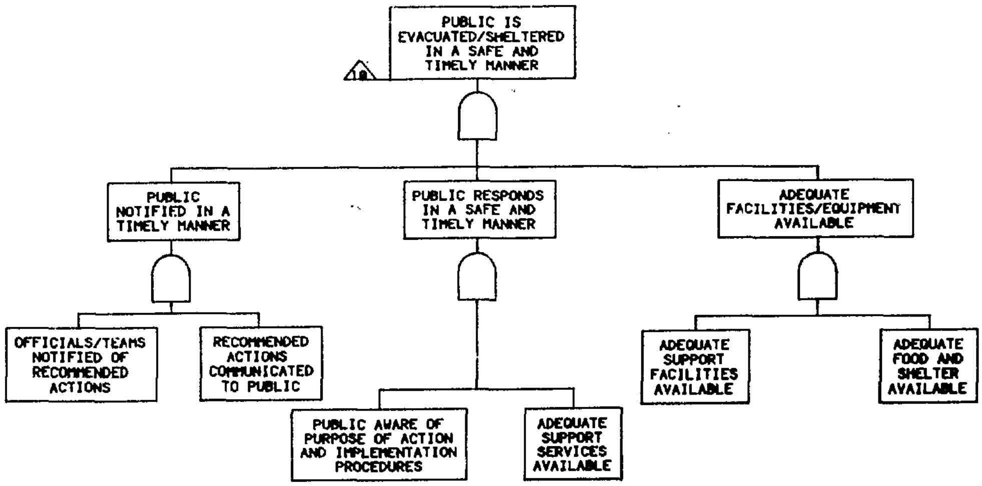
We have defined an Information System/Decision Model whose output in the form of conditional matrices specifies the particular success paths that should be utilized. Such models enter the goal tree at many locations and a sample model will be presented later. The need for particularized decisions becomes apparent in areas of high population density during inclement weather, when the road network cannot accommodate a full scale evacuation or when the presence of water bodies and terrain features could force some evacuees to travel through the radioactive plume.

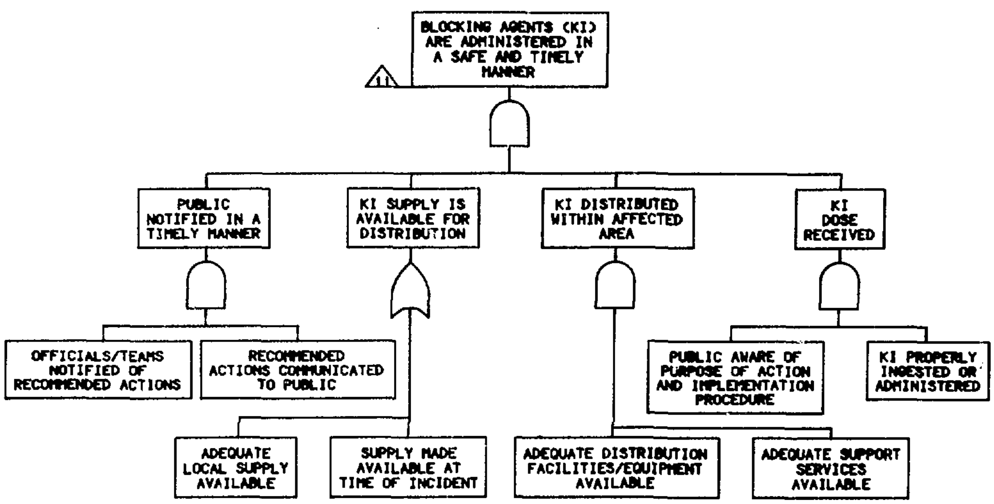
The role of evacuation, sheltering, use of blocking agents and the output of the decision model can be modified to account for an additional success path under certain circumstances. A controlled release (note the conditional displayed in Tree 9) is defined as one that can be reduced in emission rate or terminated for a period of time, or permanently. Consider a situation where the containment building contains substantial airborne radioactivity and the pressure is rising but has not yet reached an unsafe level. Plant authorities may wish to vent the containment to prevent its rupture. If the wind direction is shifting away from population centers, the decision may be made to wait, if possible, until the wind shift occurs. If the wind is steady, the decision may be made to evacuate in the direction of the wind but to hold off completely or vent slowly until the evacuation is well underway or until the public can be safely sheltered.

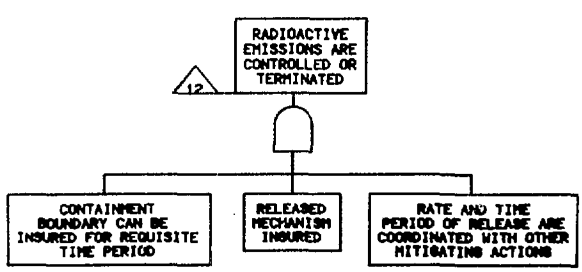
The other extreme of a controlled release is the situation that existed in the TMI-2 containment after the plant was under control. The containment pressure level was safe but venting was necessary. Radioactivity was released at a well-controlled rate during optimal meteorological conditions to maintain acceptably small levels of radioactivity in the environment. Evacuation, sheltering, or use of blocking agents were not required.

Tree 10 provides the elements common to successful evacuation and sheltering. Lower levels of the tree (not shown) identify and specify the specific tasks/facilities/equipment needed to achieve each goal. Tree 11 is a similar structure for administering blocking agents. Tree 12 applies to a controlled release. By Release Mechanism Insured we mean that the venting aperature will function properly. It is evident that lower levels of this tree are intimately linked to knowledge of plant status and other emergency response measures.

2. Control Chronic Exposure Pathways. This function is accomplished by interdiction as noted on Tree 9. Again, an Information System/







Decision Model impacts the tree. This model uses weather and monitoring data and analytical predictions to locate contaminated areas and to determine the proper course of action. Lower levels of this tree are specified in Trees 13 and 14. Again, the success paths (not shown) specify the specific means to accomplish each goal.

V. OTHER FUNCTIONS

A. Prevention of Psychological Stress.

It is not our intention to imply that the utility is responsible for the psychological well being of the public. Rather, we note that the probability of success is enhanced when panic is avoided and the public demonstrates a cooperative spirit. It is evident from Tree 15 that this is accomplished through education, by competent response efforts and by accurate and timely dissemination of information. In other words, this function is satisfied by a successful plan.

B. Human Casualty Control.

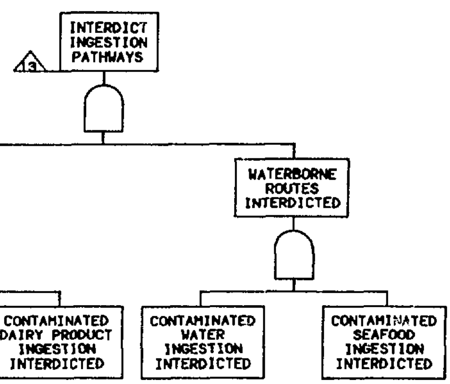
Tree 1 reveals that consideration of conventional, chemical and radiological issues for both onsite and offsite areas involves five separate functions to accomplish this goal. All have a similar top tree structure. This is shown in Tree 16. At lower levels of the tree (not shown), detailed success paths for each have been developed. Except for degree of injury/overexposure, onsite and offsite considerations are similar. The success paths give the means by which casualties are identified and diagnosed, first aid is administered and subsequent treatment is obtained. Transportation considerations include use of road, air and possibly marine transportation, depending upon the site. Specific treatment facilities are identified. Alternate success paths include facilities committed in advance and at the time of the incident.

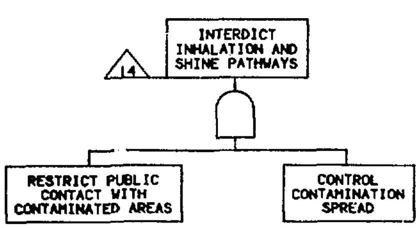
C. Property Damage Prevention.

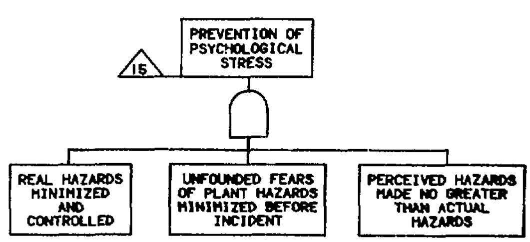
The functions/success paths which insure that property damage is prevented have many elements in common with those which insure the prevention of human injury. While certain tasks are specific to damage prevention others will be implemented by the same teams responsible for human safety. These latter tasks are included in the property tree for completeness and to emphasize to implementers the dual nature of their responsibilities. The top tree structure for conventional property damage prevention is given in Tree 17. It applies (in turn) both onsite and offsite. That for onsite chemical property damage prevention is given in Tree 18. Structures for radiological damage prevention are not given. These have not been developed in detail since the requisite subfunctions are intimately linked to termination of the incident. The offsite branch of this tree does contain provisions for sheltering of farm animals and the like.

D. Property Damage Control

This function is concerned with the







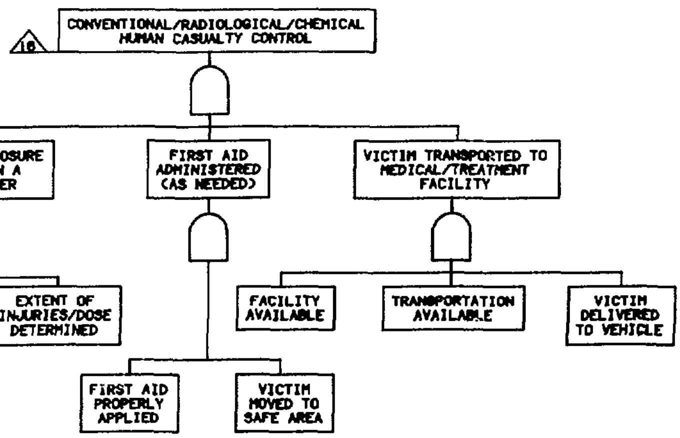
mitigation of property damage caused by the incident, the initiating event and emergency response actions. During the actual emergency, the subfunctions which satisfy conventional property damage control (Tree 19) are concerned with controlling fires and securing areas to prevent further damage. The subfunction dealing with other initiators is taken to include damage to structures caused as a result of emergency response actions. A separate subfunction deals with damaged equipment (e.g., motor vehicles) that must be moved to prevent interference with other emergency actions. Subfunctions which deal with the restoration of conventionally damaged property have not been developed since this is usually not the responsibility of emergency plan implementers.

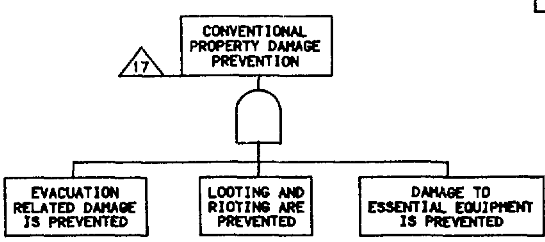
The radiological (onsite and offsite) and onsite chemical property damage control functions are concerned with decontamination and/or disposal of contaminated materials and property. Their common top structure is shown in Tree 20. The need for an Information System/Decision Model to locate contaminated areas and determine the proper course of action is noted. Of course the success paths for radiological and chemical decontamination are quite different.

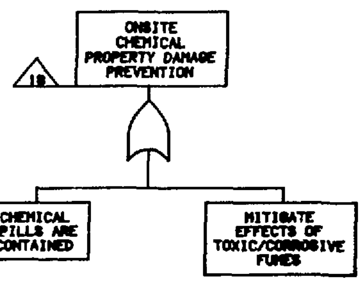
VI. DISCUSSION

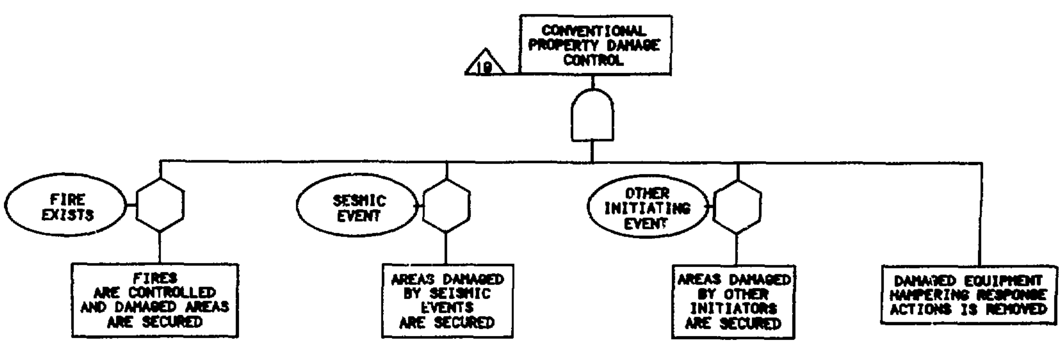
A. Task Implementation/Institutional Activities.

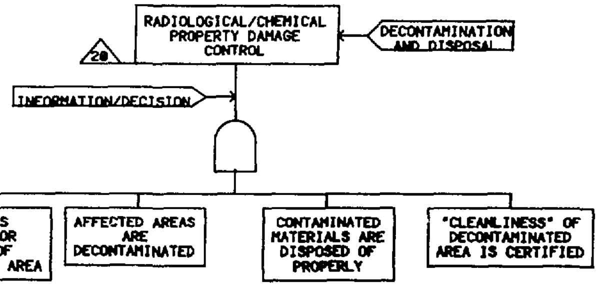
The lowest levels of the goal tree detail the specific tasks that must be implemented and the mechanisms for their accomplishment. Figure 2 illustrates the elements common to implementation of many tasks. It is at this level that institutional activities have their greatest impact on the tree. Table 1 gives the institutional activities that insure successful











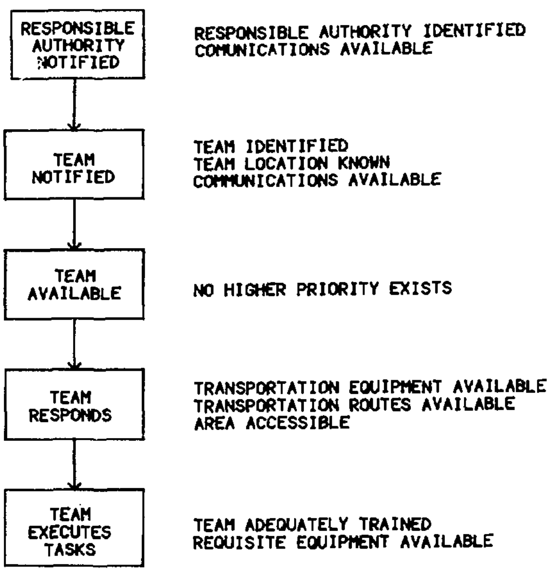


Figure 2 - Elements Common to Task Implementation

Table 1

Institutional Activities for Task Performance

1. Task Implementation Requirements
   - A. Task Designation
   - B. Implementation Procedures
   - C. Auxiliary Procedures (Communications, Notification, Assembly)
   - D. Training (Programs and Exercises)

2. Equipment Requirements
   - A. Task Specific Equipment/Facilities
     - 1. Specification and Procurement
     - 2. Storage, Installation or Construction
     - 3. Maintenance
   - B. Peripheral Equipment (Communications and Transportation) See 2.A.1 to 2.A.3

3. Disseminate Information to Users
   - A. Identification of Responsible Authorities and How to Contact Them
   - B. Notification and Response Procedures
   - C. Equipment/Facility Availability and Location

task performance.

B. Information System/Decision Model.

Figure 3 illustrates a typical Information System/Decision Model to aid in the control of acute exposure pathways (Section IV.B.1 and Tree 9). The decision model considers the availability of specific success paths at a particular time. For instance, the evacuation success path would not be available to all if certain roads were not passable. The adequacy of such a model is crucial to the success of the entire emergency plan. The model itself can be classified according to a functional/goal tree structure. The need for institutional activities in support of the Information System/Decision Model is apparent. Other information systems which enter the tree have many elements in common with that of Figure 3.

C. Tree Quantification.

The very nature of emergency response defies quantification since there is no clearly

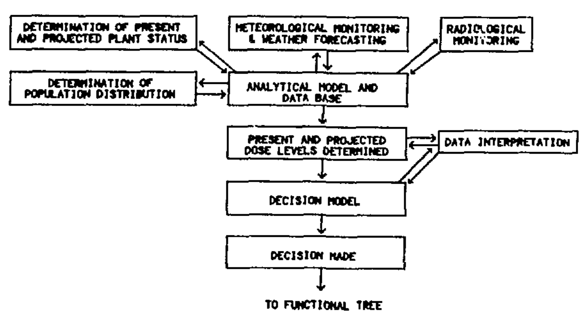


Figure 3 - Information System/Decision Model for Control of Acute Exposure Pathways

defined measure of overall success. However, it is possible to define quantitative measures of success at lower levels of the tree to test alternate success paths. For instance, a measure of success for the subfunction Control Acute Exposure Pathways (Tree 9) can be the total or maximum dose to the population for a given accident scenario. Consequence models such as CRAC and CRACIT can thus be employed to test alternate sheltering, evacuation and controlled release strategies. Proposed modifications in the Information System/Decision Model and the success paths can thereby be assessed. For instance, different evacuation paths, speed and distance scenarios can be considered. The impact of decision time can be related to warning and delay time.

D. Summary.

A goal oriented tree structure has been developed which provides a clear visual representation of the functions that must be performed to insure successful emergency preparedness. In addition to employing it as a training tool, the structure has several other potential uses. For instance, the particular information needs, implementation procedures and regulations which apply to each success path can be noted on the tree to provide a contextual overview of their importance. The adequacy of present capabilities can thereby be assessed. The tree structure can be customized to meet specific needs. Success paths that are not relevant at a particular site can be deleted. Civilian authorities and planners can delete onsite functions and those only responsible for onsite measures can remove the offsite functions.

VII. ACKNOWLEDGEMENTS

The lower branches of the tree were developed at the University of Maryland by a task group under the direction of the first author. The team members were George Harrison, Don Marksberry, Christine Reilly and Bob Starrett. Special thanks to Niall Hunt and Malcolm Patterson of Baltimore Gas and Electric Co. and Joe Braun of Combustion Engineering for many useful discussions concerning the tree structure.

VIII. DISCLAIMER

Any opinions expressed or implied herein are solely those of the authors. Several controversial functions and success paths appear on the goal tree for the purpose of completeness. The presence in no way implies that the use is supported or condoned by any other persons or agencies associated with this work.

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Perspecti[illegible] gency Planning

Results of the Early NRC Analyses of the Radiological Monitoring Data from the Chernobyl Accident

Rosemary Hogan and Thomas McKenna

ABSTRACT: This paper describes the work of one group of the NRC task force for the Chernobyl accident during the first two weeks following the accident. This group focused on organizing the available environmental measurement data into a data base which could be used to gain insights into the consequence throughout Europe and the scope of accident.

The Nuclear Regulatory Commission's Incident Response Branch, maintains an Operations Center in Bethesda to be used during a nuclear emergency involving any of its licensees, especially the nuclear power reactors. When news of the accident at the Soviet Union's Chernobyl Nuclear Power Plant reached the NRC, it was immediately realized that information, explanations and assessments would be requested from the NRC by many sources including Congress, the media and other Federal agencies. It was apparent that the NRC must be prepared to respond to these queries within a short time period. Since the Operations Center is designed to be used for this kind of situation, it was an obvious decision to activate the Center and perform the necessary monitoring and assessment activities using our emergency procedures which were already in place. This paper contains the assessment of the environmental data from the accident which was prepared for an immediate response to requests for information. As expected, the staff was asked to brief the ACRS, the Commissioners and Congressional subcommittees. Within about ten days after activating the emergency organization, the task group was able to provide an analysis of the first eight days data.

This report contains that early analysis. After their report was prepared, that part of the task group was dissolved and no further refinements of the data or computer analyses have been performed since then.

On April 26, 1986, the fourth unit of the Soviet Union's Chernobyl Nuclear Power Plant experienced a severe accident that released a large quantity of radioactivity to the environment. The first evidence of this accident appeared in Sweden during routine radiological monitoring on April 28. The elevated readings were first attributed to a Swedish source. However, further investigation indicated that the source was somewhere in the Soviet Union.

With the world alerted to these elevated radioactivity levels, Soviet officials acknowledged the accident and confirmed the release of radioactive material from the reactor. Increased radiological monitoring by many countries around the world produced large amounts of environmental data that could be used for protective action considerations by potentially affected countries. These results also could be used to better understand the accident because little information on the accident scenario and source term was available for the first several days after the accident.

A task force was formed at the U. S. Nuclear Regulatory Commission (NRC) in support of the Environmental Protection Agency (EPA) to perform accident assessments and project consequences. To accomplish these tasks, information was requested from available sources throughout the world. Most of the information received was radiological environmental monitoring data.

Radiological environmental monitoring data were sent to the NRC from European countries beginning on April 29, 1986. The analyses were performed by various state radiation agencies and nuclear facilities. Early data were received from the Scandinavian countries followed by data from West Germany, Poland, Romania, Hungary, Yugoslavia, Austria, Belgium, and the Netherlands. Later, data from Switzerland, England, Spain, Italy, Korea, Japan, Canada, Israel, and the United States were obtained. Data continually received from the Scandinavian countries and Eastern Europe during the days following the accident provided a picture of radiation levels over time. Although data are still being provided, this report contains only reports received by NRC between April 28 and May 9, 1986.

Samples of air, drinking water, milk, rain water, human food, and animal feed were analyzed by the various countries for radioactivity concentration. Ground contamination values also were provided. Although gross measurements and complete radioisotopic analyses were provided for this report, the data used were primarily limited to the iodine-131, cesium-134, and cesium-137 results.

The data were first reviewed to identify measurements that were candidates for consideration in the analysis. The team reviewed approximately 5000 measurements. Those measurements that were of interest and appeared to be complete were selected for further analysis. The data were entered on computer forms and then formatted into a data base. For each reported result, information was included on the source document, city, country, longitude, latitude, date and time of sampling, and the isotope. Any results that did not include this essential information were rejected from the data base. To simplify the data base, all human food and animal feed results were entered as forage. The results were entered in the reporting units as they were received on the original source document.

Once entered, the data were verified against the raw data. Additional reviews were performed to ensure that the longitude and latitude were consistent for each city or region and that the reporting units for each sample matrix were appropriate. Conversion factors were used to convert all the data to standard units of picocuries per liter (pCi/l), kilogram (kg), cubic meter (m^3), or square meter (m^2). The data were then sorted from lowest to highest value by sample matrix. The highest and lowest measurements were checked for "reasonableness" based on location and date. About 1000 field measurements remained in the data base after all the edits were performed.

Any conclusions derived from this analysis must consider the quality of the data. With very few exceptions, the data were received without information on the sensitivity of the measurements or the propagated error of the analyses. Without this information, all data were necessarily treated equally. In addition, some data were received in English directly from the organization performing the analyses. These data were readily understood and there was some confidence in the validity of the results. There was less confidence in data that may have been recorded by personnel who were unfamiliar with radiological nomenclature and the parameters necessary for interpretation. For example, some reports listed results of radioactivity without indicating whether the results were gross beta, gross gamma, or a specific radionuclide. Some data failed to indicate the city within the country or the English translation of the city name, thereby making it difficult to locate the city name on a map. Some data were not fully translated. As a result of the unknown quality of the data, the analysis was performed with all data treated equally and with the understanding that the conclusions should not be considered quantitative but rather would indicate relative trends during the time of interest.

The results were searched and the maximum daily concentration for ground deposition (pCi/m^2), forage (pCi/kg), milk (pCi/l), rain (pCi/l), water (pCi/l), and air (pCi/m^3) were selected for each 2° square of latitude and longitude. These values were compared with the baseline concentrations given in Table 1. The highest ratio of baseline value to measured result was calculated.

The baseline value for air is the calculated concentration in air that would result in exceeding 1.5 rem [U. S. Food and Drug Administration (FDA) preventive action Protective Action Guide (PAG) basis] to the child thyroid in 5 days based on the assumptions in NRC Regulatory Guide 1.109.

The FDA has issued recommendations* for State and local agencies in the event of accidental radioactive contamination of human food and animal feed. These recommendations or guidelines are projected doses at which responsible officials should take protective actions. There are two levels of guidelines included in the recommendations. The emergency PAG should trigger isolation of food to prevent its introduction to the public, even if the impact of such actions is significant. These drastic measures are justified because of the high projected health hazards. The preventive PAG should trigger actions on the part of the State or local officials only if there would be minimal impact.

The preventive PAGs are much lower projected doses that do not represent significant health hazard. These more conservative PAGs were used as the baseline values in the analysis of the environmental data from the Chernobyl accident. The preventive PAG is 1.5 rem projected dose commitment to the thyroid or 0.5 rem projected dose commitment to the whole body, bone marrow, or any other organ. The FDA has derived specific radionuclide concentrations for area deposition, forage, and milk that are equivalent to the projected dose commitment PAGs. The radionuclide concentrations or "response levels" were calculated used the total intake of the PAG (1.5 rem), the average daily consumption of specified foods, and the days of intake of the contaminated food. Response levels, measured in units of radionuclide concentrations, provide a convenient value for State and local officials to make protective action decisions.

The baseline values used for ground, milk, and forage are the FDA response levels for preventive PAGs. The rain and water baseline values used are the same as the FDA PAG for

* Federal Register Notice 47 FR 47073, October 22, 1982, "FDA Accident Contamination of Human Food and Animal Feeds; Recommendations for State and Local Agencies."

milk. Obviously, this is not valid for most areas because humans do not consume rain water directly.

To show when and where the baseline values in Table 1 (preventive PAGs) had been exceeded, the highest daily measurement in each country was determined. Three ranges were shown on maps (1) those that exceeded the baselines, (2) those within 50% of the baselines, and (3) those below the baseline.

The measurements used to characterize a country were validated by assuring that there were other measurements that would place a country in the same category or other measurements that were within a large fraction of the highest value. For countries with only one measurement, this measurement was compared with those from nearby countries for reasonableness.

The single measurements were used to characterize an entire country, because of when displayed on a global scale the limited number of actual measurement locations did not provide an effective briefing tool.

Examples of the resulting maps for April 29, May 1 and May 4 are shown in figures 1, 2, and 3.

If there were no data from a country for a particular day, the map indicates no markings. By following the maps by day, a general pattern where preventive actions would be considered is evident. It is interesting to note that in the early days after the accident, the air and rain PAGs were approached or exceeded. With the passage of time and distance from the accidents, the PAGs were exceeded in forage and ground. Milk results appeared as the highest measurement in Sweden on May 3, in Yugoslavia on May 5, and Switzerland on May 6.

The NRC staff also examined ground contamination for long-lived isotopes of cesium-134 (2.1 years) and cesium-137 (30.2 years). This might provide insight into the extent of possible long-term problems with agriculture. The data base contained only 28 locations with reported levels of ground contamination for cesium-134 or cesium-137. Only one location reported levels above the preventive PAG for ground contamination; all other locations reported levels less than one tenth of the PAGs.

Although data were not available from all countries, for all days and for all sample media, the available radiological measurements provided a general pattern of contamination that would affect the ingestion pathway. The maps show the extend and general movement patterns of the contamination at a level of detail consistent with the data. However, due to limitation of the data discussed in the paper, the staff always took care to assure each audience was aware of the limitations of the data.

Despite the incomplete data and the unknown quality of the data, sufficient information was available to characterize the level and scope of radiological contamination from the accident at the Chernobyl Nuclear Power Plant beyond the Soviet Union.

Table 1 Baseline values

Matrix Units	Unit of Measure	I-131	Cs-134	Cs-137
Ground deposition	pCi/m^2	$1.3X10^5$	$2X10^6$	$3X10^6$
Forage	pCi/kg	$5.X10^4$	$8.X10^5$	$1.3X10^6$
Milk	pCi/l	$1.5X10^4$	$1.5X10^5$	$2.4X10^5$
Water	pCi/l	$1.5X10^4$	$1.5X10^5$	$2.4X10^5$
Rain water	pCi/l	$1.5X10^4$	$1.5X10^5$	$2.4X10^5$
Air*	pCi/m^3	$7X10^3$	-	-

*All values except air are based on FDA preventive PAG response levels.

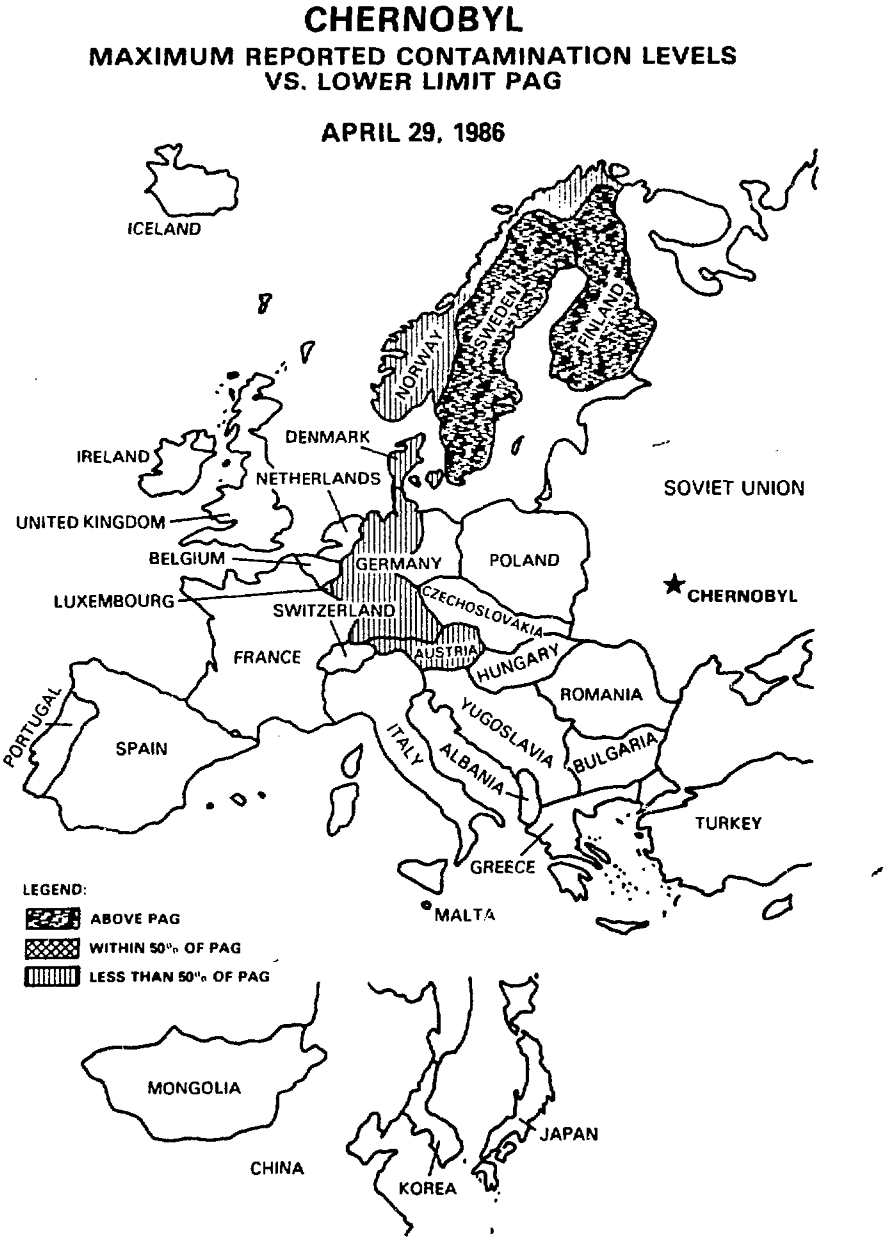
CHERNOBYL
MAXIMUM REPORTED CONTAMINATION LEVELS
VS. LOWER LIMIT PAG
APRIL 29, 1986
ICELAND
NORWAY
SWEDEN
FINLAND
IRELAND
DENMARK
NETHERLANDS
SOVIET UNION
UNITED KINGDOM
BELGIUM
GERMANY
POLAND
LUXEMBOURG
CHERNOBYL
SWITZERLAND
CZECHOSLOVAKIA
AUSTRIA
HUNGARY
FRANCE
ROMANIA
PORTUGAL
SPAIN
YUGOSLAVIA
ITALY
ALBANIA
BULGARIA
TURKEY
GREECE
MALTA
LEGEND:
ABOVE PAG
WITHIN 50% OF PAG
LESS THAN 50% OF PAG
MONGOLIA
CHINA
KOREA
JAPAN

Figure 1

CHERNOBYL
MAXIMUM REPORTED CONTAMINATION LEVELS VS. LOWER LIMIT PAG

MAY 1, 1986

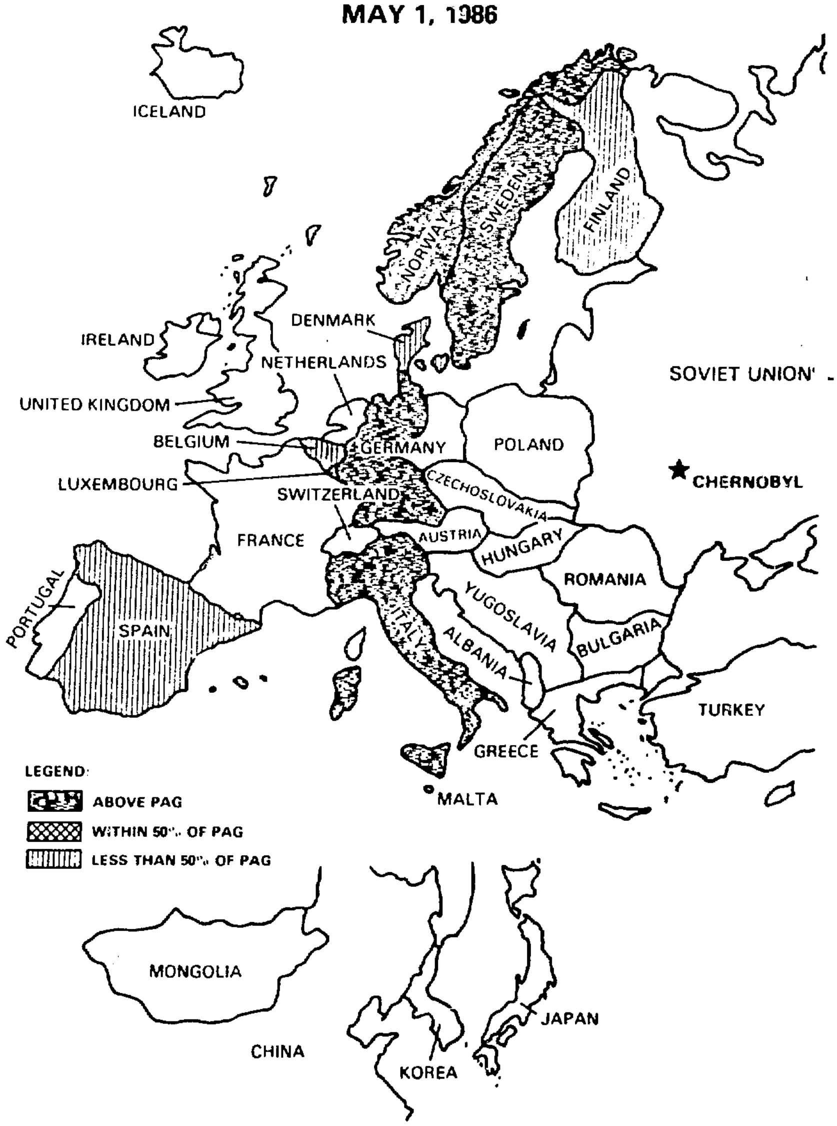

Figure 2

CHERNOBYL
MAXIMUM REPORTED CONTAMINATION LEVELS VS. LOWER LIMIT PAG

MAY 4, 1986

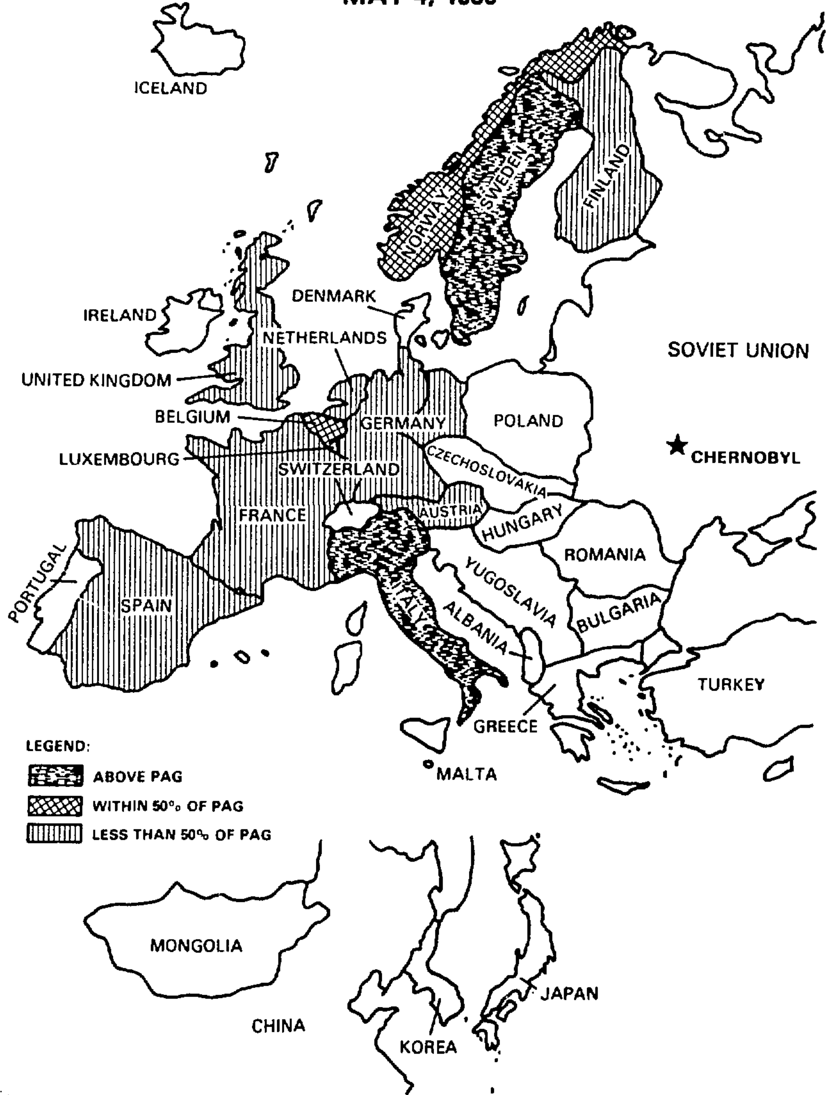

Figure 3

ARAC: A Centralized Computer-Assisted Emergency Planning, Response, and Assessment System for Atmospheric Releases of Toxic Material

M. H. Dickerson and J. B. Knox

ABSTRACT The Atmospheric Release Advisory Capability (ARAC) is an emergency planning, response, and assessment service, developed by the U. S. Departments of Energy and Defense, and focused, thus far, on atmospheric releases of nuclear material. For the past 14 years ARAC has responded to over 150 accidents, potential accidents, and major exercises. The most notable accident responses are the COSMOS 954 reentry, the Three Mile Island (TMI-2) accident and subsequent purge of ^{85}Kr from the containment vessel, the recent UF_6 accident at the Kerr-McGee Plant, Gore, Oklahoma, and the Chernobyl nuclear reactor accident in the Soviet Union. Based on experience in the area of emergency response, developed during the past 14 years, this paper describes the cost effectiveness and other advantages of a centralized emergency planning, response, and assessment service for atmospheric releases of nuclear material.

I. INTRODUCTION

The Atmospheric Release Advisory Capability (ARAC), (Dickerson et al., 1985) has developed over the past 14 years from merely a concept in 1972 to its present role as a federal emergency planning, response, and assessment resource. From the beginning, ARAC was designed to be a centralized resource of a highly trained and specialized staff devoted to all aspects of emergency response, and to reduce duplication of capabilities, software development, and maintenance. This concept was not intended to replace local functions or responsibilities; in fact, it was designed to compliment and enhance local emergency response capabilities of individual nuclear facilities.

During the development and implementation of ARAC, the Department of Energy (DOE) and Department of Defense (DOD) have been major supporters and users of the service. Presently there are approximately 50 DOE and DOD facilities connected directly to ARAC, each having on-line databases for terrain, geography, and meteorological measurement locations pertinent to their own facility. In addition to these facilities, ARAC supports the DOE response to any nuclear event capable of releasing radionuclides into the atmosphere, the Nuclear Regularatory Commission (NRC) response to nuclear power plant accidents, the Federal Aviation Administration (FAA) response to incidents in which aircraft might intercept radioactive material, and the Environmental Protection Agency (EPA) response to incidents in which radioactive material has left or might leave facility boundaries.

Advantages of a centralized emergency planning, response, and assessment system are that it:

- Avoids duplication of resources, and provides a state-of-the-art, proven response capability;
- Provides experienced staff devoted to emergency preparedness, response, and assessment;
- Is cost effective when applied to a large number of nuclear facilities and integrated into the federal emergency preparedness programs;
- Provides a standard (or criterion) for emergency response assessments while maintaining flexibility to meet site-specific and agency requirements;
- Focuses research and development on timely improvement and evaluation of emergency response resources; and
- Applies integrated research and development resources to specialized emergency response requirements in real-time, e.g., Cosmos 954, TMI, Gore, (Oklahoma), and Chernobyl events.

On the other hand, the disadvantages of a centralized system are that it:

- Is cost effective only when applied to a broad base of nuclear facility and federal agency requirements;

- Can be viewed by local authorities as a "threat" to their capabilities and responsibilities; and
- Can be viewed by local authorities as a mechanism for reducing or eliminating their responsibilities.

During the development and implementation phases of ARAC, a balance between the advantages and disadvantages of a centralized system has emerged. As stated earlier, ARAC now serves as a national emergency response resource for several federal agencies. It has developed an extensive background in emergency response by responding to over 150 accidents, potential accidents, and major exercises. The most notable ARAC responses are:

- Savannah River Plant (SRP) Tritium Release, 1974
- Train accident involving UF_6, 1976
- Chinese 200 kt and 4 mt atmospheric tests, 1978
- COSMOS 954 reentry, 1978
- TMI Nuclear Power Plant accident, 1979
- Titan II accident, 1980[a]
- SRP H_2S leak and transfer, 1981[a]
- Ginna Nuclear Power Plant accident, 1982
- Gore, Oklahoma, UF_6 accident, 1986[a]
- Chernobyl USSR Nuclear Power Plant accident, 1986

The remainder of this paper will discuss the role research and development has played in responding to accidents, both in real-time and in the model evaluation area—two significant attributes of a research and development group co-located with an operational emergency response center.

II. RESEARCH CONTRIBUTIONS TO REAL-TIME EMERGENCY RESPONSES

The various roles ARAC played during and after the TMI accident response provide an excellent basis for describing how the system is used, and the value of closely associated research and development. The five basic roles ARAC filled during and after the accident are that it:

- Provided guidance on deployment of radiological measurement systems;
- Helped interpret surface and airborne radiological measurements,
- Estimated the ^{133}Xe source term;
- Provided guidance to the FAA for air traffic safety in and out of the Harrisburg airport; and
- Estimated total population dose for the President's Commission on TMI.

A few of these roles, such as advising the measurement teams and the FAA, were relatively straightforward. Estimating the source term and modifying the MATHEW/ADPIC (Sherman, 1978; Lange, 1978a) models to estimate "man-rem" for the President's Commission required model modifications and interpretations of data by the research staff. To estimate the source term required a knowledge of both the response function of the instruments used in the aircraft to measure the ^{133}X, and of the characteristics of the ADPIC transport and diffusion estimates, which use particles to simulate radioactivity. This coupling of information led to estimates of the source term that were made available during the first four days of the accident. Later a comparison of the model calculated source term with a source term estimate provided by the President's Commission showed agreement within a factor of two to three.

During the ARAC response to the Soviet Cosmos 954 satellite reentry into Canada, the ARAC research team was able to modify a nuclear weapons fallout model (KDFOC2) (Serduke, 1978) so that it could be used to simulate the depositional "footprint" of radioactive particles generated with varying densities at altitudes between 20 and 60 km. These "footprint" simulations, together with the ground measurements, were used to define and to limit the search areas to manageable sizes. For this event, ARAC also provided guidance on the appropriate time and positioning for launching a balloon-borne measurement system into the stratosphere to observe the amount of fine particulate material (i.e., the material that remained for months in the stratospheric circulation regime). This guidance served to eliminate costly and prematurely arranged balloon flights when the regularly scheduled future flights would provide the required concentration measurements.

For the UF_6 release at the Gore, Oklahoma facility of Kerr McGee, the ARAC research team worked with LLNL chemists, Oak Ridge National Laboratory, and the Atmospheric Turbulence Diffusion Division (NOAA) scientists to define the chemical and physical characteristics of the source term and the dispersion processes. These input data were used to define and parameterize the amount of HF and UO_2F_2 released in the process, the cloud rise due to exothermic reactions, and the particle size attributed to the UO_2F_2. Without this research support, the ARAC response to this accident would not have been as timely and useful as it was to the NRC on-site assessment team.

The Chernobyl accident, because of its magnitude and limited information, has provided the largest challenge to date for the ARAC research and development team. The team was "called on" to (1) expand the MATHEW/ADPIC grid from a horizonal size of 200 × 200 km to 1920 × 1920 km, (2) estimate the vertical extent, time history, and magnitude of the source, and (3) retrieve a global particle-in-cell model (PATRIC), (Lange, 1978b) which was originally developed for estimating transport and diffusion in the stratosphere, from 6 years of storage and modify the model to simulate the upper level (tropospheric) release created by the initial explosion and fire at Chernobyl.

The first part of the MATHEW/ADPIC calculation covered a 200 × 200 km region centered on the Chernobyl reactor site (Figure 1a); it became apparent that this calculation was insufficient to answer the questions arising from the spread of radioactivity across the Soviet boundaries into the rest of Europe. Thus, the first

[a] Toxic Chemical Releases

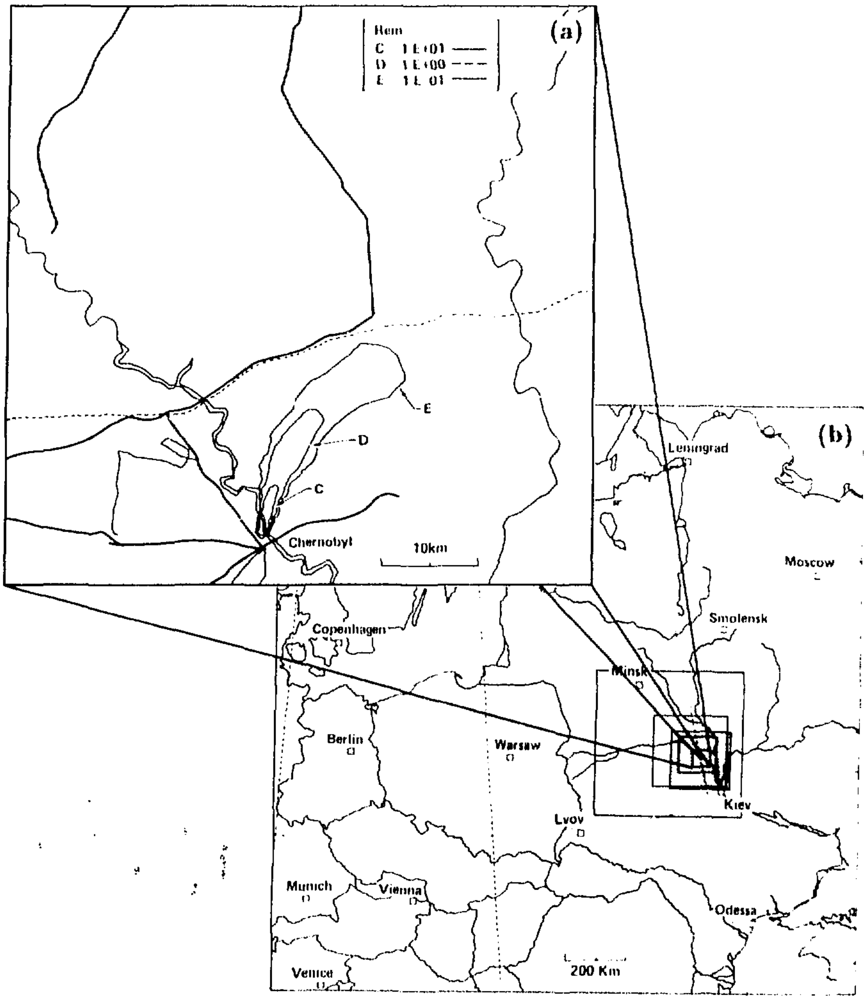

Figure 1. The calculational grids used for the MATHEW/ADPIC models: a) the calculated infant thyroid dose due to ^{131}I inhalation for a six-hour period on 29 April shown on a 60 km expanded subregion of the 200 km grid; b) the final 1920 × 1920 km grid (48 km cells) with the nested subgrids (3 km, 6 km, 12 km, and 24 km cells) outlined and the initial grid (200 × 200 km) shaded.

MATHEW/ADPIC modeling effort was terminated, as part of our ARAC response, and a second, larger-scale simulation was initiated, covering the largest area possible within the limitations of the model and available computer resources.

The MATHEW/ADPIC grid chosen was 1920 km on a side, and extended 2100 m vertically. The horizontal grid mesh was 48 km and the vertical grid spacing was 150 m. This particular grid size was chosen because it represented the largest grid that could be used for the MATHEW/ADPIC models without a major revision of the computer codes. In addition, this grid allowed for coverage of a reasonably sized area of interest for the initial dose and deposition calculations. In Figure 1b, a nested sampling grid is shown around the source (Chernobyl) for horizontal cell sizes of 3 km, 6 km, 12 km, and 24 km, respectively. These nested grids were used to sample the particles that produced surface air concentration and ground deposition estimates near the reactor site.

The source term for the reactor accident was divided into two parts, a lower and an upper cloud. The lower cloud was assumed to be produced over a period of six days as a result of heat from the burning reactor. The upper cloud was assumed to be produced by one or more of the following: explosions followed by a hot fire for several hours, convective activity near the Chernobyl reactor site associated with thunderstorms, or lifting over a warm front located between Chernobyl and the Baltic Sea. One major part of the ARAC effort during the first two weeks following the accident was associated with the determination of a lower level source term and the associated consequences.

Employing both the grid shown in Figure 1b and the initial source estimate derived from the environmental

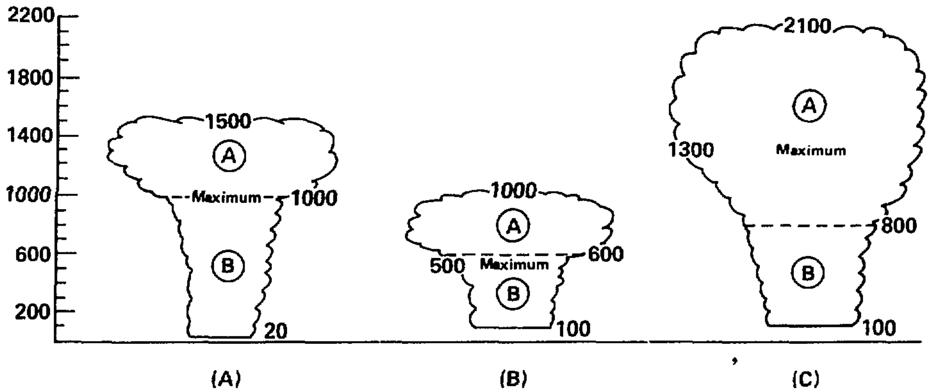

Figure 2. Depictions of the vertical distribution of source material (explosion and fire cloud): a) average distribution as used in the ADPIC calculations; b) the night and c) day, respectively, vertical distributions calculated with a non-hydrostatic cloud model.

measurements of ^{137}Cs and ^{131}I in Scandinavia and Europe, the MATHEW/ADPIC model was used to refine initial estimates of the low level source term for the first six days of the accident. Providing a reasonable description of this source term required a vertical distribution of the radioactivity, as well as a time dependence of the release.

Figure 2a shows the vertical distribution of material used to define the lower level source term in the MATHEW/ADPIC calculations. This estimate was based on prior experience simulating heated plumes within air masses—air masses which contained vertical temperature distributions similar to those shown by vertical atmospheric soundings near Chernobyl. Eighty per cent (80%) of the released material was assumed to reside between 1000 m and 1500 m, with the remaining 20% located between the surface and 1000 m. The maximum concentration was at 1000 m.

Toward the end of the two-week period immediately following the accident, the vertical distribution of the low level source term was quantitatively estimated by a two-dimensional, high resolution, non-hydrostatic cloud model. This model was originally developed to simulate thunderstorms but, more recently, has been applied to simulate plumes of smoke induced by large-scale urban fires. The estimated source strength for the model calculations was 62 megawatts, about the resident heat energy expected from the shut-down of a 3200 mw thermal reactor. Figures 2b and 2c show the vertical distribution of material, calculated by this model, for both nighttime and daytime atmospheric soundings taken near the Chernobyl reactor site near the time of the accident. During the nighttime (Figure 2b), the model estimated that 80% of the material was contained between 600 m and 1000 m. The remaining 20% of the material was located between 100 m and 600 m with the maximum concentration located at 500 m. For the daytime (Figure 2c), 80% of the material was determined to be between 800 m and 2100 m, with the remaining 20% between 100 m and 800 m. The maximum value for this case was at 1300 m. Differences between dose and deposition estimates, based on the distribution of material in the original source term estimate (Figure 2a) and the non-hydrostatic model estimate (Figures 2b, 2c), would not be large, particularly at distances of several hundred kilometers and beyond. For this reason, the dose and deposition estimates were not recalculated using the quantitatively-modeled vertical distributions of material (Figure 2).

From the PATRIC hemispheric scale model calculations, it rapidly became apparent that some radioactive material was convected or lofted (or both) to much higher altitudes than that assumed for the MATHEW/ADPIC calculations (described above). A series of calculations with material placed at 2500, 4200, and 5500 m failed to transport contamination to Japan and North America even close to the recorded arrival times—if at all. While limitations of the model reduce some of the precision desired, it presently appears that at least some material had to be injected to altitudes above 5500 m in order to account for airborne air concentration and surface rainwater and milk measurements of the radioactivity in Japan and the USA. Original estimates of dose and deposition from ^{131}I and ^{137}Cs, made during the first 3 weeks of the accident, for eastern and western Europe using the MATHEW/ADPIC model and global estimates of dispersion with the PATRIC model, are reported by Dickerson and Sullivan, 1986. Further refinements of these estimates have recently been reported by Gudiksen and Lange, 1986.

III. MODEL EVALUATION STUDIES

One of the most significant continuing research and development efforts has been in the area of model evaluation and improvement. Over the past several years, better diffusion parameterizations utilizing space varying surface roughness heights, and the use of multiple vertical wind profiles and nested grids for better con-

EXPERIMENT	TERRAIN	METEOROLOGY STABILITY	WINDSPEED (m/s)	TRACER MEASUREMENTS (km)
INEL 1971	ROLLING	C	2-6	7-80
SRP 1974	ROLLING	F-C	1-4	3-30
TMI 1980	ROLLING	F-C	1-4	40-60
ASCOT 1980	COMPLEX	F-E	0-4	1-8
ASCOT 1981	COMPLEX	F-E	0-4	1-10
EPRI 1981	FLAT	F-A	1-5	1-50
SRP MATS 1983	ROLLING	D-B	1-8	~ 20
MONTALTO 1983	COASTAL	C-B	1-6	1-6

Table 1. Summary of MATHEW/ADPIC Model Evaluation Studies

centration estimates near the source point have contributed to improving the MATHEW/ADPIC models. Many of these improvements were designed and implemented as a result of model evaluation studies which were done to define the expected accuracy of the models under various terrain and meteorological situations. Table 1 lists the evaluation studies conducted with the MATHEW/ADPIC models during the past 12 years. Contained in these studies are 26 individual experiments conducted in 6 different geographical areas. They represent approximately 3000 tracer measurements spanning a wide variety of diffusion categories. (Dickerson and Lange, 1986) The experiments shown in Table 1 utilized a multitude of tracers, including routine emissions of ^{41}Ar from the SRP nuclear reactors, the controlled venting of ^{85}Kr from the TMI containment, ^{131}I releases at the Idaho National Engineering Laboratory (INEL), sulfur hexafluoride releases from the SRP, the Montalto, and Kincaid power plant sites, and perfluorocarbon and heavy methane releases that were part of the ASCOT experiments. The releases occurred from the 62 m stacks at the SRP and TMI, and from the 187 m stack at the Kincaid power plant. The remaining releases generally occurred near the surface, except for one heavy methane tracer that was released at 60 m during the 1980 ASCOT experiments, and one perfluorocarbon tracer released in a cooling tower plume during the 1981 ASCOT experiments. The duration of the releases varied from 15 minutes to several hours. Extensive surface sampling networks were employed in each series of experiments. Maximum distances were 80 km for the 1971 INEL studies, 50-60 km for the EPRI and TMI studies, 30 km for the MATS experiments, 10 km for the ASCOT experiments, and approximately 6 km for the studies at Montalto, Italy. The experiments were supported by a variety of surface and upper air meteorological observations; data was provided by measurements ranging from normal meteorological coverage provided by the National Weather Service (NWS), to a local site tower and an extra upper air sounding during the TMI purge of ^{85}Kr. Data were supplied by a wide spectrum of measurement systems, including acoustic sounders, tethersondes, rawinsondes, optical anemometers, and towers that were an integral part of the ASCOT experiments.

It is difficult to devise a statistical process that adequately describes a model's performance when compared to tracer field data, particularly when the field data span a broad spectrum of release and sampling times, sampling distances, terrain, and meteorology. For example, the standard correlation coefficient is used sometimes; however, one point at the high end of the scale can influence the entire data set. Early on, a rigid technique, but one considered a standard, was chosen for comparisons of tracer measurements to the MATHEW/APDPIC model calculations. A factor R is computed for each pair of measurements (C_m) and model calculations (C_c) which represents the whole - number ratio between the two. For each experiment the percent of comparisons within a factor R are plotted as a function of R. The definition of R is $R = (C_m + B)/(C_c + B)$, except if ($R < 1$,) then $R = (C_c + B)/(C_m + B)$, and B is background.

Figure 3, based on the factor R, depicts a summary of the performance of the MATHEW/APIC models to date. The *best* simulation of the experimental data is given by the upper curve, which is associated with rolling terrain and near-surface tracer releases. The most *difficult* simulation is associated with complex terrain and elevated releases. Other situations provide results that are intermediate to these curves. Hence, the best results indicate that the calculated concentrations are within a factor of 2 for 50% of the measured concentrations and within a factor of 5 for 75% of the comparisons. This performance degrades to 20% and 35% for factors for 2 and 5, respectively, for the comparisons associated with elevated releases in complex terrain. This degradation of results in complex terrain is due to a variety of factors, such as the limited representativeness of measurements in complex terrain, the limited spatial resolution afforded by the models, and the turbulence parameterizations used to derive the eddy diffusivities.

IV. FUTURE RESEARCH OPPORTUNITIES

The most significant improvement in emergency response can be attained by the development and implementation of an operational mesoscale (out to 200 km) time-dependent forecast model. Technology is available today to develop a model that can be applied to a range

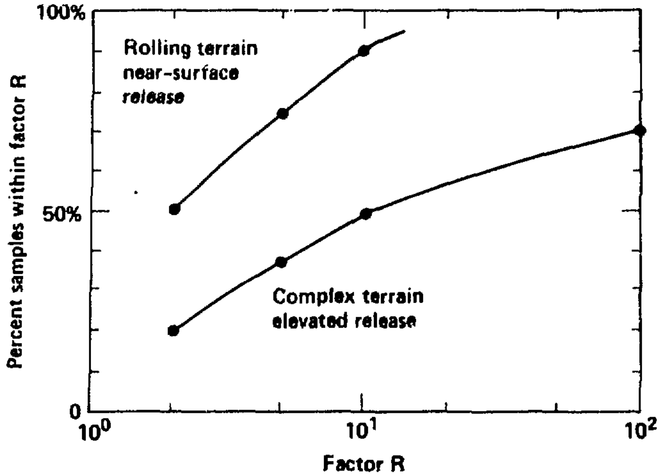

Figure 3. Percent of computed air concentrations within a factor R of measured values. The figure provides a measure of the spectrum of model evaluation results that span from near-surface tracer releases in rolling terrain to elevated releases in complex terrain.

of forecast problems and consequently would be useful (but not all inclusive) for emergency response purposes. This is definitely a high pay-off, low risk area of research and development. Chernobyl has shown the value of having the capability of simulating rainout in the transport and diffusion models. The state-of-the-science in this area has advanced to the point where rainout is technically feasible to implement, although additional research would be required to improve the simulations and make them more realistic. A logistical problem remains which involves obtaining spacially varying rainrate data, and providing a mechanism for including these data directly into the transport-diffusion models.

ARAC does provide a foundation for addressing toxic chemical response as a centralized system; however, many more technical unknowns are involved in dealing with toxic chemicals as opposed to radioactive material. Also the frequency of accidents and the number of chemicals are considerably greater and the health effects are known to a lesser degree than for nuclear material. In general, toxic chemical releases can be divided into four classes based on their physical and chemical reactivity and their density with respect to density of the ambient air. These 4 classes are: non-reactive chemical/ambient air density, non-reactive chemical/heavier-than-air density, reactive chemical/ambient air density, and reactive chemical/heavier-than-air density. Given a toxic chemical release where the chemical is non-reactive and whose density is approximately that of ambient air, the MATHEW/ADPIC models would be expected to perform as well as they do for nuclear material. If the released material is non-reactive and heavier-than-air, models are available to estimate consequences; however, considerable effort would be required to place them in an operational environment. (Gudiksen et al., 1986)

Chemically toxic releases that are both chemically and physically reactive at the source point and during the dispersion processes, would require a large research effort before the environmental consequences can be modeled with confidence. A joint research and development and implementation effort is required before ARAC or any other centralized or local emergency response system can be expected to address a range of accidental releases of toxic chemicals with any degree of confidence.

V. ACKNOWLEDGEMENT

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Digital Imagery Manipulation and Transmission for Emergency Deployment Applications

C. A. Gladden and D. S. Phillipson

ABSTRACT The Digital Imagery Transmission System (DITS) was designed as a replacement for the older slow-scan system. DITS is an interactive microcomputer graphics system capable of digitizing maps or photographs. Digitized images are stored on a local area network file server and can then be distributed to other units on the system or be transmitted over a dial-grade phone line to a remote site where other DITS terminals exist. A DITS terminal consists of an inexpensive microcomputer, an off-the-shelf RGB monitor and an electronic drawing device for entering overlay data. Hardcopy prints can be obtained in minutes.

During the mid 1970's, the Department of Energy (DOE) recognized the need for an emergency response team to provide quick response to a variety of nuclear-related emergencies. As a result, an emergency response team was formed, and the task undertaken to assemble the necessary expertise and hardware to support such a mandate. The Aerial Measuring System (AMS) for radioactive material was already in existence and provided the core for the new emergency response team.

Since the response would be to a variety of accident types, in often unknown locations, a method was badly needed for moving visual information such as maps and photographs over the international dial telephone network. A video compander (commonly known as slow scan) system was selected for this application, and while it provided a wealth of information, it also had several shortcomings. These problems basically involved the interactive capability of available slow scan equipment. While the video information could be digitized, it consisted of only one image, and no method was readily available that would allow the addition of multiple level overlay information.

With the advent of the inexpensive microcomputer, it became practical to convert the maps to a digital data base, generate the overlay information electronically, and to simultaneously implement the balance of the on-scene commander's status information system that had been in planning for several years.

The Digital Imagery Transmission System (DITS), as it was finally named, is composed of an assortment of standard and specialized software, a series of inexpensive microcomputers, monitors, cameras, and a file server. A typical DITS terminal consists of the microcomputer, a standard scan RGB video monitor, and an electronic drawing device for entering the overlay or modification information. The system is packaged in the standard deployment containers for easy movement into a field environment.

Once the area of the emergency has been defined, maps are procured and digitized using the video camera. Multiple images are digitized and stored in a contiguous data base on a common system file server. This map data base is then available to any terminal on the system for viewing and/or inputting overlay information such as radiation contamination levels and locations (actual and projected), population distribution, measurement team locations, evacuation routes, etc. Overlays may be viewed in full resolution or 1/16th resolution to show more than one at once.

A key system feature is the requirement for all overlay data or manipulations to be certified at a single scientific control point prior to distribution to any location other than the originator. Once the overlay data has been added to the image and verified, it is available for electronic distribution to local and remote users or for duplication as hard copy or viewgraphs. In order to minimize transmission times, map data base information is generally only moved once between the stations and common file server. Afterwards, only overlays or modifications are transmitted and these are compressed to further enhance the transfer times. The transmission time of a complex overlay is only minutes (typically less than 3).

In addition to such features as "roam" and two-stage "zoom", a wide variety of predefined geometric patterns are available to the user as an aid in creating overlay information. Freehand drawing is also allowed, and grid patterns with definable dimensions or distances are available to provide assistance.

This system has been tested successfully in several CONUS and OCONUS locations, utilizing the International Maritime Satellite (INMARSAT) system and the international dial telephone network. Generally, the data may be transferred using proper interface equipment and any voice grade communications circuit. Data integrity over long distances is insured by full 100 percent error correcting modems and transmission software.

Currently a DITS image consists of 16 "shots" (4 x 4) of the video camera which are assembled by the computer into one large image (2048 pixels wide by 1920 pixels high). Any section, 512 pixels wide by 480 pixels high, can be viewed in full resolution, or the whole image can be viewed in 1/16 resolution.

The IBM compatible computer used for a DITS terminal is the AT&T 6300. It features low cost, high reliability and fast operation. The current drawing device is a three button "mouse" with plans to add a light pen for additional input capability. Current hardcopy outputs are 8" by 10" and 35mm polaroid pictures and viewgraphs. Also, a color printer may be connected for producing pictures with 125 color choices. Ease of operation is accomplished with many "help screens" that consist of text mixed with graphic illustrations to better demonstrate system operation. The software is menu driven and the system prompts the user for all information needed to accomplish the task at hand.

In summation, the DITS system provides the aerial measuring system organization with an interactive digital graphics system that easily allows the principals to be kept informed of such items as contamination levels and locations, operations progress, or status. The system is lightweight, inexpensive, and based upon readily available microcomputer technology. Transfer of information over voice-grade communications circuits allows remote terminals to be located, wherever required, on a global basis. Since the information is transmitted in a digital format, it is easily encrypted using general purpose data encryption devices. The system currently consists of 7 terminals and a common file server. Provisions have been made to increase that number to approximately 20 terminals for future applications.

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Image/Data Storage, Manipulation, and Recall Using Video/ Computer Technology for Emergency Applications

James M. Thorpe

ABSTRACT Employing a blend of broadcast video and state-of-the-art computer technology the Management Emergency Response Information System (MERIS) is designed to control, manipulate, and distribute the graphic and visual information necessary for decision-making in an emergency response situation or exercise. Instant storage and recall of an extensive library of frames of video imagery allow emergency planners the time and freedom to examine necessary information quickly and efficiently.

I. INTRODUCTION AND BACKGROUND

For a number of years, EG&G Energy Measurements, Inc. (EG&G/EM) has been providing the Emergency Response and Aerial Measuring System (AMS) programs with a field-deployable photo and video capability in support of technical, management, and public affairs functions. Imagery collected from both actual operations and major exercises has been a significant tool for training team members, field managers, and other agency personnel. In response to a continuing need to provide better information handling and display methods, and for improved recall and display of all types of graphic data products, development of the Management Emergency Response Information System (MERIS) was begun. MERIS presently consists of a number of components of existing video and computer equipment and state-of-the-art image handling systems.

In MERIS, images are instantly stored and recalled, either one frame at a time or in sequence. This capability, coupled with the newest computer-controlled videodisc technology, will make the EG&G/EM library of aerial photographs, videotapes, and other imagery for all major nuclear sites available for instant recall for crisis management purposes or other urgent inquiries.

Up to 10 input stations can be used to feed information into the MERIS base station for image handling. The images are combined and titled, have graphics added, then are time-coded and stored in the MERIS library until needed for crisis management. Four large monitors are used to display the information to the crisis management teams. The choice of information on the monitors is at the control of the Crisis Manager; data can be recalled within seconds of its input.

In emergency operations, MERIS can be used to brief important visitors by recalling and displaying information stored in the MERIS library. Images can be displayed in any sequence the manager requests. Hardcopy color prints or transparencies and black and white prints can also be made of any image stored in the MERIS library.

MERIS concepts have already proved of interest to a number of emergency planners and others who must store and recall large quantities of graphic data. MERIS is an operational field-portable system that could be utilized for Accident Response Group (ARG) and Federal Radiological Monitoring and Assessment Center (FRMAC) emergency operations and exercises. Some of the unique MERIS capabilities are currently being utilized to solve specific problems in Emergency Operations Centers (EOCs) and crisis control centers.

II. DESCRIPTION OF MERIS COMPONENTS

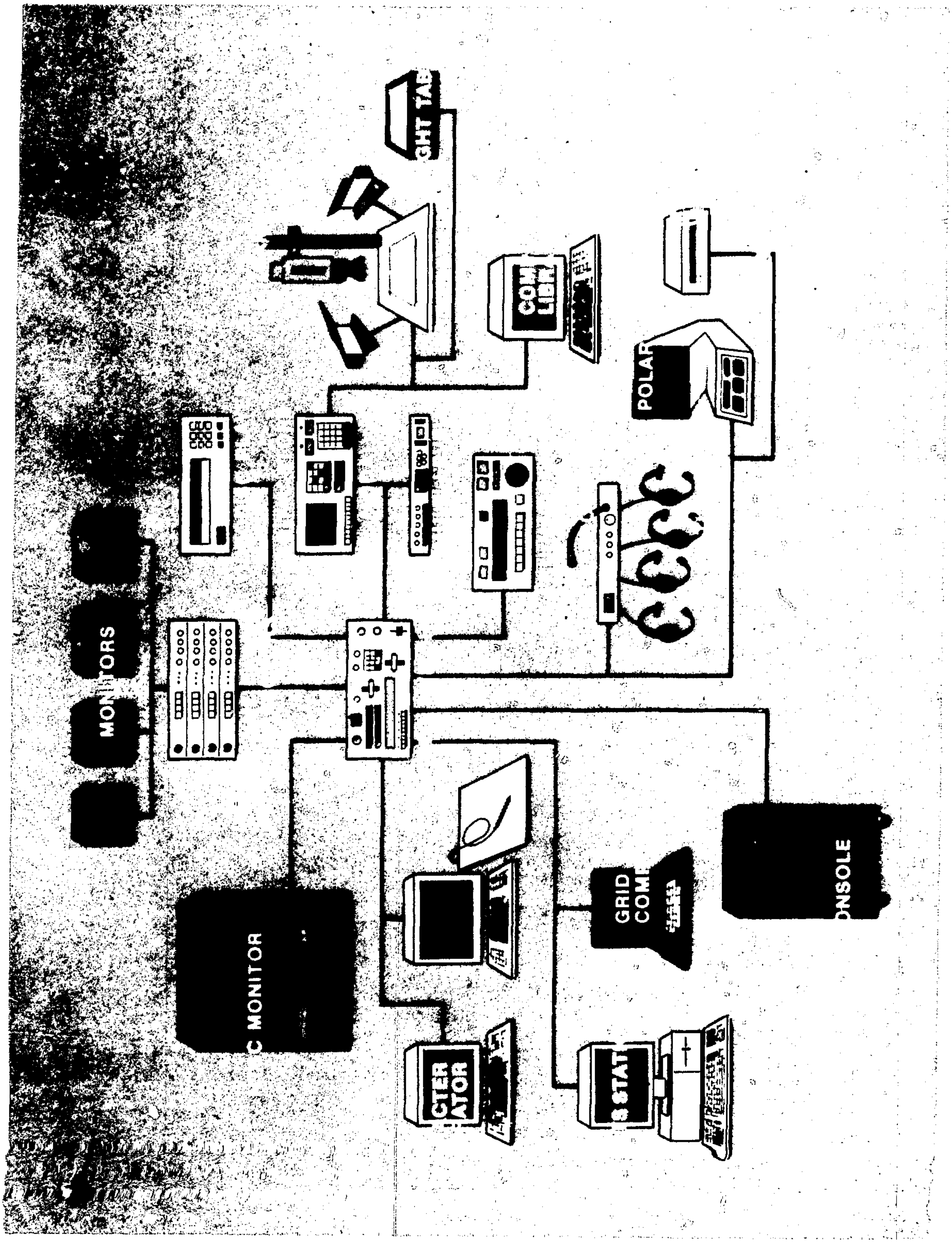
DIGITAL FRAME STORE - Stores frames of video imagery within 2 seconds. Those frames can be recalled at random or in sequential order.

VIDEO SWITCHER - Sometimes called a special effects generator. Electronically keys or overlays one video source (or image) over another. Also allows for colorization of white overlays as well as fades and dissolves among all video sources.

MULTI-LEVEL KEYER (Graham-Patten) - Works in conjunction with the video switcher to overlay or key up to six different sources or images at one time over a base image.

PAINT BOX - Electronic drawing computer used as a source that is keyed or overlayed by video switcher and multi-level keyer over a base image. Used to produce backgrounds and animation also.

CHARACTER GENERATOR - A titling and graphic background-generating computer keyed over the


MONITORS
C MONITOR
GHT TAB
COM
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POLAR
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base image via the video switcher.

COPY STAND - A floor stand unit with lights and a video camera on which graphic material can be converted into a video source for manipulation and overlay.

DIGITAL FRAME STORE LIBRARY COMPUTER CONTROL - Controls the files of the frame store's digital memory. Contains a cross-referencing recall system to allow quick access to any frame in storage.

3/4 UMATIC-VCR - Conventional news broadcast format video recorder used to record or play back an event videotape into the system. An individual frame can be stopped and stored at any time.

VIDEODISC PLAYER WITH AT&T COMPUTER - Allows storage and controller recall of an extensive library of photo and video frames (up to 54,000 per disc side). Organizes and houses a data base that allows individual frames to be categorized and labeled.

RTS INTERCOM - Allows communication between MERIS team members and emergency response management personnel.

TIME BASE CORRECTOR/FRAME SYNCHRONIZER (4) - Distributes composite images to each of the four individual channels.

VIDEO MONITOR (output) - Conventional NTSC (National Television System Committee).

III. HOW DOES MERIS PRODUCE AN IMAGE?

Since it is based on broadcast video technology MERIS can easily create a composite overlay of several pieces of graphic information within a very short time. For example, when an aerial photograph of a site is placed under the video camera on the copy stand, that image can be stored in the digital frame store with the push of a button.

When recalled, that aerial image can be overlaid by any number of sources. A character generator can overlay letters for titling, etc. The paint box can be overlaid (via the video switcher) for detailing and graphics, plus titling. The Graham-Patten keyer, along with the video switcher, allows up to six different sources (character generator, paint box, copy camera, photo, etc.) to be keyed or overlaid on the original aerial photo. The final composite image will usually have the time keyed over (in the lower right corner) as it is entered onto the still store disks. This system can store up to 400 frames of information on its built-in hard disk and 200 each on a number of removable disk cartridges.

In addition to storage and recall of individual frames of imagery, MERIS provides crisis managers with other valuable visual informational assets.

Conventional off-air broadcasts (news, press conferences, etc.) can be distributed on any channel when necessary. These broadcasts can also be recorded for later evaluation and analysis. Video images recorded by emergency response authorized crews could be similarly played back.

Public affairs briefings could also be conducted with MERIS imagery to give the significant visual detail necessary for media release.

Post-exercise or response evaluation could also be greatly enhanced with MERIS.

IV. FUTURE DEVELOPMENTS

Since broadcast video and computer technology are constantly changing, and many of those changes will have a direct effect on MERIS capabilities, constant evaluation of new technology in these areas is necessary. MERIS will reflect new ways of storing and recalling images and data as they are developed.

Several areas are currently being researched for potential MERIS applications. Fiber optic technology offers numerous distribution options without the signal degradation of conventional cable when transmission over a long distance is needed. This will enable the MERIS remote console to display images from the main system at virtually any point necessary.

This technology will also allow a video camera to be placed at a designated site for monitoring while its image is transmitted to the main terminal.

Technology advances in the storage and recall ability of videodiscs will also improve greatly in the coming years. This combined with the interactive training assets of videodisc technology will help expand MERIS and its simulation and crisis scenario training capabilities.

As MERIS is used in emergency response situations and exercises, design engineers will be able to fine-tune its response capabilities to meet changing demands.

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Using MENU-TACT for Estimating Radionuclide Releases During a Reactor Emergency

Andrea L. Sjoreen

ABSTRACT MENU-TACT is a computer code designed to estimate potential atmospheric releases of radionuclides from the reactor complex during a reactor emergency. It is part of a suite of models, the Intermediate Dose Assessment System, which is used at the U.S. Nuclear Regulatory Commission's Operations Center. MENU-TACT incorporates only those processes that can make major changes in the magnitude of a reactor release. It provides quick results of bounding calculations. MENU-TACT was written to be simple to use. Data are entered via fill-in-the-blank displays or selection of menu options. Plant-specific and generic default values for data are provided where appropriate. The code models transport of nuclides from containment through a series of volumes to the environment. The analyst controls the pathways from the volumes to the environment, timing of events, and depletion parameters, along with other required data.

I. INTRODUCTION

During a severe reactor accident the U.S. Nuclear Regulatory Commission (NRC) must first decide if plant conditions warrant taking protective actions. This initial decision is based on consideration of the consequences of a wide range of core damage accidents and is not based on dose assessements. Determination of the need for additional protective actions includes consideration of dose assessments. To produce dose assessments, NRC must first estimate the probable range of magnitudes of the release from the reactor complex. The basic tool used in the NRC's Operations Center to estimate these reactor source terms is MENU-TACT. (TACT is an acronym for Transport of ACTivity.) MENU-TACT was created by modifying TACT-III (Killough, et al., 1983). The Accident Evaluation Branch, NRC, developed TACT-III so that doses and releases from a variety of accidents could be estimated. MENU-TACT and TACT-III model transport of nuclides from containment through a series of volumes (which are also called nodes) to the environment. The number of nodes, the node volumes, and the pathways through the nodes and to the environment are all entered by the analyst. MENU-TACT and TACT-III use matrix inversion routines that solve simultaneous equations describing transport, radioactive decay, and removal processes in the volumes. Radionuclide daughter build-up is not considered in MENU-TACT and should not be important for the time period of concern when responding to an emergency (e.g., a few hours or days). The mathematical framework of TACT-III is described in Killough, et al. (1983). Modifications made to TACT-III to produce MENU-TACT serve to simplify the input required and to simplify the processes modeled.

MENU-TACT is part of the Intermediate Dose Assessment System (IDAS). IDAS also contains a data base management system and MESORAD (Sherpelz et al., 1986), an air transport and dose projection model. The source terms computed by MENU-TACT can be stored in the IDAS data base. They are read by MESORAD according to the name given to that case in MENU-TACT.

This document explains the use of MENU-TACT during an accident. The processes modeled are described. Examples of the input schemes used are presented and sample results are shown. Future enhancements to MENU-TACT are described.

II. PROCESSES MODELED IN MENU-TACT

The processes included in MENU-TACT and the treatment of reactor volumes are diagrammed in Fig. 1. This figure shows two volumes, containment and an auxiliary building. Transport processes include leakage from containment, transport through the building, recirculation through filters, and escape to the environment. Removal processes are radioactive decay, filtering, and user-specified removal processes. In Fig. 1 the user-specified removal process is spray removal. Filtering and removal apply only to non-noble gases. A single constant is entered for each process for all radionuclides affected. While this is not a very accurate treatment of removal and filtering, it is appropriate for bounding calculations.

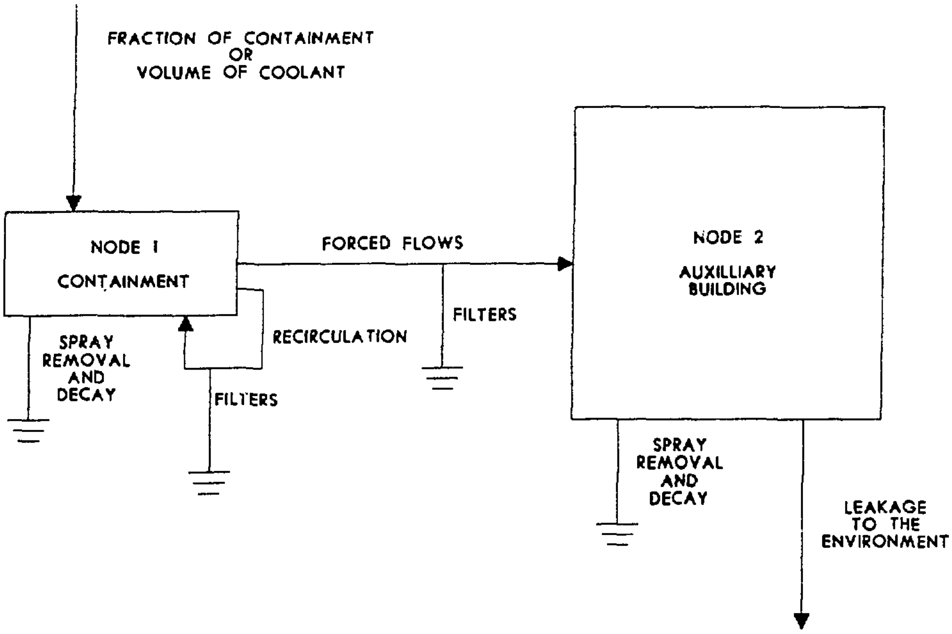

Figure 1. The processes modeled in MENU-TACT and their relationships to the volumes defined.

The timing of processes is controlled by the times entered by the analyst as shown in Fig. 2. Radioactive decay operates from reactor shutdown to end of release. (Note that decay continues during atmospheric transport as modeled in MESORAD.) Daughter build-up is not included in MENU-TACT calculations, because it was determined to be unimportant for bounding calculations. The process start time begins in-plant processes. No activity is released to the environment before the start of release time. Note that shutdown and process start time may be the same, but release start must be later, if only a few seconds, than process start. The time of the end of the release must be later than all other times.

III. MODEL TESTING

MENU-TACT has been tested both for internal consistency and against TACT-III for correctness. Each type of input data was varied from its minimum to its maximum value, with all other input data kept at their default values. The results were inspected to insure that the expected changes occured. These tests insured that MENU-TACT was modeling its processes correctly. More that 30 pairs of MENU-TACT / TACT-III runs were made to insure that both codes produced the same results. Four types of tests were run. (1) A baseline case was run with default data. (2) One and two time-step cases were run, with shutdown time both equal and not equal to process start time. (3) Two volume cases were run with combinations of large and small volumes with large and small flow rates. (Note that the matrix solution routines used in TACT-III is somewhat more robust than those in MENU-TACT in handling cases that were numerically extreme.) (4) Large and small filter efficiencies were combined with large and small removal coefficients. The TACT-III and MENU-TACT results agreed to three significant figures for all tests.

IV. USING MENU-TACT

A complete user's manual for MENU-TACT has been published (McKenna et al., in press). The following describes briefly the use of MENU-TACT in the Operations Center. The major steps in performing a MENU-TACT analysis are:

1. Preparing the release pathway form

2. Logging onto the system and branching into MENU-TACT

3. Selecting plant site (and case)

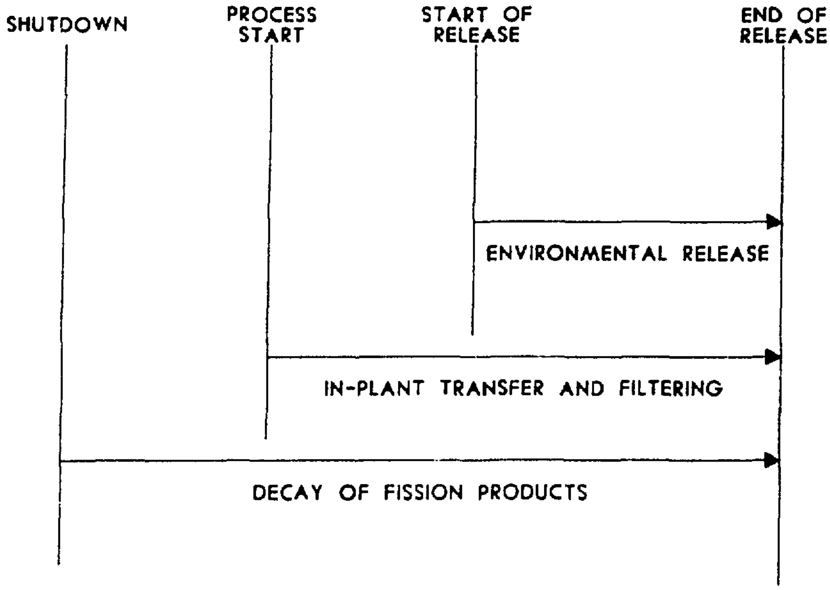

Figure 2. The relationships between the times entered in MENU-TACT.

4. Specifying basic time dependant information:
    a. case description or title
    b. power level
    c. node names and volumes
    d. shutdown time
5. Specifying core damage state, either:
    a. coolant release
    b. gap release
    c. grain boundary release
    d. core melt release
6. Specifying pathway data:
    a. event timing
    b. removal coefficient for each node
    c. portion of the release initially injected into each node.
    d. transfer rates between nodes and between nodes and the environment and filter efficiencies assumed for transfers.

All data are entered on menu screens (Fig. 3) or on fill-in-the-blank screens (Fig. 4). A default value is provided for each data item. If appropriate, the default value is plant-specific, that is, the value is read from the reactor site data base. If the default value is appropriate, the analyst can just type the NEWLINE key for that data. Each type of data has bounding values. If the analyst enters an inappropriate value, an informative message is printed and the value must be corrected before continuing. At the end of each data screen, the analyst is asked to check the entries. If any are incorrect, the analyst can return to the top of the screen to reenter the data. If a mistake is noted on the screen before it is finished, the analyst may back up through the entries with the ESCAPE key. Once all data are entered, the menu in Fig. 3 appears and the analyst may run the case or modify the data further.

The data that are plant-specific are provided as the default for that item. Once a plant site (and unit, if appropriate) is selected, the fact that it is a pressurized water reactor (PWR) or boiling water reactor (BWR) is known. PWR's are given two default volumes (or nodes), primary and containment. BWR's are given three default volumes, primary, containment, and secondary containment. The analyst can change the number of volumes, their size, and their names. The rated reactor power (Mwt) is provided. The plant design leak rate and predicted failure pressure are shown on the

```
NRC - IDAS TACT MAIN MENU
IDAS TACT  - MENU VERSION

   CREATE A NEW CASE
 MODIFY AN EXISTING CASE
 RETURN TO IDAS MAIN MENU
```

```
USE ARROW KEYS FOR SELECTION AND (CR) WHEN SELECTION IS THE CORRECT ONE
```

Figure 3. An example of a MENU-TACT menu screen.

```
             BASIC TIME INDEPENDENT DATA FOR PWR ARKANSAS 1
BASIC CASE DESCRIPTION/CASE NAME:
test
NUMBER OF NODES (NODES <=4): 2
NAME OF NODES    VOLUME IN FT**3
NODE(1) = Primary     .1000E+07
NODE(2) = Contain.    .1900E+07


REACTOR POWER (MEGAWATTS THERMAL) : 2568.
REACTOR SHUTDOWN TIME - 24 HOUR CLOCK SITE TIME
(SDTIME) (<= TIME OF FIRST CASE :    86/08/04 14:00    (YY/MM/DD HH:MM)

========================================================================
 ARK1:REFERENCE VOLUMES (ft**3)
     Primary system      9.7E+03
      Containment        1.9E+06


ARE YOU DONE WITH THIS SCREEN ?     Y__
```

Figure 4. An example of a MENU-TACT data entry screen.

Isotope	Decay 1/hr	Source Ci/Mwt	Coolant Concentration PWR Ci/gm	BWR Ci/gm	Gap fraction	Grain Boundary fraction	Core Melt fraction
I-131	3.6E-3	2.8E+4	3E-07	5E-09	2E-2	5E-1	1E+0
I-132	3.0E-1	4.1E+4	1E-07	3E-08	2E-2	5E-1	1E+0
I-133	3.3E-2	5.9E+4	4E-07	2E-08	2E-2	5E-1	1E+0
I-134	7.9E-1	6.5E+4	5E-08	5E-08	2E-2	5E-1	1E+0
I-135	1.1E-1	5.5E+4	2E-07	2E-08	2E-2	5E-1	1E+0
Kr-85	7.4E-6	1.9E+2	2E-07	0	3E-2	5E-1	1E+0
Kr-85m	1.5E-1	7.9E+3	1E-07	0	3E-2	5E-1	1E+0
Kr-87	5.4E-1	1.5E+4	6E-08	0	3E-2	5E-1	1E+0
Kr-88	2.4E-1	2.2E+4	2E-07	0	3E-2	5E-1	1E+0
Xe-131m	2.4E-3	3.7E+2	1E-07	0	3E-2	5E-1	1E+0
Xe-133m	1.3E-2	2.1E+3	2E-07	0	3E-2	5E-1	1E+0
Xe-133	5.5E-3	5.7E+4	2E-05	0	3E-2	5E-1	1E+0
Xe-135	7.6E-2	1.2E+4	4E-07	0	3E-2	5E-1	1E+0
Xe-138	2.9E+0	1.2E+4	4E-08	0	3E-2	5E-1	1E+0
Cs-134	3.8E-5	2.3E+3	3E-08	3E-11	5E-2	5E-1	1E+0
Cs-136	2.2E-3	9.2E+2	1E-08	2E-11	5E-2	5E-1	1E+0
Cs-137	2.6E-6	2.0E+3	2E-08	7E-11	5E-2	5E-1	1E+0
Te-129m	8.6E-4	6.4E+2	1E-09	4E-11	1E-4	1E-1	3E-1
Te-131m	2.3E-2	4.1E+3	3E-09	1E-10	1E-4	1E-1	3E-1
Te-132	8.9E-3	4.1E+4	3E-08	1E-11	1E-4	1E-1	3E-1
Sb-127	7.5E-3	3.5E+3	0	0	1E-4	2E-2	2E-2
Sb-129	7.4E-2	9.0E+3	0	0	1E-4	2E-2	2E-2
Sr-89	5.7E-4	3.4E+4	4E-10	1E-10	0	1E-3	7E-2
Sr-90	2.8E-6	1.7E+3	1E-11	6E-12	0	1E-3	7E-2
Sr-91	7.3E-2	3.6E+4	7E-10	4E-09	0	1E-3	7E-2
Mo-99	1.1E-2	5.2E+4	8E-08	2E-09	0	1E-2	1E-1
Ru-103	7.3E-4	4.2E+4	5E-11	2E-11	0	1E-4	7E-3
Ru-106	7.8E-5	1.1E+4	1E-11	3E-12	0	1E-4	7E-3
Ba-140	2.3E-3	5.1E+4	2E-10	4E-10	0	1E-2	2E-1
Y-91	4.9E-4	3.7E+4	6E-11	4E-11	0	0	1E-4
La-140	1.7E-2	5.4E+4	2E-10	0	0	0	1E-4
Ce-144	1.0E-4	3.0E+4	3E-11	3E-12	0	0	1E-4
Np-239	1.2E-2	5.5E+5	1E-09	7E-09	0	0	1E-4

Table 1. Nuclide-specific data used in MENU-TACT.

bottom of the screen on which the rates of transfer to the environment are entered.

The amount of the reactor inventory available for release is determined by core damage state and either plant power level and the number of Curies of each nuclide per Mwt or the number of Curies of each nuclide per ft^3 of coolant. The nuclides which may be included in an assessment are shown in Table 1. The fractions of the inventory released for each nuclide are available for a gap release, grain-boundary (TMI-like) release, or core melt. The inventory may be scaled up or down. One factor applies to all noble gases and another applies to all other nuclides. Nuclide specific scaling is not available for the above three release types. Coolant concentrations are provided for both PWR's and BWR's. For a release of coolant the analyst enters the number of ft^3 of coolant released from the reactor. As the accident progresses and better estimates of the coolant concentrations are available, the analyst has the option of revising the default coolant concentrations (Ci/ft^3) for each nuclide individually.

The transfer from the core are specified as a total number of ft^3 or a rate of ft^3/min for coolant releases and as fractions of inventory or fractions/min of inventory for all other accident types.

V. MENU-TACT RESULTS

MENU-TACT first prints a summary of the data for this case (Fig. 5). It always displays the initial inventories in the volumes, the concentrations in the nodes at the beginning of the release to the environment (Fig.6), and the total releases (Ci) to the environment (Fig. 7) for each nuclide considered for the time period of the release. More detailed results are available, if requested. If the results are needed for futher assessment in MESORAD, a file is written to the IDAS data base.

```
Site is:ARKANSAS 1     Record is: ARK1_026
          TIME INDEPENDENT INPUT
          RUN DATE  8/  5/ 86
          TITLE: test
          NUMBER OF NODES:                  2
          POWER (MWT):                      2.568E+03
          TIME OF SHUTDOWN:               86/08/04 14:00
          RELEASE FRACTION:
              PARTICULATES:                     1.000E+00
              NOBLE GASES:                      1.000E+00
          SOURCE OPTION: .             4-CORE
          PLANT OPTION:                1-PWR
Node names:
Primary Contain.
                TIME DEPENDENT INPUT
          PROCESS START TIME      END OF RELEASE
            86/08/04 14:00        86/08/04 14:10
                    VOLUMES
                NODES    VOLUME (FT**3)
          Primary           1.000E+06
          Contain.          1.900E+06
          REMOVAL COEFFICIENTS (1/HR)
                NODE   PARTICULATES        NOBLE GASES
          Primary           3.50             0.000E-01
          Contain.        0.000E-01          0.000E-01
Press Return for Transfer Rates.          =
```

Figure 5. Data summary.

ACTIVITY (CURIES) IN EACH NODE AT 86/08/04 14:00
(BEGINNING OF PROCESS)

ISOTOPE	ENVIR.	Primary	Contain.
I-131	0.000E-01	5.752E+07	1.438E+07
I-132	0.000E-01	8.423E+07	2.106E+07
I-133	0.000E-01	1.212E+08	3.030E+07
I-134	0.000E-01	1.335E+08	3.338E+07
I-135	0.000E-01	1.130E+08	2.825E+07
KR-85	0.000E-01	3.903E+05	9.758E+04
KR-85M	0.000E-01	1.623E+07	4.057E+06
KR-87	0.000E-01	3.082E+07	7.704E+06
KR-88	0.000E-01	4.520E+07	1.130E+07
XE-131M	0.000E-01	7.601E+05	1.900E+05
XE-133M	0.000E-01	4.314E+06	1.079E+06
XE-133	0.000E-01	1.171E+08	2.928E+07
XE-135	0.000E-01	2.465E+07	6.163E+06
XE-138	0.000E-01	2.465E+07	6.163E+06
CS-134	0.000E-01	4.725E+06	1.181E+06
CS-136	0.000E-01	1.890E+06	4.725E+05
CS-137	0.000E-01	4.109E+06	1.027E+06
TE-129M	0.000E-01	3.944E+05	9.861E+04
TE-131M	0.000E-01	2.527E+06	6.317E+05
TE-132	0.000E-01	2.527E+07	6.317E+06
SB-127	0.000E-01	1.438E+05	3.595E+04
SB-129	0.000E-01	3.698E+05	9.245E+04.
SR-89	0.000E-01	4.889E+06	1.222E+06
SR-90	0.000E-01	2.445E+05	6.112E+04
SR-91	0.000E-01	5.177E+06	1.294E+06
MO-99	0.000E-01	1.068E+07	2.671E+06
RU-103	0.000E-01	6.040E+05	1.510E+05
RU-106	0.000E-01	1.582E+05	3.955E+04
BA-140	0.000E-01	2.095E+07	5.239E+06
Y-91	0.000E-01	7.601E+03	1.900E+03
LA-140	0.000E-01	1.109E+04	2.773E+03
CE-144	0.000E-01	6.163E+03	1.541E+03
NP-239	0.000E-01	1.130E+05	2.825E+04

Press Return to continue   =

Figure 6. Initial concentrations computed for the data in Fig. 5.

ACTIVITY RELEASED TO THE ENVIRONMENT FROM 86/08/04 14:00 TO 86/08/04

ISOTOPE	RELEASED CURIES
I-131	1.192E+01
I-132	1.705E+01
I-133	2.506E+01
I-134	2.600E+01
I-135	2.323E+01
KR-85	9.039E+00
KR-85M	3.710E+02
KR-87	6.822E+02
KR-88	1.026E+03
XE-131M	1.760E+01
XE-133M	9.980E+01
XE-133	2.711E+03
XE-135	5.673E+02
XE-138	4.512E+02
CS-134	9.797E-01
CS-136	3.918E-01
CS-137	8.519E-01
TE-129M	8.178E-02
TE-131M	5.230E-01
TE-132	5.236E+00
SB-127	2.980E-02
SB-129	7.622E-02
SR-89	1.014E+00
SR-90	5.069E-02
SR-91	1.067E+00
MO-99	2.213E+00
RU-103	1.252E-01
RU-106	3.280E-02
BA-140	4.344E+00
Y-91	1.576E-03
LA-140	2.297E-03
CE-144	1.278E-03
NP-239	2.340E-02

Might you want to use these results in MESO RAD? =
Y :

Figure 7. Releases to the environment computed from the data in Fig. 5.

VI. PLANNED ENHANCEMENTS

Several enhancements are planned for MENU-TACT. Currently, a simpler model is being constructed which requires only the WASH-1400 (USNRC, 1975) source terms, reactor power, and event timing. In the future, another model will be added that will relate plant conditions to source terms. Also, the fill-in-the blank screens will be replaced by graphic data entry screens which will show the configuation of the selected volumes and will allow data entry on the volumes and pathways drawn on the screen.

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The MESORAD Dose Assessment Model[a]

R. I. Scherpelz, J. V. Ramsdell, G. F. Athey, and T. J. Bander

ABSTRACT MESORAD is a dose assessment code developed for use as an emergency response tool by the U.S. Nuclear Regulatory Commission and the U.S. Department of Energy. The code combines atmospheric dispersion and dose calculations to make dose estimates for releases containing as many as 50 radionuclides. It evaluates those dose pathways that are likely to be important during and immediately following a release. The pathways included are the internal dose commitment that is due to inhalation of radionuclides and the external dose that is caused by exposure to radionuclides in the air and deposited on the ground. External doses are computed using either the semi-infinite cloud approximation or the finite puff model. The finite puff model was developed to reduce computational time and is an approximation to the point-kernel integration technique found in other dose models. Initialization of MESORAD requires about 5 minutes, and simulation of the first 3 hours of a release can be completed in about an additional 3 minutes on a super-minicomputer.

I. INTRODUCTION

MESORAD is a computer code that performs dose assessments for emergency response applications. The code was developed at the Pacific Northwest Laboratory for use at the U.S. Nuclear Regulatory Commission's Operations Center in Washington, D.C. and at the U.S. Department of Energy's Unified Dose Assessment Center at Hanford, Washington. The code is used to assess the consequences prior to, during, and immediately following potential or actual releases of radioactive material. The dose pathways represented in the code are the external doses that are due to airborne and deposited radionuclides and the internal dose commitments that are due to inhalation of radionuclides.

II. ATMOSPHERIC DISPERSION MODELS

In MESORAD, atmospheric dispersion models are used to estimate radionuclide concentrations at nodes on two receptor grids. The primary grid is a Cartesian grid in which the spacing between receptors is typically several kilometers. The second grid is a polar grid with receptor nodes at 10-degree intervals that are 0.8, 1.6, and 3.2 kilometers from the release point. This polar grid provides a better resolution of concentration and dose patterns near the release point.

The primary dispersion model in MESORAD is an extension of the Lagrangian puff model in the MESOI computer code (Ramsdell et al., 1983). The model is used to estimate the concentrations at the nodes on the Cartesian grid and may also contribute to the concentration estimates at nodes on the polar grid if the wind direction reverses. A straight-line Gaussian model is used for the primary concentration estimates at polar-grid receptors. Surface contamination resulting from dry and wet deposition is estimated for all receptor locations on both grids.

III. DOSE MODELS

MESORAD assesses the radiological consequences of a release using three dose pathways: the internal dose commitment that results from the inhalation of radionuclides, the external dose that results from exposure to airborne radionuclides, and the external dose that results from exposure to radionuclides deposited on the ground. Dose estimates are computed for all receptors and are available for the most recent 15-minute period as well as the entire period since the beginning of the release.

The dose estimates can be based on as many as 50 radionuclides. Decay schemes have been included in the code, where appropriate, so that production of progeny nuclides by parent decay is represented. Radioactive decay and production are calculated during "holdup" in

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the facility (the elapsed time between the specification of the release inventory and the time of release), during transit from the release point to the receptor, and during residence on the ground (for deposited radionuclides).

A. Internal Dose Pathway

For the evaluation of the internal dose pathway, 50-year inhalation dose commitments are calculated to three organs: whole body, lung, and thyroid. The calculation of inhalation dose at each receptor point uses the ground-level air concentration multiplied by a breathing rate and a dose factor. The computational procedure is described by Scherpelz et al. (1986). The inhalation dose to thyroid is determined for the child-age group, which is generally considered to be most sensitive to thyroid doses. The doses to the other two organs are calculated for the adult-age group.

B. External Dose Resulting from Exposure to Airborne Radionuclides

Two models are used in MESORAD to treat the external dose pathway of exposure to airborne radionuclides, a semi-infinite cloud model and a finite puff model. The semi-infinite cloud model uses a simple calculation, multiplying a dose factor by the air concentrations. The code executes quickly, but provides poor estimates of the dose in cases where the radionuclides are elevated above the ground, or in the cases where the spread of the radionuclides is small compared to the range of emitted gammas. For a better dose estimate in these cases, a finite puff calculation using a discrete-point approximation is utilized. This discrete-point approximation requires more computation time than the semi-infinite cloud calculation; the accuracy of the calculation deteriorates as the spread of the radionuclides grows very large. Thus, the two models are complementary; the finite puff model offers an accurate dose evaluation when the spread is small, and the faster-executing semi-infinite cloud model can give a good estimate when the spread of the radionuclides increases.

The finite puff model used in MESORAD for the calculation of external doses is a simpli-fication of the point-kernel integration technique. In the point-kernel technique, the source volume (the puff) is divided into a number of differential volumes, and the dose at the receptor point that results from each differential volume is evaluated. The total dose equals the sum of the contributions of all the differential volumes. The quality of the computation depends on choosing the proper number of differential volumes. This selection is done iteratively, with increasingly finer divisions of the source volume. This technique also uses large amounts of computer time and is impractical for use in emergency response application.

The discrete-point approximation used in MESORAD is a simplification of the point-kernel integration technique in which the puffs that are used to approximate the radionuclide plume are immediately divided into a number of differential volumes. The iterative nature of a point-kernel integration is thus avoided, and the calculation proceeds much faster. The puffs are divided into differential volume elements using cylindrical coordinates. Each differential volume of the puff is treated as though all the radionuclides were located at a point in the center of the small volume. The dose to a receptor point is then calculated as the sum of the doses from all source points.

The discrete-point approximation was tested by comparing calculations made using a preliminary code called DISCRTPT against hand calculations of simple configurations and against calculations made using PERCS (Reece et al., 1984) and ISOSHLD (Engel et al., 1966), point-kernel codes that are used in the radiation shielding community. In these tests, the puffs were assumed to be cylinders containing air with radionuclides uniformly distributed in them, because the shielding codes needed to assume a uniform distribution in the source region. The comparisons showed that the DISCRTPT code and other methods were in general agreement within ±20% when the puffs were divided into 576 differential volume elements (8 angular divisions, 8 radial divisions, and 9 height divisions). Differences in dose estimates from different codes in the range of 20% are not unreasonable, especially when the codes use different expressions for buildup and different attenuation coefficients.

The performance of the discrete-point approximation was evaluated for use in MESORAD by varying the number of volume elements in the puff in the DISCRTPT code and comparing the dose estimates with the dose estimates computed from 576 puff elements. The results of this process were used to develop the rules that determine the number of volume elements into which puffs are divided. These rules, which represent a balance between the computational time constraints of emergency response and accuracy in dose estimates, are based on the magnitude of the characteristic horizontal dimension of the puff (diffusion coefficient) and the distance from the center of the puff to the receptor.

C. External Dose Resulting from Exposure to Contaminated Ground

The current dose rate and cumulative dose resulting from exposure to contaminated ground are computed at 15-minute intervals. These computations include the contributions of all nuclides deposited on the ground from the start of the simulation to the current time. The dose rate calculation essentially uses the product of the nuclide concentrations on the ground (corrected for radioactive decay and buildin while on the ground) and a dose-rate factor. The cumulative dose is a running summation of dose rate multiplied by duration of the dose rate (usually 15 minutes). The cumulative dose, therefore, indicates the dose that an unsheltered person standing at the receptor location for the entire duration of the scenario would receive.

IV. COMPUTER CODE EXECUTION AND PERFORMANCE

The performance of MESORAD was evaluated on a VAX-11/780 computer with Version 4.2 of the VMS operating system. The evaluation runs were made during normal computer operation periods using the system clock to determine the time required for various simulations. Both real time and central processing unit (CPU) usage were recorded to the closest 0.01 second.

All simulations were based on a common scenario, a simulation of a release following a reactor accident. The scenario assumed that the data files required for MESORAD were available when execution of the code began. This assumption corresponds to the situation where meteorological data are automatically entered into a data file and radionuclide files have been prepared for various postulated accidents. The data bases used in this evaluation include meteorological data for 9 stations and release data for 27 radionuclides. The periods simulated extended from 6 to 72 hours with a maximum release duration of 24 hours.

The real time required for interactive initialization of a research version of the code ranged from about 3.5 to almost 7 minutes. During this period, the operator specified the size of the model domain, selected from among the model options, established the source characteristics, and entered names for the data files. The length of the initialization period depends to some extent upon the model options selected. For example, the average initialization for runs that included dose computations took about 5 minutes, while the average initialization time for runs that only made dispersion and deposition computations took about 4 minutes. The CPU time for initialization was about 14 seconds for the runs with dose computations and about 9 seconds for the runs without dose computations. By careful selection of sets of default options, it should be possible to develop site-specific operational versions of MESORAD in which the initialization time is significantly reduced.

The time required for model computations following initialization varied significantly from run to run, depending primarily on number of other users on the computer. However, the CPU time required to reach a given point in the simulation showed much less variation, and this variation is easily ascribed to differences in the model options selected.

Assuming a release at about midnight in low winds and stable conditions, the full MESORAD code took 156 seconds of CPU time (following initialization) to complete the first 3 hours of simulation, and 661 seconds to complete 6 hours. Repeating the simulation without the polar grid, the CPU times were reduced to 120 and 563 seconds, respectively. The CPU times were not reduced significantly (2-second difference at the end of 6 hours) when the same simulation was run a third time, eliminating the polar grid and the tracking puffs after they left the dose computation grid. For comparison, the MESOI dispersion code (Ramsdell et al., 1983), run within the MESORAD shell, took about 2 seconds to complete 3 hours of simulation and about 6 seconds to complete 6 hours of simulation.

During these simulations, the CPU time per hour of simulation increased initially as the number of puffs on the computational grid increased. However, the time did not reach an equilibrium value as might have been expected, instead it entered a diurnal cycle that appears to be related to wind speed and stability. In nighttime low wind speed, and stable atmospheric conditions, the CPU time per simulation hour reached a maximum of about 180 seconds. When the wind speed picked up and the stability decreased during the day, the CPU time required for each hour dropped to a minimum of about 70 seconds. This variation is not reflected in the total CPU time required to complete a 3-hour simulation because the effects of the increasing number of puffs is dominant. However, it is apparent in the time required to complete 6-hour simulations. Completion of 6-hour simulations starting at about midnight, 6:00 a.m., and noon took 563, 380, and 311 CPU-seconds, respectively.

Grid spacing and release height also appear to affect computational time, but the effects of these sources of variation have not been explored in detail. It is anticipated that the general effect of decreasing grid spacing is to increase computation time because more computations are required per puff. Similarly, an increase in release height is expected to be accompanied by a decrease in computation time, because wind speeds generally increase with height, reducing the time that released material is within MESORAD's dose computation domain.

Recently, the performance of the MESORAD code has been evaluated using more detailed analysis routines. These routines identified the computational time used by various portions of the code. The programming has been optimized in several of these areas resulting in a 30% decrease in MESORAD execution time during the simulations that were timed. Further code optimization is expected.

V. SUMMARY

MESORAD is a dose assessment code developed for use by the U.S. Nuclear Regulatory Commission and U.S. Department of Energy in emergency response applications. The code's purpose is to estimate doses during and immediately following an accidental release of radionuclides. Therefore, the dose pathways treated by the code are internal dose commitment because of inhalation of radionuclides and external dose because of exposure to airborne radionuclides and radionuclides deposited on the ground. Doses accumulated during the most recent 15-minute period and since the beginning of the release are available for use.

Initialization of the code can be completed in about 5 minutes. Simulation of the initial 3 hours of a release can be completed in an additional 3 minutes, and simulation of the first 6 hours of a release can be completed in 5 to 10 minutes. Further optimization of the code can reduce these times.

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Accident Analysis Codes That Predict the Transport of Radiological Aerosol Through a Fuel Cycle Facility as a Result of an Accident

Kevin C. Greenaugh

ABSTRACT. Computer codes are presently being developed which predict the gas-dynamics, material transport, and heat-transfer induced from fire, tornado and explosion accidents in nuclear fuel cycle facilities. These codes consist of models which describe the major phenomena which occur to 0.5 to 5.0 micron particles as they are released in a room, and transported by way of the ventilation system throughout the nuclear fuel cycle facility.

I. ACCIDENT ANALYSIS CODES

Assessment of the environmental consequences of an accident in a fuel cycle facility ultimately involves calculating the atmospheric dispersion of radioactive materials, and estimating the radiation dose to the surrounding population. A radiation dose calculation of this nature requires consideration of the following parameters: (1) wind speed; (2) diffusion stability; (3) dose conversion; (4) type of release; and (5) source amount at the facility boundary. Studies have been performed to develop an understanding of the phenomena associated with each parameter.

Considerable uncertainty lies in estimating the nuclear facility source term. Codes have been developed which are comprised of mathematical models that describe the phenomena occurring during a reactor accident (e.g, HARM, TRAP-MELT), but conditions following a fuel cycle facility accident are typically different. The reactor models provide a theoretical estimation of a source release at the reactor containment boundary of radioactive material as a result of a postulated accident.

Los Alamos National Laboratory (LANL) and Pacific Northwest Laboratory (PNL) under the direction and support of the Nuclear Regulatory Commission (NRC) have been developing a family of accident codes which can be used to analyze tornado, explosion, and fire accidents in nuclear fuel cycle facilities. The developed codes are called TORAC, EXPAC and FIRAC.

TORAC models the gas-dynamics and material flows (radioactive particulate) in a facility and their perturbations due to a tornado outside of the facility. Negative pressures are associated with a tornado. TORAC is used to model backflows throughout the facility due to negative pressures at the facility boundary. A pressure time function at the facility boundary is used to simulate a tornado.

EXPAC is the computer code used to study explosion transients. The source or amount of energy and particulate released as a result of an explosion are determined by a chamber (compartment) model within EXPAC called NORDAL. Explosions can also be simulated by using pressure, temperature, or energy time functions along with mass (gas) and material injection time functions. EXPAC predicts the gas-dynamics and material flows resulting from an explosion within a facility.

FIRAC is the computer code used to study fire transients within a fuel cycle facility. FIRAC predicts material transport (both radioactive particulate and combustion products) and the gas dynamics throughout the facility. It can also predict the heat-transfer within the facility's duct work.

In addition to time functions, fires can also be modeled by an extensive compartment modeling subroutine called FIRIN. An explanation of FIRIN and the explosion compartment model NORDAL and their features will be discussed in the next section of this paper.

A general description of the gas-dynamics, heat-transfer, and material transport models incorporated in the network aspect of each of the codes will be discussed in a later section.

II. COMPARTMENT MODELS

Compartment models were developed to describe the amount of energy and particulate released in a room due to fires and explosions. These compartment models were incorporated into FIRAC and EXPAC, respectively, and are used to determine fire and explosion source terms.

A. Fire Compartment Model

Pacific Northwest Laboratory developed the fire compartment model FIRIN (Owczarski, 1985). FIRIN was developed to estimate the source release of smoke and radioactive particles from potential room fires in nuclear fuel cycle facilities. The technical bases of FIRIN consist of a nonradioactive fire source term model, compartment effects modeling, and radioactive source term models.

1. Fire source term models. Source terms required to characterize a fire include mass loss rates, heat release rates, and combustion product generation rates. The mass loss rate (burn rate) in FIRIN is the net heat flux received by the material in the fire (after taking into account the flame heat flux and the radiation loss) divided by the heat required to produce a unit mass of combustible vapors. The heat release rate in FIRIN is the product of the heat of combustion and the mass loss rate (a correction for incomplete burning is performed). Lastly, the rate of producing combustion products is obtained in FIRIN by multiplying the fractional yield of each product upon combustion by the mass loss rate.

Two burn modes are also considered in FIRIN. They are combustion and smoldering. Combustion is assumed to occur when the oxygen concentration in the room exceeds 15% and high ventilation exists. Smoldering is assumed to occur when the room has less than 11% oxygen and is under-ventilated.

2. Compartment effects. The severity of a fire inside an enclosure can be affected by compartment constraints and barriers. Compartment barriers tend to contain and retain most of the gases, mass, and thermal energy released during the combustion process. Models are incorporated in FIRIN which describe how each of these phenomena interact within the compartment.

Three major types of heat transfer mechanisms are considered in FIRIN - radiative, convective, and conductive. Radiative heat transfer is modeled as occurring from the combustion flame, and the smoke layer to the walls, ceiling, and equipment. In modeling the fire within the compartment, a two-layer regime, is assumed: a hot layer (smoke) on top and a cold layer which contains contaminant free air.

Convective heat is assumed to be carried off by fire gases into the ceiling hot layer. Also, whenever a solid body is exposed to the moving hot layer having a temperature different from that of the body, heat is transferred between the fluid and solids according to Newton's law of cooling.

Compartment floors and ceilings are temporary heat sinks for energy generated from burning materials. A simple one-dimensional form of Fourier's heat conduction equation is used to model the thermal conduction rate within compartment solid media. The conduction equation is written as:

$$\frac{\partial^2 T}{\partial x^2} + \frac{q_1}{k} = \frac{1}{\alpha}\frac{\partial T}{\partial t} \qquad (1)$$

where: k = thermal conductivity
α = thermal diffusivity ($k/\rho cp$)
q_1 = internal energy source term

The equation is solved by a finite-difference technique with convection boundary conditions. The Fourier heat conduction equation is also used to describe wall heat conduction with the following boundary condition:

$$- k \left.\frac{\partial T}{\partial x}\right| = \dot{Q}_{net} \qquad (2)$$

where $\dot{Q}_{net}$ is the heat flux from the flame divided by the wall surface area times Δt for the wall in the cold layer. The heat flux boundary condition to the wall in the hot layer must consider radiative and convective heat rates to the wall.

An overall heat balance in the fire compartment is performed to calculate the hot layer temperature. A simplified form of the internal energy and enthalpy equation is formulated to obtain the hot layer temperature from the principle of conservation of energy, and the assumption that potential and kinetic-energy terms are negligible.

Models for species inventories (carbon dioxide, carbon monoxide, water, HCL, nitrogen, oxygen, methane) of both hot and cold layers are incorporated in FIRIN. Assumptions are made in the mass balance calculations that no mixing takes place between layers, and only air is found in the cold layer. It is further assumed that mixing within each layer is uniform. The ideal gas law is employed to obtain total mass and molar information.

The three major mechanisms of fluid motion considered in FIRIN are flows caused by fire plume entrainment, ventilation, and additional flow paths. Each of these mechanisms can transport radioactive particles out of the compartment. Mechanisms which contribute to the radioactive particle depletion modeled in FIRIN are sedimentation, Brownian diffusion, diffusiophoresis, and thermophoresis.

Radioactive source term models are built into the FIRIN code in the form of subroutines allowing estimation of the mass rate and size distribution of radioactive particles becoming airborne in the event of a fire. These models are based on experimental data obtained from burning contaminated combustibles such as rags, gloves, and plastic bags in gloveboxes or solvents, extraction fluids, and cleaning fluids used in fuel cycle operations.

FIRIN is designed to provide mass and energy input (for the fire room) to FIRAC, which models mass, energy, and heat transfer through the ventilation system of a nuclear facility.

B. Explosion Chamber Model

Los Alamos National Laboratory developed an explosion model called NORDAL that was incorporated into the EXPAC computer code. The NORDAL subroutine is capable of providing the total energy released from TNT, red oil, hydrogen and acetylene explosions in the form of time functions. Parameters needed to model explosions from these materials are the total material mass, the material type, and the characteristic length of the room where the explosion occurs. The characteristic length is assumed to be the cube root of the room volume. The characteristic length along with the speed of sound is used to determine the characteristic time.

Pressure pulses occur with most explosions. After 4-8 pressure pulses, the room experiences an over-pressurization. The characteristic time is defined as the time factor after which a room over-pressurizes and a peak energy release occurs. The total energy released by the explosion caused by the chemical exothermic reaction between oxygen and the exploding material is obtained by integrating the area under the energy time function curve. This time function is used by EXPAC to describe the explosion. The gas-dynamic and material transport affects of the explosion on adjacent rooms is then determined by the network modeling aspect of the EXPAC code.

III. MODELING NUCLEAR FACILITIES

The source terms determined by compartment models or time functions are used to describe the conditions in the near-field (close to the transient). Numerous mechanisms occur which result in a difference between conditions in the near-field and conditions in the far-field (at the facility boundary). For example, particulate deposition will cause a difference in the amount of radioactive particulate released close to the source and the amount that reaches the facility's boundary.

For each of the codes, the ventilation system is assumed to be the primary path for particulate, heat, and gas transport to the nuclear facility's atmospheric boundary. Therefore, mathematical models and a network modeling approach is used to describe material and gas flows through a complex network of rooms, gloveboxes, duct work, filtration systems, and other components typically found in a nuclear fuel cycle facility ventilation system.

A. Gas-Dynamics and Network Modeling

Characteristically, nuclear facility ventilation systems are essentially one-dimensional with many branching and looping ducts. Therefore, a lumped parameter of each system element has been used as the methodology to simulate a ventilation system. This method is sometimes referred to as network modeling and consists of dividing a ventilation system into a number of system elements called branches, joined at certain points called nodes. Spatial distributions are not considered in this lumped-parameter approach. Ventilation components that exhibit resistance, such as dampers, filters, or blowers, are located within system branches. Mass flow rates across branches are calculated from the following momentum balance equation:

$$\frac{d\rho Alu}{dt} = - A\left(P_2 - P_1\right) - F + \rho A \Delta Z g$$

$$- A\left[\left(\rho_2 u_2\right) V_2 - \left(\rho_1 u_1\right) V_1\right] \quad (3)$$

where: ρ = gas density
P_2 = pressure downstream
P_1 = pressure upstream
F = friction
g = gravitational acceleration
V_1 = area out
V_2 = area in

Here the term on the left represents the total momentum. The first two terms on the right represent the pressure force and the friction force, respectively. The third term on the right represents the gravitational force, and the last term represents the rate of momentum influx and efflux by virtue of bulk motion. The last term in equation (3) is assumed to be zero denoting equal momentum flows into and out of the branch.

Frictional forces or resistances are modeled through the application of a power law relationship between force and volumetric flow rates. For example, the resistance through a duct is modeled as being proportional to the volumetric flow rate squared. The resistance coefficient can either be an input parameter that remains constant during a given simulation, or a calculated value based on the relationship between force and flow rate.

The connection points (nodes) of a system element are at upstream and downstream ends of branches. Components that have larger volumes, such as rooms, gloveboxes, and plenums, are located at nodal points. Therefore, a node may possess some volume or capacitance where fluid storage or compressibility may be accounted. The appropriate mass flow rates which are determined upstream and downstream of each node are used along with node energy and mass balances to determine node densities and pressures. The energy and mass equations used in the node calculation are the following:

<u>Energy Balance</u>

$$\frac{v}{\Delta t}\left(C_v\left(\rho_f T_f - \rho_i T_i\right) + \frac{1}{2_{gc}}\left(\rho_f u_f^2 - \rho_i u_i^2\right) + \frac{g}{g_c}\left(\rho_f h_f - \rho_i hi\right)\right.$$

$$= s_1 \rho_1 u_1 \left[C_{pT^1} + \frac{1}{2_{gc}} u_1^2 + \frac{3}{3_c} h_1\right] \text{ inlet}$$

$$- S_2 \rho_2 u_2 \left[C_p T_2 + 1/2_{g_c} u^2 + \frac{g}{gc} h_2\right] \text{ outlet} + Q - \omega \quad (4)$$

where: Q = heat added
ω = work
T_f = downstream branch gas temp.
T_i = upstream branch gas temp.
S_1 = upstream branch flow area
S_2 = downstream branch flow area

Accumulation of kinetic and potential energy are assumed to be negligible for node energy balances.

<u>Mass Balance</u>

$$\frac{v}{\Delta t}\left(\rho_f - \rho_i\right) = \rho_1 v_1 s_1 - \rho_2 v_2 s_2 + \dot{M}p \quad (5)$$

where: ρ_f = downstream branch density
ρ_i = upstream branch density

u_i = upstream branch gas velocity
u_f = downstream branch gas velocity
M_p = injected mass

Once the node pressure and density is determined, the equation of state is used to determine the node temperature.

Consequently, the unknowns that appear in the equations describing system behavior are the mass flow rates for each branch and the temperature and pressure at each node. These unknowns appear in the LANL codes as a potentially large set of simultaneous equations. Solutions to these equations are obtained by a Newton-Ralphson procedure, and decomposition of the large set of equations into independent sets of smaller numbers of equations. The calculated node and branch gas-dynamics are used in many of the material transport models presented in the following sections.

Decoupling of the equations is accomplished through omission, in the linearize energy and mass balance for a given room, of the terms which contain coefficients of variation with respect to the density and pressure of air in adjoining rooms.

B. Material Transport

The material transport models incorporated in the network aspect of FIRAC describe the types of phenomena that affect accident induced material transport of Special Nuclear Material (SNM) as it is transported through a ventilation system. These phenomena include convection, deposition and entrainment. Each of these models assume that the SNM is in the form of solid powders with a size range of 0.5 to 5.5 microns. Condensation is not considered in any of the models, which is unlike the material models incorporated in FIRIN. Therefore, radioactive particulate from an extraction column in a reprocessing facility or particulate from other liquid operations are handled in FIRAC as non-condensing media.

Other phenomena such as thermophoresis, turbulent inertia deposition, and coagulation are prevalent for the size particles considered. These phenomena are being studied, and models are being developed to upgrade the present material transport aspect of the codes.

1. Convection. Convection is the transport of particulate or heat by the circulation or movement of a fluid or gas. When a postulated accident occurs in a fuel cycle facility, small particles in the micron range may be released. The particles can be transported by the movement of a gas (air) through the ventilation system of the nuclear facility. The transported particulate and the host gas are assumed to form a multiphase, multicomponent system. It is assumed in the material transport model of FIRAC that particle sizes are small, that the small particles have a small relaxation time compared to its residence time, and that the particle mass fraction is small relative to gas mass in the same volume. Thus, the particulate would have nearly the same velocity as the gas. Under these conditions the gas and particulate are in dynamic equilibrium. The gas velocity obtained from the gas-dynamics calculation along with the continuity equation

$$\frac{d}{dt}\int_v \rho_p dv = -\int_s \rho_p up \cdot ds + \dot{M}p \tag{6}$$

where: ρ_p = particulate density based on mixture volume (equals gas density)

u_p = particulate velocity (equals gas velocity)

$\dot{M}p$ = source

can be used to determine the material density concentration.

2. Gravitational settling. A small particle falling under the action of gravity will accelerate until the drag force just balances the gravitational force. It will then continue to fall at a constant velocity known as the settling velocity. The settling velocity can be obtained by equating the drag force to the gravitational force, then solving for velocity. The drag force is obtained by solving the Navier-Stokes equation which describes fluid motion, using simplifying assumptions. The solution is referred to as Stoke's Law (Hinds, 1982).

$$F_D = 3\pi\eta ud \tag{7}$$

where: η = viscosity
u = particle velocity
d = particle diameter

Equating Stoke's law to the gravitational force produces the settling velocity equation:

$$u_s = \frac{\rho_p d^2 g}{18\eta} \tag{8}$$

The settling velocity is used in FIRAC as a sink term to account for the amount of material that is deposited due to gravity throughout the ventilation system as the material is convected.

3. Entrainment. Entrainment is the displacement of stationary particles when aerodynamic forces are equal to or exceed the threshold velocity (air speed) required to overcome particulate restraining forces.

Iverson (Iverson, 1976) conducted experiments to determine the threshold friction speed (U_{*t}) for a number of material sizes and densities for a thick bed (0.5 inches or greater) of material. The parameters U_{*t} and particle diameter (d_p) were plotted, and equations were fitted to the data. The resulting equation along with the equation which relates the friction velocity to the velocity at the boundary edge can be used to determine the friction and threshold velocities. Travis (Travis, 1975) suggested that the following equation for the particle flux (q_v) can be used to determine the mass of particles per unit area per unit time that go into suspension due to entrainment

$$q_v = q_h \left(C_v/U^3_{*t}C_h\right)\left[(U^*/U_{*t})^{\rho/3} - 1\right] \qquad (9)$$

where: ρ = mass percentage of suspendable particles

C_v, C_h = empirical constants (2×10^{-10} and 10^{-6}, respectively)

q_h = horizontal mass flux along a plane $(2.61\ (\rho/g)(u_* + u_{*t})^2\ (u_* - u_{*t})$

The amount of material suspended is used as a source term in FIRAC, and is added to the amount of material being convected.

C. Duct Heat-Transfer

The purpose of the duct heat-transfer model is to determine how a combustion gas in a system cools down or heats up as it passes through a duct. FIRAC is the only code of the three which has heat-transfer features. Duct work is the only ventilation component for which heat-transfer is modeled. The model predicts the temperature of a gas leaving a duct when the inlet gas temperature is known.

The heat-transfer model is comprised of five modes of heat transfer. Forced convection between the combustion gas and the inside duct walls is calculated by the following equation

$$T_{out} = T_{in} - \frac{Q_i}{\dot{m}\ C_p} \qquad (10)$$

where: T_{in} = temp. of gas coming in

T_{out} = temp. of gas going out

Q_i = energy transferred from gas to duct wall due to convection (Q_{ci}) and radiation (Q_{ri})

$\dot{m}$ = mass flow rate of gas coming in

C_p = specific heat of gas coming in

The convection component of Q_i is a function of the gas temperature and the duct wall temperature. The radiation component of Q_i is a function of intensity factors associated with the duct wall, gas temperatures, and gas composition. The wall temperature is calculated by solving the heat conduction equation. Natural convection heat transfer from the outside duct walls to the surroundings, and radiation heat transfer from the outside duct walls to the atmosphere are also modeled.

IV. CODE OUTPUTS

Code outputs include calculated temperatures, pressures, and material concentrations at nodal points (nodes) as a function of time. Other code outputs include predicted flows, material flow rates, pressure differentials, and material accumulations on filters as a function of time. Code outputs are written to a scratch file during the transient calculation for editing or plotting by the output processor.

Figures 1-2 illustrate sample temperature and pressure code outputs for a room where an explosion is simulated using NORDAL. Figure 3 depicts sample volumetric flow rates through branches or duct work connected to the room where the explosion was simulated.

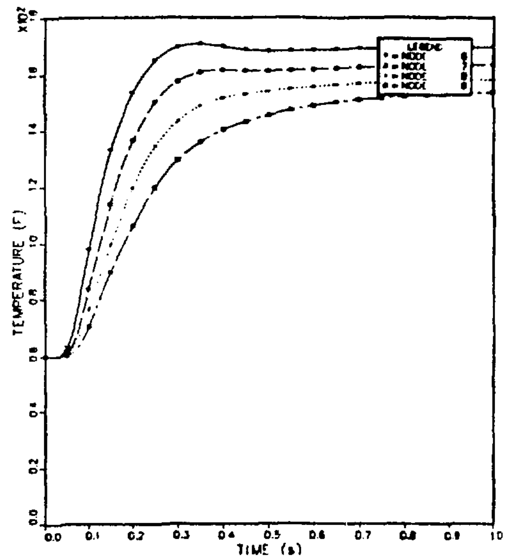

Figure 1. Plot of temperature versus time for a room where an explosion occurs.

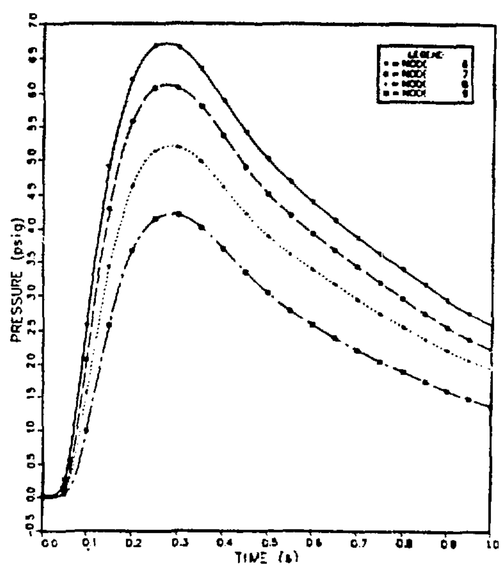

Figure 2. Plot of pressure versus time for a room where an explosion occurs.

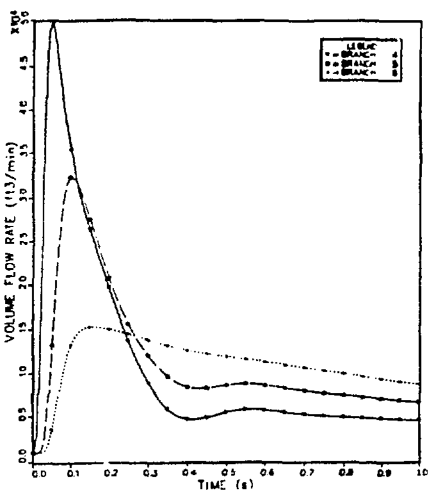

Figure 3. Plot of volumetric flow versus time for ducts connected to a room where a transient is occuring.

V. CODE SPECIFICATIONS

The LANL developed codes (TORAC, EXPAC, and FIRAC) were programmed to run on a CDC 7600 computer. Each of the codes were also modified to run on a VAX 11/780 system. A personal computer (PC) version of FIRAC has been created. The PC version of FIRAC does not have FIRIN incorporated within it. For FIRIN to be incorporated an overlay procedure must be performed. The present PC version of FIRAC must be compiled in segments due to the size limitation of most PC Fortran compilers. Each segment must be linked to produce an executible FIRAC file.

The size of each code varies due to their different features. FIRAC is the largest code. It has about 9000 lines of code (including common and comments) and 85 subroutines. Its large size is primarily due to the extensive compartment model within it (FIRIN), which consists of 2247 lines of code. EXPAC also has about 9000 lines of code and 44 subroutines. The explosion chamber model NORDAL contributes greatly to its size. TORAC is the smallest of the three codes. It consists of about 5000 lines of code and only 26 subroutines. TORAC lacks the capability of modeling multiple species and uses an incompressible flow algorithm to model gas dynamics. Each of the codes are comprised of numerous subroutines which are indicative of their modular structure which allows relatively easy modification.

Manuals that describe the use of each code are published as Los Alamos Reports (Gregory, 1985). Each manual consists of a chapter which explains in general terms each of the code features, how to develop input files, and sample problems.

VI. CONCLUSIONS AND FUTURE WORK

Codes that predict the gas dynamics, heat-transfer, and material transport induced from fire, tornado, and explosion transients in nuclear fuel cycle facilities are being developed. These codes would reduce the amount of uncertainty involved in estimating the amount of material that reaches a facility boundary as a result of an accident. The source term obtained from the codes along with plume models and other models which describe environmental conditions outside of the facility can estimate the impact on public health and safety as a result of an accident.

Because of the detailed modeling of ventilation systems incorporated in each code, the codes can be used as a tool to aide in the designing of ventilation systems.

The material transport models presently incorporated in the codes are primarily for the transport of non-condensing, spherical particles 0.5 to 5.5 microns in diameter (typical size of SNM powders). The codes are presently inappropriate to model fission product releases from spent fuel storage facilities, because spent fuel gases and particulate are smaller than the size particles modeled in the codes. Diffusion models must be incorporated into the codes to determine an accurate prediction of smaller particle deposition. The material transport models are also inappropriate for modeling accidents involving wet solutions similar to those that exist at solvent extraction operations. Though the FIRIN subroutine has the capability of modeling diffusiophoresis, which is due to condensation and Stefan flow, the network modeling approach of the codes is not in itself capable of modeling condensation.

Deposition due to thermophoresis and turbulent inertia is presently being incorporated into the codes.

An experimental facility at New Mexico State University has been built to verify experimentally the heat-transfer, material flows, and gas-dynamic models incorporated in the codes. The experimental apparatus consists of duct work connected to a cylindrical and rectangular compartment. A fan provides air flows similar to those encountered in a ventilation system. The air is heated by passing it through a furnace, which simulates hot air from an explosion or fire. The air passes through a number of dampers and filters in the duct work. Pressure transducers and thermocouples are also positioned through the duct to measure pressure drops and heat, respectively.

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REALM: An Expert System for Classifying Emergencies

Robert A. Touchton, Alan Dale Gunter, and David G. Cain

ABSTRACT

In late 1983, EPRI began a program to encourage and stimulate the development of Artificial Intelligence applications for the nuclear industry. As a part of that effort, EPRI has contracted Technology Applications, Inc. (TAI) to develop an artificially intelligent Emergency Classification System Prototype. This paper presents a description of and a status report on this recently completed prototype.

I. EMERGENCY CLASSIFICATION DOMAIN OVERVIEW

Because there exists the possibility of an accident at a nuclear facility which could cause adverse health effects in the areas surrounding the plant, a great deal of planning and preparation takes place to reduce or eliminate public health consequences of a reactor accident, should one occur. One of these planning elements involves the classification of emergencies into one of four standardized classes by use of Emergency Action Levels (EALs). The four classes of emergencies are:

Notification of Unusual Event
Alert
Site Area Emergency
General Emergency

EALs represent the observables, thresholds and logic for mapping plant and site conditions into the appropriate emergency class. A great deal of information is available in the power plant (both sensors and manual observables) to help the staff identify and classify an emergency situation. However, this wealth of data must be reduced in an expeditious manner during a time when the staff is already busy responding to an off-normal condition. This would seem to be a problem well-suited to computerization, and yet, at most plants, manual look-up tables are used in the emergency classification process. Attempts to computerize this process are hindered by the fact that the data contains significant uncertainties. There may be missing, conflicting, or ambiguous sensor readings or there may be false alarms. Therefore, **judgement** is required for proper interpretation of the information. The use of conservative assumptions within a computer program (a technique often used to solve this problem in many conventional safety-related programs) does not offer a solution in the current problems because over-reacting is just as detrimental as under-reacting.

The use of an Expert System approach can overcome these problems by incorporating Domain Knowledge (heuristics) into the computer program such that it can begin to make judgements regarding the data. Such a solution not only computerizes the data acquisition and processing, but also can appropriately compensate for the uncertainties contained in the data.

II. REALM OVERVIEW

The Reactor Emergency Action Level Monitor (REALM) is designed to provide expert assistance in the determination of the appropriate emergency status for a nuclear power plant. REALM is an EPRI-funded expert system developed by Technology Applications, Inc. to operate in a real-time processing environment. REALM embodies a hybrid architecture and thus utilizes both rules and object-oriented programming techniques. The rule-base consists of two general classes: deterministic rules, which codify the logic embodied in the current EAL tables, and heuristic rules, still under development, which will resolve ambiguities and data conflicts, identify false alarms, and draw inferences in light of missing data. The heuristic rules will go beyond the current EAL structure to address the more problematic scenarios and entail a more symbolic representation of the plant information. The heuristic rule processor (inference engine) also has the ability to accept and propagate belief information such that conclusions can include a certainty factor.

REALM has been designed to ultimately interface with the plant in some manner in order to collect sensor data. The most likely interface is with the Safety Parameter Display System (SPDS) computer, since it is already charged with the collection of safety and emergency-related data. However, during this prototype development stage, REALM utilizes simulated sensor data which is stored on disk and read in as if it were actual instrument readings. REALM must also interface with its user(s). Approximately 1/4 to 1/3 of the necessary information must be manually entered into the system. However, the manually derived inputs relevant to any given emergency is small and, thus, the user is prompted only for those relevant inputs.

REALM has four operating modes of which two are on-line (Actual and Trial) and two are off-line (Scenario Development and Training). In Actual Mode, REALM interprets the input to identify and classify emergencies in real-time. Thus, REALM will display to the user its current findings on the appropriate emergency class and the reasoning which led to those findings. The Trial Mode allows the user to make assertions in order to assess their impact (e.g. see if terminating flow in a certain system would cause a higher Emergency Class to be invoked).

The Scenario Development Mode serves as a step-wise training and exercise file builder. This mode operates off-line and in non-real-time, thus allowing a user from the Emergency Planning or Training Staff to assert a set of plant conditions, see the impact on the EALs, look ahead to find events which would cause the next higher level of emergency, and finally store on disk that set of conditions if desired. The Training Mode uses a Training File (developed using the Scenario Development Mode) to simulate the plant's behavior during an emergency in order to train operators in emergency classification and users of REALM in its operation. The Training Mode operates in real-time (as represented on the simulated accident scenario) but is off-line and includes the scoring and archiving of the trainee's response.

Within each mode, the user may request a vulnerability analysis or the system's rationale for the current plant classification. The vulnerability analysis evaluates the current knowledge context, reviewing all of the rules relevant to the next higher emergency class and determining what n (for n = 1, 2, 3, 4) events would cause that higher class to be declared. Conceptually, this involves backward-chaining inference to identify which rules are partially satisfied and to list the missing antecedents. The processor then translates the list and displays it in the Vulnerability Window. The rationale processor will collect all of the facts (and certainty factors, if applicable), process the list, and display the information in the Rationale Window.

REALM resides on a Xerox 1108-121 Lisp Machine within Intellicorp's Knowledge Engineering Environment (KEE), with extensions (i.e. methods, functions, procedures, and/or other utilities) written directly in Interlisp-D. The power of KEE's hybrid architecture for knowledge representation (frames, active values/demons, or rules) along with the powerful man/machine interfaces (graphics, windows, KEE editor, and mouse) have been utilized to produce a productive/effective/efficient emergency management decision aid.

III. *REALM PROBLEM-SOLVING APPROACH*

The process of emergency classification being developed in REALM involves a dialogue among several "mini-experts," each charged with investigating a specific aspect of the problem. First, REALM must read-in and evaluate the observables (sensored and manual). This step includes flagging of missing and out-of-expected range readings, the setting and monitoring of clocks, and trending of the data. The Conditions Expert reviews the data looking for specific conditions of interest (such as steam generator levels and pressures). Next, the Radiation Expert must assess radiation levels and location. His job is to determine abnormal amounts or locations of radiation and to monitor release rates and off-site dose projections. Then, the Critical Safety Functions Expert must evaluate the status of the various Critical Safety Functions and, likewise, the Fission Product Barrier Expert must evaluate the integrity of each of the three Fission Product Barriers. Next, the Threats Expert performs first-order diagnostics to look for known accident sequences ("first-order" means that the diagnosis only goes deep enough to classify the emergency but probably not deep enough to correct it). If a specific accident is identified, the Resources Expert checks for the operability or availability of the resources intended to counter that accident. Finally, the EAL Expert, using the input from the other experts, classifies the emergency level by evaluating parameters against their thresholds, by balancing threats against countermeasures, and by evaluating the degradation (or potential degradation) of the Critical Safety Functions and Fission Product Barriers.

IV. *REALM STATUS*

The REALM "alpha" prototype was completed in early September and is currently undergoing extensive evaluation by the TAI Project Team, as well as personnel from EPRI and Consolidated Edison (the collaborating utility). The primary focus of current efforts is on system performance (i.e. REALM response time). During the development of REALM, no effort was expended on maintaining system performance. An early prototype (completed in February, 1986) could process a "snapshot" of plant data in about 30 seconds, a duration considerably faster than could be accomplished by a human expert charged with solving the problem. However, as features and scope have been added, performance has gradually degraded to about 10 minutes. Therefore, our current mission is to get REALM's response time back to the 1/2 minute time-frame. Other current work revolves around completion and incorporation of "advanced topics" such as model-based conflict resolution and certainty factor propagation.

Finally, we have begun to actively solicit additional industry evaluations of REALM and thus extend an invitation to utilities to visit TAI (in Jacksonville, FL) or EPRI (in Palo Alto, CA) to participate in our formal evaluation program.

Section IV

Economic Issues

Chairman: *David Aldrich*
Science Applications International Corporation

Perspectives on the Economic Risks of LWR Accidents[a]

Lynn T. Ritchie and Richard P. Burke

ABSTRACT. Models which can be used for the analysis of the economic risks from events which may occur during LWR operation have been developed. The models include capabilities to estimate both onsite and offsite costs of LWR events ranging from routine plant forced outages to severe core-melt accidents resulting in large releases of radioactive material to the environment. The economic consequence models have been applied in studies of the economic risks from the operation of US LWR plants. The results of the analyses provide some important perspectives regarding the economic risks of LWR accidents. The analyses indicate that economic risks, in contrast to public health risks, are dominated by the onsite costs of relatively high-frequency forced outage events. Even for severe (e.g., core-melt) accidents, expected offsite costs are less than expected onsite costs for a typical US plant.

I. INTRODUCTION

Several studies have examined the economic risks from unanticipated events which occur, or could occur, during US LWR operation (Burke et al., 1984; Starr and Whipple, 1981; USNRC, 1975.) Models have been developed in one of these studies to estimate the economic consequences of LWR forced outages and accidents (Burke et al., 1984). These economic consequence estimates can be used together with estimates of event frequencies to calculate the "expected" losses from LWR operational incidents of various severities. Both "onsite" costs, which either occur at onsite locations or most directly affect the plant licensee, and "offsite" costs, which most directly affect the public surrounding the plant, are included in the predictions of the economic risks from LWR operation.

This paper provides a brief overview of the analytical models which have been developed to estimate the economic risks of LWR outages and accidents. Assumptions regarding post-accident offsite emergency protective measures, which are closely tied to estimates of economic consequences, are also reviewed. The results of an example analysis performed using the economic consequence models are discussed. Some broad perspectives derived from studies of economic consequences of reactor accidents which have been performed are reviewed, with comments in light of the (limited) publicly available information regarding the Chernobyl event.

II. ACCIDENT ECONOMIC IMPACTS, MODELS

A. Onsite Costs

Economic consequence models have been developed to include the following onsite costs which either occur at onsite locations or most directly affect the plant licensee:

- power production cost increases (replacement power costs) due to LWR outage time,
- physical plant capital losses caused by severe accidents,
- plant decontamination costs,
- plant repair costs,
- early decommissioning costs after severe accidents,
- plant worker health impact costs.

The power production cost increases caused by the need for utilizing generating facilities with higher fuel cycle costs during LWR outages are the dominant loss contributors for unanticipated forced outage events which do not result in core damage. These costs can be estimated using models developed at Argonne National Laboratory which account for regional and/or plant-specific variations in the mix of generating facilities used to provide

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replacement power and the costs of replacement fuels (Buehring and Peerenboom,1982; VanKuiken et al., 1984). These costs range from approximately \$200/MWe-day up to \$1000/MWe-day depending on the plant location in the US, seasonal variations, and the costs of available fossil fuels. Plant repair cost estimates for forced outage events which do not result in core damage are based on operating experience in the US. The costs of plant repair have generally been small relative to replacement power costs for events which do not result in core damage.

For more severe accidents involving core damage, power production cost increases, physical plant capital losses, and plant decontamination costs are the most important onsite cost contributors. Plant decontamination costs are modeled based on experience with the Three Mile Island Unit #2 cleanup program and engineering studies of post-accident decontamination (Murphy and Holter,1982). These costs are estimated to range between \$1B and \$2B depending on the physical progression and severity of the accident. Costs associated with decommissioning before the end of planned plant lifetime and the costs of possible plant worker health effects do not contribute significantly to the expected onsite losses from either routine forced outage events or severe accidents. In addition to the onsite cost components included in the consequence models, electric utility business costs, nuclear power industry losses, and onsite litigation costs have been addressed in the development of economic consequence models. These costs could be very important to specific organizations after severe accidents, but are not included in estimates of societal losses.

B. Offsite Costs

Economic consequence models have also been developed to estimate the offsite costs of severe accidents which might result in a release of radioactive material to the environment. The latest version of economic consequence models developed at Sandia National Laboratories are contained in the MACCS (MELCOR Accident Consequence Code System) code (Alpert et al.,1985; Chanin et al., 1986). The MACCS code models the impact of atmospheric dispersion and deposition, dosimetry, and public protective actions on offsite health and economic consequences. Health consequences and offsite costs are calculated probabilistically based on the estimated frequencies of specific radioactive releases (source terms) and the meteorological conditions at the time of a release. Economic consequence estimates have also been available in earlier reactor accident consequence models including CRAC2, UFOMOD and MARC (USNRC, 1975; Bayer et al.,1982; Clark and Kelly, 1981).

The MACCS models include the following offsite costs of public protective measures and health impacts for severe LWR accidents which might result in a significant release of radioactive material to the environment:

- evacuation costs,
- temporary population relocation costs,
- agricultural product disposal costs,
- land and property decontamination costs,
- land interdiction costs,
- permanent population relocation costs,
- health impact and medical care costs.

The costs of evacuation and temporary population relocation include outlays for food, housing, and transportation to move individuals either prior to a release of radioactivity or from areas contaminated immediately after a release occurs. The loss of productivity of individuals moved from contaminated areas is also included in the relocation costs. Agricultural product disposal costs are estimated based on the market value of contaminated goods which are not suitable for consumption. Land and property decontamination costs are based on cleanup efforts aimed at achieving specific exposure reduction factors in urban and rural areas. In addition, the cost of population relocation during decontamination is also included in the offsite cost models. The costs of land interdiction are estimated using present value discounting and the estimated tangible wealth contained within areas which cannot be decontaminated to acceptable levels for habitation. The costs of permanent population relocation from interdicted areas, including possible periods of productivity losses, are also calculated in the models. Finally, the purely economic costs of offsite health effects, including lost productivity and medical costs, are included in the models.

III. EXAMPLE ANALYSIS

The onsite and offsite economic consequence models have been combined with historical outage frequency data and estimates of severe accident frequencies to estimate the economic risks from plant operation over the remaining lifetime of a typical US LWR facility. Table 1 shows the expected losses from both routine forced outage events and severe (core-melt) accidents for the remaining lifetime of the example plant (assumed to be approximately 30 years). The present values of expected plant losses for the remaining lifetime are expressed in 1982 US dollars and are shown for discount rates of 0%, 4%, and 10%. Table 1 shows that the expected losses from severe accidents are small (\$1-\$6 million dollars) relative to the expected losses from routine outage events (\$84-\$270 million dollars) for the remaining lifetime of this plant. This results from the relatively high frequency of forced outage events and the substantial power production cost increases for LWR forced outages. Table 1 also shows that even for core-melt accidents, expected onsite losses are substantially larger than expected offsite losses.

An example of the breakdown of offsite cost

Table 1. Present Value of Routine Outage and Severe Accident Economic Risks for Remaining Life of Example Plant

Routine Forced Outage Events (No Core Damage)		Core Melt Accidents ($\approx 6 \times 10^{-5}$/reactor-year)*		
Discount Rate	($\approx$10/reactor-year)	Offsite	Onsite	Total
0%	$\$2.7 \times 10^{8}$	$\$4.4 \times 10^{5}$	$\$5.5 \times 10^{6}$	5.9×10^{6}
4%	$\$1.6 \times 10^{8}$	$\$2.5 \times 10^{5}$	$\$3.3 \times 10^{6}$	$\$3.6 \times 10^{6}$
10%	$\$8.4 \times 10^{7}$	$\$1.3 \times 10^{5}$	$\$1.7 \times 10^{6}$	$\$1.8 \times 10^{6}$

* Estimated risks for core-melt accidents based on RSS PWR core-melt accident frequencies and source terms with consequence calculations performed with new economic consequence models.

components for both a large release of radioactive material (the PWR-2 release category in the RSS) and a smaller release (the PWR-5 release category in the RSS is shown in Table 2. All costs are expressed in 1982 dollars and are mean estimates conditional upon the release. The table shows that costs associated with land and property decontamination and interdiction are the most important contributors to offsite costs for a large release of radioactive material, and that the costs associated with evacuation, agricultural product disposal, and health effects and health care costs become more important for smaller releases. However, the comparison of total offsite and onsite costs in Table 2 shows that offsite costs are negligible relative to onsite costs for the PWR-5 source term. Even for the PWR-2 source term, onsite costs are larger than offsite costs for the example site.

Table 2 also shows other attributes of the post-accident recovery program which are calculated in the economic consequence models. The estimates of population exposures avoided by protective measures are useful for performing cost-benefit analyses in developing protective action criteria. The attributes of the required decontamination program, including exposures to workers and labor requirements, are useful for determining if resource limitations would be a problem in postaccident recovery operations. For example, Table 2 shows that a large number of decontamination workers would be required to complete the recovery program in a short period of time following a very large release (PWR-2) of radioactive material.

In general, analyses performed with the offsite economic consequence models indicate that the costs of population evacuation, temporary relocation, and possible agricultural product disposal are important for severe accidents which result in only in very small releases of radioactive material to the environment. The costs of decontamination or interdiction of offsite land and property become important for severe accidents which result in larger releases of radioactive material, and dominate the offsite costs of low probability accidents with very large releases of radioactive material. The economic consequence models in MACCS are currently being employed in studies of LWR accident consequences that are based on updated source term information. The studies have shown that the offsite economic consequences of an accident are strongly dependent upon source term definition and the criteria chosen for the implementation of population protective measures including land area decontamination, interdiction, and population relocation. Changes in source term definition and the criteria chosen for the implementation of population protective measures can impact offsite cost estimates by several orders of magnitude. The site demographic characteristics also directly influence economic consequence estimates, but less dramatically.

IV. PERSPECTIVES AND DISCUSSION

The analyses performed with the economic consequence models described in this paper provide the following general perspectives and conclusions regarding the economic risks from US LWR operation:

1. In contrast to public health risks, the economic risks from US LWR operation are dominated by relatively high frequency, small consequence forced outage events. Most of the costs of these events results from reduced plant availability and capacity factors and the need for use of higher marginal cost fuel sources for the generation of electricity.

2. The economic risks from US LWR operation are dominated by onsite losses resulting from replacement power costs for short-term outages. Severe accident economic risks are also dominated by onsite losses including replacement power costs, plant capital losses, and plant decontamination costs. Only very low probability core-melt accidents with large releases of radioactive material could result in offsite costs as large as onsite costs.

3. The onsite economic risks from severe accidents may be significantly increased for plants at multiple unit sites. This is due to the possibility that a single event may result in damage and loss of power production capability at more than one plant.

4. Offsite costs for events which result in negligible or small releases of radioactive material to the environment are predicted to be small compared to onsite costs. The costs of possible population evacuation, temporary relocation, and agricultural product disposal are the most important offsite cost contributors for these events.

5. The costs of decontamination or interdiction of offsite land and property become more important for low probability, severe accidents that can result in larger releases of radioactive material.

6. Offsite economic consequence estimates are also strongly dependent upon the definition of accident source terms, including the specified radionuclide content of a release.

7. Estimates of offsite accident costs are strongly dependent upon assumptions regarding population protective measures and the criteria employed for decision making regarding the implementation of protective actions at offsite locations.

Table 2 . Onsite and Offsite Accident Costs Conditional Upon RSS Category PWR-2 and PWR-5 Releases, Example Plant and Site

	Mean Costs (1982 U. S. Dollars)	
<u>Offsite Cost Component</u>	<u>PWR-2 Release</u>	<u>PWR-5 Release</u>
Evacuation	$\$4.4x10^6$	$\$4.4x10^6$
Emergency Phase Relocation	$\$2.2x10^7$	$\$5.9x10^5$
Intermediate Phase Relocation	$\$7.6x10^7$	$\$1.7x10^6$
Agricultural Product Disposal	$\$9.1x10^7$	$\$2.3x10^6$
Population Relocation during Decontamination	$\$7.1x10^7$	$\$2.8x10^5$
Land and Property Decontamination	$\$6.4x10^8$	$\$8.9x10^6$
Land and Property Interdiction	$\$1.6x10^8$	$\$9.9x10^4$
Interdicted Population Relocation	$\$2.7x10^7$	$\$2.4x10^2$
Offsite Health Effects and Health Care	$\$1.7x10^8$	$\$6.8x10^6$
Total Offsite Cost	$\$1.3x10^9$	$\$2.5x10^7$
<u>Total Onsite Cost</u>	$\$3.3x10^9$	$\$3.3x10^9$
<u>Other Attributes from Economic Models</u>	<u>PWR-2 Release</u>	<u>PWR-5 Release</u>
Total Population Dose Incurred, 0-100 Years	$1.5x10^5$ Person-Sv	$5.9x10^3$ Person-Sv
Total Population Dose Avoided by Protective Measures	$3.1x10^5$ Person-Sv	$1.3x10^3$ Person-Sv
Total Dose to Decontamination Workers	$2.6x10^3$ Person-Sv	$1.7x10^1$ Person-Sv
Labor Required for Decontamination Program	$1.1x10^4$ Person-Years	$1.7x10^2$ Person-Years
Number of Decontamination Workers Required for Completion of Program in 90 Days	$4.6x10^4$ Persons	$6.9x10^2$ Persons

8. The analytical models and quantitative predictions of offsite economic losses from severe reactor accidents may vary significantly for different nations due to fundamental differences in national and/or regional economies. However, the philosophy employed in the development of the MACCS economic consequence models, specifically the calculation of economic consequences based on projected implementation of population protective measures, has broad applicability.

9. The economic consequence models described in this paper have been developed specifically for LWR plant accidents, and no calculations have been performed for other plant types. However, the limited publicly available information regarding the Chernobyl event seems consistent with the assumptions and philosophy incorporated into the economic consequence models, particularly in the modeling of offsite population protective measures.

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Restoration of a Radiologically Contaminated Site: SAGEBRUSH IV

Jack J. Tawil

ABSTRACT This paper describes the development of a site restoration plan for the SAGEBRUSH IV exercise held in northeastern Washington state in mid-August, 1986. DECON, a computer program developed by Pacific Northwest Laboratory, was used to produce information around which the site restoration plan was developed. The features of DECON that are demonstrated in this paper indicate its potential usefulness as a planning tool for site restoration. Strategies that are analyzed with DECON include: 1) prohibiting specific operations on selected surfaces; 2) requiring that specific methods be used on selected surfaces; 3) evaluating the trade-off between cleanup standards and decontamination costs; 4) evaluating the sensitivity of the results to various assumptions; and 5) varying cleanup standards according to expected human exposure to the surface.

I. INTRODUCTION

In August 1986, several federal, state and local agencies participated in a field exercise at a remote location in northeastern Washington state, near Kettle Falls. Called SAGEBRUSH IV, the exercise centered around a simulated mid-air collision between a C-141 and an F-16 aircraft. Nuclear weapons aboard the C-141 were released and damaged, causing dispersion of radioactive products into the environment. A site map is shown in Figure 1. This paper describes the site restoration analysis conducted during the exercise, using DECON, a computer program developed at Pacific Northwest Laboratory. In addition to the analyses performed at the exercise, the paper also is used to demonstrate other available features of DECON.

II. A DESCRIPTION OF DECON

DECON is an analytical tool that can be used to develop a site restoration plan for extensive land areas that have been contaminated with radiological products. It works on the principle of minimizing the social costs of the accident by selecting a decontamination strategy. Although not all of the off-site social costs of an accident are explicitly considered by DECON, a large majority of them are taken into account. Information reported by DECON includes 1) the least costly decontamination method that will effectively restore each contaminated surface, 2) the cost of the method, 3) its effectiveness, 4) the rate at which it can be applied, and 5) the manpower and equipment required to implement it.

DECON makes use of two data bases. The first, the reference data base, contains data on decontamination operations and methods. An operation is defined as a single technique for decontaminating a surface. Operations currently implemented by DECON and their symbols are presented in Table 1. A decontamination method is defined as one or more sequential operations. For example, the method VFR consists of the sequential operations: vacuum (V), foam (F), and remove and replace (R). The reference data base is documented in Tawil et al. (1985).

TABLE 1. Decontamination Operations and Symbols

A	Plow	P	Thin Paved Layer
B	Vacuum Blast	Q	Very Hi Pres. Water
C	Strippable Coating	R	Remove & Replace
D	Defoliate	S	Sandblast
E	Leach-EDTA	T	Sealer; Fixative
F	Foam	t	Fixative, Aerial
G	3" Asphalt	U	Hydroblast
H	Hi Pressure Water	V	Vacuum
I	Steam Clean	v	Double Vacuum
J	Wash & Scrub	W	Lo Pressure Water
K	Resurface	X	Scrape 4"-6"
L	Leach-FeCl	x	Double Scrape
M	Close Mow	Y	Deep Plow
N	Clear; Harvest	Z	Remove Structure
O	Plane, Scarify; Radical Prune		

The second data base is comprised of site-specific information, including the type of property that is contaminated, the value of the property, and the severity of contamination. The first step in preparing the site data base is to divide the accident site into a grid. The grid elements may be of any size or shape. Two criteria are applied in defining the grid elements. First, we assume that all exterior, horizontal surfaces within a grid element are equally contaminated. Second, we assume that within a grid element physically similar property has the same

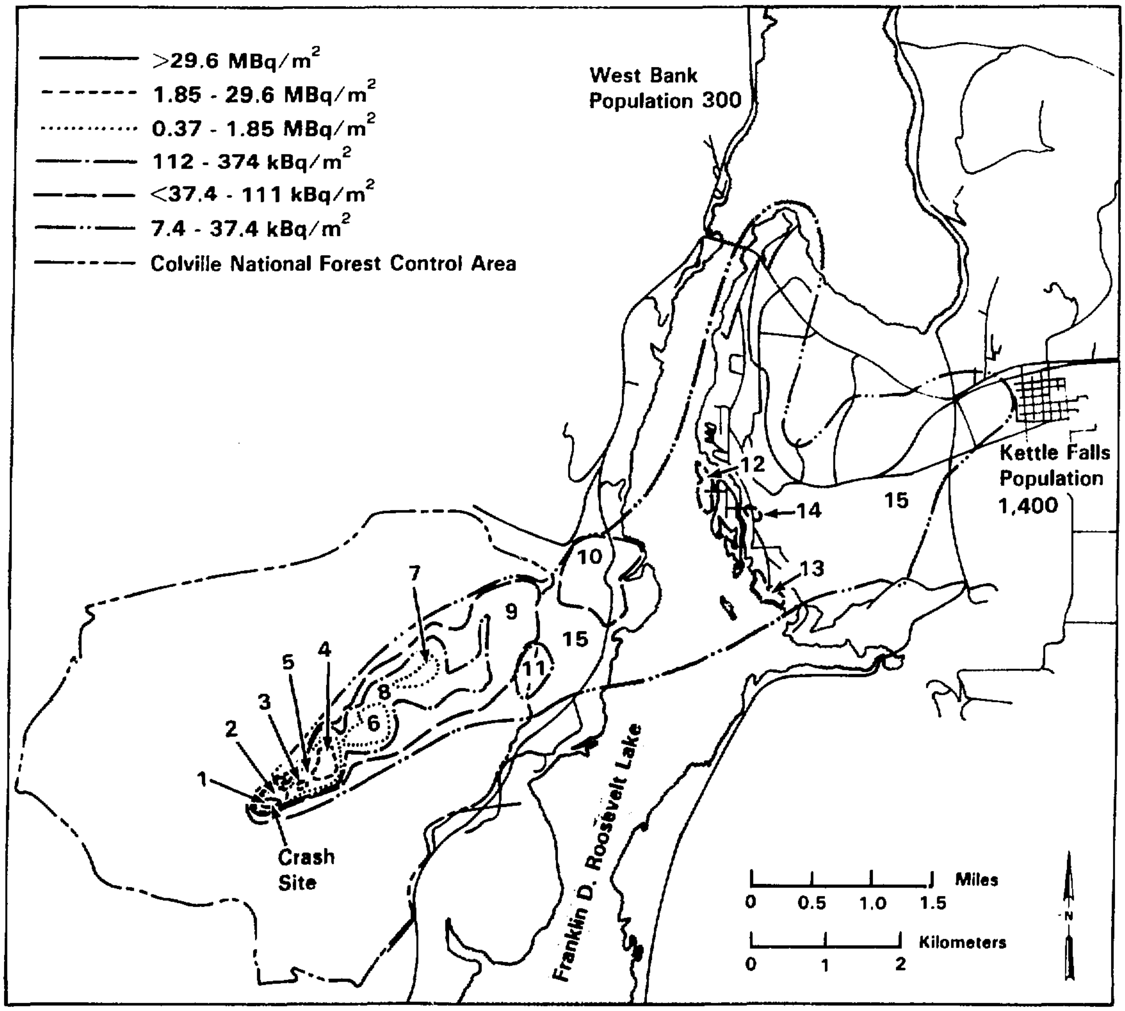

Fig. 1. SAGEBRUSH IV Exercise Site with Isopleths and Grid Elements

economic value. There is virtually no limit to the number of grid elements that can be processed by DECON. While a finer grid could be expected to give more accurate results, it would also require the user to provide a larger quantity of site-specific information.

DECON operates on the principle that identical methods can be used to decontaminate like surfaces that have been equally contaminated. The appeal of this approach is that the entire analysis can be based on surfaces. Some land uses, such as streets, wooded areas and vacant land, can each be thought of as consisting of just one type of surface. Other land use types--notably residential, commercial and industrial--are best thought of as consisting of a wide variety of surfaces. These types must be decomposed into their constituent surfaces if they are to be made amenable to the "surface" approach being suggested here. The surface types currently implemented by DECON are listed in Table 2, along with exposure factors, which are discussed later.

TABLE 2. Surface Types Currently Implemented by DECON, and Exposure Factors

Agricultural Fields	1.0
Orchards	4.0
Vacant Land	10.0
Wooded Land	10.0
Street/Roads, Asphalt	6.0
Streets/Roads, Concrete	6.0
Exterior Walls, Wood	1.5
Exterior Walls, Brick	1.5
Floors, Linoleum	0.5
Floors, Wood	0.5
Floors, Carpeted	0.5
Floors, Concrete	1.5
Interior Walls, Painted	0.5
Interior Walls, Concrete	1.5
Roofs	1.0
Lawns	1.3
Other Asphalt Surfaces	1.0
Other Concrete Surfaces	1.0

DECON makes several adjustments to property values as a result of the accident. One adjustment is to reduce the value of property because of residual contamination levels after site restoration has been completed. The size of the discount should depend on perceived health risks. The user can specify a different discount factor for each land use. A second adjustment is for deterioration of property that may occur between the time of contamination and site restoration, a period during which the property is likely to remain idle. A different deterioration factor can be specified for each land use. Finally, the costs associated with loss of use of the property are evaluated. In computing the social costs of the accident, all costs are discounted to the end of the year in which they are assumed to occur.

III. COLLECTION OF DATA FOR THE SAGEBRUSH EXERCISE

An aerial survey conducted on the first day of SAGEBRUSH IV by EG&G revealed the 37 kBq/m^2 (1.0 μCi/m^2) isopleth over the accident site; this isopleth gave a footprint of the contaminated area and facilitated the planning of a detailed aerial survey for the second day. Additional isopleths were then developed from the second day's aerial survey. Because of the distorting effect of scattered shine from an aerial survey, field monitoring teams were sent out to conduct a ground survey. A few ground measurements were taken on the first day, several on the second day, and the filling in was completed on the third day. These ground readings were entered on the site map, enabling the isopleths to be corrected for the scattered shine. The isopleths used in the site restoration analysis are shown in Figure 1. Also shown in this figure are the grid elements, which are bounded by the isopleths. The number of grid elements turned out to be 15, varying in size from 22,300 m^2 to 13,000,000 m^2.

Radiological products in the environment were assumed to consist of 90 percent ^{239}Pu and 10 percent ^{241}Am by activity. Measured activity levels ranged from less than 7.4 kBq/m^2 to over 29.6 MBq/m^2. The 7.4 kBq/m^2 figure is of special interest: it is the EPA screening limit. An individual exposed to this limit will not receive a dose higher than the individual dose limit.

A variety of land uses was represented within the contaminated area, as shown in Table 3. This information was obtained from USGS maps and from sources in the field.

TABLE 3. Land Uses in the Contaminated Area

• Residential	• Wooded Areas
• Commercial	• Agricultural
• Industrial	• Orchards
• Streets/Roads	• Vacant Land

In addition to ground concentration levels and land use information, DECON also uses information on property values. Typical values for property represented by the different land uses were obtained from the Ferry County Assessor during the exercise.

IV. SITE RESTORATION ANALYSIS OF THE SAGEBRUSH IV ACCIDENT SITE

Results relating to restoration of the SAGEBRUSH IV site using DECON are described in this section. First, DECON was run for the entire contaminated area within the 7.4 kBq/m^2 isopleth. For this "base case" we had to make several assumptions. We assumed that a Class A disposal site would be created within the contaminated area for radiological waste generated during the cleanup process. An average hauling distance of 15 miles to this site was assumed.

The decontamination costs contained in the reference data base seriously underestimate the actual costs of labor and equipment that would be used in the cleanup. First of all, the data base costs are in 1982 dollars. Second, labor and equipment costs are based on operations in a nonhostile environment. Third, an allowance of one hour per eight-hour shift is allowed for radiation control measures, and this may not be adequate. And, finally, protective clothing and other radiation control measures may reduce worker and equipment productivity. For all of these reasons, labor costs were arbitrarily increased by 70 percent and equipment costs by 40 percent.

A cleanup criterion of 7.4 kBq/m^2 was applied to all contaminated surfaces. As noted earlier, this is EPA's screening limit. The results from this base case are reported in Section A. Other cases were then analyzed and the results compared with those from the base case. First, we wanted to determine how decontamination costs varied with the cleanup criterion. This analysis is conducted in Section B. In Section C, we drop the assumption that the radwaste will be disposed of on sight. Instead, we assume disposal at the Hanford Reservation, about 250 miles away. A subarea analysis is conducted in Section D for grid elements one through nine. These grid elements are within the Colville National Forest, a wooded area which is especially costly to decontaminate. First, we consider decontaminating this area to a level of 37 kBq/m^2. Another option considered is simply to apply a fixative and restrict public access to the entire area. In Section E we compare base case results with a scenario in which operations that use water on exterior surfaces are banned. This analysis is conducted on grid element 15, a lightly contaminated area with about four percent of the land area devoted to residential use. Section F suggests an alternative restoration strategy. A decontamination analysis is conducted in which different cleanup criteria are applied to different surfaces; namely, those surfaces from which individuals typically receive the greatest exposure are decontaminated to lower residual levels of activity than surfaces providing relatively low levels of exposure. Conclusions from this paper are presented in Section 5.0.

A. SAGEBRUSH IV: Base Case

Major results for the base case are summarized in Table 4. The total cost of decontaminating the 18 million m^2 of surface area within the 7.4 kBq/m^2 isopleth is $65.6 million, for an

TABLE 4. SAGEBRUSH IV Decontamination Results: Base Case (dollars, areas and volumes are in thousands)

MINIMUM CLEAN UP LEVEL IS	7.4	kBq/m^2.
TOTAL SURVEYING AND MONITORING COSTS ARE..... $	3519.	
VOLUME OF RADIOLOGICAL WASTE IS...............	2483.	CUBIC METERS.
TOTAL DECONTAMINATION COSTS ARE.............. $	65633.	
TOTAL SURFACE AREA DECONTAMINATED IS..........	18161.	SQUARE METERS.
AVERAGE DECONTAMINATION COSTS/M**2 ARE....... $	3.61	
SURFACE AREA REQUIRING NO DECONTAMINATION IS...	70.	SQUARE METERS.
PRE-ACCIDENT PROPERTY VALUE IS .... $	8361.	
POST-DECONTAMINATION PROPERTY VALUE IS ...... $	7739.	
NET PRESENT VALUE OF PROPERTY IS .... $	41.	
TOTAL REDUCTION IN PROPERTY VALUE IS .... $	8319.	
TOTAL POTENTIAL SAVINGS FROM PROPERTY BUY-OUT		
1) AT PRE-ACCIDENT PROPERTY VALUES .... $	57325.	
2) AT NET PRESENT VALUE OF PROPERTY ...... $	65543.	
SIZE OF RESIDENT POPULATION IS	520.	PERSONS.
SIZE OF THIS AREA IS	18137.	SQUARE METERS.

average cost of \$3.61 per m^2. About 70,000 m^2 of surface--primarily, the interiors of buildings in lightly contaminated areas--required no decontamination, as they were already within the 7.4 kBq/m^2 limit. Nearly 2.5 million m^3 of radiological wastes were generated in the cleanup operations and disposed on-site. Surveying and monitoring costs were estimated at \$3.5 million.

The total market value of all real property within the contaminated area is estimated to have a pre-accident value of only \$8.4 million. This is an economically troubled area based on timber, a depressed industry. In addition, much of the land is rocky and has little economic value--the county assessor valued much of it at as little as a dollar per acre--and even agricultural lands are marginal. It is worth noting that the cleanup costs far exceed the economic value of the property. If the federal government were to provide compensation to all property owners (including itself, in the case of federal lands), social costs of up to \$65 million could be avoided by not restoring the property. (The compensation cost of \$8.4 million is a transfer, not a social cost.)

Total manpower and equipment requirements for the site restoration, including surveying and monitoring operations, are shown in Table 5.

Residual contamination levels in the base case are estimated to result in property value losses of \$622,000. This is about 7.5 percent of the total pre-accident value of the property.

TABLE 5. Manpower and Equipment Requirements, Base Case (1000s of hours)

DRIVER, HEAVY TRUCK	315.98
OPERATOR, MED. EQUIPMENT	124.41
OPERATOR, FARM EQUIPMENT	5.38
BUILDING LABORER	131.61
COMMON LABORER	1.60
CLEANING WORKER	.33
FARM LABORER	7.97
PAINTER	.12
FOREMAN	35.24
PILOT	1.43
FLIGHT CREWMAN	1.43
AIR GROUND CREWMAN	2.85
SPRAY OPERATOR	.06
FRONT END LOADER	60.72
5000 GAL. SPRAY TRUCK W/PUMP&BOOM	.10
BULLDOZER	2.31
AIRPLANE	1.43
WEED SPRAYER	1.94
ORCHARD BLAST SPRAYER	.97
TRACTOR W/PLOW	5.38
PICKUP TRUCK	1.59
CHIPPING MACHINE	30.04
HYDRAULIC EXCAVATOR	5.95
VACUUM, HAND	.33
PAINT SPRAY EQUIPMENT	.18
MOBILE STREET FLUSHER	.01
10-TON ROLLER	25.11
GRADER	30.31
DUMP TRUCK	315.88

B. Costs vs. Cleanup Levels: A Trade-Off Analysis

As expected, we found decontamination costs to increase as stricter cleanup levels were imposed. For this analysis, we re-ran the base case, but with cleanup levels of 18.5, 27.75, 37.0, 74.0 and 185.0 kBq/m^2. The resulting relationship between cleanup level and decontamination costs is shown in Figure 2. Decontamination costs are seen to drop sharply for cleanup criteria of less than 37 kBq/m^2, while above this level the decline in costs is considerably more modest. However, in order to determine the optimal cleanup level on economic grounds, one must know the dollar value of avoiding the additional adverse health effects that result from moving to a stricter cleanup level. The dollar value of avoiding health effects, including long-term fatalities, is a highly controversial issue and lies beyond the scope of this paper.

C. Hauling Radwaste Off-Site

The base case assumes that a Class A disposal site for radwaste could be made available within the accident site. In the event that the radwaste is to be disposed off-site, hauling

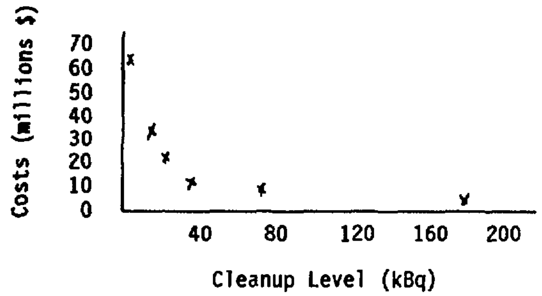

Fig. 2. Cleanup Costs vs. Cleanup Level

distances and costs could increase substantially. For example, if the radwaste is transported to the Hanford Reservation, the one-way hauling distance will increase from 15 miles to about 250 miles, depending upon the specific route taken. This longer hauling distance will increase cleanup costs to about \$250 million, as compared with \$66 million in the base case. In addition, disposal costs at Hanford would be greater than the estimated cost of \$4.86/m^3 (1982 \$) for an on-site Class A disposal pit (Tawil, 1985).

D. Alternatives to Decontaminating Forest Lands: A Subarea Analysis

Analysis of decontamination costs in the base case shows that a major share of these costs are attributable to decontaminating wooded areas. The density of mature trees in a forest makes it difficult to utilize large but efficient equipment in the decontamination process. The reliance on less capital intensive methods causes costs to increase sharply. While wooded areas comprise about 42 percent of the contaminated area, they account for over 96 percent of the decontamination costs in the base case. Grid elements one through nine consist primarily of wooded areas inside the Colville National Forest. If, instead of restoring this subarea to a cleanup level of 7.4 kBq/m^2, we apply a fixative and restrict public access, the restoration costs can be slashed from \$26.7 million to \$1.8 million. If we combine this strategy with cleaning up grid elements 10 through 15 to a level of 37 kBq/m^2, we can reduce decontamination costs for the entire site to just \$2.4 million. While this strategy obviously reduces costs sharply, its desirability depends on the willingness of those who must make this decision to accept restricted access to the subarea containing grid elements one through nine, and accepting a cleanup level--and the associated health effects--of 37 kBq/m^2 in the remaining areas. It should be mentioned that while a stricter cleanup level will reduce the expected health effects to occupants of the restored area, expected health effects to radiation workers will increase.

E. Restricting the Use of Water in a Populated Area

In this section we consider the consequences of prohibiting decontamination methods that use water on exterior surfaces. Contaminated water has the potential of creating major problems. It can penetrate the root systems of plants, crops and trees and can contaminate water treatment facilities. The benefits from using water--a cheap and effective way to reduce dosage through the external and inhalation pathways--must therefore be carefully weighed against the costs. The results of running DECON on grid element 15 with a ban on operations using water (i.e., operations W, H, Q, U, L, and E--see Table 1) are presented in Table 6. This table also shows the level of detail that DECON reports with regard to decontamination operations at the grid element level. The columns in Table 6 show the type of surface, its area in 1000 m^2, the contamination level prior to decontamination in kBq/m^2, the method used to decontaminate the surface (see Table 1), a symbol indicating whether a restriction or requirement is in effect for that surface, the residual contamination level in kBq/m^2, the unit cost of applying the indicated decontamination method in \$/m^2, the total cost (1000 \$) of applying the method over the entire surface area, and the rate (in m^2/h) at which the method can be applied. The results show that in the absence of the restrictions, water would have been applied to asphalt and concrete streets, parking and other paved areas, and to roofs and lawns. While (double) vacuuming (v) is a relatively inexpensive substitute for water, the foam (F) used on roofs at \$2.65 per m^2 and the resodding (R) of lawns at \$9.51 per m^2 are relatively costly alternatives.

F. Cleanup Standards Based on Expected Exposure to Activity

DECON allows the analyst to impose different cleanup criteria to different surfaces by selecting exposure factors. The potential usefulness of this feature lies in the fact that human exposures to different surfaces vary considerably. Housing interiors, for example, would usually offer high exposures while highways and wooded areas would tend to offer low exposures. The exposure factors are interpreted as follows. An exposure factor of 1.0 causes the cleanup standard to be identical to the user-specified value, 7.4 kBq/m^2 in the base case. An exposure factor of 2.0 causes the user-specified value to be doubled, while an exposure factor of 0.5 causes the user-specified value to be halved. To illustrate this feature, DECON was run with the exposure factors shown in Table 2.

Decontamination costs fall to just \$9.4 million, and only 8.2 million m^2 of surface area need to be decontaminated (at an average cost of \$1.15/m^2.) Arguably, this decontamination strategy could result in even less overall population dose than produced in the base case. Most interior surfaces are decontaminated to levels of just 3.7 kBq/m^2, and only vacant land and wooded areas are decontaminated to levels above 37 kBq/m^2 (both are decontaminated to 74.0 kBq/m^2). Yet decontamination costs are less than 15 percent of the base case costs.

Table 6. Detailed Surface Results with Restrictions, Grid Element 15 (dollars, areas and volumes in thousands)

SURFACE	AREA	GRND CONC	METH	RES CONC	COST/M**2	TOT COST	RATE
AGRICULTURAL FIELDS	5334	22.20	A	2.22	.0055	29.34	8500.
ORCHARDS	390	22.20	TO	6.29	1.1402	445.03	280.
VACANT LAND	1431	22.20	A	.37	.0389	55.67	1770.
WOODED LAND	5464	22.20	TN	6.29	6.3501	34698.05	266.
EXTERIOR WOOD WALLS	13	2.22	----	....	....	....	......
EXTER'R BRICK WALLS	2	2.22	----	....	....	....	......
LINOLEUM FLOORS	4	11.10	T	2.22	.4070	1.45	40.
WOOD FLOORS	4	11.10	V	1.11	.4260	1.58	69.
CARPETED FLOORS	8	11.10	V	4.44	.4260	3.46	69.
CONCRETE FLOORS	3	11.10	V	2.96	.4260	1.26	69.
INT'R WOOD/PL WALLS	25	1.11	----	....	....	....	......
INT'R CNCRETE WALLS	5	1.11	----	....	....	....	......
ASPHALT STRTS/PRKNG	130	22.20	v /	7.40	.0133	1.73	8632.
CNCRETE STRTS/PRKNG	130	22.20	v /	7.40	.0133	1.73	8632.
ROOFS	11	22.20	F /	1.48	2.6480	29.16	81.
LAWNS	113	22.20	R /	.37	9.5121	1070.89	40.
OTHR PAVED ASPHALT	1	22.20	v /	7.40	.0266	.03	4316.
OTHR PAVED CNCRETE	5	22.20	v /	7.40	.0266	.14	4316.

NOTES:
/ = RESTRICTED OPERATION(S) IS IN EFFECT + = METHOD IS REQUIRED
//// = UNABLE TO DECONTAMINATE SURFACE ---- = DECONTAMINATION NOT REQUIRED

V. CONCLUSIONS

A site restoration analysis conducted on the SAGEBRUSH IV exercise site suggests a variety of alternative strategies. Decontamination of the entire area to the EPA screening limit of 7.4 kBq/m^2 is a relatively expensive option. A trade-off analysis shows that decontamination costs fall off rather sharply with less restrictive cleanup criteria until a cleanup level of 37 kBq/m^2 is reached, after which the decline is relatively modest. Because wooded areas are difficult to decontaminate, restricting access to the contaminated areas of the Colville National Forest rather than decontaminating them can substantially cut the cost of the cleanup. A requirement that significantly increases the restoration costs is to haul the radwaste generated in the cleanup to a remote site, rather than creating an on-site disposal area. Restrictions against using water on exterior surfaces also increase decontamination costs significantly, especially in lightly contaminated areas. Finally, imposing different cleanup criteria on different surfaces, depending upon their potential for exposure, appears to be a promising way to reduce decontamination costs without increasing total population dose.

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**ANS Topical Meeting on Radiological Accidents—
Perspectives and Emergency Planning**

The NRC's Rulemaking to Require Materials Licensees To Be Financially Responsible for Cleanup of Accidental Releases

Mary Jo Seeman

ABSTRACT On June 7, 1985, the U.S. Nuclear Regulatory Commission (NRC) published an advance notice of proposed rulemaking (ANPRM) in the Federal Register to address funding for cleanup of accidents and unexpected decontamination by certain materials licensees.

The NRC asked for public comment to help them determine whether to amend its regulations to require certain materials and fuel cycle licensees to demonstrate that they possess adequate financial means to pay for cleanup of accidental releases of radioactive materials. If licensees lack adequate financial resources and funds are not available for prompt cleanup, the consequences could be potentially significant for the public, the licencee and the federal government.

The purpose of this paper is to explain the purpose and scope of the Commission's proposed regulatory action, as well as describing several accidents that made the Commission consider this action. Additionally, the paper will address other regulatory precedents. Finally, the paper will conclude by generally characterizing the public comments and items of concern raised by commenters.

Good afternoon, ladies and gentlemen. Today, I would like to briefly outline the scope and purpose of a proposed regulatory program under review by the Commission. If implemented, such a program would establish financial assurance requirements for cleanup of accidental releases by certain categories of non-reactor licensees. After describing the scope of this effort, I plan to briefly identify issues raised by the public in this area, and present the staff's schedule for addressing these items of concern.

On June 7, 1985, the U.S. Nuclear Regulatory Commission published an advance notice of proposed rulemaking (ANPRM) in the Federal Register to address funding for cleanup of accidents and unexpected contamination by certain materials licensees.

In the ANPRM, NRC asked for public comment to help them determine whether to amend its regulations to require certain materials and fuel cycle licensees to demonstrate that they possess adequate financial means to pay for cleanup of accidental releases of radioactive materials, i.e., an authorized release of radioactive materials due to human error, employee sabotage, system failure, an act of God, or defective components.

The proposed regulatory program would amend current NRC regulations (10 CFR 30, 40, 61, 70, and 72) requiring certain fuel cycle and materials licensees to demonstrate that they possess adequate financial means to pay for cleanup of accidental releases of radioactive materials onsite and offsite. Licensees under consideration for this ANPRM include radiopharmaceutical manufacturers, pool irradiators, industrial radiographers, users of gauging devices, gas chromatography, well-logging, nuclear medicine diagnosis and radiation therapy. Other NRC licensed operations under consideration include fuel cycle activities such as uranium milling, UF-6 production, and fuel processing and fabrication.

Regulated waste management activities include commercial low-level waste disposal, independent spent fuel storage facilities, and persons disposing or storing their own waste under special license conditions. If implemented, it is anticipated that the financial responsibility program would exempt the following U.S. Department of Energy facilities: High-level waste repositories (licensed under 10 CFR Part 60), independent spent fuel storage facilities, and monitored retrievable storage facilities (both licensed under 10 CFR Part 72). The staff took this approach in the ANPRM because in the event of an accident, DOE has access to public funds to pay for cleanup. In the ANPRM, the Commission solicited public comments on the advisability of having NRC require financial responsibility for prompt cleanup of radioactive materials both on-site and off-site after accidental or unexpected contamination by fuel cycle and other materials licensees. Any environmental restoration required following an accidental release into or upon the land or water would be covered.

The financial assurance program for non-reactor licensees under consideration by the Commission is intended to address cleanup for accidental releases of radioactive materials. The proposed program would not address authorized and predictable activities normally associated with decommissioning or closure of a non-reactor licensee's operations. The latter is being addressed in a separate, ongoing Commission rulemaking on decommissioning.

The financial assurance program being considered by the Commission in this ANPRM is also separate and distinct from the compensation program mandated by the Commission regulations issued pursuant to the Price-Anderson Act, which does not provide funds for cleanup per se. At the time the ANPRM was issued in 1985, that program applied only to nuclear reactors on a mandatory basis and to plutonium processors and fuel fabricators on a discretionary basis.

As many of you are probably aware, the Commission has authority under section 170 of the Atomic Energy Act to apply Price-Anderson indemnification to any category of licensee, whenever the Commission deems it advisable in the exercise of its licensing authority. However, the Commission has not chosen to extend Price-Anderson indemnification to materials licensees with the single exception of those engaged in the use of plutonium in a plutonium processing and fuel fabrication plant. These activities and uses of materials would not be included in the proposed program. Rather, the program proposed in this ANPRM is not intended to provide compensation to persons for personal injury or property damage and is, therefore, not a public liability program.

The Commission is considering such a proposed regulatory program in part, because it appears that CERCLA, or the Comprehensive Environmental Compensation and Liability Act of 1980 (P.L. 96-510) would not necessarily provide funds for releases involving NRC licensees. In a Federal Register Notice issued on September 8, 1983, the U.S. Environmental Protection Agency (EPA) made the following policy statement:

EPA has also chosen not to list releases of source, byproduct, and special nuclear material from any facility with a current license issued by the Nuclear Regulatory Commission (NRC), on the grounds that the NRC has full authority to require cleanup of releases from such facilities. The EPA provided written comments to the Commission regarding the ANPRM, and in their written comments stated, "EPA strongly supports the concept proposed in this rulemaking."

I now want to briefly provide some background information regarding the scope of the proposed effort. In the ANRPM, the Commission staff noted that although there was little information available on the financial condition of NRC fuel cycle and materials licensees, they believed that most of these licensees already had some financial resources or insurance coverage for on-site and off-site cleanup as a prudent business practice. However, if a licensee did not have adequate financial resources and an accidental or unexpected contamination did occur, the staff felt that there could be both short and long term adverse public health and safety consequences from the radioactive contamination, as well as loss of use of the contaminated property.

For the purposes of initial discussion in the ANPRM, the Commission noted that it was considering a $2,000,000 baseline as the required maximum amount of financial responsibility for materials and fuel cycle facilities. This figure was chosen because it is in the range of known cleanup costs for NRC licensees and of other dollar amounts of State and Federal financial responsibility requirements for cleanup of accidental releases. The amount would be changed to reflect changes in inflation and technology. Types of possible acceptable financial assurances that the Commission staff is considering include liability and property insurance from the conventional and nuclear insurance pools, cash or negotiable securities held by a third party, a financial test, a surety or performance bond, and a corporate guarantee from a parent company.

The issue of financial responsibility for cleanup of accidental releases of radioactive materials caused by non-reactor licensees has also been an issue with the States. Some States have developed financial assurance programs covering their non-reactor licensees to address this issue. Additionally, the Conference of Radiation Control Program Directors recognized the need for Federal standards in this area. The Commission sent a draft of the ANPRM to officials in all fifty States asking for their comments. Staff also discussed the ANPRM with State radiation control program officials earlier. State comments generally supported the development of the rulemaking, and specific comments were incorporated into the ANPRM.

The need to consider such a proposed regulatory program was based on concerns regarding accidental releases of radioactive materials. Both NRC and Agreement State licensees have had accidental or unexpected releases of radioactive materials that have been costly to cleanup. State and Federal estimates for cleanup costs have been estimated at up to $2,000,000 for a single event. Some recent examples that the NRC staff is aware of include the following:

In 1983, a cesium-137 sealed source was accidentally ruptured. Workers inadvertently spread the contamination into residences and public buildings. The cost for cleanup of of this contamination was estimated to be at least $500,000.

In 1982, an americium-241 sealed source in use in a well-logging operation was inadvertently ruptured, resulting in contamination of both on-site drilling equipment and off-site homes and commercial residences. Cleanup costs were estimated to be up to at least $1,000,000.

During 1979 and 1980, a tritium manufacturer's operations in Tucson, Arizona, resulted in releases both on and off-site. State officials estimated that the State spent approximately $2,000,000 in labor and capital costs for removal and cleanup of the tritium.

The NRC staff have had a difficult time establishing the scope of the problem of costly cleanups required as a result of accidental releases of radioactive materials. Relatively little data on cleanup costs for accidental releases of licensees is available in NRC records, mainly because this type of information is not required to be submitted to the agency.

However, a recent review by the staff of NRC unusual events reports for radioactive releases by materials and and fuel cycle licensees indicates that from 1980 to 1983, accidental or unexpected releases from licensees' operations believed to involve significant cleanup costs (more than a few thousand dollars) involved fewer than one percent annually of the total fuel cycle and materials licensees authorized to possess and use byproduct, source and special materials.

Besides lacking a comprehensive data base for licensee cleanup costs for accidental releases, the Commission also has no available records to determine if licensees lack (or previously lacked) adequate funds to provide for prompt cleanup of accidental releases. Accordingly, in the ANPRM, the Commission solicited input from the public and the industry on the scope and magnitude of the problem.

However, even given this lack of a strong data base, the NRC staff does not believe it is prudent for the Commission to wait until an event occurs which requires expensive cleanup to consider the development of such regulations. Other agencies such as the U.S. Environmental Protection Agency, have gone forward with financial responsibility requirements in the absence of a large documented accident data base.

Additionally, several NRC-funded studies have presented cost estimates for cleanup of NRC licensed fuel cycle facilities related to emergency planning issues and the Price-Anderson Act. See a 1979 paper authored by J.P. McBride, and entitled "Economic Consequences of Accidental Releases from Fuel Fabrication and Radioisotope Processing Plants", prepared by Oak Ridge National Laboratories for the U.S. NRC (NUREG/CR-0222). Another example can be found in a 1983 working paper by H.K. Elder entitled "Technology, Safety and Costs of Decommissioning Reference Nuclear Fuel Cycle and Non-Fuel Cycle Facilities Following Postulated Accidents", Pacific Northwest Laboratories, (NUREG/CR-3293).

However, since these studies were prepared for different purposes and assumptions, it is difficult to compare the results or to use their conclusions as the basis for estimating cleanup costs following an accidental release caused by a non-reactor licensee's operations. Accordingly, in the ANPRM, the staff noted that they proposed to use the limited but actual, past cleanup cost experience (discussed previously) as the basis for setting the amount of financial responsibility coverage for non-reactor licensees.

In the ANPRM, the NRC staff stated that they would consider at a later date, the issue of financial responsibility for the small number of licensees who have the potential to be involved in the significantly more costly cleanups postulated in the NRC-funded studies.

The Commission's consideration of such a financial assurance program is in line with the regulatory precedents enacted by a variety of State and Federal agencies charged with protecting the public health, safety, and the environment. For example, federal agencies have enacted requirements pursuant to the Motor Carrier Act, the Resource Conservation and Recovery Act, the Federal Water Pollution Control Act, the Surface Mining Control and Reclamation Act, the Outer Continental Shelf Lands Act, and the Deep Water Port Act.

The stipulated dollar requirements vary from $10,000 to over $5,000,000 for these different programs. As an example, the 1984 minimum levels of financial responsibility for motor carrriers transporting hazardous substances ranges from between $1,000,000 to $5,000,000.

A number of commenters responding to the ANPRM suggested that the NRC staff should take a closer look or even follow the U.S. Environmental Protection Agency's (EPA) regulatory approach with regards to requiring financial responsibility. To briefly bring you up to speed, the U.S. EPA regulations for liability coverage required owners and operators of treatment, storage, and disposal facilties to demonstrate liability coverage for sudden and accidental occurrences in the amount of $1 million per occurrence and $2 million annual aggregate, all exclusive all legal defense costs.

Owners and operators of surface impoundments, landfills, and land treatment facilities were required to demonstrate liability coverage for nonsudden accidental occurrences in the amount of $3 million per occurrence and $6 million annual aggregate, exclusive of legal defense costs.

These commenters pointed out that after EPA had promulgated these requirements for liability coverage, a variety of parties informed EPA that the current state of the insurance market prevented them from obtainging insurance. Accordingly, EPA was faced with considering whether to revise or even eliminate their financial responsibility requirements.

Accordingly, NRC staff looked at how EPA addressed the problem of a contracting insurance market after they had promulgated insurance requirements. NRC staff examined the July 1985 testimony of Gene Lucero, who at the time had the title of Director of EPA's Office of Waste Programs Enforcement. In his testimony, Lucero noted that many approaches had been proposed to address the pollution liability insurance shortage. He noted that EPA believed that no one approach, but rather a combination of several approaches was needed to solve the insurance shortage. These included a return to careful underwriting practices through use of environmental audits, premiums that reflect risks, and contracts that address policy limits in unambiguous approaches.

Additionally, NRC staff also examined the July 21, 1985 EPA Notice of a Proposed Rulemaking, which asked for comments to enable EPA to determine if revisions to their regulations were necessary in light of the current state of the insurance market. EPA noted that they were considering taking one or a combination of the following five regulatory actions in response to the problem:

0 Maintain existing liability requirements;

0 Clarify the required scope of coverage and/or lower the required levels of coverage;

0 Authorize other financial responsibility mechanisms;

0 Authorize waivers; and

0 Suspend or withdraw the liability coverage requirements

At this time, EPA has apparently chosen the third alternative. In July of 1986, they issued an interim final rule stating that they had decided to authorize owners and operators to use a corporate guarantee as another mechanism to comply with the liablity coverage requirements.

As I noted earlier, the NRC staff has, and will continue to monitor and assess the status of the EPA liability program in their consideration of a NRC financial responsibility program for cleanup of accidental releases of nonreactor licensees.

Finally, I would like to briefly summarize the statistics regarding who commented and what their concerns were. The Commission used the ANPRM to solicit input from all interested parties regarding the scope and nature of such a proposed regulatory program. Besides mailing the ANPRM to all non-reactor parties holding a specific license, the NRC also targeted a direct mail to NMSS licensees, public and special interest groups, as well as trade associations representing the business, finance, and insurance communities. The ANPRM generated a great deal of interest, particularly among NRC materials licensees, and State and Federal regulatory agencies. NRC received 159 written responses from licensees, insurance and surety trade organizations, federal state and local governments, and members of special interst groups and the public.

The ANPRM generated a great deal of interest, particularly among NRC materials licensees, and State and Federal regulatory agencies. NRC received 159 written state and local governments, and members of special interest groups and the public.

Approximately 80% of the commenters were either NRC or Agreement State licensees. The majority of these commenters identifying themselves as licensees expressed concern regarding the cost impacts of such a program. Others felt that such requirements were unnecessary because they were not aware of any real problem regarding the inability of licensees to pay for accidental releases or abandonments. Others suggested that the NRC should consider exempting operations similar to theirs from the coverage of this rulemaking.

Around sixteen percent of commenters were identified as governmental organizations, such as municipal, state, or federal agencies. Many supported the need for this rulemaking. Some felt that federal financial responsibility standards were necessary to protect the public health and safety and the environment. Others indicated that such a program was appropriate because responsibility for paying for cleanup of accidents and abandonments should rest with the licensees, and not with the taxpayers.

The remaining four percent of commenters were identified as members of the public, environmental groups, and trade associations representing insurance and surety companies. The trade associations pointed out that surety or insurance coverage for the program intended by NRC was either not available or too costly. One surety trade association suggested that the NRC staff allow flexibility in developing standards so that affected licensees would have the ability to demonstrate financial assurance through a number of different instruments. In general, environmental groups and the public supported the rulemaking, although one felt that NRC would not go far enough in setting up such a program.

In their review and analysis of the public written comments, the staff identified a number of major items that were raised by

0 What type of history has there been for accidental releases?
What history has there been of accidents involving nuclear materials where the licensee has been unable for cleanup?

0 How does the Commission intend to determine coverage for such a program?

0 What are the cost impacts to licensees and the public for such a proposed program? What are the cost impacts of not implementing such a proposed program?

0 What amount of coverage is appropriate? Which licensees should be covered under such a program? Who should be exempted?

0 Is liability insurance available for this type of risk? What will premiums cost?

0 What types of financial assurance should be allowed?

The staff has reviewed and analyzed these questions and is continuing to assess how these issues will be resolved. It would be premature at this time to provide conclusions with regard as to how the staff and the Commission intends to address these issues. However, I can say that the staff recognizes that the issues regarding who should be covered and cost impacts of such a program were complex and required a more in-depth analysis.

Accordingly, the staff has obtained technical assistance in these areas. The first project is intended to help staff better assess the extent to which accident frequencies can be used to determine amounts of coverage for different licensee categories. Appropriate exemptions for different licensee categories will also be examined. Secondly, the staff also has obtained technical assistance to help them arrive at an evaluation of the benefits and costs of such a proposed financial assurance program to all parties. The staff hopes to use the results from both studies to help them determine how to address these two concerns.

Where do we go from here? The approved schedule for this rulemaking is found in the Agency's Regulatory Agenda, which is published periodically in the Federal Register. That schedule calls for the staff to submit a proposed rulemaking package for consideration to the Commission in early 1987. If the Commission decides to proceed, then the proposed regulations will be published in the Federal Register for review and comment later that spring.

In closing, I think that the issue of financial assurance for cleanup of accidental release caused by non-reactor licensee is an important priority. I appreciate the opportunity today to bring this work to your attention, and I will try to answer any questions you may have.

Applicability of Comprehensive Environmental Response, Compensation, and Liability Act of 1980 (CERCLA) to Releases of Radioactive Substances

Steven R. Miller[a]

ABSTRACT The Comprehensive Environmental Response, Compensation, and Liability Act of 1980 (CERCLA), commonly called "Superfund," provided a $1.6 billion fund (financed by a tax on petrochemical feedstocks and crude oil and by general revenues) for the cleanup of "releases" of "hazardous substances," including "source," "special nuclear" or "byproduct material," and other radioactive substances, from mostly inactive facilities. The U.S. Environmental Protection Agency (EPA) is authorized to require private "responsible parties" to clean up releases of hazardous substances, or EPA, at its option, may undertake the cleanup with monies from the Fund and recover the monies through civil actions brought against responsible parties. CERCLA imposes criminal penalties for noncompliance with its reporting requirements. This paper will overview the key provisions of CERCLA which apply to the cleanup of radioactive materials.

The Comprehensive Environmental Response, Compensation, and Liability Act of 1980 (CERCLA), Pub. L. No. 96-510, 42 U.S.C. §9601 et seq., commonly called "Superfund," provided a $1.6 billion fund (financed by a tax on petrochemical feedstocks and crude oil and by general revenues) for the cleanup of "releases" of "hazardous substances," including "source," "special nuclear," or "byproduct material," and other radioactive substances from mostly inactive facilities. "Federally permitted releases" from active facilities are generally regulated under the Resource Conservation and Recovery Act, (RCRA), 42 U.S.C. §6901 et seq. (1976) and other environmental laws.

The term "release" is extremely broad (i.e. any leaking, spilling, emitting, etc.) and covers the U.S. Department of Energy (DOE) sites where hazardous substances are in contact with the environment (i.e. air, water, soil) even within federal or private industry property boundaries.

The term "release" excludes releases of "source," "special nuclear," and "byproduct material" resulting from a "nuclear incident," subject to the financial protection requirements established by the Nuclear Regulatory Commission under section 170 of the Atomic Energy Act of 1954, as amended, 42 U.S.C. §2011, §2210. Also excluded, are releases from uranium mill tailings sites being cleaned up by DOE under Title I of the Uranium Mill Tailings Radiation Control Act of 1978, Pub. L. No. 95-604, 42 U.S.C. §7901 et seq.

CERCLA imposes criminal penalties for noncompliance with its reporting requirements. Under section 103a of CERCLA, 42 U.S.C. §9603a, any "person" (including federal agencies and private companies) "in charge of" an onshore or offshore facility must report releases of hazardous substances in excess of "reportable quantities" to the National Response Center as soon as the "person" has knowledge of the release, and to publish information concerning such releases in local newspapers. The failure to promptly report such releases can subject responsible parties, including employees, to criminal penalties. Exempted from the reporting requirements are "federally permitted releases" (i.e. releases permitted under other federal environmental laws), which include releases of "source," "special nuclear," or "byproduct material" in compliance with legally enforceable licenses, permits, and orders, issued pursuant to the Atomic Energy Act of 1954, as amended, 42 U.S.C. §2011 et seq.

Until the U.S. Environmental Protection Agency (EPA) promulgates reportable quantities for radioactive materials, the minimum reportable quantity is one pound.

Continuous releases must also be reported at least annually.

[a]The views expressed in this paper are those of the author and do not necessarily represent those of the U.S. Department of Energy.

There are no other affirmative legal requirements, for federal agencies or private parties, except as specified by EPA.

Owners, operators, generators and transporters of hazardous substances are jointly, severally, and strictly liable for the cleanup of all releases.

EPA can either require private parties to clean up releases of hazardous substances or can proceed to undertake the cleanup with monies from the Fund. Any Fund monies so expended by EPA can be recovered through civil actions brought against the "responsible parties." Monies from the Fund are available for both emergency "removal" actions and long-term "remedial" actions on private sites listed on CERCLA's "National Priorities List."

CERCLA applies to federal agencies procedurally and substantively, as it does to private parties EXCEPT: (1) Monies from the Fund are generally not available for the cleanup of federal facilities; and (2) EPA cannot sue other federal agencies.

CERCLA assigned all response authorities to the President. By the terms of Exec. Order No. 12,316, 46 Fed. Reg. 42,237 (1981), reprinted in 42 U.S.C. §9615NT.,EPA was generally delegated all CERCLA response authorities. The U.S. Department of Defense was, however, specifically delegated all CERCLA response authorities at its own sites. Exec. Order No. 12,316 permits EPA to redelegate its authorities to other federal agencies.

EPA's CERCLA authorities with respect to federal facilities include: the ability to undertake assessments and feasibility studies and conduct emergency response actions, where an imminent and substantial endangerment to public health and safety exists. EPA can issue administrative orders against federal agencies for noncompliance. Such orders cannot, however, be enforced in court. With respect to nuclear releases, however, CERCLA's implementing regulation, "The National Contingency Plan," 40 C.F.R. 300 et seq. provides that DOE give assistance in responding to these releases, regardless of whether they be on federal or private lands.

Cleanup standards are established by EPA for federal and private facilities on a case by case basis. The National Contingency Plan requires the application of "applicable" or "relevant and appropriate" federal standards and for consideration to be given to the use of state and other standards.

There is no state enforcement role under CERCLA and no waiver of sovereign immunity that would make federal agencies responsible for complying with state laws and regulations. However, EPA can enter into cooperative agreements with affected states for the undertaking of cleanup actions on private sites. States can sue both federal agencies and private parties for damages to natural resources and for recovery of costs incurred in any cleanup actions the states have undertaken for which the federal agencies and private parties are responsible.

CERCLA is currently up for a five-year Congressional reauthorization because the authority of the IRS to collect the tax which finances the Fund expired on September 30, 1985. A \$7.5 billion reauthorization bill has passed the Senate (H.R. 2005, as passed by the Senate, 99th Cong., 2d Sess., 131 Cong. Rec. S12184 (1985). A \$10.2 billion reauthorization bill has passed the House (H.R. 2005, as passed by the House, 99th Cong., 2d Sess., 131 Cong. Rec. H11619 et seq. (1985). House and Senate Conferees are currently meeting to resolve the differences between the bills. It is expected that a Superfund reauthorization bill will pass Congress by Fall 1986. As of July 1986, the House and Senate Conferees have agreed to a \$9 billion level of funding but have not yet agreed on the funding sources.

The Administration has indicated that it would support the reauthorization of Superfund at a level of \$5.3 billion. Statements of Administration Policy have indicated that any bill that passes Congress with a broad based "value added" tax similar to one that passed the Senate or crude oil and petrochemical taxes at the levels contained in the House passed bill may be vetoed by the President.

In March of 1986, the House and Senate Conferees agreed on federal facility provisions which imposed specific cleanup requirements and EPA oversight on federal facilities. Under these provisions, releases of "source," "special nuclear," or "byproduct material" that are being cleaned up under other laws would not need to be docketed by EPA under CERCLA (or presumably otherwise subjected to the federal facilities requirements) provided EPA concurs that the cleanup undertaken by the federal agency is consistent with CERCLA.

Other provisions being considered by House and Senate conferees affecting radioactive materials include provisions concerning indoor radon, naturally occurring radon, the temporary storage of radium wastes currently located at certain New Jersey sites, a research and development project on innovative and alternate technologies to characterize radioactive mixed waste at the Hanford site in Washington, and a "grandfather provision" which is intended to exempt the radioactively contaminated Weldon Spring site in St. Charles County, Missouri, currently owned by DOE, from the federal facility requirements of the new law. The final language on these provisions will be contained in a conference report which was not available when this paper was written.

Price-Anderson—Where We've Been, Where We're Going

Ira Dinitz

ABSTRACT The Price-Anderson Act, which became law on September 2, 1957, as part of the Atomic Energy Act of 1954, provides a system to pay funds for claims by members of the public for personal injury and property damage resulting from a nuclear accident. The Act as it now operates entails a two-part insurance system for large utility licensees. The first consists of $160 million in primary nuclear liability insurance purchased by utilities operating large nuclear power plants. Under the second part, these utilities could be assessed up to $5 million per reactor per accident for damages exceeding $160 million. With 101 large power reactors now licensed, the primary and secondary insurance presently totals $665 million. The present Price-Anderson Act expires on August 1, 1987. There are presently two bills H.R. 3653 and S. 1225 being actively considered by the Congress for modification and extension of Price-Anderson. Hearings have been held and the two bills have been marked up and reported out by the Senate and House oversight committees.

The past three years have been particularly active ones for those of us involved in the renewal of the Price-Anderson Act. Over the past eighteen months, various Congressional Committees with jurisdiction over nuclear matters have been considering a number of bills to modify and extend the Price-Anderson Act which expires on August 1, 1987. This shared jurisdiction over atomic energy activities has led to different versions of the same bills being reported out by oversight committees. I will go into detail on this a bit later in the paper.

The Price-Anderson Act was originally enacted in 1957 as a ten-year law, and has been renewed twice with similar ten-year expiration dates. The Act had two main objectives: to ensure that the public would be compensated if an accident occurred at a nuclear facility, and to set a limit on the liability of private industry in order to remove a major deterrent to private participation in the development of nuclear energy.

Price-Anderson provides a system to pay funds for claims by members of the public for personal injury and property damage resulting from a nuclear accident. The Act requires utility holders of licenses of large commercial nuclear power plants to provide proof to the NRC that they have the maximum amount of private nuclear liability insurance--generally referred to as financial protection -- that is available. That financial protection, currently $665 million, consists of a primary layer of nuclear liability insurance of $160 million and a secondary retrospective premium insurance layer. This secondary layer works in the following way. In the event of a nuclear accident causing damages exceeding $160 million, the licensees of each commercial nuclear power plant would be assessed a prorated share of damages in excess of the primary insurance layer up to $5 million per reactor per incident. With 101 large commercial reactors under this system today, the secondary layer totals up to $505 million.

The Price-Anderson Act authorizes the Commission to enter into indemnity agreements with reactor licensees. These agreements specify the amount of financial protection, if any, required of licensees and the obligation of the federal Government to provide funds when a nuclear accident exhausts private liability insurance or when no private liability insurance is required. This government obligation to provide funds is called "government indemnity."

The Price-Anderson Act places a ceiling on the total amount of public liability in an accident. This ceiling or "limit of liability," for large commercial nuclear power plant licensees is currently tied to the maximum amount of insurance available through private sources. For many years, the limit of liability was $560 million. In the 1975 amendments to the Price-Anderson Act, Congress provided that the limitation on liability would grow, once the total protection of the primary and secondary layers of insurance reached $560 million. In November 1982, the $560 million level was reached and the government's indemnity was essentially eliminated for large reactors. The present limit of $665 million will continue to increase in increments of $5 million for each new commercial reactor licensed to operate. In the 1975 amendments Congress also explicitly provided that "in the event of a nuclear incident involving damages in excess of the amount of aggregate liability, the Congress will thoroughly review the particular incident and will take whatever action is deemed necessary and appropriate to protect the public from the consequences of a disaster of such magnitude."

Both the private nuclear liability insurance polices and the indemnity agreement that the Commission enters into with licensees are "omnibus" in nature. That is, in recognition of the requirement of the Act to provide coverage for the licensee and other "persons indemnified," the policies cover not only the utility licensees but also any other person liable for the accident. The scope of Price-Anderson coverage includes any accident in the course of transportation of nuclear fuel to the reactor site, in the storage of nuclear fuel at the site, in the operation of the reactor, including discharge of radioactive effluents, in the storage of nuclear fuel and nuclear waste at the reactor site, and in the transportation of nuclear fuel and nuclear waste from the reactor.

The insurance industry formed two insurance pools to provide nuclear liability insurance capacity to the utility industry at the time of passage of the Price-Anderson Act. One pool, American Nuclear Insurers is composed of investor-owned stock insurance companies. The other pool, Mutual Atomic Energy Reinsurance Pool is made up of policy-owned mutual insurance companies. About half of each pool's total liability capacity comes from foreign sources. Member companies comprising the pools decide independently the amount of capacity they wish to commit to nuclear risks. The Facility Form policy is for the owners and operators of nuclear facilities and when provided as financial protection is a formal part of the Price-Anderson system. The pools also participate in the secondary part of financial protection required by Price-Anderson by issuing policies that set forth the terms, conditions, and obligations of the parties to cover the secondary part of insurance protection. The pools are authorized to charge the utilities and then pay out the collected premium funds on behalf of the utility which had the accident.

While Congress placed particular emphasis in the 1957 enactment of Price-Anderson on the basic principles of protection of the public and encouragement of the industry, other principles were considered. Congress also recognized that the substitution of government indemnity for private insurance should be a short-term expedient--that is, one limited at least initially to ten years. The substitution of private insurance for government indemnity was statutorily recognized in the 1965 extension of Price-Anderson by stipulating that government indemnity would be reduced to the degree that financial protection was provided above $60 million. The process of shifting the burden of nuclear liability risks from the Government to the nuclear industry culminated in the 1975 amendments to the Act which established a secondary retrospective premium insurance layer to provide funds for excess insurance after an accident exhausted the primary insurance layer. As mentioned, as the secondary layer continued to increase, government indemnity was gradually phased out and in November 1982, was eliminated.

Prior to 1977, the Joint Committee on Atomic Energy had exclusive jurisdiction over atomic energy issues. Upon the dissolution of the JCAE, however, a number of different House and Senate committees were granted jurisdictional authority over atomic energy. Regarding the most recent Price-Anderson extension, this shared authority has meant hearings on bills to modify Price-Anderson before four subcommittees with bill markups by five subcommittees and five full committees. This situation contrasts quite dramatically with the 1975 extension of Price-Anderson Act. At that time, only one bill was introduced to modify and extend Price-Anderson with that bill marked up and reported out only by the Joint Committee.

Nine bills have been introduced in the 99th Congress to modify and extend Price-Anderson. Only two of the bills, H.R. 3653, introduced by Congressman Udall, and S. 1225 introduced by Senators McClure and Simpson have been reported out. H.R. 3653 was reported out by the House Interior and Insular Affairs, Energy and Commerce and Science and Technology Committees. S. 1225 was reported out by the Senate Energy and Natural Resources and Environment and Public Works Committees.

Not surprisingly there are significant differences between H.R. 3653 and S. 1225. There are also differences, however, in the different versions of each of these two bills as reported out by the respective oversight committees. I would now like to discuss some of the major provisions of H.R. 3653 as well as point out some of the differences in the two versions of the bill. I will then follow with a similar discussion of S. 1225.

First, as reported out by the House Interior Committee, H.R. 3653 increases the funds available to pay public liability claims arising out of nuclear accident assuming the operation of 101 large commercial nuclear reactors from $665 million to approximately $6.5 billion. This would mean that the deferred retrospective premium assessed against large commercial power reactors would increase from $5 million per reactor per accident to $63 million per reactor per accident with not more than $10 million per reactor collected in any one year. Second, large commercial power reactors would maintain not less than $200 million of primary financial protection. Third, in order to pay public liability claims arising out of a nuclear accident on a timely basis, the NRC would be able to borrow funds from the Treasury or request Congressional appropriation of additional funds to pay the entire aggregate maximum retrospective premiums on behalf of utility licensees. These licensees would then reimburse the Commission over a period of years. Fourth, a future Congress could enact a revenue measure to recover from reactor licensees, funds paid out in compensating victims of a nuclear accident above the liability limit. Fifth, a study commission would be created to examine and report to Congress on alternative measures of compensating victims of a nuclear accident above the liability limit. Sixth, primary insurance and secondary deferred retrospective premiums would not be used to pay the costs of investigating, settling and defending claims for damages arising out of a nuclear accident. Seventh, the costs of a precautionary evacuation would be compensable if liability exists under state tort law. Eighth, the 20-year statue of limitations would be eliminated and a three-year from discovery rule would be substituted. This rule would recognize any claim filed within three years of discovery of damages. Ninth, DOE contractor liability protection would be equivalent to the financial protection made available for NRC reactor licensees.

There are two major differences in H.R. 3653 as reported out by the Interior and Insular Affairs, Energy and Commerce and Science and Technology Committees. First, as I mentioed, in the version of H.R. 3653 reported out by the Interior and Insular Affairs Committee, accidents involving high level radioactive waste would not be subject to a limitation of liability. This provision was deleted by the Science and Technology Committee in its markup so that high level radioactive waste activities would be subject to the same limitations of liability as other DOE activities. The Energy and Commerce Committee further complicated this issue by exempting all DOE contractor activities from a limitation of liability. Second, the Science and Technology Committee removed the NRC from the three-member panel that would be convened to determine the reasonableness of precautionary evacuations for incidents involving Department of Energy contractors.

Turning now to the Senate, S. 1225 as reported out by both the Senate Energy and Natural Resources and Environment and Public Works Committees provides among other things for Department of Energy indemnification of its contractors to the same level as that required of NRC licensees; confirms that the Department of Energy must indemnify contractors involved with its nuclear waste programs; provides for federal compensation of claims arising from incidents involving nuclear material that has been illegally obtained from unknown sources; and allows for the reimbursement of the cost of precautionary evacuations resulting from incidents involving DOE contractors.

Two of the provisions in the version of S. 1225 as reported out by the Energy and Natural Resources

Committee differ from the version reported out by the Environment and Public Works Committee. First, in the Energy Committee version, the Nuclear Regulatory Commission is directed to increase by rulemaking the present $5 million retrospective premium charged each reactor following a nuclear incident to a premium of not less than $15 million nor more than $20 million. The Commission would be required to annually adjust this premium based on the effect of inflation.

Under the version of S. 1225 reported out by the Environment and Public Works Committee, however, the maximum deferred premium that could be assessed against a nuclear reactor per incident would be $60 million with no more than $12 million collected in any one year. Second, S.1225 as reported out by the Energy Committee deletes the 20-year statue of limitations and adopts a 3-year from discovery rule. In the Environment Committee version of S.1225, a fixed statute of limitations provision is retained but increased to 30-years.

It is still too early to predict whether H.R. 3653 and S. 1225 will be brought to the floor of both chambers during the last weeks of the 99th Congress. If this does not occur, it would still be possible for the bills to be considered in a lame duck session if such a session were to be convened. If Congress does not enact Price-Anderson extension legislation this year, however, new bills to extend and modify Price-Anderson would have to be introduced in the next session of Congress. If a new round of hearings and markups were held, it is difficult to predict whether legislation to extend Price-Anderson could be enacted by the August 1, 1987 expiration date of the present Act.

Section V

Institutional Issues

Chairman: ***Robert Wilkerson***
Federal Emergency Management Agency

Protective Action Guides: Rationale, Interpretation, and Status

Joe E. Logsdon

ABSTRACT: The Environmental Protection Agency (EPA) is developing Protective Action Guides (PAGs) for responding to radiological emergencies. The existing interim guidance for plume exposure pathways is 0.01 to 0.05 Sv (1 to 5 rems) whole body dose equivalent and 0.05 to 0.25 Sv (5 to 25 rems) committed dose equivalent to the thyroid from a 2 to 4 day exposure. Interim PAGs for ingestion exposure pathways are 0.015 to 0.15 Sv (1.5 to 15 rems) committed dose equivalent to the thyroid and 0.005 to 0.05 Sv (0.5 to 5 rems) committed dose equivalent to the whole body or any other organ from a one-year exposure. Draft Relocation PAGs have been proposed as a range of 0.01 to 0.05 Sv (1 to 5 rems) committed effective dose equivalent from the first year exposure to deposited radioactive material from all exposure pathways except ingestion of food and water.

The guidance (i.e., PAGs for plume, ingestion, and relocation; limits for emergency workers and other persons entering restricted zones; and guidance for development of recovery criteria) is reviewed with regard to status, values, rationale, interpretation and implementation.

I. BACKGROUND

The Environmental Protection Agency (EPA) is responsible for developing radiation protection guidance for emergency responses to nuclear accidents. This guidance is published in interim form in a document entitled "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents" (PAG Manual) (EPA, 1975). After all PAGs are developed and some experience is gained in their application, we plan to revise them as necessary and publish them in final form.

In the early 1960's the Federal Radiation Council (FRC) developed PAGs for radioactivity in food due to fall- out from nuclear weapons tests (FRC, 1964 and 1965). These were not appropriate for the early phase of a nuclear accident, when the concern is direct exposure to an airborne plume of radio-active materials, and in 1975 EPA developed recommended PAGs for airborne plumes.

Because of the urgent need to provide guidance to the States, EPA elected to issue these PAGs on an interim basis, with the intent of accumulating experience in their application before final promulgation. A similar development process has continued for subsequent PAGs that apply to the intermediate phase. The Food and Drug Administration (FDA), in cooperation with EPA, developed recommendations on radioactivity in food and animal feed and published them in October 1982. The interim PAGs for plumes and recommendations for radioactivity in food and animal feeds have since been incorporated into State radiological emergency response plans (RERPs).

As a result of experience gained over a period of years from developing and testing RERPs, conducting training programs, and the accident at Three Mile Island, the need for some revisions and additions became apparent. It also became apparent that additional PAGs are required for exposure to deposited and resuspended radioactive materials and that guidance is needed for developing criteria for cleanup during recovery.

Prior to the accident at Chernobyl, revised plume PAGs, and ingestion PAGs incorporating the FDA's recommendations for food with EPA's guidance for drinking water, had been developed and reviewed by Federal and State agencies and the nuclear industry. Draft guidance had also been developed for exposure to deposited and resuspended radioactive materials and was in the process of being developed on how to establish long-term radiation protection criteria for recovery. As a result of current experience being gained from Chernobyl and participation in the development of guidance for the International Atomic Energy Agency, some further needs have been identified. These include simplification of the decision process for protective actions and improved, more explicit response levels for radioactivity in food.

II. STATUS

A. Plume PAGs

Plume PAGs were issued in 1975 for whole body external exposure and thyroid inhalation exposure. A proposed revision to include PAGs for other organs for inhalation pathways and the technical rationale for selection of the plume PAG values was sent in 1985 for comment by State and Federal agencies and the nuclear industry.

The most important changes based on those comments are:

1. Revised units are being proposed from "dose equivalent" to "committed effective dose equivalent" based on International Commission on Radiation Protection (ICRP, 1979) dosimetry. The revised PAGs would be a range of 1 to 5 rems committed effective dose equivalent subject to a special limitation for thyroid dose. This change would not require revisions to existing emergency response plans for nuclear power plants, as explained later.

2. The urgency of implementing early protective actions based on plant conditions instead of dose calculations at the time of the accident is given greater emphasis.

3. Simplified dose calculation procedures using nomograms are removed from the guidance, and users are encouraged to employ computer technology in their emergency plans for dose calculations. The simplified procedures are retained as a backup in Appendix D to the PAG manual and for training programs on accident assessment.

4. The cost analyses supporting PAG value selection are revised to eliminate planning costs as a consideration.

5. BEIR-3 (NAS, 1980) analyses are used as the primary basis for evaluation of risk of long-term health effects.

6. The 75-rem whole body dose limit for lifesaving activities was eliminated, and exposure for such activities at projected doses above 25 rems are recommended only for volunteers aware of the risk involved.

Transmittal of the plume PAGs to the Federal Radiological Preparedness Coordinating Committee (FRPCC) for their concurrence is being delayed pending incorporation of some guidance on derived response levels under development by EPA in cooperation with the International Atomic Energy Agency.

B. Ingestion PAGs

Recommendations regarding radioactivity in food and animal feed were developed by FDA in cooperation with EPA and published in 1982 (FDA, 1982). EPA adopted these recommendations (1.5 to 15 rems to the thyroid and 0.5 to 5 rems to the whole body or other organs) as interim PAGs and added drinking water as a category of food. These PAGs, along with implementation guidance, were transmitted for review by State and Federal agencies and the nuclear industry in 1985. The only major changes required as a result of the comments relate to simplification of the dose projection process and the inclusion of example calculations.

As a result of the Relocation Tabletop Exercise in December 1985, it became apparent that, while ingestion PAGs are adequate for application to situations in which supply of food is an emergency need, an emergency for food and drinking water would not exist unless the implementation of protective actions would create a food or water shortage. Stated another way; it would not be reasonable to expect persons to consume food or water that is contaminated to the level of the PAGs if other noncontaminated food or water is available at a cost that is justifiable on the basis of risk avoided. On the other hand, there should be some level of dose above which cost should not be a consideration. This dose is logically the upper range of the PAG (the "emergency PAG"). EPA is currently evaluating the need to develop additional guidance for non-emergency situations following accidental releases. This will require additional peer review before transmittal to the FRPCC for their concurrence. Derived dose conversion factors for all major radionuclides and food types are also being developed.

C. Relocation PAGs

Draft relocation PAGs have been transmitted for review and comment by State and Federal agencies and the nuclear industry. They are recommended as a range of 1 to 5 rems committed effective dose equivalent to individuals in the general public from exposure to deposited radioactive materials via all exposure pathways except ingestion of food and water. The exposure and/or intake period to be assumed for dose projection purposes is one year and the guidance is applicable for up to one year after the accident. If special radiation protection criteria for recovery are not developed within one year, the existing Radiation Protection Guides (FRC, 1961) for individuals in the population will apply beginning with the second year. Persons temporarily reentering the restricted zone established on the basis of the relocation PAGs would have their exposure controlled according to established occupational exposure limits.

After resolution of comments from those presently reviewing the draft guidance, the relocation PAGs will be sent to the FRPCC for their concurrence prior to their transmittal for use as interim guidance.

D. Recovery Guidance

The selection of radiation protection criteria for recovery must consider the existing environmental, social, and economic circumstances. Since the public would already be protected by way of relocation, recovery becomes primarily an economic issue. We do not plan to establish generally applicable numerical recovery criteria. Instead we are developing procedures that may be used after an accident to conduct a cost/risk analysis of the affected area. This analysis would provide a primary basis for selection of radiation protection criteria for recovery.

Draft technical reports dealing with cost/risk analyses and alternative exposure pathways are under development. These will form the basis for procedures that can be used in the development of radiation protection criteria for recovery.

III. RATIONALE

The following three principles represent the primary considerations in establishing values for the PAGs:

1. Acute radiation effects (those effects that would be observable within a short period of time, approximately 60 to 90 days, following exposure and that have a dose threshold below which such effects would not occur) are to be prevented.

2. Delayed effects (primarily cancer and life span shortening) must be minimal and limited to levels that EPA judges to be adequately protective of the public health, and

3. The guidance must be practicable in terms of cost and risk incurred in implementing it.

We have concluded that PAGs in the range of 1 to 5 rems meet these three principles for evacuation to avoid short term (two to four days) exposure to the plume and deposited materials and also to avoid longer term (up to one year) direct exposure to deposited materials. Values for protective actions for ingestion pathways and for sheltering should generally be lower.

No lower limit has been specified for short term (a few hours) sheltering to avoid plume exposure since the cost and risk are small and generally indeterminate. A lower limit for sheltering should be established by local planners or responders based on considerations other than dose (e.g., communication needs, social concern, and political boundaries).

The lower limit of 1.5 rem to the thyroid and 0.5 rem to whole body or other organs from emergency ingestion of food and water are justified primarily on the basis of comparative risk acceptable to the public. Lower values can be justified on the basis of cost/risk analyses for some combinations of radionuclides and foods, but perhaps not for others. The possibility of establishing ingestion exposure criteria based on cost/risk analyses for response under conditions not constituting an emergency is being evaluated, as mentioned previously.

IV. INTERPRETATION

PAGs are stated as a range of values to allow for informed judgment, taking local constraints and conditions into account. They are tools to be used in planning and decision aids for use in actual response situations.

Although there are some minor differences in the interpretation of the ranges for the three categories of PAGs, generally speaking, the lower end of the range is the projected dose at which plans should be made to implement simple or low-impact protective actions (sheltering, use of stored cattle feed, and simple decontamination techniques) and to consider implementation of high-impact actions (evacuation, food restriction, and relocation). The higher impact actions should be implemented at the lower end of the range unless constraints (e.g., weather, costs, and competing emergencies) make their implementation impracticable. Low-impact actions should be considered and implemented, where practicable, at levels below the lower range of the PAGs. No level is established below which low-impact protective actions should not be implemented. This decision is left to local authorities based on existing conditions and the generally accepted public health practice of limiting radiation exposure to as low as reasonably achievable levels.

The upper end of the range is the projected dose above which decisions should usually be made to implement high impact protective actions without consideration of cost or difficulty. Protective actions should not be implemented in any case, however, where the health risk associated with the protective action will equal or exceed the health risk from the radiation that would be avoided.

V. IMPLEMENTATION

The technical portion of PAG implementation involves a determination of the projected dose as a function of location in the environment, the selection of the actual PAG values to be used, and the identification of areas where the selected PAGs will be exceeded. (Realistically, dose projections in the early phase of an accident based on radiological information are not likely to be available at the time of protective action decisions. Such decisions are expected to be made in most cases on the basis of dose estimates related to existing or forecasted plant and/or environmental conditions.)

Dose conversion factors and example dose projection methods, as well as guidance for selecting appropriate PAG values, are provided in the implementation portions of the PAG manual. Users are encouraged to adopt existing computer techniques or to develop their own to supplement the simplified dose projection methods presented.

Since "committed effective dose equivalent" is being adopted as the unit for plume exposure pathways and for direct exposure to deposited radioactive materials, the thyroid, which exhibits a high ratio of curable to fatal cancers, requires special consideration. This is because effective dose considers only the risk of fatal cancers. To provide a simple method of accounting for the high risk of curable thyroid cancers, dose conversion factors for inhalation are derived from the "annual limits for intake" (ALIs) from ICRP-30 (ICRP, 1969). Because this ALI is bounded by the non-stochastic limit of 50 rems, this results in a dose to the thyroid that is one third that calculated on the basis of effective dose. Although the PAGs for pathways other than ingestion of food and water are expressed in terms of effective dose, the implementation guidance results in projected doses that are higher than the true effective dose in cases where thyroid inhalation dose is involved.

Most States already have RERPs based on whole body and thyroid dose equivalent PAGs, and there was concern that the change to effective dose equivalent would force major changes in existing plans. However, if one uses the dose projection techniques provided with the previously issued dose-equivalent plume PAGs and bases decisions on those projected doses and PAGs, calculations show that the effective dose PAGs will be satisfied for most postulated reactor accidents. In those cases in which the effective dose PAGs would not be satisfied, calculations indicated that the effective dose PAG would be exceeded only slightly (10 percent). Therefore, it is not considered necessary to convert RERPs for nuclear reactor accidents to effective dose in cases when the previously issued PAGs and calculational techniques are used.

During the intermediate phase of the accident, monitoring data should be available particularly for decisions to return evacuees or to relocate individuals based on relocation PAGs. The draft now undergoing peer review recommends relocation PAGs in the range of 1 to 5 rems committed effective dose equivalent from the first year exposure to and intake of deposited radioactive materials. It also provides guidance for selecting a value from within the range. In general, the lower end of the PAG range will be bounded by the acceptability of costs for continued relocation relative to the risk avoided. A range of costs for risk avoidance that EPA has used as one factor in its development of environmental standards is recommended as a basis. The range of costs presently used is $400,000 to $7,000,000 per statistical life saved for implementation of an environmental standard. Based on these criteria, a lower bound on the PAG range of less than 1 rem is not expected to be cost-effective.

The upper range relocation PAG is limited by the lower of (a) 5 rems in the first year or, (b) by a dose that could be reduced during the first year so that residents would receive no more than the RPG for individuals (FRC, 1961) during the second year. Based on lessons learned at the Relocation Tabletop Exercise, it was concluded that a gradual return to contaminated areas should be allowed. Return for occupancy at doses greater than the lower range of the relocation PAG should be permitted only after completion of experiments showing probable effectiveness of dose reduction efforts during the first year.

Our intent is to allow for flexibility in the implementation of PAGs. However, above the upper PAG range there is significant risk to the exposed population; and protective actions should normally be considered mandatory, recognizing that when an accident actually occurs, unforseen conditions or constraints may require overriding professional judgment to protect the public.

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The Federal Radiological Emergency Response Plan

Vernon Adler

Abstract - This paper describes the development of the Federal Radiological Emergency Response Plan and its application.

I. GENESIS OF THE FEDERAL PLAN (FRERP)

Work began on the Federal Radiological Emergency Response Plan (FRERP) in 1980 with an effort to bring together Federal resources into a consolidated plan for response to a peacetime nuclear emergency. Plan development became a priority subject following the emergency in March 1979 at Three Mile Island.

A major recommendation of the Kemeny Commission was that there should be centralization of emergency planning and response in a single agency at the Federal level with close coordination between it and State and local agencies, and that there should be a single agency that has the responsibility both for assuring that adequate planning takes place and for taking charge of the response to the emergency.

Additionally, P.L. 96-295 (June 30, 1980) directed the President to prepare and publish a National Contingency Plan. Three months later (September 29, 1980), E.O. 12241 delegated this authority to the Federal Emergency Management Agency (FEMA).

FEMA published the National Radiological Emergency Preparedness/Response Plan for Commercial Nuclear Power Plants (Master Plan) in December 1980, which applied to nuclear power station radiological emergencies.

II. DEVELOPMENT OF THE FEDERAL PLAN

The interagency Federal Radiological Preparedness Coordinating Committee (FRPCC) undertook development of a more detailed plan, following Master Plan publication. This FRPCC effort, led by FEMA, involved 12 Federal departments and agencies. The scope of the Committee's effort was expanded to include all major peacetime radiological emergencies; in April 1983, draft planning guidance was published by FEMA for use by other Federal agencies.

The draft Federal Plan (FRERP) was published for comment in January 1984 and served as the basis for the first Federal Field Exercise (FFE-1) in March 1984 at St. Lucie, Florida. The overall concept of operations for the Federal Plan was tested first in the tabletop exercise (HIEX-82).

The St. Lucie Exercise (FFE-1), established the viability of the draft Federal Plan. Exercise of the Federal Plan was conducted as part of a full field Radiological Emergency Preparedness (REP) exercise which involved the utility, State and local government. Approximately one thousand people participated in the FFE-1 Florida exercise; FEMA's cost of planning for and conducting the exercise exceeded $0.5 million. FFE-2 is to be the second in a series of planned triennial field exercises of the Federal response to a major radiological emergency. It will simulate an accident at the Zion Nuclear Power Plant and will involve coordination of Federal efforts, the State of Illinois and Wisconsin and the counties of Lake and Kenosha.

The Federal Plan became fully operational following concurrence of all twelve participating Federal agencies and its publication in the Federal Register on November 8, 1985.

III. SCOPE OF THE FEDERAL PLAN

The Plan covers a wide range of peacetime nuclear emergencies requiring Federal response in support of State and local government requests for Federal assistance. It is primarily an offsite plan, applicable within the United States and its territories to the Federal response in the event of a nuclear reactor, nuclear weapon, radiological transportation, or other significant accident involving radiological material.

IV. PLANNING ASSUMPTIONS FOR THE FEDERAL PLAN

State and local governments have primary responsibility for protection of public health and safety. Consequently the Federal government will respond when requested by a state or when required to fulfill statutory responsibilities.

The Federal Plan creates no new Federal authorities. It provides for the coordinated implementation of response by Federal agencies operating under each agency's existing authorities.

V. CONCEPT OF OPERATIONS

The FRERP provides for sharing the "lead agency" responsibilities for responding to a major radiological emergency, based on various statutory responsibilities, authorities, and capabilities. The Cognizant Federal Agency (CFA) is the agency that owns, authorizes, regulates or is other wise responsible for the facility on radiological activity causing the emergency, and that has authority to take action on site. For example, the CFA, which is the Nuclear Regulatory Commission in the case of a commercial nuclear power station, operates from the utility's Emergency Operations Facility (EOF), as a general rule. FEMA coordinates among Federal agencies and between the Federal government and State/local governments; its operational center is the Federal Response Center (FRC) through which Federal resources are coordinated in response to State requests for assistance.

FEMA, as principal coordinator for the Federal family, is attentive to numerous interfaces, including coordination among the Federal agencies; coordination between Federal and State/local governments; coordination of offsite with onsite efforts; and coordination between Washington, D.C., and onscene (between the FEMA Emergency Support and Response Teams).

The Department of Energy applies its technical expertise and resources for offsite radiological monitoring and assessment; DOE operates from the Federal Radiological Monitoring and Assessment Center (FRMAC), which is led by a DOE Offsite Technical Director (OSTD).

VI. FEDERAL AGENCY RESPONSE FUNCTIONS UNDER THE FEDERAL PLAN

The Cognizant Federal Agency (CFA) is either the Nuclear Regulatory Commission, Department of Defense, or Department of Energy. It is also probable, in circumstances where the organization or entity responsible for the source of radioactivity is unknown or not of U.S. origin, such as in the case of nuclear satellite reentry, that FEMA would coordinate the Federal response using the technical assistance of U.S. Federal agencies like DOE, NRC or EPA.

DOE may also serve as the CFA in weapons and certain transportation accidents. The Nuclear Regulatory Commission is the CFA in a nuclear power station accident. The Department of Defense can be the CFA in nuclear weapons accidents and otherwise provides support activities.

The other Federal agencies which are currently responders under the Federal Plan include:

a. Health & Human Services - provides health care;
b. Department of Commerce (National Weather Service)-meteorological information;
c. Housing and Urban Development - housing support;
d. Department of Interior - public lands, ground waters;
e. Department of Transportation - coordination and support of transportation;
f. Environmental Protection Agency - long-term radiological monitoring;
g. National Communications System-communications support if required.

VII. EXERCISES OF THE FEDERAL PLAN AND LESSONS LEARNED

The dozen Federal agencies, involved in the Federal radiological emergency response exercise program, are cooperating in a joint effort to conduct major exercises like the Federal Field Exercise of 1984, on a triennial schedule with Federal tabletop exercises held in the year following the field exercises; consequently the second Federal Field Exercise (FFE-2) will be conducted in 1987 with a tabletop effort planned for the following year.

The Federal Field Exercise (FFE-1) in March 1984, and the Relocation Tabletop Exercise in December 1985, provided "lessons-learned" in several areas. The mock accident in FFE-1 followed a scenario of much greater severity than was experienced at Three Mile Island. Eight hours into the FFE-1 emergency, the State of Florida requested Federal Assistance. Major Federal assistance to the State and local government was provided through the Federal Response Center. The FFE-1 included testing of 40 interfaces, grouped into seven broad areas for evaluation including notification, activation and deployment, onsite/offsite activity coordination, public information coordination, White House and Congressional coordination. The FRERP Field Exercise Evaluation Report of June 1984 documents lessons learned as the result of the first Federal Field Exercise (FFE-1) of the Federal Plan.

Reentry and recovery were not examined in FFE-1, and so a relocation tabletop exercise, the RTE, was developed and conducted in December 1985 at FEMA's Emergency Management Institute in Emmitsburg, Maryland. Lessons learned are documented by the March 12, 1986, RTE After-Action Report. For example, it became clear from this exercise that the concept of a Joint Information Center (JIC) needs further definition of goals and guidelines, and that substantial benefits of information exchange were obtained through liaison officers to response centers; this should be encouraged.

An American Nuclear Insurers (ANI) representative in the Federal Response Center provides the Senior FEMA Official rapid access to important data concerning insurance industry response initiatives. It became clear through the RTE that, at present, Federal agencies may or may not choose to implement their respective statutory authorities through the onscene organization, that is, through the SFO or CFA. They may choose instead to implement these authorities through their own headquarters or regional offices.

This and other important policy questions which came into sharper focus in the exercise are being examined through the FEMA-led Federal Response Subcommittee of the Federal Radiological Preparedness Coordinating Committee. Answers to these questions and development of appropriate procedures among Federal agencies in support of the Federal Plan are an ongoing effort.

The second Federal Field Exercise (FFE-2) will be held June 23-25, 1987, at the Zion Plant. The 10-mile Emergency Planning Zone for Zion includes both the State of Illinois and Wisconsin. This exercise will be part of a full REP exercise by these States and the Commonwealth Edison utility. The FFE-2 will include a complete accident scenario, beginning on site and extending to an including key post-accident recovery considerations.

Role of the Federal Radiological Monitoring and Assessment Center (FRMAC) Following a Radiological Accident

John F. Doyle III

ABSTRACT The Federal Radiological Emergency Response Plan (FRERP) calls for the Department of Energy to establish a Federal Radiological Monitoring and Assessment Center (FRMAC) immediately following a major radiological accident to coordinate all federal off-site monitoring efforts in support of the State and the Cognizant Federal Agency (CFA) for the facility or material involved in the accident. Some accidents are potentially very complex and may require hundreds of radiation specialists to ensure immediate protection of the public and workers in the area, and to identify priorities for the Environmental Protection Agency (EPA) long-term efforts once the immediate protective actions have been carried out. The FRMAC provides a working environment with today's high technology tools (i.e., communication, computers, management procedures, etc.) to assure that the State and CFA decision makers have the best possible information in a timely manner on which to act. The FRMAC planners also recognize an underlying responsibility to continuously document such operations in order to provide the State, the CFA, and the EPA the technical information they will require for long term assessments. In addition, it is fully recognized that information collected and actions taken by the FRMAC will be subjected to the same scrutiny as other parts of the accident and the overall response.

I. BACKGROUND

At the time of the Three Mile Island accident in March 1979, the U.S. experience in responding to major nuclear accidents was limited to a few earlier accidents involving nuclear weapons and the joint Canadian - U.S. response to the reentry of the 40KW nuclear reactor on the COSMOS 954 satellite over the Northwest Territory of Canada in January of 1978. Since 1960, the Radiological Assistance Program (RAP), consisting of a dozen or so agencies in the U.S., had been assisting the State and local governments with routine responses to transportation accidents involving trucks, trains, and aircraft, as well as problems in industry and hospitals with lost sources, spills, or wet packages. The typical Radiological Assistance Teams, consisting of five to ten personnel trained in Health Physics and equipped with appropriate monitoring equipment and protective clothing, are well prepared to handle these types of problems. The Canadian experience, however, taught us that a nuclear accident involving dispersal of fission product contamination of potentially high levels over large geographic areas presents major response management problems. Further, it was thoroughly demonstrated in the COSMOS 954 response that aerial radiation measurement technology would be extremely valuable for rapidly defining the affected areas.

Therefore, when the Three Mile Island accident occurred, DOE immediately responded to almost simultaneous requests for assistance from both the State and NRC with both RAP teams and the Aerial Measuring System capability. By Thursday, the second day of the accident, DOE on-site management offered to host an ad hoc discussion of off-site measurement results thus far, for all interested parties including State, NRC, EPA, other federal agencies, and local public safety organizations. Following this meeting, DOE decided to bring in additional resources to establish and support a federal technical coordinating center for off-site measurement and analysis activities. By Saturday this Command and Control Center was fully equipped, and ten RAP teams plus the EPA and other agencies were coordinating a significant off-site monitoring and assessment activity in conjunction with the State Bureau of Radiation Health. Both the State and the NRC were the major users of the results. Following the Three Mile Island accident, the approach and procedures performed on an ad hoc basis were formalized into the present Federal Radiological Emergency Response Plan (FRERP). In addition to coordination of the off-site monitoring and assessment effort, the present FRERP plan assures that all federal support to the state and local authorities is fully coordinated under the direction of the Federal Emergency Management Agency (FEMA).

II. THE FRMAC ROLE

The Federal Radiological Monitoring and Assessment Center has a crucial role to play in response to a major radiological accident. The lesson learned from the Three Mile Island response was not only that one must rapidly assemble accurate off-site measurements, but that these results must be clearly and quickly placed in the hands of the local decision makers and must represent all off-site data being collected. Unfortunately the Chernobyl accident, with its extensive spread of fission product contamination, clearly underlines the need for the role described above. The complexity of the measurement problem cannot be underestimated. Alpha, beta, and gamma effects must all be accounted for over potentially large geographic areas, and the whole process may be further complicated by a continuing release of radiation as well as changing environmental factors. The calibration and assessment functions share the same complications as the data collection. The bottom line, however, is that in spite of the very complex technical processes which must be carried out, information critical to the protective action processes must be placed in the hands of the local decision makers and the CFA at the earliest possible time. These results must also be available in a format that the local authorities and the CFA can immediately use to inform nontechnical authorities and the public in a clear and unambiguous way. Ultimately the news media will be the primary means for communicating directly with the public concerning the status of the situation and recommended actions for public protection. There is a very real technical concern for having an adequate amount of time to make accurate measurements, cross-check the results, prepare assessments, and complete the protective action judgments before making public statements. The major challenge for the FRMAC staff, working hand-in-hand with the State and CFA, is to create a scientific work environment for the off-site measurement effort to compress the information turn around time to the shortest possible time. Emergencies which are technically complex require extensive preplanning and a major investment in technical resources in order to meet the problem head on.

III. THE FRMAC RESOURCES

A. Management

Senior level managers with extensive operating experience in field situations are needed to quickly size up the situation, determine the needed major resources, and implement deployment plans to begin the response. Interfaces with State and local authorities and the CFA must be immediately established to determine those resources they feel are most urgently required.

B. Monitoring Personnel and Special Equipment

For a large-scale release of radiation which may contain particulates, a large number of trained monitoring personnel and the Aerial Measuring System resources will be placed on full alert and deployed if the release has already occurred or is imminent. RAT teams from the closest DOE laboratories and contractors and the EPA teams will be among the first federal personnel to reach the scene. The aerial radiation measurements can provide a very rapid assessment of a large area, locate plumes, and provide isotopic identification. The aerial data can assist the monitoring manager in dispatching ground monitoring teams that are best equipped to make in-situ measurements, set up air sampling stations, etc.

C. Assessment/Data Base Personnel and Equipment

At the earliest possible time after monitoring is initiated, an assessment team of health physics specialists must establish a common map grid for collecting and reporting data, specify the preferred units for reporting results, assure the integrity of all monitoring instruments (i.e., set up a small calibration range), and prepare maps with results clearly shown. The assessment team must consist of State, CFA, DOE, and EPA personnel. Before any data leaves the FRMAC, each organization must indicate that they have knowledge of the data and the FRMAC Off-site Technical Director must formally release the data to be transmitted or hand carried to the State and CFA senior officials. Protective action recommendations are not provided by the FRMAC since they are the prerogative of the State and CFA. If asked by either the State or CFA for technical assistance, the FRMAC will respond.

D. Other Technical Support

1. Command Post Support

The DOE support team can fully equip a command post to support an operation involving 200 to 400 personnel for around-the-clock operations. Included are logging and dosimetry badging for all FRMAC staff, word processing equipment, copy machines, office supplies, a technical reference library, signs/labels for all FRMAC functions, and mundane but essential items such as coffee pots.

2. Communications

A comprehensive communication system capable of providing telephone, VHF radio, HF radio, and satellite links can be airlifted to even a remote site to assure command and control of the operation. In addition, the communication system plays an essential role in the rapid dissemination of FRMAC results to the State and CFA emergency centers.

3. Photo/Video

A scientific and documentary photo/video capability is prepared to support requirements for aerial photo maps, technical documentation of recovery and assessment operations, and copying of documents for volume dissemination. Teams are trained to work in full anti-contamination clothing if necessary. Graphic arts personnel also support this effort to prepare maps or appropriate graphics for use in media briefings at the Joint Information Center.

4. Electro-Mechanical

Large turbine powered generators to fully support a stand-alone FRMAC site, mechanical systems to repair and support FRMAC, and other major support items may be deployed if required.

IV. FRMAC SITE SELECTION

The selection of a FRMAC site depends on a number of factors, but the most significant are the details of the accident. Although there will be some pressure to prelocate a potential FRMAC site, the specifics of the accident may rule out such a site. All of the resources described above are packaged for military airlift and would be trucked from the nearest airport to the FRMAC site. For a worse-case situation, over 200,000 pounds of equipment which would fill seven or eight 40-foot flatbed trucks may be deployed to the site. The equipment and personnel can be completely self-supporting except for housing and food. Even this support could be provided by the military if required. The FRMAC site must not be in the path of any released radiation, and the distance from the site can vary from five to twenty miles depending on the severity of the accident. The FRMAC advance party team is prepared to make this selection at the time a response is required. The FRMAC advance party could be expected to arrive on-site within six to eight hours, and a full FRMAC can be in operation within 24 hours. Some of the important criteria for a FRMAC site are:

- o Space providing 10,000 square feet in a single or separate rooms. (i.e., school, National Guard Armory, tents, trailers, etc.)
- o High ground for communications repeaters
- o Road access for large flatbeds and RVs equipped for analysis
- o Nearby housing for 200 to 400 personnel
- o Available fuel for aircraft and generators
- o Many other items dictated by the accident

If the accident is the result of a natural disaster such as a flood, earthquake, tornado, etc. then many roads, utilities, etc. could be unusable.

V. FRMAC SUMMARY

A major radiological accident creates a potentially complex set of problems for off-site and on-site response teams. For the off-site effort alone, most State and RAT team resources will be exhausted after the first 24 to 48 hours of continuous operations. The FRMAC concept permits some thirteen federal agencies to provide a carefully coordinated and centrally managed support as a backstop for the State and CFA for the off-site radiation monitoring activity. Through the FRMAC, the State and CFA may obtain comprehensive technical assistance drawn from the extensive resources of the DOE national laboratories and contractors. Under the FRMAC concept, each agency is also permitted to carry out their individual statutory requirements. The concept has been fully tested in a full-field exercise held in Florida at the St. Lucie reactor in the spring of 1984. A second full-field exercise is planned for the summer of 1987. The resources described above continually support the DOE in several major emergency programs and, in these other roles, carry out a dozen or more exercises per year in addition to actual responses to more limited emergencies.

Exercising the Federal Radiological Emergency Response Plan[a]

Kathy S. Gant, Martha V. Adler, and William F. Wolff

ABSTRACT Multi-agency exercises were an important part of the development of the Federal Radiological Emergency Response Plan. This paper concentrates on two of these exercises, the Federal Field Exercise in March 1984 and the Relocation Tabletop Exercise in December 1985. The Federal Field Exercise demonstrated the viability and usefulness of the draft plan; lessons learned from the exercise were incorporated into the published plan. The Relocation Tabletop Exercise examined the federal response in the post-emergency phase. This exercise highlighted the change over time in the roles of some agencies and suggested response procedures that should be developed or revised.

I. INTRODUCTION

The Federal Radiological Emergency Response Plan (FRERP) was developed to define the roles of various federal agencies when assisting a state during a radiological incident and to ensure the coordination of the federal assistance. The FRERP was first published in the Federal Register as an interim, but operational, plan on September 12, 1984 (FEMA, 1984). It was republished in a slightly revised form on November 8, 1985 (FEMA, 1985), after concurrence by all the participating agencies. The plan has been tested in exercises, so that needed changes could be identified. This paper will concentrate on two of those exercises, the Federal Field Exercise in March 1984 and the Relocation Tabletop Exercise in December 1985.

A basic understanding of the FRERP requires some brief identification of the major federal participants and their roles. One is the Cognizant Federal Agency (CFA), the agency that owns, licenses, or regulates the material or facility involved in the accident. The CFA has the primary responsibility for the events at the site of the accident and makes federal recommendations to the state regarding protective actions or reentry. In some cases, there will be no CFA, or a state regulatory agency will have onsite responsibility. For an accident involving a commercial power reactor, the Nuclear Regulatory Commission (NRC) will be the CFA.

The Federal Emergency Management Agency (FEMA) coordinates all nonradiological assistance to the state. FEMA plans call for establishing a Federal Response Center (FRC) near or colocated with the state response center, so that the federal agencies and their state counterparts can work together. FEMA also assists the CFA in presenting federal recommendations and, along with the CFA, has responsibilities in the coordination of public information and Congressional and White House liaison.

The radiological monitoring and assessment assistance is initially coordinated by the Department of Energy (DOE) under the portion of the FRERP called the Federal Radiological Monitoring and Assessment Plan. DOE may set up a Federal Radiological Monitoring and Assessment Center (FRMAC) from which the state and participating federal agencies can coordinate their data collection and analysis. The radiological assistance portion of the plan may be implemented separately to provide federal radiological monitoring and assessment assistance in less severe incidents. After the emergency phase of the response ends, the Environmental Protection Agency will assume the coordination of any needed federal monitoring assistance.

The other federal agencies will carry out any statutory responsibilities, as well as contributing their resources to provide radiological assistance or other assistance to the state through the FRC or FRMAC.

Exercises were an important ingredient in the development of the federal plan. In October 1982, at the time when the interagency Subcommittee on Federal Response was still working on the planning guidance for the FRERP, member agencies of the Subcommittee held a command post exercise to clarify the notification and communication requirements which would have to be articulated in the FRERP. The differing ways that agencies interpreted the planning guidance became apparent in this exercise, but it offered the opportunity to deal with the differences systematically by clarifying them as the plan development proceeded.

[a]Based on work performed at the Oak Ridge National Laboratory, operated for the U.S. Department of Energy under Contract No. DE-AC05-84OR21400 with Martin Marietta Energy Systems, Inc.

II. THE FEDERAL FIELD EXERCISE

The first major exercise of the FRERP was held March 6-8, 1984, at the St. Lucie Nuclear Power Plant on the eastern coast of Florida. Approximately 1000 people from Florida Power and Light Company, the state of Florida, Martin and St. Lucie counties, volunteer groups, and representatives from 11 federal agencies and their contractors participated in the exercise. The goal of the exercise was to test the FRERP, which had been published in the Federal Register in final draft form, in terms of the interfaces between different agencies, the usefulness of the assistance to the state, and the adequacy and compatibility of the agency plans.

Two preliminary drills preceded the actual exercise. A limited number of participants from all the participating agencies took part in a tabletop exercise on December 1-2, 1983, at the utility's Emergency Operating Facility. This drill tested some of the notification procedures, information flow, and agency interfaces using a simple scenario. The second preliminary drill, called the "Dry Run", was held January 24-25, 1984, using the emergency facilities that would be used in the exercise. Communications and facilities were evaluated with a second simple scenario; the control and evaluation procedures for the actual exercise were also tested.

The first day of the three-day March field exercise was the plant's annual compliance exercise. A well-developed scenario described a large release of radioactive material which went outside the plant boundaries. The state notified the appropriate federal agencies and requested federal assistance; the responding agencies simulated their arrival and began setting up operations so that full federal play began by the second day. The second day tested the federal agencies' abilities to communicate and work together to meet the needs of the state and local governments. On the third day, the scenario shifted to the fifth day after the accident and the emergency phase of the response began to scale down.

All federal equipment and personnel were prepositioned for the exercise to reduce costs. The FRC was established in trailers adjacent to the state's Field Emergency Operating Center. DOE established the FRMAC in a hanger at the Stuart, Florida, airport, and the state moved its radiological monitoring activities to that location. The onsite response was directed from the utility's Emergency Operating Facility; a Joint Information Center (JIC) was located in the same building.

The Federal Field Exercise demonstrated that the FRERP was an operable plan that could be implemented. Some of the necessary interactions among the federal agencies and between the federal agencies and the state were tested for the first time. The exercise play showed the importance of preplanning and developing good working relationships with counterparts in other agencies before an emergency occurs. Minor problems, such as the need for all federal agencies involved in monitoring to be present in the FRMAC and clarification of the data flow, were then addressed in the plan before it was published.

III. RELOCATION TABLETOP EXERCISE

As with most emergency response plans, the FRERP is most specific with regard to the emergency phase. While the plan was designed to extend beyond the initial response period, the mechanisms for communication and coordination described in the plan had never been tested during an extended response. The Relocation Tabletop Exercise provided the first detailed exercise experience with the FRERP in the period beginning several days after an accident.

The Relocation Tabletop Exercise was conducted on December 9-11, 1985, at the National Emergency Training Center in Emmitsburg, Maryland, in conjunction with the Commonwealth of Pennsylvania and Duquesne Light Company. Duquesne Light operates the Beaver Valley Power Station, the site of the simulated accident. Representatives of the Pennsylvania Emergency Management Agency, the Pennsylvania Bureau of Radiation Protection, Beaver County Emergency Management Agency, Duquesne Light, the American Red Cross, the Institute for Nuclear Power Operations, and American Nuclear Insurers joined the federal players from 13 agencies for the exercise.

The Relocation Tabletop Exercise differed from the St. Lucie exercise in many ways. It was conducted as a tabletop not a field exercise and dealt with the return-relocation phase following a large power reactor accident, a sensitive time for which the FRERP guidance is less specific. There were no media, no visitors, and few observers. Although Duquesne Light Company actively participated in the exercise, it was not part of any required emergency activity. One workshop, to discuss the types of issues that would arise during the exercise, preceded the actual event.

The exercise scenario had postulated a substantial release of radioactive materials from a fuel handling accident at the Beaver Valley Power Station in Shippingport, Pennsylvania, leaving radioactive materials deposited over part of the surrounding area. The exercise assumed that over 100,000 people would have been evacuated (according to the Pennsylvania plan) and that federal assistance would have been requested. When play began, postulated to be the fourth day after the simulated release, the plant was stable, and no further releases of radioactive material were anticipated. It was assumed that there was increasing economic and political pressure to allow the evacuated people back into the area.

Instead of playing under one long scenario, the general scenario was divided into a series of nine miniscenarios, each of which focused on a single problem. The players worked from tables which represented operations centers or centers for federal assistance that would have been established under the FRERP. The staffs at different tables consulted with each other as necessary. At the end of each miniscenario, there were general discussion of the issue and a report from each facility.

The "no-fault" exercise was intended to identify issues and problems which needed consideration or procedures which might need to be developed for this postaccident phase. An important exercise goal was to document any changes that might take place in the federal roles and determine whether these changes were adequately covered in the federal plan. The Environmental Protection Agency (EPA) used the exercise to see whether the concepts described in the Relocation Protective Action Guidelines (PAGs) being developed (EPA, 1985a; EPA, 1985b) for making relocation decisions could be applied.

While most of the evaluators found no need for major revisions to the FRERP itself (FEMA, 1986), several situations arose which indicated a need for more study and possibly for the development of additional procedures for use in the later stages of a response.

For example, after a serious accident at a nuclear power plant, and when the plant is stable and onsite recovery is underway, the Nuclear Regulatory Commisssion will want to focus its efforts onsite. Some of the responsibilities it holds under the FRERP as the CFA may be less appropriate or less in its areas of interest and expertise. For example, the NRC does not have expertise related to environmental questions and therefore does not feel comfortable coordinating the technical aspects of this phase. The exercise was unclear as to what effect this change in focus would have on the CFA's responsibilities.

The FRC, the coordinating facility for the nonradiological federal assistance, did not have as much play as expected during the exercise. This may have been a result of the exercise structure, where problems were handled one at a time, resulting in little strain on the state resources. The lack of full representation by the appropriate state agencies may have also contributed to the reduced FRC participation. The question of how the FRC role may change as the response moves into the recovery phase was therefore not fully explored.

The exercise showed that the technical radiological assistance provided by the federal government was needed by the Commonwealth of Pennsylvania beyond the emergency phase of the response. The activities of the FRMAC in this regard were praised by most participants.

EPA's draft Relocation PAGs (EPA, 1985a; EPA, 1985b) contributed to making decisions about how people would be allowed to return to their homes. The Commonwealth of Pennsylvania, however, chose to select a lower projected one-year dose than the lower bound of the guidelines as the criterion for immediate return. More measurements and study would take place before other areas where projected doses from the radiation levels were approaching the projected doses of the PAGs would be opened for return. EPA considered the Commonwealth's approach and incorporated it into the draft Relocation PAGs (EPA, 1986a; EPA, 1986b; EPA, 1986c) that were released for public comment.

One problem identified in the exercise is how federal agencies, such as EPA, the Department of Agriculture, or the Food and Drug Administration, should go about fulfilling any statutory responsibilities when they are also participating in the FRC or the FRMAC. Procedures need to be developed so that agency recommendations to the state can be made as required without appearing to conflict with or to bypass the communication paths outlined for the CFA, FRMAC, or FRC in the federal plan.

Play at the Joint Information Center (JIC) was added late in the exercise planning. No public information representative was involved in the planning. Consequently, the JIC activities were not closely integrated into the exercise, limiting the usefulness of any observations. The Commonwealth's plan calls for public information activities to be located in several different places. Although colocation of the JIC staff is desirable, ways of coordinating the release of information, when all those with the authority to approve the release cannot be colocated, should be explored.

Detailed advanced planning is probably not appropriate for the postemergency response phase. Each accident will be based on a unique set of initial response plans as well as present different problems, necessitating tailored solutions. The principles of the FRERP can still guide the extended federal response, but it is useful to think ahead about how the response might function and what kinds of problems could arise. The Relocation Tabletop Exercise was successful in beginning this process. During the coming months, the interagency planning group will be discussing some of the problems identified in the exercise.

IV. CONCLUSION

The FRERP was never intended to be a static document; additional changes will probably be made as the participating agencies get more experience in responding under the Plan. In this regard, it is important for the federal agencies to be involved in state emergency preparedness exercises whenever possible and for the states to include provisions for federal assistance in their plans.

The FRERP is a generic plan, applicable to all sorts of radiological emergencies. The major exercises of the FRERP in the past have involved scenarios dealing with power reactor accidents. Another major field exercise of the FRERP is to be held in 1987, in conjunction with a power reactor exercise. Exercises involving other types of accidents, different agencies as the CFA, and more complicated situations where there may not be a designated CFA will help individual agencies see how their plans must be adapted to their changing roles in the response. Also, such exercises will help to produce a workable, generic federal response plan.

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Emergency Response to Transportation Accidents

Cheryl Sakenas

ABSTRACT - The Nuclear Regulatory Commission (NRC) has issued a policy statement regarding its role in transportation accidents involving the release of radiation. Each State has the primary responsibility for protecting the health and safety of its citizens. States have chartered certain agencies with the responsibility of responding to radiological emergencies.

When informed of a transportation accident, the NRC will notify the designated State agency to ensure that they have been informed of the incident and will offer NRC assistance. The NRC also will notify the U. S. Department of Energy (DOE), the U. S. Department of Transportation (DOT) and any other affected agency to ensure a coordinated Federal response.

Under an existing NRC/DOT Memorandum of Understanding, NRC is responsible for investigating any accidents involving packages regulated by the NRC. NRC staff may be dispatched to an accident scene involving packages not regulated by NRC whenever significant amounts of radioactive material were or might be released and NRC activities will generally be limited to information collection unless assistance is requested. DOE would be available to assist State and local responders in monitoring and decontamination of affected areas. In an accident in which Federal assistance was being provided to the State, portions of the Federal Radiological Emergency Response Plan would be activated to coordinate the provision of that assistance. An example of recent experience in responding to a transportation event will be discussed.

On March 23, 1984 the Nuclear Regulatory Commission issued a policy statement regarding the agency's role in the response to transportation accidents involving radioactive material. This was published in the Federal Register on March 29, 1984 at 49 FR 12335. The purpose of the policy statement is to state clearly the extent of the NRC's participation and involvement in responding to a transportation accident or incident. NRC's role and responsibilities for responding to any potentially threatening incident involving NRC-licensed activities are delineated in the NRC Incident Response Plan, NUREG-0728, Rev. 1, April 1983.

The response to transportation accidents is less predictable than the emergency response to radiological accidents at licensed sites because of the uncertainties surrounding (1) the location where the accident occurs, (2) the diversity of authority of those who will be responding, and (3) the likely limited radiation knowledge of the first-on-scene responders (who are usually local officials). Each State has the primary responsibility for protecting the health and safety of its citizens from public hazards.

The Commission views the States as appropriately having the lead in the overall direction of response to transportation accidents. In the United States, we experience approximately 10-20 transportation accidents per year involving radioactive materials. These are routinely handled by a State-designated agency.

Most carriers of shipments of materials licensed by NRC are exempt from NRC regulations while in transit and come under the regulatory authority of the U. S. Department of Transportation (DOT). The DOT operates a National Response Center which serves to relay information to State and Federal agencies concerning transportation incidents involving hazardous materials. DOT regulations require a carrier, to promptly notify the National Response Center after an incident occurs in which, fire, breakage, spillage, or suspected radioactive contamination occurs. Each notification of a transportation incident of any kind is relayed by the National Response Center to the Regional Office of the Environmental Protection Agency (EPA) for incidents on land or to the U. S. Coast Guard Captain of the Port for incidents in navigable waters. When a reported incident is known to involve radioactive material, notification also is made to the Regional Coordinating Office for Radiological Assistance of the U. S. Department of Energy (DOE) and to the NRC Operations Center. The NRC also may become aware of a transportation incident through other channels, such as the shipper, the carrier, or the police or highway patrol.

When informed of a transportation accident, the NRC will inform the agency designated by the State as soon as practicable to ensure that the State agency has been informed of the incident. The NRC will offer technical assistance in the form of information, advice, and evaluations. The NRC also will ensure that DOE, DOT, and other affected agencies, including the Federal Emergency Management Agency, are made aware of the incident.

Under an existing NRC/DOT Memorandum of Understanding, NRC is responsible for investigating all accidents, incidents, and instances of actual or suspected leakage involving packages of radioactive material regulated by the NRC. Accordingly, NRC inspectors will normally be dispatched to the accident scene whenever a report is received that such packages (i.e., Type B containers) were involved in a significant accident that could have released or threatens to release radioactive materials. To maintain awareness so that meaningful technical assistance may be provided to the State, NRC staff also may be dispatched to an accident scene involving packages not regulated by NRC (i.e., Type A containers) whenever a significant amount of radioactive material might be released. The NRC will provide information on packaging characteristics in response to any query regarding NRC-approved packages.

Any NRC personnel at the scene of a transportation accident will notify the State/local government on-scene coordinator of his/her presence and make clear that, unless NRC assistance is requested by the on-scene coordinator, NRC activities will be primarily limited to information collection and assessment. NRC personnel will pass along recommendations to emergency response personnel on radiological issues only if NRC assistance is requested by the on-scene coordinator or if NRC personnel at the scene believe additional actions are needed to protect emergency response personnel or members of the public. NRC's role, however, is not to evaluate State and local government emergency response actions. NRC will respond to requests for information on NRC activities in connection with the event. Requests for specific information on an accident normally will be referred to the appropriate State agency, or to the DOE if the situation relates to DOE activities. This policy relates solely to radiological concerns. Responding to any attempt to steal or sabotage a shipment of nuclear material is a responsibility of the Federal Bureau of Investigation (FBI) as delineated in the NRC/FBI Memorandum of Understanding dated April 27, 1979, and published in the Federal Register on December 20, 1979 (44 FR 75535).

A Federal Radiological Emergency Response Plan (FRERP) has been developed for use in peacetime radiological emergencies and defines the authorities and responsibilities of each Federal agency involved in a significant emergency response. In most cases Federal response will not be needed. State and local governments may request Federal assistance as needed. In most cases this would involve field monitoring and cleanup support from DOE or EPA and technical assistance from NRC. If contamination of food-stuffs is involved, assistance may be requested from the Departments of Health and Human Services and Agriculture.

Since the FRERP is concerned primarily with Federal support to State and local governments beyond the immediate site of the emergency, State and local governments will define an area "onsite" at the time of the accident and manage all actions within that area. If the accident involves materials shipped by DOD or DOE, these agencies will define and control the onsite area. There are some exceptions to this, depending on the type of material, i.e., spent fuel, and whether the involved State was an Agreement State, in which case the State may define and control the onsite area.

An example of how the NRC responds to a transportation event involved the spill of uranium oxide (yellowcake) following a collision between the tractor trailer carrying the material and a train on August 27, 1985 in Wells County, North Dakota. Of the 53 drums containing the material, 30 ruptured, contaminating a 5000-square foot area, including a 2-mile stretch of highway which was isolated by the responders.

The first responders, local fire department, State and local law enforcement, and local rescue team were backed up by the State Hazardous Materials Emergency Response Team which included State Health Department and Radiation Control Program personnel. The day after the accident the State requested NRC assistance in the cleanup and recovery efforts. The Region IV office responded by sending three health physics inspectors to the accident scene to assist the State by providing recommendations on decontamination and recovery of the area. The mining company also dispatched a team to take care of the cleanup. Oak Ridge National Laboratory performed bioassay analysis on contaminated response team members. The site was returned to normal in 15 days.

New Source Terms and the Implications for Emergency Planning Requirements at Nuclear Power Plants in the United States

Geoffrey D. Kaiser and Michael C. Cheok

ABSTRACT This paper begins with a brief review of current approaches to source term driven changes to NRC emergency planning requirements and addresses significant differences between them. Approaches by IDCOR and EPRI, industry submittals to NRC and alternative risk-based evaluations have been considered. Important issues are discussed, such as the role of Protective Action Guides in determining the radius of the emergency planning zone (EPZ). The significance of current trends towards the prediction of longer warning times and longer durations of release in new source terms is assessed. These trends may help to relax the current notification time requirements. Finally, the implications of apparent support in the regulations for a threshold in warning time beyond which ad hoc protective measures are adequate is discussed.

I. INTRODUCTION

In 10CFR50.47, the Nuclear Regulatory Commission (NRC) requires that no operating license will be issued for a nuclear power station unless the NRC finds that there is reasonable assurance that adequate protective measures can be and will be taken in the event of a radiological emergency. Thus, utilities are required to develop an emergency plan for each nuclear power plant. Generally, the zone within which emergency plans must be developed for prompt protective measures against exposure to the plume (known as the plume exposure emergency planning zone (EPZ)) consists of an area with a radius of about 10 miles.

The reasons why 10 miles was chosen for the radius of the EPZ are given in NUREG - 0396 (USNRC, 1978) and are summarized as follows in NUREG - 0654 (USNRC, 1980).

1. Projected doses from the traditional design basis accidents would not exceed Protective Action Guide (PAG)[a] levels outside the zone.
2. Projected doses from most core-melt accident sequences would not exceed Protective Action Guide levels outside the zone.
3. For the worst core-melt sequences, immediate life threatening doses would generally not occur outside the zone.
4. Detailed planning within the EPZ would provide a substantial base for the expansion of response efforts in the event that this proved necessary.

As is explained in NUREG-0396, the second and third criteria were applied within a risk-based framework that relied upon the Probabilistic Risk Assessment that was done in WASH-1400 (USNRC, 1975), including the frequencies of accident sequences and the corresponding characteristics of the fission product source terms. The WASH-1400 fission product source terms were input into the CRAC (Calculation of Reactor Accident Consequences)

[a]The PAGs were defined by the U.S. Environmental Protection Agency (US EPA 1980) and are expressed as a range: 1-5 rem for whole body exposure and 5-25 rem for thyroid exposure of the general public. The lower value should be used if there are no major local difficulties or risks that would arise in carrying out protective actions at that level. In no case should the higher bound be exceeded in determining the need for protective action. Reference (USEPA, 1975) states that these recommendations are for use by Federal and State agencies in their emergency planning activities. It also states that the PAGs are specifically directed towards protective actions by civil authorities in the event of inadvertent releases of radioactive material from fixed nuclear facilities. Thus, the PAGs are envisaged for use in both emergency planning and response.

code, which was the consequence modeling code developed for WASH-1400, and "Dose vs Distance" curves were calculated as shown on Figure 1. These curves give the probability of exceeding a given whole body dose as a function of distance, conditional on the occurrence of core-melt.

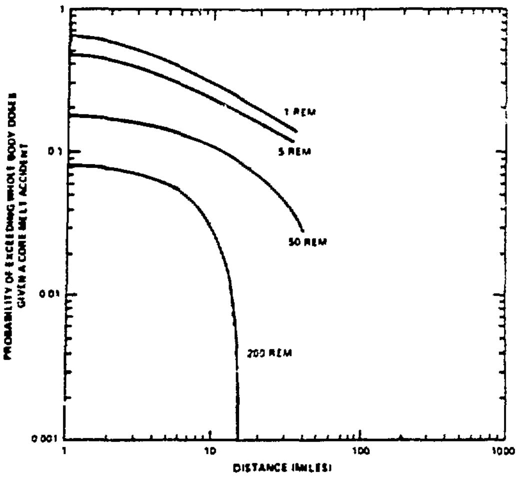

FIGURE 1 REPRODUCTION OF FIGURE I-11 FROM NUREG-0396

Briefly, criterion 2 may be roughly translated into the quantitative equivalent that the probability of exceeding the PAGs at 10 miles is no more than about 0.3 for "most core-melt sequences." The 200 rem curve on Figure 1 is dominated by the "worst core-melt sequence" and shows a pronounced knee at about 10 miles. Since 200 rem is roughly the threshold for early fatalities (see the PRA Procedures Guide (USNRC, 1983)), the 200 rem curve in Figure 1 is the approximate quantitative expression of the third criterion.

In summary, the original reasons for the choice of a radius of 10 miles for the EPZ contained the following important elements.

1. Risk-based calculations of the consequences of core-melt accidents, including consideration of both the "worst case" accident and the consequences of all core-melt accidents.

2. Source terms with characteristics that were determined using the methods of WASH-1400.

3. The EPA's Protective Action Guides.

4. A criterion that addressed the consequences of design basis accidents.

5. A criterion that addressed the need for flexibility of response.

II. *NEW SOURCE TERMS AND THE EPZ*

Recent work on fission product source terms has shown that, in several cases, the magnitude of the source term is considerably smaller than was thought to be the case at the time that WASH-1400 was written (APS, 1985), although this statement cannot be shown to be true for all source terms in all reactors and the NRC has stated that generalizations are inappropriate (USNRC, 1986). However, many have found recent progress to be so promising that they have already addressed the question of what the implications of the new source term results are for emergency planning.

The Baltimore Gas and Electric Company has submitted a request to the NRC for an exemption from the requirement that the EPZ should have a radius of 10 miles (Montgomery, 1986). Using a Calvert Cliffs-specific source term for Event V, the interfacing systems LOCA, which is the worst case source term for this particular PWR, BG&E essentially used criterion 3 of the four from NUREG-0654 to show that a radius of no more than 2 miles is needed for the EPZ. They reinforce this conclusion by showing that the Calvert Cliffs DBA does not lead to doses exceeding the PAGs at this distance. However, they do not address the question of whether most core melt accidents lead to the PAGs being exceeded at 2 miles, or whether planning inside 2 miles allows sufficient flexibility to expand beyond that distance if need be.

The Electric Power Research Institute and its contractor, NUS Corporation, (EPRI, 1986) have undertaken an exercise to see what can be concluded if the new Industry Degraded Core Rulemaking Program (IDCOR, 1984) source terms are used within the framework prescribed by NUREG-0396 and NUREG-0654. Using criterion 2 (most core-melt sequences and the PAGs) and criterion 3 (worse case core-melt sequences and life-threatening doses) EPRI has shown that the radius of the EPZ for the four IDCOR reference plants (Zion, Sequoyah, Peach Bottom and Grand Gulf) need be no more than 3 miles. The criterion relating "most" core melt sequences to the PAGs at the EPZ boundary is by far the most restrictive and this is significant, as will be discussed later in this paper. EPRI did not discuss the influence of DBAs on the radius of the EPZ. This is reasonable, because the current DBA source term is recognized to be conservative and is being revised using the NRC's modern source term code package (USNRC, 1986a; 1986b). Finally, EPRI did not address the flexibility question except to say that, with the IDCOR source terms, the probability that there will be a need to expand prompt emergency response beyond 10 miles, or even as far as 10 miles, is very small.

IDCOR itself has undertaken to devise a new framework for the definition of the radius of the EPZ (IDCOR, 1986). Briefly, IDCOR defines "Limit Lines" on a plot of frequency vs mean individual whole body dose. These limit lines correspond to an individual risk of early fatality of 10^{-7} per year in the region of the line in which the whole body dose exceeds 200 rem and an individual risk of latent cancer fatality of 10^{-7} per year in the region of the line in which the whole body dose is less than 200 rem. Using the IDCOR source terms and also the NRC source terms calculated with the NRC's Source Term Code Package, IDCOR shows that the EPZ need have a radius of only 2 miles. IDCOR's work is therefore risk-based and relies on new source terms. However, the PAGs play no role in defining the radius of the EPZ and the flexibility criterion is not addressed. The DBA also plays no role.

The owners of the Seabrook plant have also carried out extensive work on EPZ reduction and have been engaged in discussion with the NRC. At the time of writing, no published work was available. However, the authors understand that the work involved extensive new analysis of the source terms for risk dominant accident sequences and that the approach adhered fairly closely to that of NUREG-0396 and NUREG-0654.

Finally, other authors have considered the question of the relationship between the size of the EPZ and the characteristics of the source terms. There is not enough space to review these in detail here. However, Kaiser (1986) has looked at the relationship between the radius within which evacuation is necessary to avoid life-threatening or injury-threatening radiation doses and the source term magnitude and has concluded that, if the largest average volatile fission product release fraction is less than about ten percent, a 2 mile EPZ would be sufficient to prevent such doses. Sofer (1985) has discussed the concept of the "graded approach" in which the radius of the EPZ is not reduced but in which planning for prompt evacuation is confined to the first two miles. He shows that the graded approach can be justified even without new source terms.

III. ROLE OF THE PAGS

In the approaches adopted above, the authors all retain a risk-based approach and they all use new source terms. Not everyone uses a DBA-based criterion, but there seems to be a general consensus that this is not a serious problem because the DBAs are to be revised and will look like severe core damage sequences in the category of "most core-melt accident sequences" anyway. The flexibility criterion is universally ignored or played down, and may deserve more attention than it has received hitherto. However, the most significant difference between the approaches is the attitude towards the use of the PAGs.

Essentially, there are two approaches to the use of the PAGs in defining the radius of the EPZ. One is to abandon the second of the NUREG-0654 criteria and not to consider at all whether "most" core-melt accidents exceed the PAGs at the boundary of the EPZ. In the authors' view, this approach is highly desirable. For example, the 5 rem PAG corresponds to a probability of about one in a thousand that the affected individual would develop cancer over the remainder of his/her lifetime. From Figure 1, the probability of occurrence of the 5 rem PAG at the 10 mile boundary of the EPZ is about 0.3. The WASH-1400 core-melt frequency is about 5×10^{-5} per year. A further factor of 0.1 can be applied to allow for the probability that the wind will blow towards a particular individual. Multiplying all of these factors together, emergency planning is being driven by an individual risk of not more than one in a billion per year. This is six to seven orders of magnitude below the normally occurring individual risk of latent cancer fatality.

However, while technical arguments can certainly show that the risk associated with the PAGs is small, there is no certainty at the present time that any scheme that is ultimately introduced to justify a reduction of the radius of the EPZ - if, indeed, any such scheme is introduced at all in the forseeable future - will be able to avoid the incorporation of the PAGs. Doubtless, Federal Agencies other than the NRC will be consulted, and the opinions of a host of State and local authorities will have to be taken into account, and there is no predicting what consensus may emerge from these discussions.

If the PAGs cannot be excluded in the context of the impact of "most" core melt accidents, what then? The immediate difficulty is that, if the second criterion of NUREG-0654 is applied to a source term consisting entirely of noble gases, there may still be a probability of greater than 30 percent of exceeding the 1 rem PAG at 10 miles. In other words, even if all source terms are so small that the noble gases dominate the offsite doses, the second criterion of NUREG-0654 may still be very restrictive and may make it difficult to show that the radius of the EPZ can be reduced to as little as 2 miles. This point is discussd further below.

The next sections of this paper will develop a chain of reasoning that will show that there is a way of removing this restrictive result without diminishing the importance that the PAGs play in the underlying rationale for determining the radius of the EPZ. The foundation stone of the argument is the concept, which was not developed in NUREG-0396 or NUREG-0654, that there is a warning time beyond which ad hoc protective measures are sufficient to protect the health of the public. As is shown below, there is support for this concept in the regulations.

IV. REGULATIONS AND AD HOC EVACUATION

In the discussions of the final rule on emergency planning and preparedness in the Federal Register of July 13, 1982, the NRC has a section devoted to the risks associated with low power operation, where it states: "For issuance of operating licenses authorizing only fuel loading and low power operation (up to 5% of rated power), no NRC or Federal Emergency Management Agency (FEMA) review and determinations concerning the state of adequacy of site emergency preparedness shall be necessary" because: "The risks of operating a power reactor at low power are significantly lower than the risks of operating at full power" and "Even for a worst-case, low likelihood sequence which could eventually result in the release of fission products accumulated at low power into the containment, the additional time available *(at least ten hours* (authors' italics)) would allow adequate precautionary actions to be taken to protect the public near the site."

Thus, this chain of reasoning lends support to the concept that there is some threshold in time beyond which ad hoc emergency response is adequate. In NUREG-0396, however, the doses which are the basis for Figure 1 were calculated by assuming that people continue their normal activities for 24 hours after the passage of the plume, no matter how long the warning time and or the duration of the release. It would be logical to calculate Figure 1 on the basis of doses accumulated only up to the time beyond which ad hoc response is deemed to be adequate, since the purpose of Figure 1 is to give guidance on what kind of planned emergency response might be required.

This concept of a threshold in time beyond which ad hoc response is adequate has relevance to the results of new source term research, see Table 1, which compares the warning times and durations of release derived for the IDCOR reference plants with those from WASH-1400. Briefly, most of the IDCOR warning times exceed those of WASH-1400, and all of the IDCOR durations of release exceed those of WASH-1400.

	IDCOR[d]		WASH-1400	
	Warning Time	Duration of Release	Warning Time	Duration of Release
Peach Bottom				
TW	10	80	2	3
TC(V)[a]	12	50	2	3
TC(NV)[a]	4	50	2	3
S_1E	21	30	2	3
TQVW	10	30	-	-
Grand Gulf				
T_1QUV	46	10	2	3
AE	57	10	2	3
$T_{23}QW$	10	80	-	-
$T_{23}C$	1.5	50	2	3
Sequoyah				
TMLB'	26	4	1	0.5
$S_2HF(DO)$[b]	8.5	6	2	1.5
$S_1HF(DO,IC)$[c]	23	6	2	1.5
V^2	0.5	5	2	3
	17	4	1	0.5
Zion				
TMLB'	30	13	1	0.5
TMLB'(IC)[c]	1.5	7	2	3
V	23	4	1	0.5

[a]Failure to shut down with venting in the wetwell airspace (V) or without venting (NV)

[b]DO - Drains open; DC - drains closed

[c]IC - Impaired containment

[d]See IDCOR Summary Report (IDCOR, 1984) for definition of symbols T, W, etc. The sequences listed above are those used to calculate the consequence analysis results in the above reference. The failure is by gross rupture of the containment unless otherwise stated.

TABLE 1
SUMMARY OF WARNING TIMES AND DURATIONS OF RELEASE (Hr)

Figure 2 was derived with a source term that consisted principally of the noble gases. The warning time was 1 hour and the duration of release was also 1 hour, and the doses on which Figure 2 was based were accumulated up to 24 hours after cloud passage as in NUREG-0396. As can be seen, there is a chance of 0.8 that the 1 rem PAG will be exceeded at 10 miles. This highly conservative case - a release consisting mainly of noble gases would be expected to have a longer warning time and duration - illustrates the point that was noted above, that criterion 2 of NUREG-0654 may still be restrictive even with small source terms.

Figures 3A and 3B show how Figure 2 changes if warning times and durations of release are longer (3A) and if doses are only accumulated up to a threshold time of either

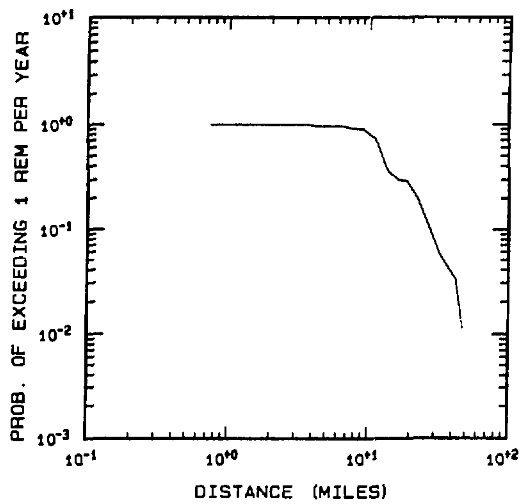

FIGURE 2 DOSE VS DISTANCE CURVE FOR NOBLE GAS RELEASE - CONSERVATIVE CASE

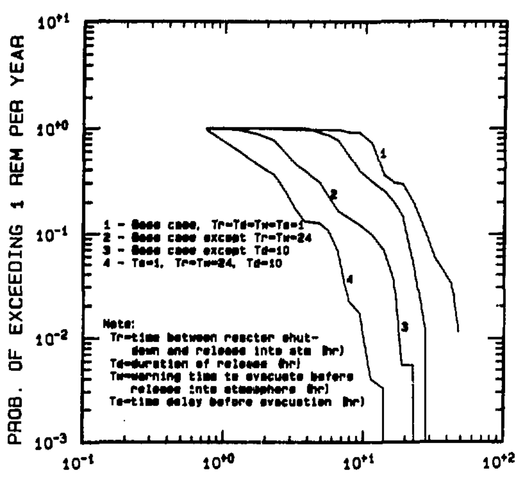

FIGURE 3A DOSE VS DISTANCE - EFFECTS OF WARNING TIMES AND DURATIONS OF RELEASE

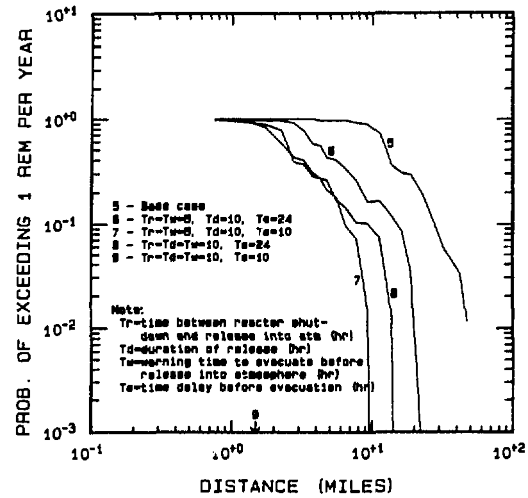

FIGURE 3B DOSE VS DISTANCE - EFFECTS OF EVACUATION TIMES

24 hours or 10 hours (3B). By a logical chain of reasoning, which is not reproduced in detail here, it can be shown that (i) if most core-melt sequences for a particular reactor are small (i.e. noble gases dominate the off-site doses); (ii) the warning times and/or the durations of release are long; and (iii) there is a threshold in time (say 10 hours) beyond which ad hoc emergency response is adequate, then the requirement that "most" core-melt sequences should not exceed the PAGs at the boundary of the EPZ is no longer the most restrictive. Thus even though the importance of the PAGs in the underlying philosophy would not be diminished, the PAGs would no longer drive the radius of the EPZ as seems to be the case at present.

V. NOTIFICATION TIME REQUIREMENTS

In the discussion above, it appears to be possible to make some constructive suggestions relating to the impact of the PAGs on the radius of the EPZ by focusing not only on the magnitude of the new source terms, but also on the new timing characteristics. As shown below, this focus on timing may also have a positive impact on the notification time requirements.

The notification time requirement stems originally from NUREG-0396 and NUREG-0654 and is clarified in FEMA 43 (FEMA, 1983). The alert and notification system should be capable of notifying the public within the plume exposure EPZ within 15 minutes of a decision having been taken to implement a protective action.

The combination of longer warning times, longer durations of release and generally reduced release magnitudes have significant implications regarding the process of making a decision about the most suitable protective actions and the subsequent need to notify the public within 15 minutes. The discussions leading to a decision will, for most sequences, be made without the pressure that arises from the expectation that a matter of minutes can make an important difference to the health and safety of the public. For the same reason, it is not likely to be crucial to be able to notify the public within 15 minutes. Thus, the IDCOR results from the Technical Summary Report illustrate a trend in the timings associated with the results of new source term research which, if confirmed, suggest that, even for severe core melt accidents, the current 15 minute notification requirement may not be needed in order to implement effective emergency actions.

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Coordination Between NRC and FEMA: Emergency Preparedness Issues of Current Interest

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ABSTRACT The framework for cooperation between the Nuclear Regulatory Commission (NRC) and the Federal Emergency Management Agency (FEMA) is formally defined in an April 18, 1985 Memorandum of Understanding. Basically, FEMA coordinates all Federal planning for the offsite impact of radiological emergencies and takes the lead for resolving offsite emergency preparedness issues and NRC does the same for onsite emergency preparedness. This paper describes the efforts of both agencies to coordinate offsite and onsite emergency preparedness issues through the NRC/FEMA Steering Committee on Emergency Preparedness. The current status of several site-specific and generic emergency preparedness issues, such as offsite medical services, nighttime emergency notification, fuel cycle facility emergency planning, consideration of earthquakes in emergency planning and ingestion pathway guidance, will be described.

I. INTRODUCTION

In April 1985 FEMA and NRC entered into a new Memorandum of Understanding (MOU) relating to radiological emergency planning and preparedness. The new MOU preserved the NRC/FEMA Steering Committee on Emergency Preparedness as the focal point for coordination of emergency planning, preparedness, and response between the two agencies.

The Steering Committee consists of an equal number of members to represent each agency (usually 4 members from each agency) with one vote per agency. There is a provision to escalate issues that cannot be resolved by the Steering Committee to NRC and FEMA management, although this has not been necessary to date. Each agency designates a co-chair and they appoint their respective members. The present co-chairs are Ed Jordan, NRC and Richard Krimm, FEMA. The Steering Committee maintains a record of each meeting. A meeting cannot be held without at least the co-chairs or two assigned members for each agency.

The NRC members have the lead responsibility for licensee planning and preparedness and the FEMA members have the lead responsibility for offsite planning and preparedness. The Steering Committee assures coordination of plans and preparedness evaluations and revises acceptance criteria for licensee, State, and local radiological emergency planning and preparedness as necessary. This entails extensive coordination of site-specific issues. In addition, the Steering Committee is the principal forum for coordination and resolution of generic emergency planning and preparedness issues.

As a practical matter, the Steering Committee does not take votes. The Steering Committee operates by consensus. The reason this works is that each agency has unique technical expertise in its assigned area of responsibility, i.e., NRC onsite and FEMA offsite. The following five examples of generic issues coordinated by the Steering Committee all concern offsite matters. Where these examples use the terms "staff" or "NRC staff" for convenience, you should read "NRC staff and FEMA."

II. OFFSITE MEDICAL SERVICES

In 1980 the Commission promulgated a regulation, 10 CFR 50.47(b)(12), to require that "arrangements are made for medical services for contaminated injured individuals." In 1982 the Atomic Safety and Licensing Board (ASLB) for San Onofre found that the term "contaminated injured" encompassed "offsite exposed" persons. This was overturned by the Atomic Safety and Licensing Appeal Panel (ASLAP), which found that "contaminated injured" meant an onsite or emergency worker who was traumatically injured and also contaminated. At that time the NRC Staff supported the ASLAP position stating that the term "contaminated injured" did not encompass "offsite exposed" persons and that *ad hoc* arrangements facilitated by a list of medical facilities was sufficient for offsite exposed persons. In a 1983 policy statement (CLI-83-10), the Commission stated that planning standard (b)(12) extended to members of the general public who may be exposed to dangerous levels of radiation and that a list of medical facilities capable of providing appropriate treatment was sufficient to meet the standard. The U.S. Court of Appeals in 1985, after decision reviewing CLI-83-10, GUARD v. NRC, held that the Commission did not reasonably interpret its own planning standard when it said that a list of existing treatment facilities was sufficient in terms of constituting "arrangements" within the meaning of the regulation. The Commission asked for the staff's views on whether the Commission should: (1) reinterpret the phase "contaminated injured individuals" to eliminate exposed individuals from the scope of planning standard (b)(12) or (2) determine what additional arrangements are necessary for exposed individuals or (3) amend planning standard (b)(12) to make a list of medical facilities the sole planning requirement for exposed individuals.

The staff favored option 2 and proposed some additional arrangements for individuals exposed offsite. The Commission approved the staff recommendation in June 1986, and in July 1986 the Commission adopted a policy statement that planning standard (b)(12) requires pre-accident arrangements for medical services--beyond a list of treatment facilities--for individuals who might be severely exposed to dangerous levels of offsite radiation following an accident at a nuclear power plant. In that statement the Commission gave the staff some general guidance that satisfactory arrangements should include: (1) a list of local or regional medical facilities or other unique characteristics; (2) a good faith reasonable effort by licensees or local or State governments to facilitate or obtain written agreements with the listed medical facilities and transportation providers; (3) provision for making available necessary training for emergency response personnel to identify, transport, and provide emergency first aid to severely exposed individuals; and (4) a good faith reasonable effort by licensees or State or local governments to see that appropriate drills and exercises are conducted to include simulated severely exposed individuals. The NRC and FEMA are preparing guidance to licensees and State and local governments on how to implement the new Commission policy.

III. NIGHTTIME EMERGENCY NOTIFICATION

The Shearon Harris ASLB wrote the Commission in November 1985 regarding the Board's concerns about the possible generic implications of certain portions of the evidence in the Shearon Harris record. They were concerned about the adequacy of nighttime emergency notification of residents in the plume exposure emergency planning zones (EPZs) surrounding nuclear power plants. The siren systems accepted under FEMA criteria were based on daytime conditions and the Board felt that substantial numbers of EPZ residents would not be aroused from sleep should notification be necessary between midnight and 6:00 a.m., particularly if bedroom windows were closed. Specifically, the Board believed there was evidence that in EPZs where sirens were the primary means of notification, such notification would not be "essentially complete" within 15 minutes under some typical (summer) nighttime conditions. At the Commission's request, the staff and FEMA responded to the Board's concerns in December 1985 and again in February 1986. The responses stated that there are no generic safety problems involving nighttime emergency notification and that the FEMA study conducted for the Shearon Harris case confirms NRC/FEMA their judgment that siren systems designed and evaluated in accordance with NUREG-0654/FEMA-REP-1, Rev. 1, and FEMA-43 meet the NRC requirements for daytime and nighttime alerting. Specifically, FEMA found the the Shearon Harris siren system can be expected to arouse and alert approximately 90% of the EPZ residents during the hours from 2 a.m. to 6 a.m., the "worst" summer nighttime hours. This conclusion was based on three major considerations: (1) the effect of nighttime activities, so that some percentage of people are awake, (2) the effect of intrafamily networking, and (3) the effect of social, interfamily networking.

The ASLB issued its decision in the Shearon Harris case in April 1986 and found that the alerting system, where sirens are to be supplemented by a tone-alert radio in the bedroom of each residence in the first 5 miles of the 10-mile EPZ, meets the applicable requirements under summer nighttime conditions. In a May 16, 1986 letter to the Commission, the Board stated that it would not accept a 90% alerting level (Shearon Harris) as meeting the Commission's requirement of "essentially 100%" alerting in the first 5 miles of an EPZ (NUREG-0654). Rather, the Board held that "essentially 100%" meant a notification system capable of alerting greater than 95% of the EPZ residents in the first 5 miles and that this required the addition of tone-alert radios in the Shearon Harris case. In addition to raising this "greater than 95% alerting within 5 miles" as a generic issue, the Board raised winter nighttime conditions, with its greater percentage of windows closed, as a more limiting case than the summer nighttime case addressed at Shearon Harris. To illustrate its point the Board gave data from a 1982 staff analysis titled "Evaluation of the Prompt Alerting Systems at Four Nuclear Power Stations," NUREG/CR-2655, PNL-4426, which projected alerting rates under nighttime conditions. Among other data cited, was the study prediction that 53% of the residents at Indian Point would be alerted during a winter night with snow cover.

Two members of the Board, who had heard the Indian Point Special Proceeding, wrote the Commission in June 1986 that, had they known about this alerting rate estimate, they would have recommended that the Commission require a backup system for prompt alerting, such as tone-alert radios. The utility responded to the Commission in July 1986 stating among other things, that NUREG-2655 was based an 88-siren system as opposed to the present 151-siren system. The two Board members acknowledged in July 1986 that this assuaged their major concern about the adequacy of the siren system at Indian Point, but that they remained concerned about the adequacy of siren systems on snowy winter nights. These two Board members also were concerned because the staff had never notified them about NUREG/CR-2655.

In August 1986 the Chairman requested the staff to provide by the end of September 1986 a point-by-point response to the generic safety concerns in the May 1986 letter from the Shearon Harris Licensing Board as well as recommendations on whether Commission rules or guidance documents need revision or clarification. Further, the Chairman requested that the staff provide a response to the June and July 1986 letters from members of the Indian Point Licensing Board which raised a related concern regarding the adequacy of alert systems during stormy winter nights. The staff and FEMA are preparing a response to the Chairman's request.

IV. FUEL CYCLE FACILITY EMERGENCY PLANNING

The staff issued orders in 1981 to certain fuel cycle and other materials licensees to submit radiological contingency plans. Also in that year, the Commission published an Advance Notice of Proposed Rulemaking (46 FR 29712) proposing to codify the radiological emergency requirements set forth in the orders. The staff reviewed the 18 letters responding to the advance notice and in 1986 submitted to the Commission proposed amendments that would formally require emergency plans for certain fuel cycle and other radioactive material licensees. These proposed regulations would require about 30 licensees to have emergency plans and would require about 30 other licensees to either (1) submit a plan, (2) submit an evaluation showing that a significant release is not plausible, or (3) amend their licenses to reduce their possession limits. The staff estimates that few of these latter 30 licensees would submit plans; about half would submit an evaluation and about half would reduce their possession limits The 30 licensees that would need an emergency plan are the same licensees that submitted Radiological Contingency Plans under the orders issued in 1981.

NUREG-1198, "Release of UF_6 From a Ruptured Model 48Y Cylinder at Sequoyah Fuels Corporation Facility; Lessons Learned Report" was published in June 1986. This report contained many recommendations that are currently being evaluated by the staff; these recommendations could amplify, expand, or clarify some of the emergency preparedness requirements listed in the staff proposal to the Commission. Hence, at the Commission's request, the staff submitted in July 1986 proposed changes to that would solicit public comment on certain recommendations in NUREG-1198.

V. CONSIDERATION OF EARTHQUAKES IN EMERGENCY PLANNING

Following the San Onofre and Diablo Canyon proceedings, the Commission ruled in December 1981 that its regulations do not require consideration of the potential complicating effects of earthquakes on emergency planning. The Commission asked the staff to initiate a rulemaking to determine on a generic basis whether the regulations should be changed to require such consideration. In December 1984 the Commission published a proposed rule that ". . .Emergency response plans. . .need not consider the impact on emergency planning of earthquakes which cause or occur proximate in time with an accidental release of radioactive material from a facility...." In August 1985 the staff forwarded to the Commission for approval (SECY-85-283) a proposed final amendment to the regulations that would have required "limited consideration" in emergency planning of the complicating effects of "severe, low-frequency natural phenomena." In April 1986, in San Luis Obipspo Mothers for Peace v. NRC, rehearing en banc, 789F.2d 26 (1986) the U.S. Court of Appeals for the District of Columbia affirmed the Commission is interpretation of its emergency planning rules; i.e., that the rules do not require consideration of the potential complicating effects of earthquakes on emergency planning. Although the Commission disapproved the staff proposed final amendment in June 1986, the matter is not resolved at this time.

VI. INGESTION PATHWAY GUIDANCE

Draft FEMA Guidance Memorandum (GM) IN-1, "The Ingestion Pathway," provides planning considerations that include areas of review and acceptance criteria for the protection of the human food chain, including animal feeds and water, which may become contaminated following a radioactive release from a commercial nuclear power plant. The guidance is addressed to State and local government emergency planners within a 50-mile radius of nuclear power plants, i.e., the ingestion exposure emergency planning zone. The implementation would be one year from the date of publication of the final (GM) IN-1 as part of the annual plan update. Principal features of the guidance are (1) public education and information (2) protective response on the basis of the Food and Drug Administration response level for dealing with accidental radioactive contamination of human food and animal feed (3) and exercises and drills. Because the guidance is still in draft, it is not appropriate to provide more detail. FEMA expects to issue IN-1 in final form later this year.

VII. CONCLUSION

The NRC/FEMA Steering Committee is an effective mechanism for coordinating emergency preparedness issues between the two agencies. This paper has described some complex generic issues that have been coordinated between the two agencies. In addition, there are many site-specific issues that are addressed by the Committee.

Interactions Between Multiple Organizations Responding to a Reactor Accident in Switzerland

Martin Baggenstos and Hans-Peter Isaak

ABSTRACT. The poster deals with the problems that emerge from the distribution of responsibilities in the case of a nuclear accident. It is demonstrated how clear and simple procedures and good co-ordination between the on-site and off-site emergency organizations are necessary for the smooth handling of such a situation. A good communications network is particularly essential. The special problem of co-ordination between different organizations performing measurements of the contaminated air, ground and food is also discussed. From the experience gained during the Chernobyl accident, it has been established that only those organizations, which are correctly trained and have a perspective over the complete accident situation, can perform valuable work. At the beginning of the accident for example, there was no organization responsible for taking samples covering the whole of Switzerland. A first quick assessment of the radiological situation could therefore only be made on the basis of measurements performed in the direct vicinity of the laboratories or from single samples which had to be transported over large distances.

I. DISTRIBUTION OF RESPONSIBILITIES BETWEEN THE PLANT LICENSEE AND AUTHORITIES

A. General Principles of Responsibility

In the case of an accident in a nuclear power plant, the emergency organization of the plant and several off-site authorities are involved. In determining the responsibilities for the protection of the population, attention must be payed to the following principles: (1) Who has the best means for making certain decisions? (2) How can the decision reach the population in the fastest way possible, over a minimum number of authorities? The application of these principles can lead to a case, where in an accident situation, certain official authorities are bypassed. Experience gained from exercises has shown that this is generally no problem, if such an arrangement is clearly settled in advance.

B. Distribution of Responsibilities

During the course of an accident, there are many technical and radiological decisions to be made. There is a connection between the technical and radiological measures to be taken in the plant and the radiological consequences in the environment. Fig.1 shows how the on-site and off-site responsibilities are distributed among the on-site staff, the off-site emergency management and the Nuclear Safety Inspectorate, which acts as a focal point for the other organizations. The responsibilities can be summarized as follows:

Responsibility On-site. The general responsibility for the technical and radiological decisions within the plant lie with the emergency staff of the plant.

Responsibility Off-site. The decisions for protective measures in the environment are made by the off-site emergency management (and not by the plant licensee), because the responsibility for the protection of the population rests in the hands of the off-site authorities.

Nuclear Safety Inspectorate. The Safety Inspectorate is the only authority, which has expert knowledge of both the technical and radiological behavior of the plant, as well as knowledge of the requirements for the protection of the population. The inspectorate is therefore able to supervise the on-site actions taken by the plant staff and to give expert advice to the off-site management of the accident.

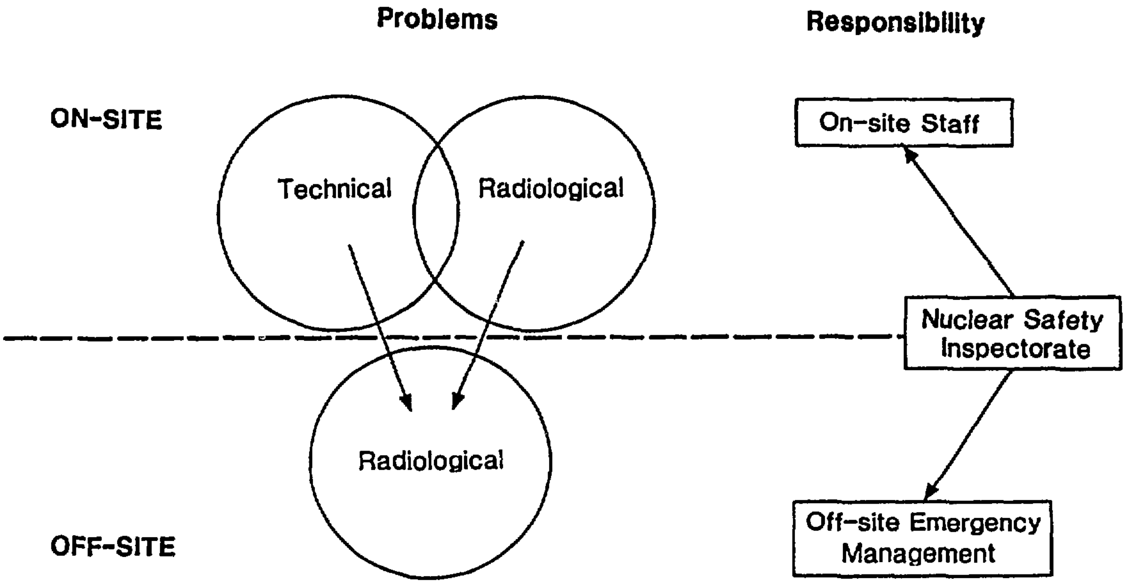


Fig.1 Principle Distribution of On-site and Off-site Responsibility

1. **Detection of the Accident.** Many incidents occur in a nuclear power plant, that do not necessarily require protective actions in the environment. It is however possible, that such accidents may occur which make protective measures indispensable. For several reasons, the distinction between the two cases is not always easy. (1) A severe nuclear accident may develop from a simple failure. (2) The judgement of what is relevant for the protection of the population depends somewhat on the point of view of the on-site and off-site organizations. (3) In certain cases the population may take independant protective actions based on their own assessment of the situation (e.g. in the case of an evident fire in the plant).

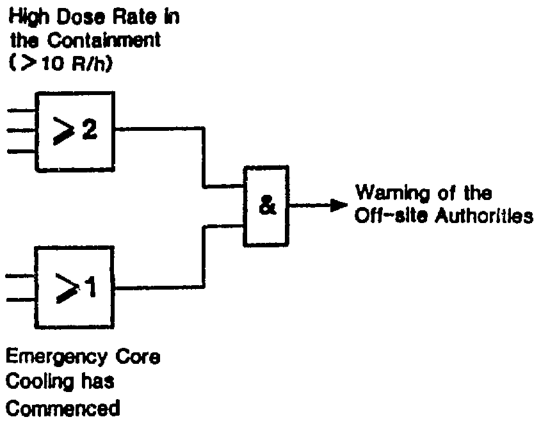
The main problem is how to set the trigger level for alerting the different emergency organizations. Different solutions are conceivable. (1) The off-site emergency management is alerted by the plant staff very early in the course of an accident. The advantage of this solution is that the authorities have lots of time to build up their organization and are ready should they be needed. The disadvantage is that in the case of an over-cautious assessment of the on-site situation, false protective actions could be taken. The damage may be larger than the benefit of the action. Furthermore the off-site emergency management does not have such detailled technical knowledge of the plant as the plant staff. (2) The off-site emergency management is not informed before serious radiological consequences for the population become probable. The advantage of this solution is that the authorities know that protective measures must be implemented at once. The disadvantage is that the authorities may have to react without much time available. The following solution has been chosen in Switzerland:

Detection of the Accident. The detection of an accident and the correct assessment of the on-site radiological situation lies within the responsibilities of the plant staff.

Notification of the Safety Authorities. The Nuclear Safety Inspectorate will always be notified in the following cases: if the reactor has been shut down (scram), if the situation is optically perceivable by the population (cooling tower no longer operates or fire in the plant) or if the accident has technical or radiological implications within the plant.

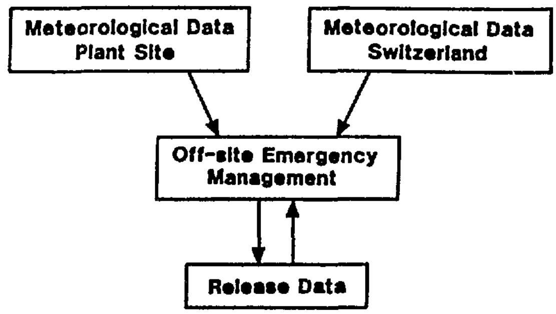
Notification of the Off-site Emergency Management. The off-site emergency management will only be informed, when the accident threatens to develop in such a way that protective measures for the population become inevitable.

The criteria for alerting the off-site authorities are chosen in such a way that the plant operator in the main control room recognizes (with the help of a clearly visible and audible signal) that an incident with damage to the fuel cladding has occured. In order to prevent false alarms, the criteria for alerting the off-site authorities are two-fold, namely high dose rate in the containment and initiation of emergency core cooling. The combination of these two signals is transmitted automatically to the main control room and is held fast optically and acoustically.



2. Assessment of the Radiological Danger to the Population. In order to assess the potential radiological danger to the population, specific data concerning the source term (time scale and amount of radioactivity released) and meteorology must be available. In principle the responsibility for the correct assessment of the off-site radiological situation could lie with either the on-site or the off-site organizations. (1) The emergency staff of the plant could assess the off-site situation and communicate this to the off-site authorities. The plant staff would need information on the release, the meteorology of the environment and knowledge of dispersion calculations in the atmosphere. The advantage of this solution is that the plant staff would be in the best position to judge the source term and the meteorological situation at the plant site. The disadvantage is that the plant staff would have no first-hand information on the meteorological situation for the whole of Switzerland. (2) The off-site authorities could assess the off-site situation. In this case the advantage is that the off-site authorities would have expert knowledge of dispersion calculations and would be in a better position to judge the need for protective measures. The disadvantage is that release and meteorological data would have to be communicated from the plant and from other meteorological stations to the off-site authorities. In an accident situation this would function only if a good communications network can be guaranteed. In Switzerland the following solution has been chosen:

The off-site National Emergency Operations Center is located at the Swiss Meteorological Institute in Zürich. This institute has a complete knowledge of the meteorological situation in Switzerland. Furthermore specialists are always available to correctly interpret the meteorological data. The meteorological data from the plant site and other meteorological stations are automatically transmitted to the Emergency Operations Center on a routine basis. Concerning the release data (time scale and behaviour), these would have to be collected from the plant as needed with the help of a telephone or telefax transmitter/receiver. No automatic transmission of the release data is forseen at the moment.



3. Recommendation and Realization of Protective Measures. In principle, protective measures for the population may be realized on the basis of either a prognostic dose, which gives an estimation before or during the release of radioactive materials, or an expected dose, which is calculated after the release has occured. If an acute dose is expected within a short time, then protective measures should be established from the prognosive dose. On the other hand long-term protective measures should be determined from the expected dose, based on actual measurements of the radiological situation. Nowadays many sophisticated computer programs exist, which can calculate the prognostic dose on-line. The question is whether or not such an on-line computer program or a simple Gauss-Model for hand calculation is best suited for emergency applications. The on-line program can process the actual weather data and many other parameters and calculate different dose pathways. With a simple calculation one is restricted to a few typical weather situations and parameters.

The decision which type of program is to be used, also depends on the user, namely the organization which is responsible for the recommendation or realization of protective measures. The responsibilities for recommendation and realization of protective measures for the

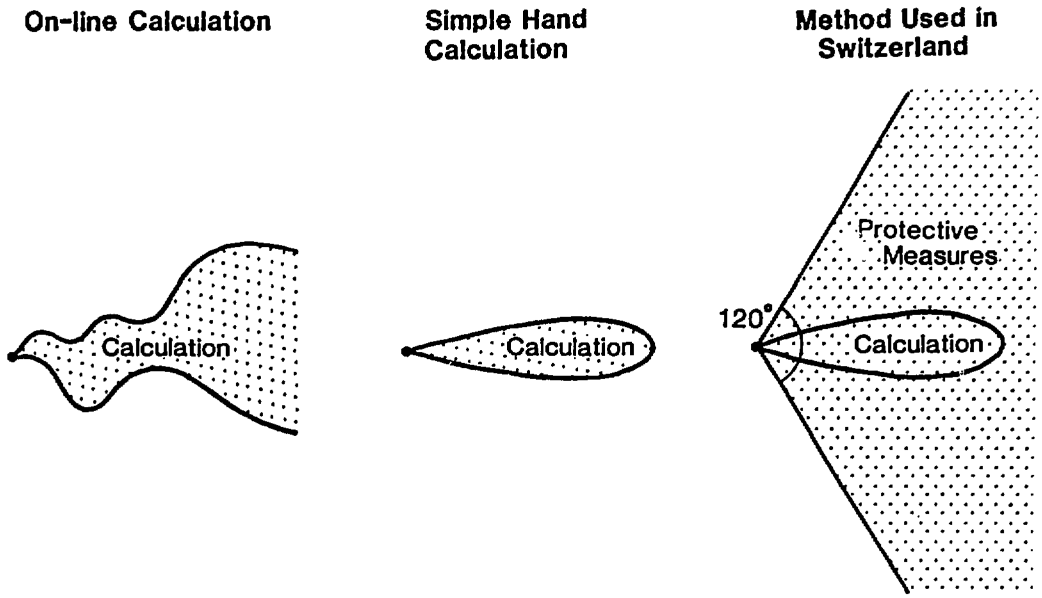
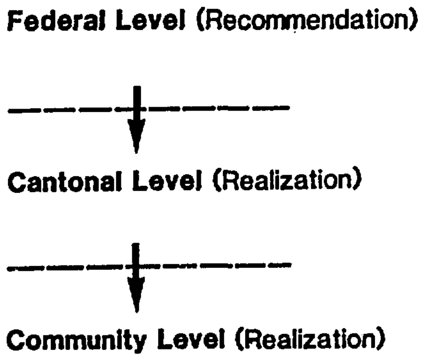


Fig.2 Examples of Possible Prognostic Dispersion Calculations

population are distributed within different federal, cantonal and community levels. Radiation protection experts are generally only available on the federal level. However, the cantonal and community levels must understand, what the federal level is recommending.

Responsibilities for the Recommendation and Realization of Protective Measures



Dose calculations performed with a sophisticated on-line program can correct for many special effects such as plume meander. This type of calculation certainly has many advantages. The problem is that the authorities responsible for the execution of protective measures may be tempted to overvalue the accuracy of the calculations and then implement protective measures only within the calculated zone. Dose calculations performed using a simple Gauss-Model do not allow for correction of effects such as plume meander. On the other hand the character of the estimation is more visible and the executive authorities are not tempted to ascribe to much importance to the accuracy of the calculation. The solution which has been chosen in Switzerland is shown schematically in Fig.2, along with the two basic methods of dose calculation. In Switzerland the basic calculation is performed with a simple Gauss-model. Protective measures are, however, implemented within a much larger area covering an angle of at least 120 degrees.

4. **Lessons Learned from the Chernobyl Accident.** Although in the case of the Chernobyl accident, Switzerland only experienced the ingestion phase of the accident, several important lessons could be learned concerning the distribution of responsible emergency organizations. These are:
Emergency Reference Levels. The definition of Emergency Reference Levels should be undertaken by the same authority, which in the case of an accident must implement the reference levels. All authorities which propose or execute protective measures should be involved in the planning phase.
Recommendations. Official recommendations must be announced to the population by recognized authorities, not by so-called experts.
Information Exchange. Information exchange must be co-ordinated in advance between the authorities that propose and those that execute protective measures. This includes information of the public. A good communications network is essential.

II. Co-ordination of Radiological Surveillance in the Environment after a Nuclear Accident

A. General Principles of Co-ordination

In the case of an accident in a nuclear power plant it is customary, that the normal routine radiation surveillance personnel is supported by additional civil and military survey teams. For a good co-ordination between the many survey teams performing measurements, the following principles should be adhered to. (1) Survey teams measuring the same quantity should be equipped with the same measuring instruments. (2) Survey teams should use the same supporting documents for sampling, documentation and transmission of data. (3) The measurements performed should be reproducible in the sense that the places of measurement are easily found and accessible to motor vehicles (places should be documented with a photograph). (4) If possible, civil resources should be limited to certain more costly and time consuming detailled measurements. (5) Military resources should be used for a rapid survey of the fallout situation. (6) The parameters to be measured should in all cases be adapted to the degree of actual danger (for instance the amount of strontium in milk is not important in the first week after an accident).

B. Utilization of Measuring Organizations in the Different Phases of an Accident

1. Priority of Measurements.

In the case of an emission of radioactive particles one can differentiate two distinct phases of the accident, namely an acute phase and a medium/long-term phase. Fig.3 shows how in the first phase of the accident the radiological danger to the population is great for a relatively short period of time. The second phase represents a lesser radiological danger but over a much longer period of time.

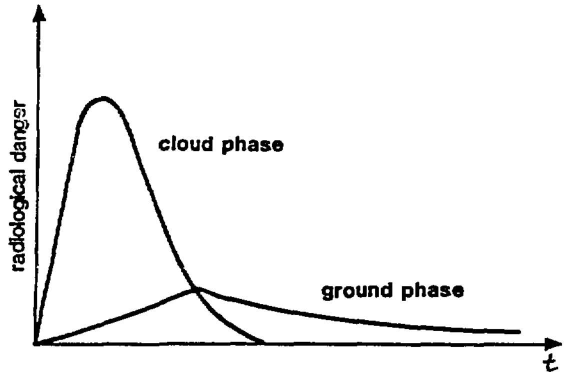


Fig.3 Potential Radiological Danger to the Population

Table 1 shows the different organisations which perform measurements during the various phases of the accident. Basically, permanent civil systems, which are always operational, are employed in the cloud phase of the accident. Measurement of the ground contamination in the medium-term ground phase of the accident are performed by organisations, which must first be mobilized. In the long-term ingestion phase of the accident, all available laboratories are used.

Time	Cloud Phase	Ground Phase	Measuring Organization
↓	EXTERNAL RADIATION FROM CLOUD		Fixed measuring systems, helicopter 1), survey teams 1)
	INHALATION OF CLOUD		survey teams, laboratory 1)
		EXTERNAL RADIATION FROM GROUND	Fixed measuring systems, helicopter 2), survey teams 2)
		INGESTION OF CONTAMINATED FOOD	Laboratories 1) and 2)

1) Permanent civil systems which are always operational

2) Civil or military systems that are not routinely operational, but must first be mobilized to install their laboratories.

Table 1 : Potential Radiological Danger and Measuring Organization

2. Utilization of Different Measuring Organizations. There are many different organisations involved in an accident situation, each of which has a well designated responsibility.

Survey Teams of the Nuclear Power Plant Affected. The on-site and off-site responsibilities are clearly distinguished in Switzerland. The survey teams of the plant are exclusively responsible for the safety of the plant personnel and for the measurement of the radiological situation in the plant. The plant survey teams are rapidly available in an accident situation, but are needed most urgently within the plant. The plant is therefore not responsible for performing measurements outside of the plant site.

Civil Measuring Systems which are Permanently Operational. Switzerland is covered by a permanent network of dose rate measuring units (density is about 1 probe per 1000 km2). This network quickly gives a first assessment of the external radiological situation. Moreover there are several research institutions, which are in possession of permanent laboratories for measurements of radioactivity. Some of these institutions possess motor vehicles equipped with measuring instruments. In the case of an accident, these laboratories spontaneously take samples of the radioactivity in the air, on the ground and in food. Helicopters equipped with measuring systems can also be rapidly engaged. The helicopters are flown by military personnel. A civil survey team performs the measurements and takes the samples. All these systems are organized in such a way that they are ready for action within about 1 to 2 hours and can measure and transmit data independently.

Civil and Military Measuring Systems Which Must First be Installed. After the radioactive cloud has passed, additional means are needed in order to measure the radiological fallout situation within a reasonable time of about 1 to 2 days. A large measuring capacity is required in order to be able to produce a detailled chart. The measurements must be performed for the whole of Switzerland, because any background measurement in an area far away from the site of the accident is also important when determining protective measures. With the requirement that one measurement should be performed per 100 km2, this means for the military survey teams and helicopters that about 400 points must be measured. For larger distances, the density of measured points can be reduced somewhat if the weather situation can be judged reliably. For a complete determination of the situation concerning contaminated food, over 30 laboratories covering the whole of Switzerland may be mobilized. These laboratories must be able to measure a few hundred samples each day, and transmit the data to the Emergency Operations Center.

3. Lessons Learned from Chernobyl. The Chernobyl accident has shown that certain problems result in the co-ordination and data transfer between the different measuring organizations. Within the scope of the Chernobyl accident the following measuring systems were in operation in Switzerland: fixed measuring systems, civil ground survey teams, survey helicopters, and civil and military laboratories. The Chernobyl accident showed, that only a well organized emergency organization can in the case of an accident make correct decisions and implement them. All authorities concerned must be involved in the planning of the emergency organization. This is especially true for the local communities, which must implement the decisions taken. Specifically the following problems were encountered:

Fixed Measuring Systems. Of the 50 planned probes, only 11 were in operation at the time of the accident. A rough estimation of the external radiological situation was therefore missing in the beginning.

Civil Survey Teams. The mobile ground survey teams have a fixed charge to measure within the vicinity of the nuclear power plants. Because the taking of samples was not organized at the beginning, the mobile teams also had to take samples from far away places and bring them to the laboratories. These mobile units were not efficiently used.

Helicopter. Helicopters were the most efficient means of taking samples during the first few days. The co-ordination between civil and military personnel presented no problems. The results were immediately available.

Civil and Military Laboratories. The permanently available laboratories performed quickly and efficiently. After a few days they however became overcharged, because of lack of co-ordination between the organizations taking samples. A uncontrolled yield of samples occured. The subsequently mobilized laboratories were mobilized too late. A few days were needed to manage the large amount of data coming in.

Acknowledgment. The authors would like to thank Laszlo Babocsay for his help in preparing the figures and for his critical reading of the manuscript.

Utility Perspective on Emergency Preparedness

George J. Giangi

ABSTRACT

This paper discusses some of the major lessons learned from the TMI-2 accident with respect to emergency response and preparedness. A description of the GPU Nuclear Corporation emergency preparedness program is also provided to illustrate the significant differences in the program before and after the accident. While major lessons have been adopted by both government and industry there continues to be emergency preparedness related regulations and concerns that promote the misperception that nuclear power plants are unsafe. In addition, annual exercises send a message to the public and news media that most, if not all, nuclear plant emergencies will result in evacuation or sheltering of the surrounding population.

Prior to the March 28, 1979 accident at the Three Mile Island station, the typical emergency preparedness program at utilities consisted of a simplified emergency plan with several implementing procedures. The responsibility for maintaining the program rested with one individual with resources that were shared from other departments. Although drills and exercises were conducted on an annual basis, the frequency, level of participation and scenario scope were generally far less than current programs.

The NRC regulations and guidance governing emergency preparedness at that time were minimal in comparison to the volumes currently in effect. (1) In 1970, 10CFR50.34 was the first Emergency Planning rule. In 1975, Regulatory Guide 1.101, which was approximately 4 pages, provided guidance for onsite Emergency Plans and offsite emergency planning (for the LPZ area only). 1980 marked the start of significant Emergency Planning rules and guidelines. The rules required onsite and offsite Emergency Plans (for the two EPZ's) with required exercises as a condition for maintaining the plant license.

(1) Atomic Industrial Forum "Background Info" June, 1986

During the TMI-2 accident several areas were of significant concern. The communications system in place during the accident did not adequately allow for proper information flow among the utility and Government emergency facilities. Over a matter of days, the telephone company installed additional phone lines to enhance the existing system. Outside lines were not always available due to the stress placed on the telephone system from public use.

The ability to properly assess the radiological impact on the plant and the environment during the accident was also of significant concern. Certain areas of the plant had very high (approximately 100 Rem/hour) radiation levels due to noble gases that made it difficult to monitor with hand held radiation detection instruments. The dose projection process consisted of manual calculations using isopleths which was cumbersome and time consuming.

The declaration of the emergency and notification to offsite agencies were delayed predominantly due to the lack of understanding by the operators as to why the plant was behaving the way it was. The current Emergency Plan, consistent with regulatory guidance requires the operator to promptly declare the emergency and initiate notifications whenever certain values "emergency action levels" are reached or exceeded . This action initiates the implementation of the Emergency Plan simultaneous with plant mitigation. In addition, these Emergency Action levels have been included into the Emergency Operating Procedures to assure this process.

The planning and response process, as well as the responsibility for maintaining the emergency preparedness program, have significantly changed from the time of the accident. The TMI Emergency Preparedness program today has seven full time staff members. Their disciplines include operations, maintenance and radiological controls. Over a dozen drills are conducted annually which include unannounced shift drills to test the plant crew on backshifts, full activation quarterly drills and an annual exercise. Scenarios are completely developed in-house and employ the use of computers and simulators for more accurate and timely data production. Drills and exercises are often based on real industry events and are conducted using audio and visual aids, moulage kits, radiation sources and actual meteorology. Provisions are in place for ensuring the viability of the program on a day-to-day basis. Examples of this include periodic inventory of emergency facilities, monthly testing of emergency communications systems and updating letters of agreement.

Attachment 1 provides specific information pertaining to the GPUNC Emergency Preparedness Department including budget and annual exercise costs.

Additional examples of significant improvements which have been made to the emergency preparedness program since the accident include:

- Major plant modifications to improve reliability and assessment capability
- Enhanced operator training program
- Expansion of in-plant and effluent radiation monitors as well as portable radiation detection instruments
- Dedicated Emergency Response Facilities
- Dedicated Emergency Communications System with redundant features
- A sophisticated computerized dose projection model that is terrain specific and tracks the plume in a reasonably predictable manner
- A well defined emergency classification system that is integrated with plant emergency procedures.
- Installation of a Prompt Public Notification System
- Improved Public and News Media Education Program
- Closer coordination and interface with Federal, State and Local Government officials
- Utility supported radiological training for offsite emergency workers.

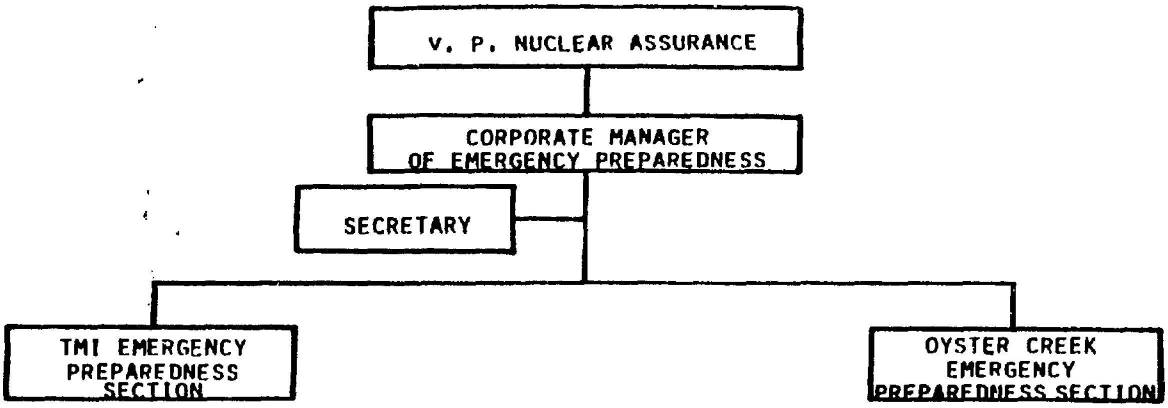
While a significant number of lessons learned have been addressed, it is still necessary to review the consequences of the TMI-2 accident in order to put it in perspective. In some areas this has not been done. Perhaps one of the most important examples is simply that the amount of Iodine that is released to the environment during a core melt scenario in a Light Water Reactor is significantly less than previously postulated. In fact, approximately 20 curies of Iodine escaped to the environment while it was thought that 20 million curies would escape. This information, along with recently published source term information, suggests to some that relaxation of Emergency Planning rules is in order. For example, this may be in the form of a reduced Emergency Planning Zone.

The nuclear industry readily accepted and corrected one of the lessons learned from the TMI-2 accident. Specifically, more education on nuclear power was needed for the public and news media. This has been implemented via annual emergency public information brochure, annual news media training, aggressive nuclear plant speakers programs and plant tours. However, the required annual exercises which receive widespread media attention continue to promote the misperception that all nuclear plant emergencies will lead to a protective action of evacuation or sheltering due to the harmful levels of radiation. FEMA and NRC should allow the states and utilities to conduct more realistic exercises which test the implementation of the plans without catastrophic plant failures and lethal doses. In addition, FEMA Guidance Memoranda continue to be developed which impose new requirements upon the state and local governments and ultimately on the utilities. (eg; increasing requirements for the ingestion pathway 50 mile EPZ).

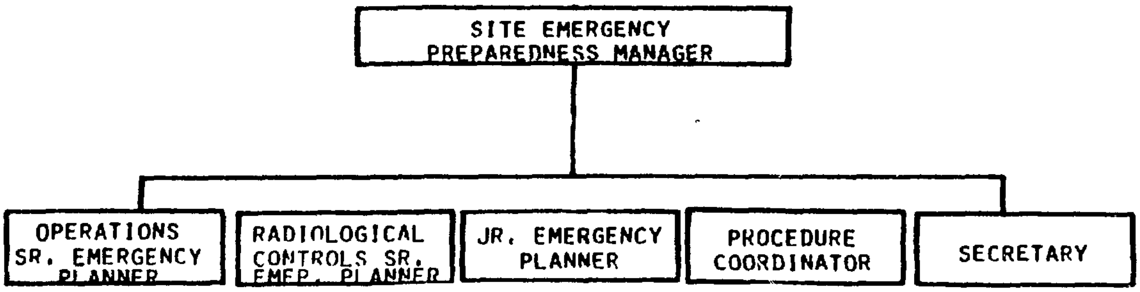
Except for the nuclear industry no other industry is required to have an outdoor warning system. Although recent evidence suggests that the probability for the occurrence of a nuclear plant accident posing harm to residents is extremely low compared to other natural or manmade hazards it is disappointing to note that recent events challenge the adequacy of siren systems which meet or exceed NRC and FEMA requirements to alert residents that may be indoors under night time conditions.

In summary, State and Local Governments and Utilities are in a better position to respond to a nuclear plant accident today than was the case on March 28, 1979. This is a dynamic process which continually requires updating. If nuclear power is to remain a viable energy source, regulations, guidance and licensing proceedings should be re-evaluated and modified to reflect the latest source term studies and annual exercises should be allowed to be more realistic.

GPU NUCLEAR EMERGENCY PREPAREDNESS DEPARTMENT



GPU NUCLEAR EMERGENCY PREPAREDNESS SECTION STAFF



TYPICAL CREDENTIALS OF EP STAFF MEMBERS

SITE EMERGENCY PREPAREDNESS MANAGER	BS DEGREE (OR ADVANCED DEGREE) WITH 10 OR MORE YEARS OF EXPERIENCE IN APPLIED HEALTH PHYSICS/EMERGENCY PLANNING AND MANAGEMENT
OPERATIONS SR. EMERGENCY PLANNER	PREVIOUS SRO CERTIFICATION WITH 5 - 10 YEARS OF EXPERIENCE MAY HAVE AN AS OR BS DEGREE
RADIOLOGICAL CONTROLS SR. EMERGENCY PLANNER	BS DEGREE IN SCIENCE OR ENGINEERING WITH 5 - 10 YEARS OF EXPERIENCE IN APPLIED HEALTH PHYSICS/EMERGENCY PLANNING
JUNIOR EMERGENCY PLANNER	AS OR BS DEGREE WITH 0 - 3 YEARS OF EXPERIENCE IN A TECHNICAL RELATED FIELD
PROCEDURE COORDINATOR	AS OR BS DEGREE WITH 0 - 3 YEARS OF EXPERIENCE

SITE EMERGENCY PREPAREDNESS SECTION BUDGET

ITEM	(COST IN THOUSANDS OF $)
STAFF PAYROLL (INCLUDING OVERHEAD)	275
DEPARTMENT SUPPLIES	10
EMERGENCY EQUIPMENT AND FACILITY UPGRADE	100
MAINTENANCE AND REPAIR OF OFFSITE SIREN SYSTEM (APPROXIMATELY 80 SIRENS)	75
OFFSITE TRAINING AND RERP MAINTENANCE	125
OTHER CONTRACTOR SUPPORT (E.G., SPECIAL PROJECTS, SIMULATOR TIME)	50
TOTAL	635K

ANNUAL EXERCISE COSTS

- EMERGENCY PREPAREDNESS DEPT:

  STAFF 5 X 3 MONTHS → 1.25 MAN YEARS

- EMERGENCY RESPONSE PERSONNEL:

  100 SITE PERSONNEL

  25 CORPORATE STAFF

  125 TOTAL PERSONNEL

  PARTICIPATION IN:

  PRACTICE DRILLS
  TABLE TOP EXERCISES
  ANNUAL EXERCISE

  } 5 DAYS

125 PERSONNEL X 5 DAYS (250 DAYS/MAN YR) → 2.5 MAN YEARS

UTILITY PERSONNEL   3.75 MAN YEARS X $50K/MAN YR → $190K
(INCLUDES OVERHEAD)

TOTAL $190K

Improving Emergency Management Through Shared Information Processing—Considerations in Emergency Operations Center Design

Richard E. DeBusk and J. Andrew Walker

ABSTRACT An Emergency Operations Center (EOC) is a shared information processing facility. Although seemingly obvious, many EOCs are designed and operated based on other criteria. The results, measured in terms of response effectiveness, are difficult to determine. A review of some recent disasters reveals a pattern of poor performance for the EOCs involved. These conclusions are tentative because so little research has been done on the design, operation, or evaluation of emergency operations centers.

The EOC is not an onsite response command post but a facility removed from the response where tactical and strategic decisions are made based on information from the response site and elsewhere. The EOC is therefore the central focus of emergency information processing and higher-level decision making. Examining existing EOCs, several common functions emerge. These functions can be described in terms of shared information processing. However, many factors impact the design and operation of EOCs. Politics, budgets, and personal ambition are only a few such factors.

Examining EOC design and operation in terms of shared information processing operationalized in the seven principal functions within the EOC provides a framework for establishing principles of EOC design and operation. In the response to emergencies such as Bhopal or Chernobyl the stakes are high. Applying new techniques and technologies of management systems can improve the probability of success. This research is a beginning step--to understand how EOCs function, to define the system. Predictive or prescriptive analysis must wait until sufficient empirical data is available to complete a descriptive model for EOC operations.

INTRODUCTION

Bhopal, Challenger, Chernobyl: major emergencies are alarmingly frequent. Typically, these emergencies require complex, intergovernmental, multiorganizational responses. In this environment, effective information sharing among the responding organizations is a basic requirement for success. Chernobyl is the most recent example of a major emergency. The response to Chernobyl has been considered inadequate by most authorities and poor information sharing has been identified as a major factor contributing to the poor response. Poor information sharing seems to have occurred between the reactor operations staff and the central Soviet government and between the Soviet government and its neighbors.

Additionally, better information sharing could have improved the response to Bhopal, the Beirut terrorist attack, and the Mexico City earthquake. William Petak (PAR, 1985:12) challenges emergency managers to develop proactive approaches. Increased preparedness, increased cooperation between cognizant organizations, and increased integration of hazard prevention (engineered safety systems) and hazard response (emergency management systems) requires improved information sharing.

Emergency management is a broad, multidisciplinary topic and one long neglected. Research in many areas of emergency management can, we believe, help emergency managers meet the significant challenge posed by major emergencies.

We have chosen to examine one function in emergency management, (centralized command and control in a facility normally called an Emergency Operations Center), in the context of a concept applicable to all functions in emergency management (shared information processing). To do this, research is organized in three parts. Part one reviews the concepts of and requirements for shared information processing. Part two establishes characteristics of EOCs and relates these characteristics to information sharing. Part three, the conclusion, suggests improvements in EOC design built on information sharing concepts and recommends other emergency management improvements.

I. INFORMATION SHARING AND EMERGENCY MANAGEMENT

What is information? What is sharing? Why is sharing information important? We use Blumenthal's definition of data and information: "A datum is an uninterpreted raw statement of fact. Information is data recorded, classified, organized, related, or interpreted within context to convey meaning." (Blumenthal, 1969: 27). Thus a fact such as, "The radiation dose inside the reactor containment of a nuclear facility is 2,500 rem/hr.," is data. When related to standards and interpreted, the information emerges that the radiation dose is too high for human exposure.

Sharing is both borrowing and lending data and information. Effective results from sharing only come about when we don't hoard or protect that which our neighbor needs. However, too much information is just as bad. Filtering may be necessary to extract the most relevant information and not overload emergency managers. Further, sharing may be counterproductive if we don't return the borrowed item in original condition; otherwise it may not be usable again.

A. Why Is Shared Information Processing In Emergency Management Important?

The manager's need for information is the forcing function in this age of information. Increasing levels of technological sophistication make more information available to more organizations and people. Organizations specialize in order to cope with the ever-increasing complexities of technology. Specialization leads to interdependence and the increasing need to share information.

We are also realizing now, more clearly than before, that our resources are finite. Management Systems Laboratories' hypothesis is that organizations previously acted to reduce information sharing by operating with slack resources and operating in self-contained tasks. Organizational interdependencies and a broader base of those participating in decision making combined with resource constraints are now forcing organizations to redesign their operations based on shared information processing.

The manager's need for information is also the forcing function in emergency management. Industry has produced new hazards to add to natural disasters. The public has access to more information about these hazards and demands that the government provide protection. If government wishes to maintain its credibility and if public administrators and elected officials wish to maintain their offices, they must respond to this challenge.

Previously, when the lack of global, instantaneous communications made the world a larger place, informal policies in emergency management were possible. Information about catastrophic emergencies hundreds or thousands of miles away didn't seem threatening. Today, the world has become very small. The media intensively covers a wide range of potential hazards and the public demands protection from risks that seem very near. Comprehensive, integrated capabilities are required to respond to the demand for the better management of emergencies. Interdependence, broadened decision making, and constrained resources are, therefore, forcing information sharing in emergency management.

B. Emergency Response And Emergency Management Are Different.

Emergency response is a technical, hands-on activity. The operation involves physical items such as bulldozers, aircraft, radiological monitoring equipment, and medical equipment. The operation of emergency response cannot be seen in isolation. Successful emergency response depends on successful emergency management.

Emergency management includes the activities that support and direct the response. The operation of emergency management involves data, information, information processing, and decision mechanisms. The importance of emergency management to emergency response was illustrated in Mexico City. The data providing the number of collapsed buildings and the number of trapped people was not received, sorted, manipulated, and portrayed to Mexican government officials until forty-eight hours after the earthquake. By that time most of the trapped people could not be helped. Resources from within Mexico could not be redirected and resources from outside Mexico could not be requested until the government knew the scope of the problem. Better information sharing among the agencies of the Mexican government may have saved thousands of lives.

C. Effective Decisions Require Good Information.

Complex technological or natural disasters require a cooperative, integrated response from many response groups at all levels of government. Data must be quickly and accurately gathered, stored, manipulated, portrayed, and communicated as information to many groups. Information must be shared.

The "place" where data becomes information is a critical component in the effective management of emergencies. This place is the Emergency Operations Center. Raw data is not usually forwarded up the reporting chain. Even after the EOC extracts information from emergency incident data, the information is both funneled and filtered before being forwarded.

Funneling involves summarizing, collecting, interpreting, and disseminating dissimilar but related data. This data manipulation requires great technical expertise and high ethical standards (to prevent hoarding data to make one group, usually one's own, look good) to avoid misinterpretation and distortion. Filtering is the process of reflecting trends or patterns in information. Funneling and filtering extracts the most relevant information.

This discussion shows that the quality of the information available to most government officials and to the public is dependent on the operation of the EOC as an information processing and information sharing facility. The EOC links the complex management chain together. Effective information processing and sharing in the EOC is a necessary condition for effective emergency management.

II. INFORMATION SHARING IN THE EMERGENCY OPERATIONS CENTER

Many individual characteristics are important in the analysis of emergency operations centers. We found it useful to combine many of the individual characteristics into broad categories. This method permits comparison of EOC characteristics. However, prior to our discussion of EOC characteristics we believe it helpful to provide a definition of what an EOC is and what its functions are.

A. What Is An Emergency Operations Center?

An EOC is a facility, which may be fixed or mobile, that serves as the primary point of command, control, and communications in emergency response. EOCs, and the personnel in them, are normally removed from the physical site of the emergency and so do not directly respond to the incident. Generally, EOCs are not on-scene command posts used for direct operational control of emergency response elements. Usually, that direct operational control is delegated to security, health protection, operational, or fire department units which follow established emergency procedures or direction from the EOC.

Because EOCs are most often remote from the incident site, and the actual emergency response elements, emergency managers depend on communications systems to supply information about the emergency. EOCs consist of equipment, personnel, and methods of operation to process information and make and direct the implementation of decisions that terminate emergencies.

An activated EOC utilizes information to coordinate and manage an emergency response at a higher-than-operational level. This may include ordering specific actions by response elements, notification of appropriate government jurisdictions and the media, and requests for assistance. In summary, an EOC processes information, and based on the superior knowledge that the information provides, determines response and supporting actions to terminate an emergency.

To illustrate the complex paths emergency information and decision making follow, we have included Figure 1, Successful response requires well-defined and tested paths for emergency information flow and decision making. This figure represents typical emergency flow for U.S. Department of Energy (DOE) facilities. Notice that the decision path, which includes both the actions for response elements and requests for assistance, is simpler than the information (notification) path. The information path may involve many peripheral organizations (e.g., the media) that could significantly impact the responding organization over the long term.

The EOC is often the major source of information to the public and the media. The importance of this communication cannot be overestimated. History often judges the success of an emergency response more on the perceptions created by the media than the reality buried in official records. Therefore, although more complex and often time consuming during emergency response, the information (notification) requirements cannot be overlooked. The primary purpose of any EOC is to manage information and decisions. The quantity of information is great and both information funneling and filtering are required to ensure high quality, relevant information and decisions result.

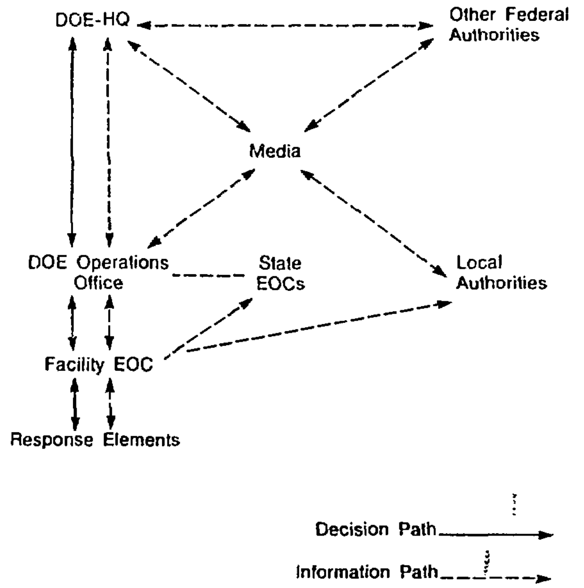


Figure 1. As shown in this DOE example, successful response requires well-defined and tested paths for emergency information flow and decision making.

In the information processing role, which includes the orderly flow of relevant information and decision making, EOCs are the primary interface point between on-scene response and the outside world. The EOC serves as a two-way clearinghouse for information: for information from the incident scene and for supporting information from outside the response organization. This concept is critically important for information sharing in EOCs. As technology permits faster and faster access and processing of information, there is a tendency to be inundated by information. Post-incident evaluations of many emergencies typically have revealed that the management systems contained much significant information that was not used in the response. This occurs because of a lack of ability to focus on what is relevant (Information Technology for Emergency Management, 1984: 152-153).

We find that every EOC, as a two-way clearinghouse, must include a carefully designed and implemented functional process that can handle a great deal of information and separate it by relevance to the response.

First, by funneling (or channeling) information is directed to the appropriate place for filtering. Second, and most important, is the filtering of information. Information from the incident (internal) as well as information from organizations outside the response (external) must be filtered. Only in this way is relevant information obtained for accurate and appropriate decision making. Our concept is represented in Figure 2, The funneling and filtering of information is necessary to support decision making based on the most relevant information. It is becoming more and more important for EOCs to facilitate the sharing of information among those in the facility and between the EOC and the great number of outside organizations with a stake in a coordinated and effective emergency response. Simply put, this requires EOCs defined as information sharing facilities.

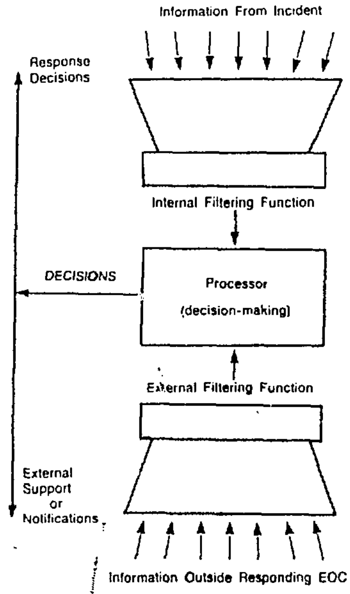


Figure 2. The funneling and filtering of information is necessary to support decision making based on the most relevant information.

B. Characteristics of EOCs

We have combined many individual characteristics into several broad categories relevant to information sharing. These same categories will permit application and design criteria to be established.

1. The Emergency Organization

The emergency organization executes emergency management responsibilities. Differences and similarities between facilities and organizations define unique domains of emergency management responsibility. Similarities in potential emergencies, policy guidance, and overall mission make facilities and their organizations comparable. Differences in geographic proximity to the facilities managed, the organizational history and culture, and organization structure and procedures used provides a substantial and meaningful contrast between EOC emergency organizations.

When activated, the EOC emergency staff finds it difficult to make the transition from their normal management responsibilities to their emergency roles. Managers have collateral responsibilities and this must be taken into consideration in emergency planning and preparedness. Managers must be trained and exercised in their emergency roles and in the process of working with unfamiliar organizations. The role of training cannot be overstated. Managers cannot devote much time to emergency preparedness and their complex emergency management roles require orientation and familiarization. Emergency procedures require drills in order to be used proficiently in the high-stress environment of a response. Creative and innovative training is a must for a smoothly functioning EOC emergency staff.

Shared information processing can be enhanced or hindered by the physical layout of the EOC. Typically, a technical, specialized support cadre is available to receive, sort, manipulate, and portray emergency information. The management cadre uses this information to make critical emergency decisions.

The process of receiving, sorting, manipulating, and portraying emergency information is noisy. This area must be separated from the area where the management cadre works. In this way, internal (to the EOC) information is filtered by the support cadre for use by the management cadre. Well-rehearsed procedures, trust, confidence, and teamwork from people not used to working as a team are required for an emergency organization to work at all. The degree of success depends on the dedication of both the people involved and their parent organization, and on the organization of emergency planning and preparedness.

2. Internal Notifications

An internal notification system gets the right people to the EOC quickly. Following the report of the emergency and the decision to active the EOC, the communications staff in the EOC contacts members of the emergency organization. People cannot share information until they can be matched up with the information. A delay in assembling the EOC emergency organization can have catastrophic consequences.

3. Internal Information Flow

Accurate, reliable, and timely information about an emergency is necessary for a successful response. The internal information flow ensures information reliability.

EOC information flow usually parallels the structure and physical positioning of the emergency organization. From the incident to the EOC, and once in the EOC, from functional group to functional group, the path information follows determines its efficacy. If information passes outside its functional path, e.g., security information processed by health physicists, the information becomes contaminated with loss of confidence and biased perceptions. Without strict control over information flow from source to decision point, the decision maker loses confidence in the reliability of emergency information.

4. Emergency Information Display

Although part of internal information flow and management, the display of emergency information in the EOC is so critical it is considered separately. The display of emergency information must support the next decision. For this reason, status rather than event information is usually portrayed in EOCs. Event information displays show multi-incident information or the chronological history of event status. Status information is sorted and displayed by function. Status information lessens the tendency of the emergency manager to get caught up in the story that is portrayed on an event display.

Emergency information may be displayed by a variety of means of varying degrees of technological sophistication. The technological sophistication has little to do with the effectiveness of the display. Of critical importance is the placement of the display: can the emergency managers see the information they need? Additionally, the accuracy and timeliness of the information displayed is critical. If the emergency manager does not have faith in the display process, the manager will waste valuable time confirming information.

5. External Information Flow

External information flow supports both the information path and the decision path illustrated in Figure 1. The information display techniques are also critically important here. Information should be portrayed in exactly the same format in a hierarchically separate but organizationally similar EOC. Thus, the information displays in the facility, operations office, and headquarters EOCs of DOE, as one example, should be identical. (They are not!)

Well-developed and rehearsed procedures are necessary to ensure that the media, and through the media, the public are provided incident status quickly and reliably. This is difficult to achieve in reality because of the need to protect classified information and because most top managers will not delegate the approval process for press releases. The stakes are too high.

6. Emergency Communications Systems

Control over the flow of communications internally and externally ensures the flow of data and information is fast and reliable. Communications systems support all other functions, but particularly the internal and external information flows. Specialists in the EOC normally control the communications systems. These specialists receive and sort data (funnel and filter). Data may be sorted by function or priority. In order to ensure that only high-quality, relevant information ends up on the displays the management cadre needs, the specialists must properly process the information as it enters the EOC.

The communications systems are critical to the information processing function of the EOC. Indeed, information processing cannot occur without good communications. These systems could include telephones (there are a great variety of these, some quite exotic and expensive), radios (usually HF, VHF, and UHF), teletypes (which can operate from radios or telephone lines), runners (which transfer information within the EOC or for short distances outside the EOC), and finally other systems which could include microwave or satellite or other more exotic types.

These communications systems represent a significant investment in any emergency response system. Effective planning and preparedness to best use them is essential.

7. Maintaining the Historical Record

Emergency response provides valuable feedback that supports training. Maintaining a historical record of the response allows emergency response personnel to review their actions, decisions, and assessments and improve them. Legal and reporting requirements are also met.

III. CONCLUSION

Complex organizations, especially public ones, are driven by many forces. A beginning step in an analysis of any such complex public organization is to define the organization's operation, that is, what is being managed. The emergency operations center is a facility that manages emergencies. Such facilities do not execute emergency response actions but direct, control, oversee, and coordinate the actions of one or many response groups. Additionally, actions outside of the response may impact the EOC. Reporting requirements dictated by law or departmental procedure, press relations, and the concerns of the public must be addressed. Almost any emergency above the trivial will become complex and require the activation of special organizations and facilities to respond and manage the response to such emergencies. The EOC is the central focus of emergency management activities above the operational level.

Examining the response to recent emergencies, such as Three Mile Island, Bhopal, Mexico City, or Chernobyl, it is obvious that improvements in all functions of emergency response and emergency management are desperately needed. If we are to support emergency management above the operational level, we must begin with the EOC.

The EOC is a central command, control, and communication facility in emergency management. We have described the actions of the EOC in terms of seven critical functions: the emergency organization, internal notifications, internal information flow, information display, external information flow, communications systems, and maintaining the historical record. How these functions are executed largely determines the success or failure of emergency management actions. We believe that by describing these functions in terms of shared information processing, the probability of success in emergency management can be improved.

Sharing information begins at the operational level. The operational response personnel must collect and forward data about the incident to the EOC. This takes time. What impetus do these people have to do a good job at this data collection and communication? Because operational responders are dependent on the EOC for support they are willing to share information with the EOC. Interdependence leads to information sharing. The operational personnel must have confidence that the EOC will provide the necessary support or they will not effectively share information, will not take the time to collect and communicate incident data to the EOC, and will not execute the directions of the EOC.

When the emergency begins, the emergency organization is activated in the EOC. Information from the incident begins to pour in, as do requests for information from higher levels in the organization, other organizations, the press, the public, and employees. The EOC must make decisions about response actions, resource allocation, press releases, and a host of other critical matters. We have observed that these decisions are made. All of the functions of the EOC identified above are performed. We have also observed that some EOCs do better than others.

At this point it is only possible to say that empirical evidence suggests that EOCs designed and operated as shared information processing facilities do a better job of managing emergencies. As one example, some EOCs separate their managers from those personnel who receive and process the incoming data and convert that data into useful information.

Separating personnel in the EOC on the basis of their information processing function reduces confusion and focuses attention on individual responsibilities. Further, technical specialists usually process information and determine what is important and what is not (funneling and filtering). These specialists categorize information and add a bias. Managers also have difficulty not looking over the shoulder of these specialists. Being aware of the garbage-in/garbage-out syndrome, managers are concerned about the ability of these specialists to give them quality information.

In some cases, budgets limit the size of the EOC and everyone is crowded into one or several rooms. These organizations often save thousands of dollars, but if a major emergency occurs the organization will pay for this economy. Another example of an economy-driven criterion is the often-noted lack of concern over the ability of the EOC to maintain a historical record. The historical record is the best method available to review actions and suggest improvements. The use of flight data recorders in determining the cause of aircraft accidents and preventing future accidents is evidence of the need for a similar capability in EOCs.

EOC design and operation based on shared information processing will significantly increase the probability of successful emergency management. An emergency preparedness program including adequate emergency management documents, training, drills and exercises, and evaluations and audits is necessary for organizations to refine the skills of EOC personnel and mold these personnel into a team. A team of EOC professionals will be more effective if their operating procedures are based on processing information, because processing information is the operation of EOCs.

EOCs collect data, process that data into information, and then use that information to make decisions and direct response actions. EOCs will be more successful if their organization, activation processes, internal information flow, information display, external information flow, communications, and record keeping operate for maximum efficiency and effectiveness in processing and using information. The higher the level of preparedness in these functions, the more effective the EOC will be. In future papers, a descriptive model of EOC operation should be developed. Eventually, predictive and prescriptive models will allow us to say with more confidence what will improve EOC capability, provide a measurement and evaluation of that capability, and improve EOC operations.

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The NRC Operations Center's Function

Eric W. Weiss

ABSTRACT The Nuclear Regulatory Commission has maintained a 24-hour-a-day, 365-days-a-year, manned Operations Center since the emergency incident at the Three Mile Island Nuclear Power Plant in 1979. The Center functions as the NRC's point of direct communication through dedicated telephone lines for reports of significant events at licensed nuclear power plants and certain fuel cycle facilities. The Center has become a key element in the agency's emergency preparedness.

The effectiveness of the NRC Operations Center depends in large measure on complete and accurate reports from the licensees. The information provided is used to: identify generic safety issues and precursor events that may compromise plant safety; develop licensee performance trends that are used to adjust NRC regulatory emphasis; and, evaluate and provide for the appropriate NRC response to events in a real time mode.

I. INTRODUCTION

The Nuclear Regulatory Commission has maintained a 24-hour-a-day, 365-days-a-year, manned Operations Center since the emergency incident at the Three Mile Island Nuclear Power Plant in 1979. The Center, which is located in Bethesda, Maryland, functions as the NRC's point of direct communication through dedicated telephone lines for reports of significant events at licensed nuclear power plants and certain fuel cycle facilities. The Center also has a commercial telephone number which is (301)951-0550. At night and on weekends and holidays when the NRC regional offices are closed, the main telephone numbers for the NRC regional offices play a recorded message to call the Operations Center in the event of an emergency. The NRC Operations Center receives a wide variety of calls and has become a key element in the agency's emergency preparedness.

The staff at the Operations Center evaluate the telephone notifications and, depending on the safety significance of the particular event, notify other appropriate NRC personnel and other Federal agencies. Examples of Federal agencies that are sometimes notified include the Federal Emergency Management Agency (FEMA), the Department of Energy (DOE) and the Department of Transportation (DOT). In all cases the NRC regional office responsible for the facility reporting the event is notified. Response to an event may vary from simply recording the circumstances of the event for later evaluation to immediately activating response teams within headquarters and the appropriate NRC region. Events are monitored by the NRC while they are in progress from the standpoint of actions to protect the health and safety of the public.

During the early hours of an emergency, the Operations Center becomes the focal point for action by the NRC. The NRC monitors the licensee's actions during an emergency to assure appropriate protective action is being taken with respect to offsite recommendations. When requested, the NRC supports the licensee with technical analysis and coordinates logistics support. The NRC supports offsite authorities, including confirming the licensee's recommendations. The NRC keeps other Federal agencies and entities informed of the status of the incident. The NRC keeps the media informed of the NRC's knowledge of the status of the incident, including coordination with other public affairs groups.

Not every event telephoned into the Operations Center is an emergency; some non-emergency events have important generic implications for other operating nuclear power plants that could lead to an emergency, if left unchecked. Consequently, each event called into the Operations Center or reported by a regional office is evaluated to determine any generic implications for similar facilities. Event reports are screened during the first hours of the first working day following the receipt of a notification. Events that may be significant from a generic standpoint then receive additional in-depth evaluation. For events found to have significant generic implications, the NRC issues an Information Notice or a Bulletin to the appropriate licensees and construction permit holders.

II. OPERATIONS CENTER PERSONNEL AND EQUIPMENT

The Operations Center is continuously manned by a Headquarters Operations Officer who is an engineer or scientist specifically trained for that job. The NRC considers events analysis an important part of the function of the Headquarters Operations Officer. By immediately analyzing the events that are reported, a serious event can be recognized at its earliest stages and events with generic implications can be treated appropriately.

Many of our Headquarters Operations Officers have advanced degrees and they all receive in-depth training on reactor design and operations at the NRC Technical Training Center in Chattanooga, Tennessee. Typically the course of instruction for a Headquarters Operations Officer lasts about 12 weeks in Chattanooga plus a couple more weeks in Bethesda on the equipment and procedures for the Operations Center.

The Headquarters Operations Officer has special equipment to aid him in his function. The Operations Center has two dedicated computers that serve to record, disseminate, and analyze events. The telephone system for the Operations Center is more complicated than that used by most telephone operators. There are two dedicated sets of telephone lines to each licensed nuclear power and fuel facility. The one most often mentioned is the Emergency Notification System (ENS) that appears as a red telephone in the control room of each nuclear power plant and rings in the Operations Center once it is picked up. The other dedicated telephone system is the Health Physics Network (HPN). The HPN is used during an emergency to establish conference calls between health physicists involved in the event. Besides the two dedicated telephone systems the Headquarters Operations Officer has access to the commercial telephone system and the Federal Telecommunications System (FTS).

The Operations Center also has a dedicated telephone line to FEMA. The dedicated telephone lines are a real advantage during an emergency when ordinary phone systems tend to become overloaded.

The Headquarters Operations Officer is able to control all of the various phone systems available by using a electronic PBX (private branch exchange). This system offers the Headquarters Operations Officer six telephone bridges to establish conference calls during an event. During an emergency, there are usually at least four telephone bridges established. One is used for the ENS, one for the HPN, and then there are two counterpart links established between headquarters and regional personnel so that headquarters and regional personnel can discuss an event without interrupting the flow of information from the site.

The Headquarters Operations Officer who is usually alone in the Operations Center, is joined by several teams during an emergency. Without attempting to describe everyone who has a role in the Operations Center, three teams are worth mentioning to give an overall picture of the function of the Operations Center during an emergency. The Reactor Safety Team follows the course of the event and attempts to predict future plant responses, e.g., Is the break getting bigger; what alternatives are there for reaching cold shutdown? The Protective Measures Team follows the course of the event from a radiological point of view, e.g., Will there be any health consequences; is evacuation warranted? The Executive Team directs the agency response to the emergency.

III. PROBLEMS AND TWO EXAMPLES

The chief problems with reports made to the NRC Operations Center have been lack of completeness in the reports and inadequate qualifications of some of those attempting to make reports to the Operations Center.

Two recent events, one at the Davis-Besse nuclear power plant near Toledo, Ohio and the other at the Point Beach nuclear power plant near Two Rivers, Wisconsin, highlight the need for more complete reports and improved licensee response to emergency classification.

A. Davis-Besse

At 1:35 a.m. on June 9, 1985, the Davis-Besse plant experienced a complete loss of main and auxiliary feedwater for nearly 12 minutes. The emergency plan identified the loss of feedwater event as a Site Area Emergency, which is the second most serious of the four emergency classes defined by the NRC's regulations. However, it appears that all knowledgeable personnel in the control room were occupied with stabilizing the plant and, thus, were not able to classify the event as a Site Area Emergency and activate the emergency plan. Had the plant not been brought to a stable condition quickly and had plant safety further degraded, it is possible that the efforts of all knowledgeable personnel in the control room would have been required for recovery efforts, further delaying initiation of appropriate onsite and offsite emergency response.

At 2:11 a.m., which is 36 minutes after the loss of all feedwater, the shift technical advisor (STA) called the NRC Operations Center from the control room using the Emergency Notification System to report the event pursuant to the Commission's regulations, which allow a maximum of 1-hour from the time that an emergency is declared until it is reported to the NRC Operations Center. At the beginning of the event, the STA had been in his quarters in the administration building, which is outside the protected area about a half mile from the plant. Although the STA mentioned the trip of the main and auxiliary feedwater pumps, the STA did not describe the length of time that the plant was totally without feedwater or the difficulty the plant had in restoring auxiliary feedwater. No Emergency Class was declared, nor was the fact conveyed to the NRC that plant conditions which warranted the declaration of a Site Area Emergency had existed for nearly 12 minutes.

At 2:26 a.m., the STA informed the NRC that an Unusual Event, the lowest of the 4 Emergency

Classes, had been declared at 2:25 a.m. The STA also informed the NRC that although the emergency plan identified the total loss of feedwater event as a Site Area Emergency, the plant was no longer in this emergency action level at this time. At 2:29 a.m., the licensee informed the county that an Unusual Event had been declared. The licensee depended on a procedure that required the county to notify the state representative. However, because the county could not reach the local state representative, the State of Ohio was not notified of the Unusual Event declaration until after the event had been terminated, more than 6 hours after its declaration.

At Davis-Besse, the emergency plan is initially implemented by the shift supervisor, who also has primary responsibility for ensuring that the plant is maintained in a safe condition. Because of competing priorities of (1) directing attention to necessary recovery actions to obtain a safe and stable plant and (2) reviewing the emergency plan and initiating its actions, there was a substantial delay in declaring an Emergency Class and implementing the emergency plan. If the June 9 event had progressed in severity, valuable time needed to initiate appropriate onsite and offsite response to the emergency would have been lost.

Corrective actions being undertaken by the licensee as a result of this event include a number operational and procedural changes that include but are not limited to the following: Changing the STA shift schedule from a 24-hour duty day to rotating 12-hour shifts. Having the STA spend the entire shift within the protected area, and having the STA office located within 1 to 2 minutes of the control room. Training the STA as an Interim Emergency Duty Officer to advise the shift supervisor in event classifica tion and protective action. Having the licensee make emergency notifications directly to the State of Ohio.

B. Point Beach

On July 25, 1985, at 7:25 a.m. (eastern time), Point Beach Unit 1 experienced an event involving loss of offsite power. Point Beach Unit 2 continued to operate normally during this event. Because of the incomplete understanding of the event by those making the notification to the NRC Operations Center, the NRC Operations Center was not made aware of the details of the event. At 7:37 a.m., a security guard called the NRC Operations Center to notify the NRC that Point Beach Unit 1 had declared an Unusual Event. The explanation for the Unusual Event was that the plant had a turbine runback. This did not make sense to the Headquarters Operations Officer because a turbine runback is neither an emergency nor even a reportable non-emergency event in itself. When the NRC Headquarters Operations Officer asked questions, the security guard was unable to provide additional information because of his limited technical knowledge of the plant and because the call was made from a location outside the control room where the security guard could not obtain additional information from the operators involved.

The Headquarters Operations Officer called the control room, and as a result of asking questions learned that a station transformer had been lost. However, not until 2½ hours later, when the plant notified the NRC Headquarters Operations Officer that the Unusual Event was terminated, did the Headquarters Operations Officer learn that there had actually been a loss of offsite power. A loss of offsite power at a nuclear power plant is a serious event warranting the declaration of an Unusual Event.

On October 15, 1985, the NRC issued Information Notice (IN) 85-80 to describe these two instances when an emergency condition was not classified and declared in a timely manner. Although events such as these are the exception rather than the rule, the NRC has counseled licensees that it is the licensee's responsibility to ensure that adequate personnel, knowledgeable about plant conditions and emergency plan implementing procedures, are available on shift to assist the shift supervisor to classify an emergency and activate the emergency plan, including make appropriate notification, without interfering with plant operation.

IV. CONCLUSION

The effectiveness of the NRC Operations Center depends in large measure on complete and accurate reports from the licensees. The information provided is used to: identify generic safety issues and precursor events that may compromise plant safety; develop licensee performance trends that are used to adjust NRC regulatory emphasis; and, evaluate and provide for the appropriate NRC response to events in a real time mode.

If you are planning a trip to Bethesda, Maryland, we would be delighted to make arrangements for a tour of the Operations Center.

Section VI

Medical Issues

Chairman: ***Clarence Lushbaugh***
Oak Ridge Associated Universities

Dose-Rate Models for Human Survival After Exposure to Ionizing Radiation[a]

Troyce D. Jones, M. D. Morris, and R. W. Young

ABSTRACT This paper reviews new estimates of the LD_{50} in man by Mole and by Rotblat, the biological processes contributing to hematologic death, the collection of animal experiments dealing with hematologic death, and the use of regression analysis to make new estimates of human mortality based on all relevant animal studies. Regression analysis of animal mortality data has shown that mortality is dependent strongly on dose rate, species, body weight, and time interval over which the exposure is delivered. The model has predicted human LD_{50}s of 194, 250, 310, and 360 rad to marrow when the exposure time is a minute, an hour, a day, and a week, respectively.

I. BACKGROUND

A. Mechanisms of Death

When mammals are exposed to high doses of ionizing radiations, blood lymphocytes and stem cells of the active bone marrow are killed. Mammals may die from infection or hemorrhage when these cell populations drop below certain critical levels. The time to death depends upon the number of cells killed; the species and strain of the mammal; and the cage/hospital care given, including therapeutic support, barrier nursing, nutrition, etc. This mode of death is commonly referred to as hematologic syndrome (or death from hemapoietic depression).

Hematologic death and modes of death resulting from gastrointestinal (GI) and central nervous (CN) system damage are described in Langham (1967) and many other sources (e.g., Baum et al., 1984). However, this review will discuss, in some detail, the biological and physical conditions contributing to hematological depression, the relevant animal studies from radiation biology, and human radiation accident and therapeutic experiences. This background section will establish the justification for new analytical tools to be used in modeling the LD_{50} for man.

Bergonié and Tribondeau (1906) proposed that the level of radiosensitivity of an organ or tissue is related to (1) the degree of differentiation of its cells morphologically and physiologically, (2) the mitotic activity, and (3) the length of time that the cells remain in an active stage of proliferation, which includes the number of divisions between the youngest precursor cell and the mature functional (or differentiated) cell. This proliferation of cells, which has commenced terminal differentiation, is commonly referred to as amplification and is especially important in maintaining sufficient numbers of lymphocytes (to fight infection) and platelets (to prevent post-irradiation hemorrhage). Because stem cells of the marrow are more radiosensitive than the rapidly proliferating crypt cells of the GI system or the highly differentiated cells of the CN system, survival of the organism is dependent upon the bone marrow although the time to death of a lethally irradiated animal may be determined by damage to the GI or CN systems (Langham, 1967; Baum et al., 1984).

According to Alper (1979), the killing of animals by radiation is determined by the death of "target cells" in "target tissues," and this concept is now a "basic part of the framework of radiological thinking." The United Nations Scientific Committee on Effects of Atomic Radiation (UNSCEAR, 1982) has further broadened this concept by proposing the basic premise that the nonstochastic response of a tissue depends upon the level of cell killing. ("Nonstochastic" is used to describe radiation-induced injuries where both the frequency of occurrence and the severity of the injury are porportional to the radiation dose.) Thus, for hematopoietic deaths, there is no controversy about the sequela of effects that precede death. Following whole-body exposure to lethal doses of radiation

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(below about 1000 rad given over a short time interval), death in small mammals occurs within 30 days and in larger mammals within 60 days.

When the radiation is delivered over a longer time, the magnitude of the lethal dose is increased, and the time to death becomes longer and less sharply defined. These changes can derive from two processes. First (if the radiation dose is from photons), subcellular or enzymatic repair of sublethal lesions in nuclear DNA can occur before a second photon (or photon-induced lesion) can further damage the site and kill the cell. Second, when the dose rate is sufficiently low (i.e., the dose is given over a time interval greater than 30 minutes), compensatory cellular proliferation begins in an attempt to restore tissue homeostasis. Thus, for low-dose-rate exposures to either high or low linear energy transfer (LET) radiations, the survival time can be quite long. The process has been described by Bond, et al. (1965):

> Proliferating cell systems like those found in hemopoiesis can, in operation, be likened to a retail dealer vending several items. If the factory manufacturing a particular item is damaged so that production is reduced or stopped, the retail dealer is not affected until stocks in the chain of supply lines are exhausted. Each individual item (cell) in the store even then remains as good or as "functional" as ever, and the segment of the vendor's business (organ) represented by that item is not affected until the number of individual items falls below a critical level or is exhausted. The entire business (mammal) is not seriously hurt or destroyed unless that particular item represented a major part of (was vital to) his operation. Survival will then be possible only if the factory can be put back into operation reasonably quickly (rapid regeneration), or if a substitute source or product can be used temporarily (symptomatic or substitution therapy) until restoration of the factory eventually takes place.

When humans have been accidentally or therapeutically exposed or when test animals have been irradiated, mortality is commonly an "all or none" event with respect to proportion killed in a population of individuals. That is, there is some high sublethal dose where no subjects die and some dose about twofold higher where very few, if any, individuals survive. The narrow transition zone (wherein some individuals die) is defined by a dose range where the upper dose is only 2 to 3 times that of the lower dose. Thus, because the mortality function is extremely steep, the dose that is lethal to 50% of the exposed individuals can in effect characterize the entire response function for most practical considerations with respect to nuclear safety, civil defense, and radiotherapy.

In spite of the large numbers of documented human exposures (Lushbaugh, 1969), there are inadequate human data to serve as a basis for promulgating an LD_{50} for man or to study how the human LD_{50} changes with different biological and physical conditions.

Historically, the most commonly accepted LD_{50} value for man has been that of the National Council on Radiation Protection and Measurements (NCRP) (NCRP, 1974). For high dose-rate exposures such as those that might occur following the explosion of a nuclear weapon, the NCRP has promulgated a value of 450 R (in air), which has been taken by the NCRP to be about 315 rad (to marrow). This NCRP LD_{50} value "is the median of a number of educated guesses made by a group of U.S. experts in 1949," and there is still insufficient human data to substantiate or change the 1949 value.

Lushbaugh et al. (1967) treated 93 terminally ill patients with a dose rate between 0.75 and 1.6 R/min. They also gave therapeutic aid to seven victims of the Y-12 radiation accident. From this combined population, Lushbaugh et al. derived an LD_{50} of 425 R (in air) or 281 rad (to marrow). Of course, all of the 93 terminally ill patients died, but 18 died within a time interval that suggested that the radiation treatment may have predominated slightly over the progression of death naturally associated with the disease. In contrast to studies of all-accidental human exposures, this study was supported by accurate dosimetry (Beck et al., 1971), and the numbers of patients were sufficient to permit a good statistical analysis of mortality.

In Lushbaugh's study, the patients were terminally ill, a state frequently speculated to result in increased radiosensitivity. On the other hand, the patients were given state-of-the-art hospital care, which would be expected to decrease radiosensitivity (NRC, 1975). Thus, although this study is without doubt the only accurate source of data that measures the LD_{50} of man, there is much uncertainty as to how to extend the results to different dose rates or to "non-sick" humans exposed under accident conditions.

Mole (1984) chose to consider the Y-12, Vinca, and Ewing Sarcoma patients of Rider and Hasselback (1967). Mole bases his LD_{50} of 600 R (in air) and 450 rad (to marrow) on survival of 28 individuals from a population of 29 exposed. Mole's analysis is strengthened somewhat because he used animal data to describe the shape of the mortality curve in the dose-normalizing technique of Jones (1981).

However, doses to accident victims have been determined from calculations and experimental "mock up" and thus are quite unreliable on an individual basis. Mole made a series of assumptions that combine to build a very high LD_{50}. These assumptions include Mole's belief that barrier nursing, antibiotics, platelet transfusions, and marrow grafts did not enhance survival. There are, however, several sources of experimental data to suggest that such procedures are likely to enhance survival by a significant amount, but these studies were not acceptable to Mole (NRC, 1975; Evans et al., 1985).

As a basis for evaluating the LD_{50} of man, Rotblat studied the atomic bomb experience in

Hiroshima. He presented his analysis at a meeting of the Institute of Medicine at the U.S. National Academy of Sciences (NAS, 1985). Rotblat found an LD_{50} of 220 rad tissue kerma in air (~250 R) and 154 rad (to marrow). Although the numbers of deaths and exposed individuals are adequate for a good statistical analysis (i.e., 201 deaths in 765 exposed), Rotblat's analysis is quite controversial because of:

1. extremely inaccurate analytical methods (as only one of several possible examples, Rotblat goes to great effort to compare the shapes of survival curves in a semiquantitative manner by transforming the scales on both the ordinate and the abscissa of only one curve; he then remarks that the similarity between the transformed curve for humans and the untransformed curve for mice "is striking"),

2. an assumption that all deaths after the first day were due to radiation and that combined injury from thermal burns and blast did not increase mortality, and

3. the fact that the survival curve is fivefold flatter than any ever observed in any mortality study (Jones, 1981; Mole, 1984) so that a few humans would die at doses as low as 50-75 rad but some could survive at doses twofold greater than the LD_{50}. Both of these effects have no basis of support in the vast literature on radiation therapy and radiation biology (Jones, 1981; Mole, 1984).

Many others have promulgated human LD_{50} values over time, but the four studies reviewed here illustrate the problem adequately. One is faced with the choice of extrapolating from sick humans to "normal" humans or analyzing data on normal humans where the doses and the number of deaths per number exposed are unreliable. Even with the best analytical methods, a reliable LD_{50} value must depend upon accurate dosimetry and sufficient numbers of exposed individuals and deaths. Some experienced investigators express a great reluctance to use sick humans as analogs of normal humans. However, it is our view that mechanisms of death may not be changed greatly in some populations of sick humans and that mortality models based on carefully done therapeutic populations provide a technically accurate estimate comparable to analytical models based on many species of test animals.

Thus, from the human data collected to date, it is not possible to define, with acceptable confidence, an LD_{50} value (or a mortality response function) or to anticipate how human mortality varies with dose rate and numerous other physical and biological variables.

Hence, this paper will draw on (1) the vast amount of animal mortality data published in the literature, (2) well-known principles from radiation biology that will help to fit each of these many animal studies into the proper perspective, and (3) simple unifying dose-response models (Jones, 1981, 1984) that permit all data on all species and experimental and biological factors to be analyzed simultaneously instead of the conventional approach of sequentially analyzing a few studies, which then usually cannot be merged into a coherent model that can be evaluated for man.

B. Physical and Biological Conditions That Affect Death

Myelopoiesis is the processes whereby a marrow stem cell divides in order to maintain the homeostatic population of stem cells and to supply differentiated cells to blood, bone, and thymus/lymph. Myelopoiesis is strongly dependent upon species and may vary to a lesser degree within individuals or strains. Myelopoiesis occurs rapidly in small mammals and more slowly in large mammals. The rate within an individual also varies according to homeostatic equilibrium or the need for compensatory cell proliferation to repair tissue injury. Figure 1 is an illustration of the species variation of myelopoisis (Bond et al., 1965).

From Fig. 1 it is seen that new cells in rat can be observed in the blood within about 3 days following exposure, whereas in man the time increases to about 8 days. Also, cell turnover is more rapid in the smaller species. Blood cell counts reach nadirs at shorter times in the small species so that large species survive longer after low lethal doses. But, because of rapid cell removal kinetics, the smaller species can survive higher sublethal exposures.

Thrombopoiesis results in production of megakaryocytes that secrete platelets, which are essential to prevent hemorrhage. As seen in Fig. 2, fully differentiated megakaryocytes are produced in about 7 days and have a mean lifespan of about 10 days in peripheral blood (Szirmai, 1965).

During thrombopoiesis, megakaryoblasts can increase in number, and this amplification is also common in processes of erythropoiesis and granulopoiesis (Szirmai, 1965). These partially and fully differentiated blood cells are more radioresistant than the stem cells so that, at doses that are just adequately lethal, time to death may be extended through amplification of surviving cell populations. However, lymphocytes (which amplify their numbers greatly within peripheral blood) are quite radiosensitive. As the magnitude of the lethal dose is increased, the survival time is shortened greatly; death results when lymphocytes are killed instead of when stem cells are unable to match the homeostatic demand for new cells. Burros have been found to be quite radiosensitive and die within just a few days following superlethal doses of radiation. Death of lymphocytes may help explain why burros are extra radiosensitive and may die in 5-10 days whereas most hematologic mortality in other large animals is seen at times greater than 10 days. Of course, 5-10-day survival times may be observed in all other species if the treatment dose is sufficiently large.

Radiosensitivity is directly related to cell cycle kinetics. The typical cell cycle is illustrated in Fig. 3. When the need for new cells is low (or zero), cells cease proliferative activity and are commonly referred to as

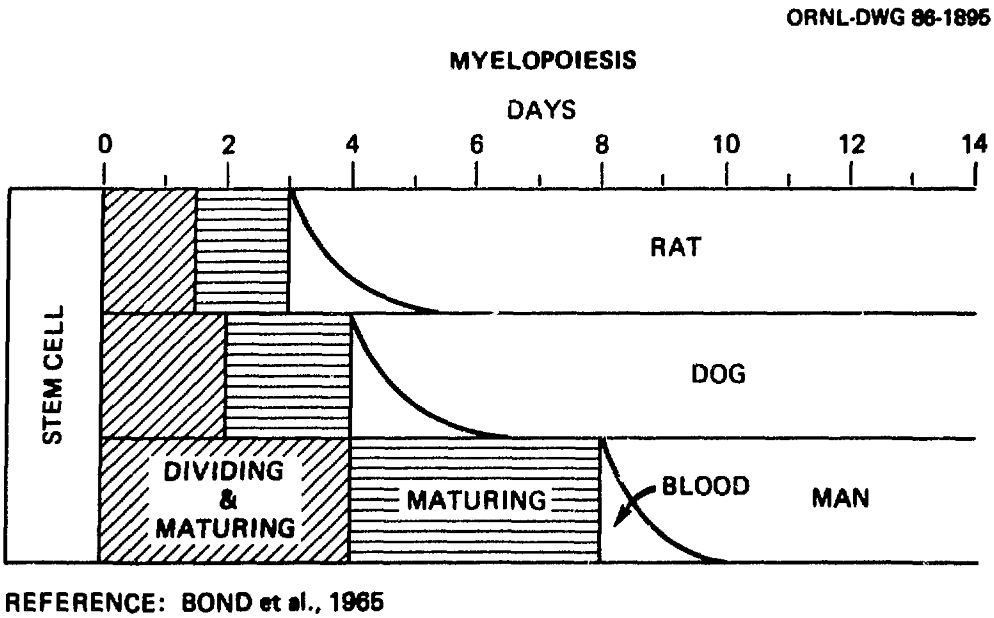

Fig. 1. Variation of myelopoiesis by species. (Reference: Bond et al., 1965.)

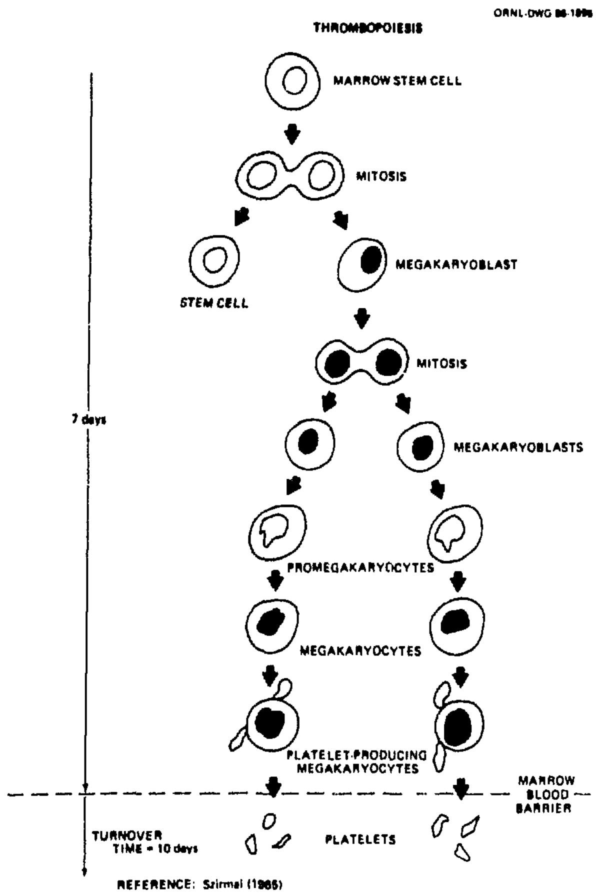

Fig. 2. Process of thrombopoiesis in man. (Reference: Szirmai, 1965.)

being in a resting or G_0 state. As seen in Fig. 3, a cell with the normal amount of nuclear DNA is said to be in the G_1 state (i.e., mitotic inactivity gap with diploid DNA). Following this 12-hour period new DNA is synthesized, and the cell drops into an inactive state with tetraploid DNA preceding binary fission into two new cells.

As illustrated in Fig. 4 (Case I), many radiogenic lesions in non-dividing DNA have a high probability of repair because the enzymes can read the complementary strand of DNA and repair the damage site before the DNA helix separates and a new helix of DNA is synthesized. In each new cell, a DNA helix contains one new and one old strand of DNA. As seen in Case II, radiogenic lesions immediately preceding DNA synthesis have a very low probability of repair, so that it is likely that one of the daughter cells is killed or functionally altered. For most practical considerations, DNA repair ceases when the nuclear DNA replicates. However, about half of the damage can be repaired within 1 hour preceding replication. Thus, in 4 hours between synthesis and mitosis the residual damage could be reduced to about $(1/2)^4$, or 6%. Cells can be killed by damage to other organelles such as mitochondria or membranes, but cells can be ten-fold more resistant during interphase periods than during metaphase periods.

It is obvious that a great many physical and biological factors can affect intracellular lesions, cell death, and, thus, death of the animal. Important physical factors commonly include: type (or LET) of radiation, dose level, dose rate, fractionation of dose with time, dose distribution within the marrow, etc.

Important biological factors that affect intracellular lesions, cell death, and death of

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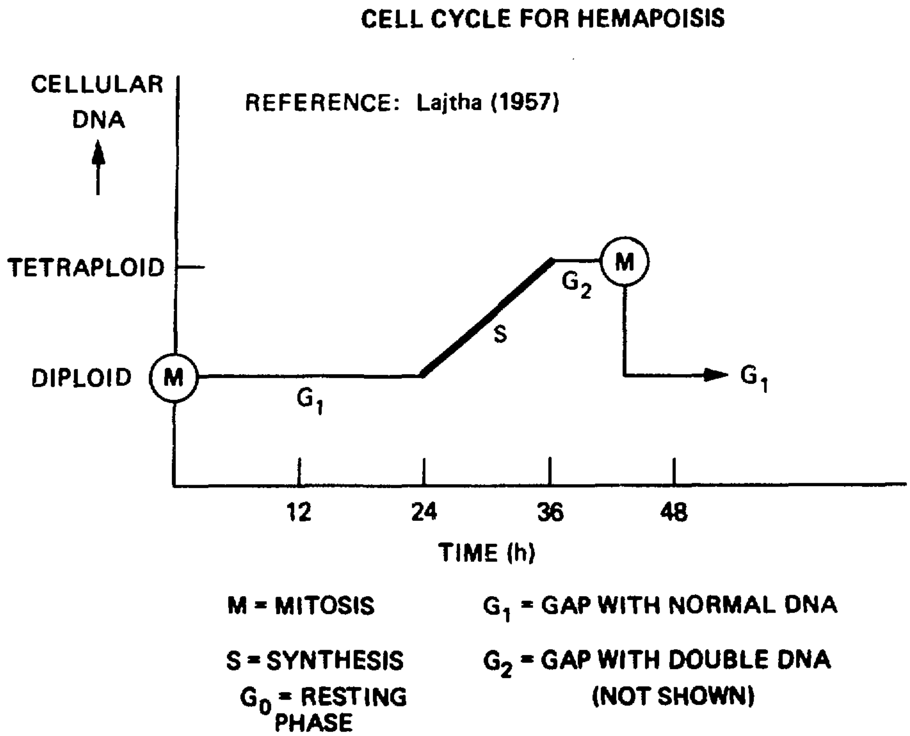

Fig. 3. Cell cycle for hematopoiesis. (Reference: Lajtha, 1957.)

the animal include: DNA content per cell; time periods of various phases of the cell cycle under different in vivo conditions; sensitivity of the individual, strain or species to infection and/or hemorrhage (physical conditions can host the opportunity for infection); capacity of the animal for compensatory repopulation of platelets and granuolocytes; etc.

II. METHOD

Lethality data were collected from many different animal studies. Experiments selected to evaluate the model were restricted to penetrating photons and any combination of body size and irradiation geometry that resulted in a uniform dose profile to all parts of the active marrow. A total of 224 different mortality studies were included in the data base. Data included: 13 different species; body weights from 25 g to 375 kg; sheep, goats, swine, and calves with body weights near man (i.e., 70 kg); radiation sources of ^{60}Co, ^{182}Ta, ^{95}Zr, atomic bombs, and X rays from several voltage potentials and many different moderating materials; and exposures from bilateral, multiple sources, unilateral, rotational, free-moving animals, 4π, and quadrilateral geometries, as long as the dose was uniform over the active bone marrow.

Typically, physical factors can be quantified or ordered on a numerical scale. But biological factors vary with species, strain, age, etc., and with the composite force from the collection of other biological and physical factors. Most biological factors can be treated as "classification" type variables in a regression analysis but usually cannot be quantified. However, the variance in the data set resulting from classification variables can be quantified. Regression analysis methods were used to quantify how much variance is left in the data set after physical factors are considered. Then, we evaluated how much of this variance can be due to individual "classification" variables (such as species) and, finally, how much of the species effect is left unaccounted for after body weight is considered.

Within each species, a model that linearly related the log of LD_{50} to the log of dose rate fitted the data fairly well. Across species, the slopes of these lines were relatively consistent, but their intercepts differed significantly. As evidence of this, a model fitted to all data with a single intercept and slope accounted for less than 1% of the variation in the data ($R^2 < 0.01$). A model that included a separate intercept for each species but only a common slope for the log of dose rate accounted for 84% of the variation; a model that allowed separate intercepts and slopes for each species improved this only slightly, to 86%. Furthermore, when intercepts were fitted for each species, there was a clear inverse relationship between species body weight and the value of the intercept (heavier species had relatively higher response values at a given dose rate). In fact,

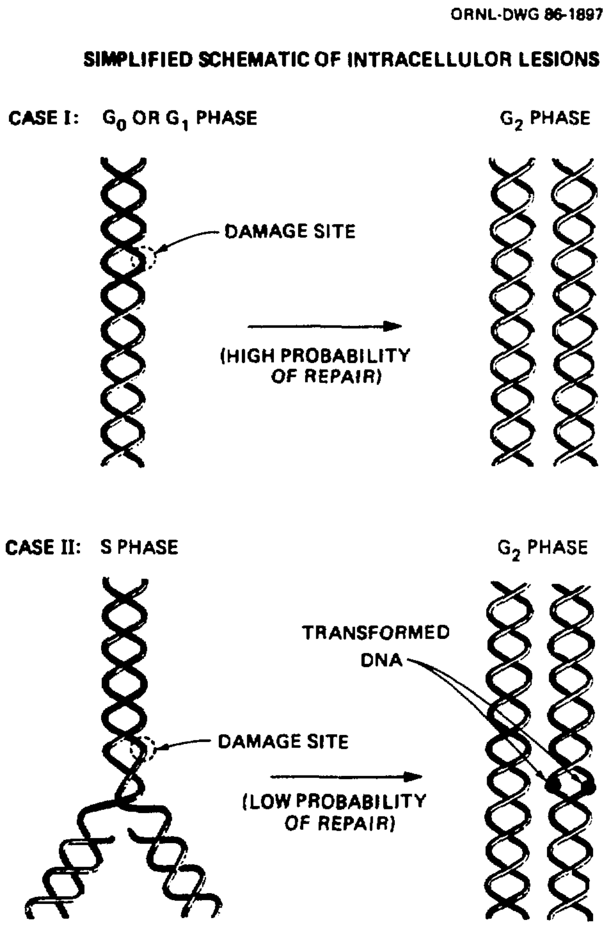

Fig. 4. Simplified schematic of intracellular lesions.

a model that contained a single intercept, a term for the log of dose rate, and a term for standard body weight accounted for 64% of the variation in the data.

In consideration of the above, a two-stage model was adopted.

Stage 1: For species i, a model of form

$$[\log_{10}(LD_{50})]_i = \alpha_i + \beta\cdot\log_{10}(\text{dose rate}) + \gamma\cdot\log_{10}(\text{body weight})_i + \varepsilon$$

was used. β and γ are regression coefficients assumed to be valid across species, α_i is an intercept specific to species i, and ε represents experimental or unexplained error in the reported values of $\log_{10}(LD_{50})$, assumed to be distributed with mean zero and variance σ^2.

Stage 2: Across species, α_i was assumed to be distributed with mean α and variance δ^2. This random intercept can be thought of as a species effect, after correction for species body weight. In order to use the model to predict results for a species not included in this data set (e.g., man), it was assumed that the applicable value of α_i would be a new (unobserved) value from this same distribution.

Using a computational technique described by Laird and Ware (1982), maximum likelihood estimates were calculated for the parameters of the model as follows:

$\hat{\alpha} = 2.743$
$\hat{\beta} = -0.070$
$\hat{\gamma} = -0.161$
$\hat{\sigma}^2 = 0.0096$
$\hat{\delta}^2 = 0.0194$

So, a point estimate of the LD_{50} for an unspecified or new species is

$$\text{estimated } LD_{50} = 10^{[\hat{\alpha}+\hat{\beta}\cdot\log_{10}(\text{dose rate}) + \hat{\gamma}\cdot\log_{10}(\text{body weight})]},$$

or

$$\text{estimated } LD_{50} = 10^{[2.743 - 0.70\log_{10}(\text{dose rate} - 0.161\log_{10}(\text{body weight})]}.$$

For man, a species having a 70-kg body weight, this reduces to

$$\text{estimated } LD_{50} = 281\ (\text{dose rate})^{-0.070}.$$

A common slope of −0.070 was used for all species, and the intercepts (i.e., α_i's) are as follows: mouse (2.946), hamster (2.925), rat (2.875), guinea pig (2.508), rabbit (2.967), primate (2.782), dog (2.493), goat (2.490), sheep (2.393), swine (2.500), man (2.743), burro (2.410), and cattle (2.205). Thus, mouse, hamster, rat, primate, dog, swine, goat, burro, and cattle seem to demonstrate a consistent monotonic relationship with body weight, but sheep and guinea pig are more radiosensitive than most other species and rabbit is more radioresistant, on a relative basis.

These formulae are compared with the experimental data, sorted according to species, in Fig. 5. It should be remembered that, although the experimental data in Fig. 5 reflect many other biological and physical variables, only one equation was used for all species. However, the model seems remarkably accurate for each species. This consistency is unique (Baverstock et al., 1985) and offers almost unlimited potential to model human response from the extensive data basis available on test animals.

The midlethal dose for man plotted against dose rate is given in Fig. 6. Marrow dose was converted to tissue kerma in air according to Jones (1977). The equation for man [viz, $281/(\text{dose rate})^{0.07}$] was solved for midlethal dose for continuous dose rates given in 1 minute, 1 hour, 1 day, and 1 week. Results are in Table 1.

III. DISCUSSION

Baverstock et al. (1985) analyzed animal data but did not use a dose-rate dependent model. Instead, they selected exposure times of one hour or less. They found a lack of homogeneity within species. However, according to our analysis, the LD_{50} given in one minute is about 190 rad to marrow, and the LD_{50} given in one hour is about 250 rad. These estimates are for a 70-kg body weight; the spread could be considerable larger for smaller species with

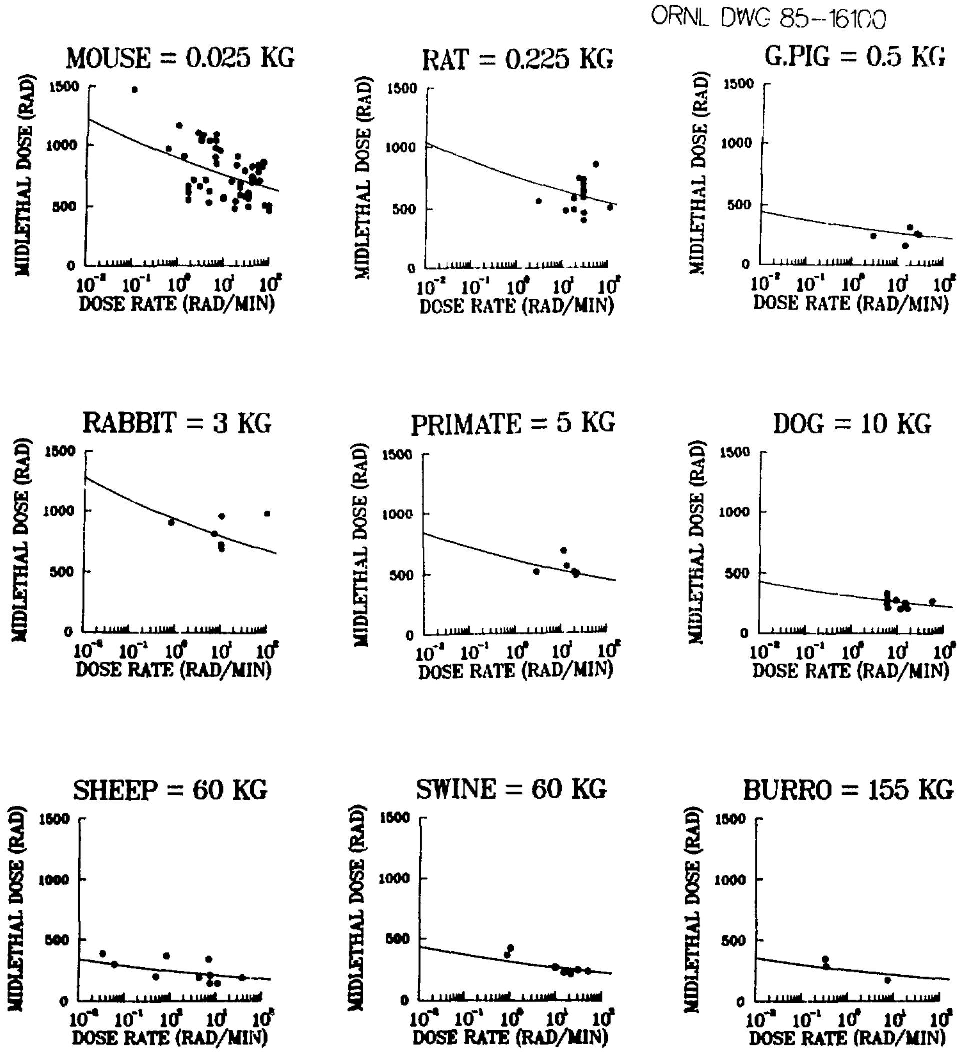

Fig. 5. Comparison of the statistical dose-rate model with experimental data for nine species.

faster mitotic rates. The model of Baverstock et al. did not discriminate between dose rates when exposure times were short compared with cellular turnover times (i.e., dose-rates above 10 R/min were considered equal). However, for low LET radiation (which was being modeled), enzymatic repair must also be considered. Such repair has been found to be significant in times much shorter than cell turnover times (Terzaghi and Little, 1975). Thus, the Baverstock et al. model does not seem well suited to analyze exposure intervals ranging from one minute to one hour. In addition, we have made estimates of dose to marrow upon which our model is evaluated. Although the Baverstock et al. model was evaluated twice--once in terms of exposure and again in midline tissue dose--no attempt was made to estimate marrow dose. Because of these significant differences in the two models and because of our much larger data base, it seems that our model has found a coherent pattern in interspecies LD_{50}. Experiments using low LET exposures (to all species) that resulted in uniform marrow dose all seem compatible with a simple interpolation model based on dose rate and body weight.

IV. CONCLUSIONS

There is no unique or practical LD_{50} for man because mortality varies strongly with dose rate and with several physical and biological factors. The NCRP LD_{50} in air may be about 25% too high, and the NCRP marrow LD_{50} may be about 60% too high--excellent agreement considering what was known in 1949 when the NCRP value was promulgated. Mole's LD_{50} in air (Mole, 1984) seems about 70% too high, and his value for marrow seems about 130% too high. However, Mole's mortality curve has the correct shape because he derived it from the animal data in the method of Jones (1981) (i.e., treatment doses were normalized to a multiple of the LD_{50} for that particular experiment).

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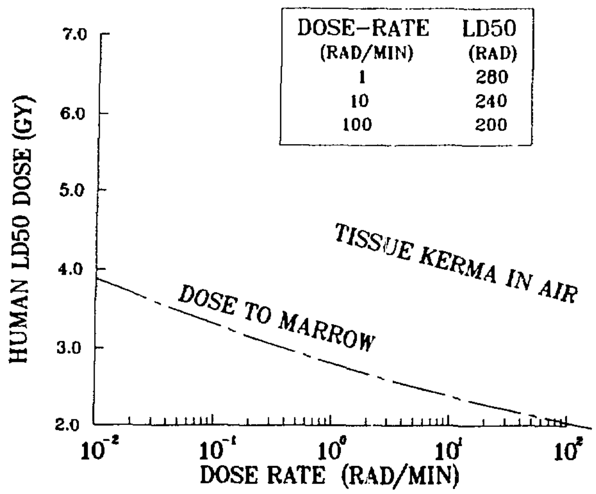

Fig. 6. Midlethal dose for man as a function of dose rate according to the statistical model derived from animal data.

Rotblat's LD_{50} (NAS, 1935) seems about 20% too low, but the slope of his mortality curve is fivefold too flat (Jones, 1981; Mole, 1984).

Lushbaugh's estimate of 281 rad to marrow for dose rates between 0.75 and 1.6 R/min seems acceptable. The mathematical model based on the animal data gives 281 rad for 1 R/min and a 70-kg body weight (Lushbaugh et al., 1967). If Lushbaugh had not included the seven Y-12 victims, who were exposed to high dose rates, with his 93 patients treated at a low dose rate, his LD_{50} estimate would probably be a bit higher than the published value of 281 rad.

The United Nations Scientific Committee has undertaken a recent analysis of human mortality. Although that analysis has not been finalized, the analysis described in this paper and our previous experience support several important issues discussed in the UNSCEAR 1982 report. Those issues include:

Table 1. Midlethal dose for man exposed to continuous dose rates given in 1 minute, 1 hour, 1 day, and 1 week

Exposure time	Dose rate (rad/min)	Photon midlethal dose	
		Marrow (rad)	Tissue kerma in air (rad)
1 Minute	194	194	350
1 Hour	4.2	250	450
1 Day	0.22	310	560
1 Week	0.035	360	650

- differences in effects due to different photon energies are considered to be negligible (p. 572),
- sublethal damage can normally be repaired in a few hours (p. 573),
- "For a variety of different types of treatment, different dose rate and LET, the reduction in the proportion of surviving cells resulting in 50% death of the mice was the same for all treatments" (p. 573),
- there is little or no enzymatic repair above 100 rad/min (p. 575) so the LD_{50} should be constant at dose rates above 10^2 or 10^3 rad/min.

Conclusions presented in this manuscript are expected to be firm, but numerical results presented at this time are subject to small changes when a final reporting of this study is made in 1987. Because of the success of this exploratory statistical model, a comprehensive effort is now under way to collect data on all individual dose treatment groups that contributed to the 224 different LD_{50} values analyzed in this study. Also, other studies are being added to the data base. This more comprehensive data base will be analyzed for mortality response as a function of treatment dose expressed as multiples of the LD_{50} value appropriate for a particular study. Thus, a universal mortality function will be derived (Jones, 1981), and 95% confidence limits will be evaluated.

When that effort has been completed, the body weight of man and different dose rates of interest can be used to calculate tables of dose-response values.

V. ACKNOWLEDGMENTS

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ANS Topical Meeting on Radiological Accidents—
Perspectives and Emergency Planning

Radiation Carcinogenesis Following Low-Dose or Low-Dose-Rate Exposures

R. L. Ullrich

ABSTRACT. A variety of dose responses have been observed for cancer induction following low linear energy transfer (LET) radiation. In general, however, the response is curvilinear, with a rapidly rising component in the intermediate dose range followed by a plateau or decline in incidence at high doses. The response is more linear at low doses, whereas the response at intermediate doses is approximated by a dose-squared relationship. Models for this response are based on the biophysical theory of cellular effects. However, many types of effects contribute to the tumorigenic processes, and host factors play a major role. At low dose rates the carcinogenic effect is generally reduced, which is caused by a dimunition of the dose-squared component and results in a linear response. Effects of fractionation can vary with total dose, fraction size, and fraction interval. High LET radiation is more tumorigenic. The dose-response relationships are more nearly linear and are less dose-rate dependent. The relative biological effectiveness (RBE) varies with dose, dose rate, fractionation, and target tissue.

The primary risk of exposure to low doses of radiation is cancer induction. While most exposures to radiation occur at low doses or low dose rates, most information on the carcinogenic effects of radiation for human populations is for high dose and high dose rate exposures. Statistically reliable information at low doses is very difficult to obtain because of the sample sizes required to detect increased cancer risks which are small compared with the total cancer risk in the population. Since it is not possible to measure directly effects at low doses and dose rates, estimates of risk are based upon extrapolation by use of mathematical models of the dose response relationship (Upton, 1977). The two models most often applied are either a linear model or a linear quadratic model. Although the question of dose response models is fundamental, available human data are insufficient to allow a choice between these models and estimates of risk can vary depending upon the model used.

The linear quadratic model has as its basis biophysical concepts of radiation effects within cells (Kellerer and Rossi, 1978; Kellerer and Rossi, 1972). Biophysical theory proposes that effects at the cellular level are a result of the interaction of two sublesions within the nucleus. These sublesions can be produced independently by two ionization tracks or, when sufficient energy is deposited, by a single ionization track. For low-LET radiation in the intermediate dose range the sublesions are generally produced by two independent tracks in close proximity in time and space, thus the response increases with the square of the dose. When the dose is low, the probability of two independent tracks passing in close enough proximity to produce interacting sublesions is reduced. At these low doses the dose-squared response is absent and the response is linear. This linear response is a result of the production of both sublesions by a single ionization track. For high-LET radiation the two sublesions are theoretically produced by a single track over a wide dose range because of the dense nature of the ionization events. As a result of this, effects on cells show a linear relationship with dose. While this model appears to be adequate for many cellular effects, its applicability to tumor formation is less clear. It should be appreciated that this model is based on the induction of initial cellular events but may not be adequate for the complex, apparently multistage process of carcinogenesis.

Animal studies have been useful in examining the question of low dose and dose rate effects. In studies with low LET radiation a variety of dose responses have been observed ranging from those with a threshold to those in which the response appeared linear (Fry, in press; Ullrich and Storer, 1979a; Ullrich and Storer, 1979b; Ullrich and Storer, 1979c; Ullrich, 1983). In spite of this range of responses, in most instances the dose response has been found to be curvilinear. These curvilinear dose responses can generally be described by a linear-quadratic regression equation in which there is a rapidly rising (dose squared) component in the intermediate dose range and a more shallow (linear) component in the low dose range (see Figure 1).

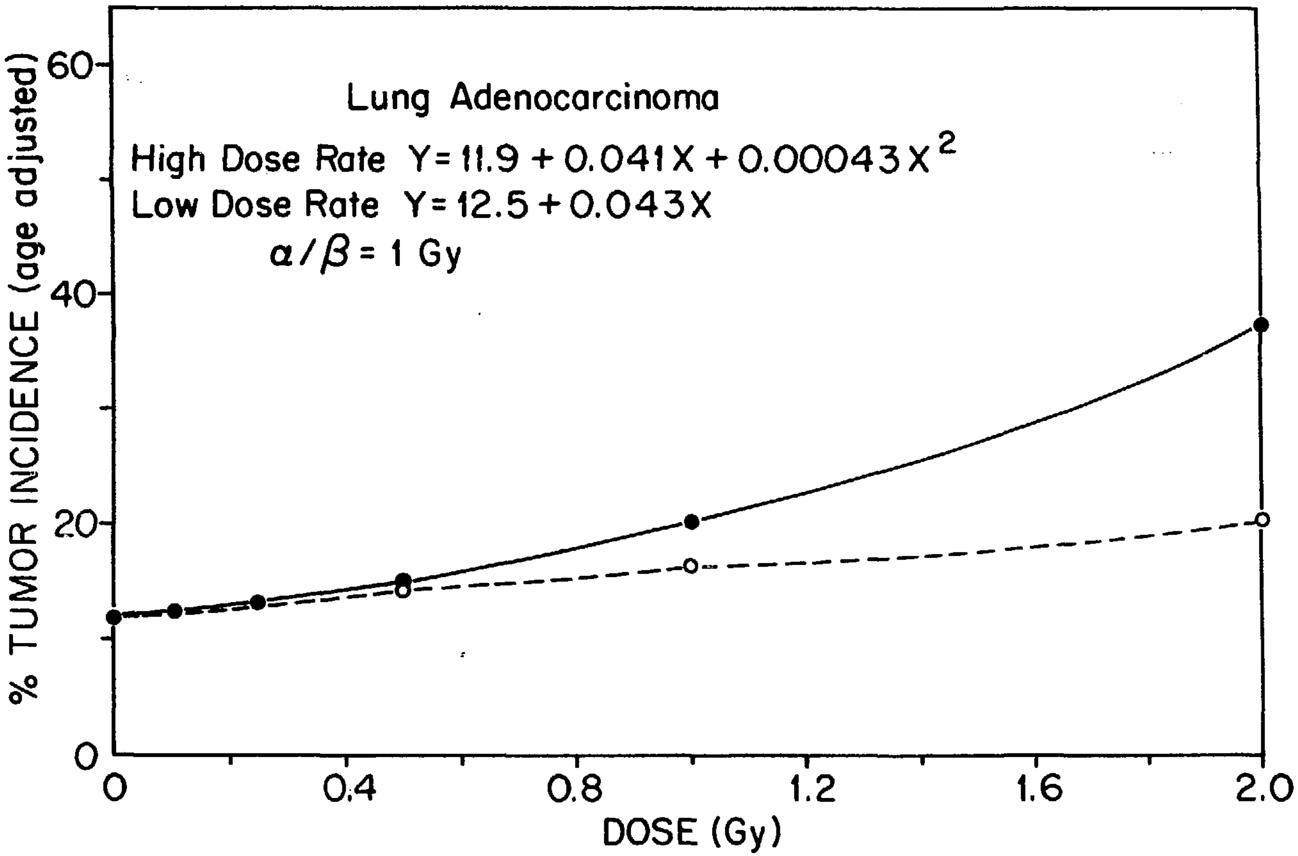

Figure 1. Incidence of lung tumors following high (●) or low (O) dose rate γ-ray irradiation.

While current models of dose response emphasize initial cellular events, many types of radiation effects can contribute to the process of tumor development. Some effects may play a role at the level of initiating events, transforming a normal cell into a cell capable of forming a tumor. Other effects, some operating at the cellular level and others at tissue, organ, or systemic levels, influence whether the carcinogenic potential of transformed cells is realized by modulating their expression. In fact, it may be that host factors play a major role in determining the expression of initiated cells. This suggestion is supported by observations in experimental animals. The incidence and spectrum of tumors in different strains of mice vary widely in nontreated and irradiated animals. Susceptibility to radiation within a strain varies with both age and sex and can be altered by environmental or endocrine manipulation.

In experimental studies, when low-LET radiation exposure occurs at a low dose rate, the carcinogenic effect of the radiation has been found to be reduced in virtually every instance (Ullrich, 1980; Ullrich and Storer, 1979b; NCRP, 1980; Ullrich, 1983). The primary effect of lowering the dose rate has been found to be a reduction of the effect in the intermediate dose range. This reduction diminishes the quadratic component of the dose response as would be predicted by assuming dose squared terms come from interacting independent events and makes the response shallow and more nearly linear over a wide dose range. In those instances in which it has been examined, the slope of the linear response following low dose rate exposures and the slope of the linear component from the linear quadratic dose response following high dose rate exposure have been found to be the same (see Figure 1).

When the radiation dose is fractionated the effects on tumor development can vary depending upon the total dose and the fraction size. If the dose/fraction is a dose which is on the predominantly linear portion of the dose response curve the response is similar to that following low dose rate exposures. If the dose/fraction is in a region where the quadratic component predominates the response may be similar to, or in some instances greater, than that for a similar single dose. This is illustrated in the following table:

Table 1

Incidence of lung tumors after a 2 Gy dose delivered as acute fractionated or protracted exposures

Exposure Regimen	Observed Incidence
High Dose Rate	38.6%
Low Dose Rate	21.4%
Low Dose Fractions	21.3%
High Dose Fractions	32.9%

High LET radiation (e.g., neutrons) is more carcinogenic on a dose for dose basis than low LET radiation (Upton et al., 1970; Ullrich et al., 1976; Ullrich et al., 1977). Recent data for tumor induction indicates a linear dose response following high LET radiation over the 0 to .25 Gy dose range (sometimes 0-.5 Gy). At higher doses the dose response tends to bend over. Because of the quadratic nature of the dose response in the intermediate dose range for low LET radiation and the linear response for high LET, the relative biological effectiveness (RBE) for high LET radiation increases with decreasing dose to a point where the low LET response also becomes linear (Fry, in press). At this dose level the RBE becomes constant. The value for this constant RBE for tumor induction varies with tissue. The carcinogenic effects of high LET radiation have generally been found to be less dose rate dependent than for low LET radiation. Dose rate independence is expected for cellular effects assuming that effects are produced by single tracks. Recently, however, data has suggested an enhanced tumorigenic effect following low dose exposures in certain tissues even at low doses (Ullrich, 1984). The general applicability of this observation is not known at the present time.

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A National Emergency Medical Assistance Program for Commercial Nuclear Power Plants

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ABSTRACT - Radiation Management Consultant's Emergency Medical Assistance Program (EMAP) for nuclear facilities provides a twenty-four hour emergency medical and health physics response capability, training of site and off-site personnel, and three levels of care for radiation accident victims: first aid and rescue at an accident site, hospital emergency assessment and treatment, and definitive evaluation and treatment at a specialized medical center. These aspects of emergency preparedness and fifteen years of experience in dealing with medical personnel and patients with real or suspected radiation injury will be reviewed.

Radiation injuries are classified according to those resulting from excessive exposure, those resulting from contamination and those resulting from combined exposures with or without trauma or serious illness. The contaminated and injured or seriously ill patient presents the only unique administrative and operational problem for emergency treatment. The proper management of these patients involves the use of procedures not normally utilized in emergency departments. Because of this, special planning, training and practice are necessary so that emergency care can be delivered without spreading contamination. Persons sustaining external exposure injuries can be evaluated and, if necessary, treated in the normal setting of an emergency room. However, the definitive evaluation and treatment of individuals who have received serious external exposures can tax the most advanced medical center.

Because of the infrequency of accidents and the complexity and cost to provide a complete medical program for the evaluation and treatment of radiation injuries, Radiation Management Consultants (RMC) established an Emergency Medical Assistance Program (EMAP) in 1970 to serve the commercial nuclear industry. This program is based on a regional approach to the management of radiation injury, with three levels of care: first aid and rescue at the nuclear facility, emergency treatment at a nearby hospital, and definitive evaluation and treatment at a large medical center. Unlike many medical emergencies, the clinical manifestations of radiation injury unfold over time - usually days to weeks. Consequently, patient care can be carefully managed and controlled at each of these 3 levels without impairment or detriment to the treatment and recovery of the patient. The nature of radiation injury is such that at each level of care priority can be given to more life-threatening trauma or illness that may accompany the radiation injury.

EMAP is a nationwide program coordinated and directed from RMC's offices in Philadelphia and Chicago. Currently it covers 27 nuclear power plant sites owned and operated by 20 utilities in 13 states across the country. RMC's EMAP staff includes full-time professionals in radiation medicine, nursing, health physics, radiochemistry, environmental science and emergency medical services.

A specialized accident radiobioassay laboratory, including mobile whole body counters, is available at both offices to assist the staff in dose evaluation and patient management. Associated with this in-house accident response capability are two definitive clinical care centers: Hospital of the University of Pennsylvania in Philadelphia and Northwestern Memorial Hospital in Chicago. EMAP maintains its close affiliation with these centers through staff appointments, teaching responsibilities and membership on the respective centers' Radiation Accident Coordinating Committees. In addition, the two offices maintain and exercise the twenty-four hour availability of a Radiation Emergency Medical Team (REM-Team) that can be dispatched to a plant site or local hospital to assist in the initial evaluation of the patient(s) and in the orderly transfer to one of the clinical care centers. A typical REM-Team consists of a radiation medicine physician, a certified health physicist, and radiation technicians who are supplied with emergency instrumentation. A calibration laboratory is maintained in Philadelphia to insure that instruments are properly calibrated and functioning at all times. A twenty-four hour "hot-line" connects each participating site and associated hospital throughout the country with the offices in Chicago and Philadelphia. EMAP participants are encouraged to use this line to obtain advice and/or assistance anytime a radiation injury occurs. RMC has long recognized that even minor problems can sometimes have medical-legal signficance.

In addition to providing a centralized emergency response capability, the EMAP staff assists personnel at nuclear power plant sites and associated hospitals in establishing and maintaining preparedness at the local level. This assistance includes the development of procedures, design of facilities, placement of equipment and supplies, training of personnel at plant sites in first aid and rescue of radiation accident victims, training ambulance service personnel in transportation; and training local hospital personnel in emergency treatment. To the extent possible, procedures, equipment and supplies are standardized at all sites. Specialized equipment has been designed to facilitate treatment and decontamination of patients while controlling the contamination to a designated area of the hospital.

Preparedness is maintained at all levels of care through training and drills. Since 1970, EMAP has conducted over 500 drills and exercises. This includes the development of scenarios, moulaging accident victims and evaluating and/or controlling the drills. Most of these drills are videotaped from the beginning of the simulated accident in the plant to the completion of the drill at the hospital. These video tapes are used in the critique at the hospital immediately following the drill and are subsequently edited with narrative for use by the plant and hospital personnel for interim training.

Because of EMAP's emphasis on providing the best patient care in the most effective manner possible, the entire program is further strengthened and coordinated through annual medical seminars and an educational newsletter, "Curie-Osity", special medical staff lectures, periodic meetings of the EMAP professional staff and monthly communication checks. The annual medical seminar, sponsored by one of the definitive clincial care centers, focuses on occupational health and current issues affecting emergency medical planning. This seminar provides an opportunity for medical staff and plant health physics personnel to exchange ideas and learn about recent accidents and developments in patient evaluation and care.

During the past 15 years RMC has evaluated and, when necessary, treated more than 300 individuals with real or suspected radiation injuries. Not all these patients were from commercial nuclear power plants. In fact, only one of the patients requiring medical evaluation and treatment for overexposure came from a nuclear power plant. Most of these individuals had radiation injury due to industrial radiography or medical radiotherapy sources. (One of the patients undergoing treatment, for example, received 100,000 rads to his thumb and two fingers over a period of eight years.)

Because of major involvement with nuclear power plants, RMC's personnel have developed considerable expertise in dealing with external and/or internal contamination problems. Probably the most complex contamination problem occurs when an individual becomes ill or is injured in a contaminated area of a nuclear facility. Fifty-nine incidents such as this have occurred in which patients were taken from a nuclear power plant to a nearby hospital for emergency treatment. Eighty-one percent of these cases involved injuries typical of those occuring in other heavy industries. These included falls (38%), injuries due to lifting and turning (19%), injuries caused by machine tool use (9%), injuries caused by chemical explosion/blast (6%) and others (9%). Contusions, lacerations, puncture wounds, open and closed fractures, soft tissue injuries, blast injuries, burns, eye injuries, skull fractures and central nervous sytems injuries have been some of the results of these accidents. The most serious injuries have been due to falls. These have resulted in two deaths and two instances of paraplegia (i.e., paralysis of the lower body).

Nineteen percent of incidents involved serious illnesses of endogenous and exogenous origins. In three instances individuals required treatment for cardiac problems. Problems associated with diabetes led to another needing hospital emergency treatment. Two deaths occurred both due to heart attacks. Illness of exogenous (environmental) origin is not uncommon, and severe heat stress is the chief culprit.

Personnel illnesses occurred "round-the-clock", while the majority of serious injuries occurred during off-hours. Protective clothing and respirators were implicated in accident and illness causation in several instances. (Loose clothing can be caught in rotating machinery, can lead to tripping, and contributes to retention of body heat. In addition, peripheral vision is lost with some respirators.)

Most of the accidents or illnesses that occurred in contaminated areas involved only one patient. Twenty percent of the instances involved two patients, but in no case were more than two individuals involved. In these incidents, the highest level of contamination on a patient was 20mR/hr. The highest level of attendant exposure from handling these patients was 14mR received over a period of two and one-half hours. In no case did the contamination compromise patient care. In no case was contamination spread beyond the controlled area in the hospital. Patients were taken to the hospital because of their injuries or illness - not because of their contamination (which tended to be at "nuisance" levels).

Although no serious radiation injuries occurred at TMI, RMC's EMAP provided needed assistance during and after the accident. Support was provided in laboratory analysis, whole body counting, counseling employees and their families, as well as members of the public.

In order to determine if a "systems approach" would be effective in the management of a multi-casualty situation, the accident at Chernobyl was carefully studied by RMC personnel. The Russian medical response system at Chernobyl appeared to be well planned. It, too, was based on a 3 tiered "regional approach", with on-site care, local support hospitals, and definitive care centers. It differed from the EMAP program in that a permanent staff of physicians, nurses and support personnel were available in an on-site "clinic" having 115 beds. Although taxed by the magnitude of the medical emergency situation, Chernobyl's two support hospitals and both of their definitive care centers responded appropriately to provide care to the sick and injured.

The strengths of RMC's EMAP lies in its full-time, in-house, twenty-four hour capability to respond and assist plant sites, local hospitals and definitive care centers in the complete evaluation and treatment of radiation injuries, in its association with large university medical centers, in its familiarity with personnel and specifics of nuclear power plants, and in its fifteen years of experience with radiation medical problems at nuclear power plants and facilities.

Section VII

Accidents and the Public

Chairman: *Russell Dynes*
University of Delaware

Evacuation Behavior in Nuclear Power Plant Emergencies: An Alternative Perspective

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ABSTRACT - It is alleged that the Governor's advisory at TMI led to spontaneous evacuation and overresponse. Several researchers have called this the "shadow phenomenon." This phenomenon, however, rests on assumptions that are challenged by this paper. Instead, it is concluded that evacuation behavior at TMI was consistent with our general understanding of human behavior in disasters. Emergency planners should be more concerned with underresponse to a nuclear power plant emergency than with overresponse.

I. INTRODUCTION

Since the accident at TMI, social scientists have been attempting to predict how the public would respond to a future nuclear power plant accident (Perry and Lindell, 1983; Johnson, 1983; Johnson and Ziegler, 1983; Johnson and Ziegler, 1984). Much of their attention centers on the issue of evacuation behavior. The central thrust of the research is directed towards resolving emergency response issues. The results of the research have significant implications for future policy decisions concerning the feasibility and content of emergency planning in the nuclear industry.

Several questions lie at the heart of this effort:

(1) How did people behave during the Three Mile Island accident?
(2) Why did people behave in the manner observed?
(3) Can we infer future behavior based on the observations at TMI?
(4) What will cause people to evacuate in a future nuclear power plant emergency?
(5) Are nuclear power plant emergencies a unique category unto themselves or may they induce behavior similar to other types of emergency?

The purpose of this paper is to present a different view of this issue other than that being currently advanced. In doing so, the paper attempts to sort out the answers to the questions. To initiate this debate, a review of the predominant line of current social science thinking about evacuation in nuclear power plant emergencies is presented.

II. THE "SHADOW PHENOMENON" AND OVERRESPONSE

A popular social science view holds that the experience at TMI demonstrated a unique phenomenon never before witnessed on such a large scale in an emergency situation. Researchers labeled this the "shadow phenomenon" (Ziegler et al., 1981). Stated simply, this phenomenon is that more people evacuated at TMI than was recommended in an advisory issued by the Governor of Pennsylvania. It is estimated that 144,000 people within a 15-mile radius left the area (Flynn, 1979). It is pointed out, 144,000 people within a 15-mile radius left the area (Flynn, 1979). It is pointed out, however, that only 2,500 people met the evacuation criteria set by the Governor. Those criteria were that the person either be pregnant or five years of age and under, and live within a five-mile radius of the plant. Thus, the conclusion reached is that 141,500 people "overresponded" to the advisory; that is, evacuated without being told to do so by the Governor's advisory. Presumably, the "shadow" extends for some distance away from TMI, that distance being established by the presence of evacuees. Based on this rather arbitrary way of measuring proper response, it is concluded that people will "overrespond" to a nuclear accident. Overresponse is synonymous with taking action when no recommendation from an official source is made. Often this is referred to as "spontaneous evacuation." A corollary of the shadow phenomenon theory is that spontaneous evacuation, hence, overresponse will always occur when a warning is issued for a nuclear plant accident due to people's innate fear of radiation. Currently, this theory is being promoted in interventions concerning emergency preparedness at Atomic Safety Licensing Board

Hearings on several nuclear power plants. It is the thesis of this paper that this theory is misleading and based on unfounded assumptions. Adoption of this view may actually jeopardize public safety if an emergency does occur.

III. ASSUMPTIONS OF THE "SHADOW PHENOMENON"

Advocates of the shadow phenomenon rest their proposition on several assumptions that are never explicitly defined. These assumptions warrant identification

A. The Information Assumption.
For this "overresponse" to occur, it is implicitly assumed that the Governor's advisory was the only information reaching the public that could be validly used to make a decision to evacuate. Since the advisory provides the conditions by which overresponse is measured, it can be construed that in order to be "rational" people could only respond to the advisory.

B. The Insider Assumption.
The phenomenon assumes that "overresponse" is an objective condition that can be measured using universally accepted criteria. In doing so it implicitly assumes that the insiders, that is, the people who are part of the system under stress, hold the same definitions of appropriate response as those who observe response after the emergency. It also assumes that insiders define risks and risk levels in the same way as the outsiders.

C. The Knowledge Assumption.
The shadow phenomenon assumes that the language for presenting the criteria and information in the Governor's advisory were compatible with the knowledge systems of people receiving the information. That is, people knew where the 5-mile radius was, people could distinguish the health risk to pregnant versus non-pregnant women and could comprehend the use of five years of age as an evacuation criterion.

D. The Individualism Assumption.
The shadow phenomenon assumes that people make decisions and behave as individuals within an undifferentiated social space. That is, that social groupings play no role in behavioral response to an emergency.

How valid are these assumptions? In the next section I attempt to provide an answer to this question by addressing some of the basic questions posed in the introduction to this paper.

IV. ANALYSIS OF THE ASSUMPTION

A. The Information Assumption.
While it is true that more people evacuated than was recommended by the Governor's advisory, their actions are consistent with scientific knowledge on how people behave in disasters. People at TMI responded to information available to them at that time, and their behavior is understandable given the nature and content of that information.

First, the Governor's advisory did not tell the entire population what to do, only a specific subgroup. Without information directed at other groups not included in the pregnant women and small children categories, it is quite inaccurate and misleading to say that people "did other than what the Governor's advisory suggested." Without information from the Governor they acted on the basis of other information. Support for this is found in the NRC telephone survey (Flynn, 1979). Only 14% of those who evacuated said that the Governor's advisory was critical in making their decision. Thus, they made decisions on the basis of other factors.

Of the families that did not evacuate 71% stayed because they were not told to leave. Thus, many who stayed were waiting for guidance from official sources, and many who left acted on sources of information that filled the gap created by the lack of information in the advisory. Had the Governor's advisory given information to other groups of people, a different pattern of response would have likely occurred.

The theory of spontaneous evacuation assumes that the Governor's advisory was the only information people acted upon. This simply was not the case. According to the NRC survey the major piece of information that was critical in deciding to evacuate was the situation with the hydrogen bubble (30%). People did not behave contrary to the advisory, but acted on the basis of their situational perceptions of the accident.

Further evidence that raises questions about the validity of this assumption is provided by a study of news media coverage of the accident (Stephens and Edison, 1982). The data in the study by Stephens and Edison show splits in both local and national news coverage between reassuring and alarming statements on a range of safety issues including emergency preparedness, evacuation, radiation exposure, reliability of information, plant conditions, the chance of a meltdown, and the hydrogen bubble. Some news sources were stating the need for a "total evacuation of the immediate area" and many emphasized the likelihood of an explosion or meltdown.

Furthermore, people were exposed through the media to the discusisons of other recommendations and advisories by various officials. For example on Thursday, March 29, the second day of the accident, a radio announcer mistakenly interpreted an interview with Dr. Steinglass of the University of Pittsburgh as an order for pregnant women and preschool children to evacuate rather than a cautionary recommendation. Friday morning, the Dauphin County Civil Defense Director informed the public that an evacuation could be ordered; however, directions on where to go and what to do also accompanied the statement. That same morning the Governor made a 10-mile sheltering recommendation. Later in the day the official evacuation recommendation was made. On Saturday, the Nuclear Regulatory Commission (NRC), one of the most credible information sources on the accident, went on record in a news conference that a 20-mile evacuation order was being considered.

In light of this evidence, the Governor's advisory is a poor standard by which to judge the appropriateness of response and was fairly limited in its influence on evacuation behavior. People responded to the range of information in the system. Furthermore, it appears that it was the uncertainty of the information that encouraged evacuation and not deep-seated fears of radiation.

B. The Insider Assumption.

The preceding argument underscored the inappropriateness of using arbitrary criteria to judge behavior. Even if the imposition of a 5-mile/pregnant women and small children criterion is accepted as valid, the shadow phenomenon still assumes that the response of the people to a disaster can be judged as being correct or incorrect by people who did not experience the disaster. This outsider, however, has different information than the insiders did at the time of the emergency. Often the situational factors that may lead to evacuation cannot be captured in the research methodologies used to measure and understand behavior. Risk perceptions of the outsider do not necessarily mesh with those of the insiders.

Some empirical support for the questionable validity of this assumption is provided in a follow-up study of the TMI residents (Houts et al., 1980). In a reinterview of a sample of people drawn from the NRC survey sample, respondents were asked if they would evacuate if a similar accident occurred. The percentage of people indicating they would evacuate was similar to the actual levels of evacuation behavior in the accident. If people had thought they had overresponded we would have predicted much lower percentages for response to this behavioral intent question. It seems likely, therefore, that researchers who judged people to be overresponders simply failed to appreciate the cultural and social context of the emergency. At the time TMI presented distinct risks with great uncertainties. To use an analogy, if the possibility of a dam failure confronted flood plain occupants would it be valid to label people making precautionary evacuations as "overresponders" when the dam did not fail? Would it be due to deep-seated fear of water? Unlikely. The mystique of radiation to researchers is probably far greater than for excess precipitation.

C. The Knowledge Assumption.

The shadow phenomenon assumes that people, in arriving at an evacuation decision, understood the context of and basis for the Governor's advisory. Research shows, however, that the public has a poor technical grasp of nuclear power and ionizing radiation concepts (Reed and Wilkes, 1976). Furthermore, the public generally fails to even distinguish between various stages of the fuel cycle or between peaceful and military applications of fission technology (Lindell et al., 1978). Thus, it is difficult to believe that the public would grasp the full meaning of differentiating two high risk groups in the evacuation recommendation. While it may be of societal value to protect pregnant women and children, it usually is used as a means of prioritizing protection and not to exclude other groups from a margin of safety.

Furthermore, while it is much more likely that people understand the concept of "miles" of distance, it does not necessarily follow that they are always good estimators of distance or view the five miles as a precise boundary. Considerable research has been conducted on how people perceive their environments and spatial organization of these environments (Golledge and Moore, 1976; Saarinen, 1976; Rapoport, 1977; Lynch, 1960). One finding from these studies is that people are often poor estimators of distance and, certainly, variable in the way they bias distance estimates. Moreover people do not always organize space in terms of concentric radii. Rather they use roads, landmarks, natural boundaries and other natural and human features of the landscape as reference points. In addition, it is unlikely that even with a good reference point for the five-mile boundary, the radius was judged by people to be a fixed or accurate barrier. People may not perceive that airborne radiation would somehow obey this artificial designation of a barrier that the Governor implied was intransversible by radiation.

D. The Individualism Assumption.

The shadow phenomena assumes people will evacuate as individuals and not in a larger social context. This simply is not borne out by a large range of evacuation studies in a variety of disasters. The propensity of most people is to evacuate in a group context (Drabek, 1969; Drabek and Stephenson, 1971; Perry et al., 1981;). For example, the Drabek studies indicated that of families intact during a flood disaster, 92% evacuated together and 64% of those apart evacuated after reuniting. In other situations people also behave in a collective or group context that does not necessarily involve families but friends, neighbors, or strangers as well (Gruntfest, 1977).

In the case of TMI it is difficult to assume that people meeting the evacuation criteria imposed by the Governor would behave as individuals. A preschool child can hardly evacuate without assistance; an older sibling would not be required to stay at home; and husbands and pregnant wives would be unlikely to part. While it has been acknowledged that in a family context there would have been at least 10,000 legitimate evacuees (Perry and Lindell, 1983), this still ignores the effects of extended family evacuation, peer group evacuation, and the influence of social processes in evacuation.

To gain a better picture of family evacuation at TMI, we reanalyzed the NRC survey data (Flynn, 1979) to estimate how many households behaved as family units and how many were fragmented by the evacuation. The results show that fragmentation was highest within 5 miles of TMI and decreased with distance from the plant. This suggests the evacuation recommendation may have contributed to more fragmentation than would normally be expected. Still, the majority of families did behave as single units. Furthermore, as is documented, a significant correlate of evacuation at TMI was having a friend or neighbor do so (Cutter and Barnes, 1982).

V. AN ALTERNATIVE EXPLANATION

The evacuation at TMI is largely explainable in light of scientific knowledge of how people respond to warnings of an impending threat (Mileti, 1975; McLuckie, 1973; Mileti et al., 1981; Quarantelli, no date). The two major factors which seem to consistently affect behavior, regardless of hazard, are the situational perception of risk and emergency information. The self-reported data in the NRC survey support this pattern. The two major reasons cited by TMI evacuees for leaving were "the situation seemed dangerous" (91%) and "information on the situation was confusing" (83%). At the time, people were hearing information that would lead them to believe their safety was threatened. They were also hearing vastly conflicting information about the risks, what was being done to remedy the situation, and what they should do. The confusing information was a signal that emergency managers were not controlling the accident. Evacuation was a prudent response that did not need to follow from an "official order." A sequential or lexiographic model of decision making in response to warnings provides an alternative explanation to spontaneous evacuation due to deep seated fear of the "shadow phenomenon" model.

In a simplified form, this model suggests the following factors play a sequential role in determining response: awareness of a risk, belief that the risk is real, personalizing the risk, evaluating alternative actions, and deciding on a course of action. At TMI, a stereotypical pattern of thoughts behind a decision to evacuate could be characterized as follows:

(1) A person became aware that a problem existed.
(2) That person sought more information on the accident situation.
(3) Information conveyed a sense of personal threat to that person.
(4) The situation was not only threatening, but it was unknown whether the hydrogen bubble would explode.
(5) A good means of protection was to get away from the reactor.
(6) No other form of protection was seen to be effective.
(7) The weekend situation gave that person mobility to evacuate.
(8) Other people were perceived to be evacuating.

This decision process reflects a normal and predictable course of behavior. People did not leave merely because they dread radiation. Moreover, they did not panic. In fact, the movement was slow and drawn out over a four- or five-day time frame, hardly what could be described as spontaneous. In sum, people responded in a way consistent with the disaster situation that was conveyed to them.

VI. TMI AND PREDICTING EVACUATION BEHAVIOR

Based on the evidence presented in this paper, it is certainly possible to use TMI to predict some patterns of behavior in a future accident. This is feasible because TMI was not a totally unique phenomenon. Rather, the patterns of response were largely consistent with the general premises of human behavior in emergencies. People behaved in a manner consistent with the information they received, the situational perceptions of risk they formed, and the social context of the accident situation. The notions of "shadow" and of "overresponse" are largely unfounded and best interpreted as creations of researchers rather than any social reality. This conclusion is both encouraging and disheartening. The conclusion that human response to a nuclear accident is basically similar to a range of other types of emergencies is encouraging because we know from experience how to issue warnings to minimize the problems that were experienced at TMI. While the inherent uncertainty in that particular situation made emergency management functions difficult, the mere fact that a warning was issued fails to account for the uncertainty and confusion. An accident at the Ginna Nuclear Power Plant outside of Rochester, New York, in 1982 helps underscore the importance of an effective warning system. Despite a release of radiation, no evacuation ensued. This can be attributed in part to the successful response to the steam tube rupture in which operators returned the plant status to safety. It is also attributable to more effective communication links to the local emergency response network and the public, and, to the issuance of concise and credible information. If evacuation behavior in a nuclear power plant accident were governed by innate fears of radiation, some evacuation would have been observed at Ginna regardless of emergency management effort. Instead, the warning system served to guide behavior in a manner consistent with the risks.

On the other hand, our conclusions are somewhat disconcerting, as well. If emergency response to a nuclear power plant emergency is subject to the same problems encountered in other emergencies, a truly serious accident will result in human casualty. This is far more likely to result from failure to heed a warning than from overresponse. Future litigations over emergency planning after a serious accident, assuming a viable industry, would concern underresponse, not a shadow of fear. Emergency planners should take steps to develop effective warning systems which seek to minimize both overresponse and underresponse. This is feasible. Attempts to deal solely with a shadow phenomenon could have deleterious effects. That shadow is invisible; the consequence of non-response will not be.

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Warning Human Populations of Technological Hazard

George O. Rogers and Jiri Nehnevajsa

ABSTRACT[1] Warning people of an impending hazard seeks to make them aware of the threat and to elicit actions that would minimize the dangers to life and property.[2] Because technological and natural hazards differ in important ways, the alerting and notification process for technological and natural hazards is also different. One of the differences rests in the ability of people to detect many natural hazards in a direct sensory manner; technological hazards often make such detection more difficult. For example, detection of radiological releases without instrumentation is nearly impossible, but even with tornadoes where warning is notoriously difficult, people are at least able to use their senses to detect the potential for hazard. Hence, warning for technological hazards is in some ways more problematic, generally representing a rather rapid shift from normalcy to emergency. This paper builds on the significant foundation of natural hazard warning research in developing a model of warning suitable for technological hazards. This model specifically examines immediate cascading, or networking, of the warning signal and message, so often reported in the natural hazard literature. The implications for technological and natural hazard warnings systems are examined.

I. INTRODUCTION

Warning people of an impending danger may be partitioned into two somewhat distinct aspects. The first deals with alerting the public that something is wrong, that some hazard is imminent. The second concerns the ability of emergency officials to communicate the warning message to prompt appropriate action. The primary issues of alerting revolve around the ability to make people aware of the threat. This alerting often involves the technical ability to develop, construct, maintain and use a warning system, which may consist of sirens, bells, whistles, television and radio broadcasts, telephone systems and even social organization. The primary notification issues center on the public's interpretation of the warning message. The interpretation of the warning message is fundamentally important in the selection of appropriate action in response to warning. The focus of this paper is on the social processes associated with alerting the public to potential danger.

Hazards are broadly cast as technological and natural. Technological hazards are generally characterized by the failure of a technological system(s). While technological hazards are generated by human interaction with the environment, natural hazards may be viewed as "acts of God." Hence, questions of culpability are often associated with technological hazards. The developers and operators of the technological system become legally responsible for the safety of its operation. Such culpability is seldom attributed to natural disasters. At least for some kinds of technological crises the potential impact area is predictable. While even the best meteorologist has difficulty determining the pathway(s) and point(s) of impact for an approaching tornado, hazardous facilities that are geographically fixed, thus there is the advantage of being able to establish emergency planning zones in proximity of the fixed facility. On the negative side, at least some technological hazards are less detectable than any natural hazard. Nuclear exposures, for example, are not detectable by any of the five human senses, some toxic chemical releases are similarly undetectable. However, other technological hazards make possible the use of many of the same detection criteria as natural disaster threats. For example, the danger of dam failures is often brought on by heavy rains and results in flooding that is detectable by the same mechanisms as other forms of flooding. This paper focuses on the social aspect of the alerting process, primarily for hazards of a fixed, or at least known geographic location. However, the social principles applied to these fixed geographic locations apply equally well to other technological hazards and even natural hazard alerting situations.

II. BACKGROUND

Warning messages pass through a variety of pathways which may color their meaning. Some of these pathways involve cognitive functions, others have to do with social structural considerations. An individual's interactions with others form social networks. Even though these networks have many forms, their routine and established nature has led to widely accepted empirical generalizations about how they function in society at large (e.g., Parsons 1951, Coleman et al. 1957, Granovetter 1973, Blau 1977) and in particular how they function during emergency warning. Two general propositions are strongly supported by the disaster literature (Williams 1964): First, that people respond to emergency warnings in a context of their prior experience, extant social and physical environment and existing conditions which interact with the warning message. And second, that the degree to which the warning message is received depends on the nature of the message, taken in the context of the social network, and the prior behaviors of all social actors in processing such information. Hence, people in social networks in specific locations have extant estimates of the threat presented by the environment in which they live. These estimates and their experience vector provide the data base from which the selection of behavior is derived--the decision to accept, ignore, disseminate, challenge, or confirm the warning message (Baker 1979).

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[2]The former is the alerting function of warning systems, and the latter constitutes their notification function.

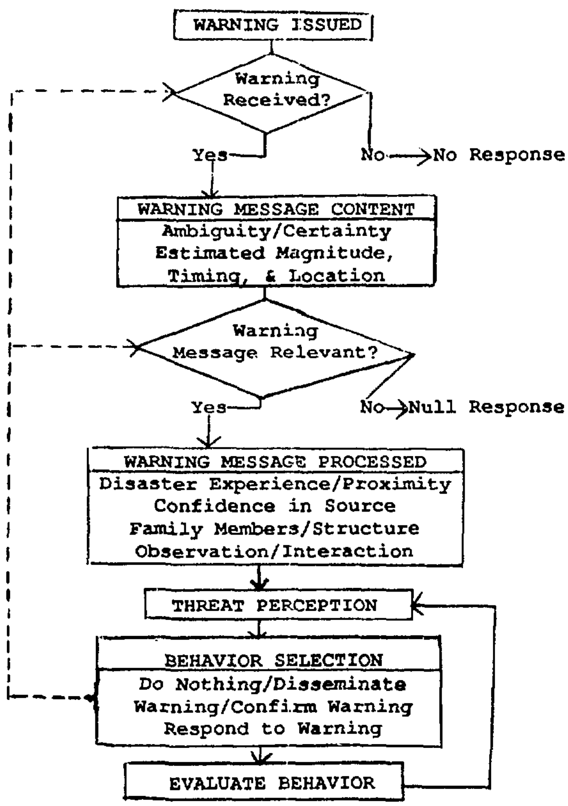

Figure 1 - Warning Process

Emergency warnings may result in the recognition of threat which creates a psychological discomfort. One important mechanism to alleviate this involves efforts to reduce uncertainty. The warning process (Figure 1) involves both factors affecting the message and characteristics of the receiver. Once the warning is received the content is evaluated in terms of the certainty and ambiguity of the estimated severity, timing and location of impact. Essentially, "Is it likely to effect me? When will it occur?" The evaluation of the warning message results in the determination of its relevance. If the message content is deemed irrelevant, no emergency response is likely. However, should the warning message be considered relevant, the message is further processed in the context of prior disaster experience, relative proximity, confidence in the source of warning, interpretation and discussion with members of the social network. The warning message is processed in the context of the existing social structure, which results in, at least, the initial perception of threat. The cumulative process provides the foundation for the selection and evaluation of emergency behavior.

The social process which is then triggered also serves to further disseminate the warning message. When an individual receives, recognizes, verifies and believes the warning message and deliberately disseminates it to others in the social network, a purposive warning dissemination takes place. An incidental warning takes place when the individual in the process of seeking confirmation of the warning message, inadvertently gets in touch with someone who has not been alerted yet. Warning confirmation and dissemination through the social network helps warn previously unwarned people, if indirectly, and confirms the meaning of the original warning for those having received it previously. "Instead of trying to stop...people [from] calling one another...ways ought to be found to take advantage of such calls so as to improve the dissemination of warning messages..." (Kendrick 1979:346). Furthermore it should be added that both confirmation and dissemination of warning through the social network improve response to emergency warnings (Quarantelli and Dynes 1976, Perry et al. 1980, Drabek and Boggs 1968, Mileti and Beck 1975, and Rogers and Nehnevajsa 1984).

Effective warning messages have been described by Janis (1958) as requiring a balance between fear-arousing and fear-reducing statements. Fear-arousing statements describe the impending danger in sufficient detail as to evoke vivid mental images of the crisis, reducing the possibility for surprise as the disaster evolves. Fear-reducing statements realistically present the mitigating factors of the impending situation, and provide information regarding realistic actions to be taken by authorities and individuals, both independently of one another and jointly. The fear-arousing content of the warning message alerts the public to the potential for harm, while the fear-reducing statements consist of notification of appropriate mitigating action.

III. THE ALERTING PROCESS

People are alerted to the potential for danger by a variety of sources. These sources of warning are broadly classifiable: warning by authorities, from the mass media, and those transmitted through the social network. Drabek (1969) and Perry et al. (1981) refer to warning from authority as those messages which are generated from and disseminated by emergency-services organizations (e.g., police or fire departments, civil defense organizations or the national guard), while mass media warnings usually come from radio and television, although in slowly-evolving disasters, the print media also play a crucial role. The social network provides warning through relatives, friend and neighbors. Perry et al. (1981) report that 41.2% of first warnings for riverine floods in four communities came from authorities. While there was apparently no time for mass media warnings in two communities, the mass media accounted for 8.1% of warning alerts. Nearly half of the first warnings in these four communities stemmed from the social network--37.6% from friends and neighbors, and 13.0% from relatives. While the distinction between social network alerts that purposely disseminate the warning, and those that incidentally warn others is not possible on the basis of the present evidence, the receipt of first warning is often made through personal contacts with members of the social network (Perry 1981, and Mileti 1974).

Around Three Mile Island more people received their first warning (were alerted) via social networks than expected to receive such warnings. Flynn (1979) found that only 6% expected to be alerted by friends, neighbors and relatives, but Brunn et al. (1979) and Barnes et al. (1979) report 18% to 25% actually claimed to have received their first warning from social network sources. Of those people receiving warning on the first day of the accident, 22% were alerted through the social network, and for those with the highest saliency (living within 6 miles), 43% were alerted by people in their social network (Barnes et al. 1979). This is consistent with other research which suggests that "word-of-mouth" warnings are more likely among people most likely to be affected by the impending danger (e.g., closest to the threatened area (Diggory 1956). Hence, for fixed-site technological hazards, where the saliency for nearby residents is fairly clear, social networks may be more effective than in situations where the proximity of hazard is less clear. In natural disasters, in which the probable impact area can be ascertained reasonably well, significant proportions of people are also alerted through the social network. For example, Perry (1981) reports 31.7% and 38.6% of the people received their first warning from others in their social network in connection with the volcanic activity and floods respectively.

The overall emergency alerting process can be considered as comprised of two basic processes. The warning alert process determines the capability of the warning technology (e.g., sirens, bells) to deliver the warning message to the public. The effectiveness, of course depends on factors of the physical environment and the system technology, both constrained by natural laws. Siren sound coverage, ambient noise levels, warning signal attenuation, biological hearing capability, acoustical properties of the alerting signal are among the salient considerations. To the extent that human activities alter such parameters, such as sleeping, operating equipment or listening to music, social behavior is clearly critical to the actualized initial receipt of the message. The dissemination of the warning alert takes place through the household and neighborhood alert processes. The household or "area" process involves the intra-household dissemination of the warning message, while the neighborhood process represents the inter-household dissemination of warning.

The warning alert process results in some households being completely alerted by the initial warning signal, others may have at least one person alerted, and in some households no one will be alerted by the initial warning signal (Figure 2). The household alerting process characterizes the distribution of the warning signal within a household that is partially alerted by the initial warning signal (i.e., at least one person). Households where everyone is alerted, either by the initial warning signal or through intra-household dissemination, become the potential warning message transmitters in the neighborhood process.

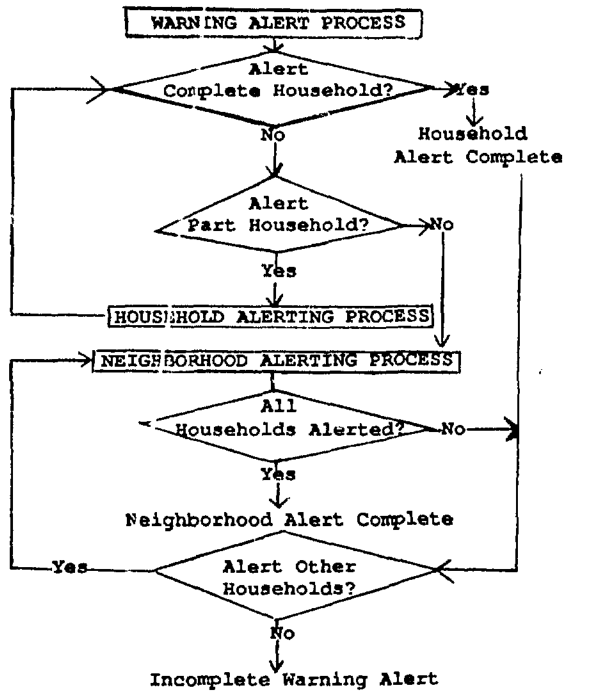

Figure 2 - Alerting Process

Households that initially remain unalerted may receive the warning through the neighborhood alerting process. The household alerting process is consistent with family re-unification for household emergency response (Rogers and Nehnevajsa 1984, Frazier 1979, and Drabek 1969), which finds that households prefer to respond to crises as a unit. Both the household and the neighborhood alerting processes provide consistent confirmation of warning that often takes place during emergency warnings (Rogers 1985).[3]

IV. WARNING ALERT MODELS

Considering a late-night (i.e., 12 midnight to 6 a.m.) warning, Nehnevajsa (1985a) incorporates three major factors in assessing warning alert: 1) The effect of non-sleeping activities, 2) the effect of intra-household networking, and 3) the effect of inter-household networking. Activity probabilities are based on a detailed study of time use conducted by the University of Michigan (Juster et al. 1983) which results in late-night probabilities of being awake of .092 and .058 between midnight and 2 a.m., and 2 a.m. and 6 a.m. respectively (Humnon et al. In Press). Intra-household networking is considered on the basis of household by size and composition, peak alerting signal levels as a function of distance from the warning signal source, attenuation rates for different types of houses and residential conditions. Somewhat conservatively assuming that about half of those alerted by the initial warning signal will make a single contact with another household (even though 87.5% of respondents in a recent University of Pittsburgh study expect their neighbors to contact others, even in the middle of the night, to warn them of impending danger), initial "acoustic" alerting of 69.0%, would be augmented to 79.3% of the residents, given these basic considerations. A second acoustic signal, resulting in 72.8% of the people alerted would be enhanced to 82.2% alerted.

Even an elementary model, which only accounts for arousal probabilities by household size significantly reduces the proportion of people left unwarned (Nehnevajsa 1985b). Assuming that only one in four people alerted by the initial signal would attempt to make contact with others in their social network the initial warning signal leaves 15.5% of the people unalerted. However, a single social network contact decreases the proportion of unalerted people to 11.6%, and a second networking attempt reduces the unalerted proportion to 8.6%.

One significant limitation of these models revolves around the timing of the networking process (Landry and Rogers 1982 and Nehnevajsa 1985b). The warning process that incorporates both the initial alerting system, which is technologically (e.g., sirens, bells, television, or radio) based, and diffusion of warning through the social network, involves initial alerting via a "broadcast" process,[4] and subsequent alerting via a "birth" process[5] (Lave and March 1975). Both processes are time oriented (t) and limited by the size of the population to be alerted (N). The broadcast and birth processes are represented respectively by

$$dn/dt = a_1(N-n), \text{ and } dn/dt = a_2 n(N-n),$$

where n, is the alerted population at the beginning of each period (t_0, t_1, t_2..., and a_1 and a_2, are proportions summarizing the diffusion properties of the respective processes. Combining these two processes into a single warning system

[3]This paper does not examine the significant issues associated with the notification process, which involves the belief and interpretation of the warning message and the selection of behavior. The significant issues of behavioral contagion, will people take action independently or in conjunction with the behavior of friends, neighbors and relatives, and the effect of source of warning confirmation, or type of warning notification on behavior selection are excluded.

[4]This process rests on the broadcast of the warning message by technical mechanisms, such as television, radio, sirens, bells, whistles or a combination of these specific technologies in combination with organizational assistance. It is referred to as broadcast because the message is broadcast from a relatively centralized source to the public.

[5]This process rests on the dissemination of the message among people. It relies on a less centralized warning dissemination, where each recipient passes or at least attempts to pass the message to others in the network.

$$dn/dt = k\{a_1(N-n)\} + (1-k)\{a_2n(N-n)\},$$

where k, is the proportion receiving the warning alert signal, and (1-k) represents the proportion not alerted by the broadcast signal.

Using this classical model of the dissemination of warning, the timing of warning over the initial period can be examined. Suppose the broadcast warning system only operates in the first three minutes of the warning period, even though no warning system that we are aware of operates only in the first few minutes without being reactivated in later periods. Further suppose that k is equal to the proportion of non-sleepers. This is equivalent to saying that arousal from sleep need not be considered for those that are not asleep. Finally consider a broadcast process efficiency (a_1) of only .5, and a birth process effectiveness (a_2) of .3. This broadcast efficiency is well below the acoustical warning rate reported by Nehnevajsa (1985a), and the contagion effect is substantially below people's expectations and reported incidents. Even assuming these conservative system parameters, the warning system alerts 76.2% in the dead-of-night (2 a.m. to 4 a.m.) in the first 15 minutes (Figure 3).[6] Given the drastically larger proportion of non-sleeping people between 8 a.m. and 10 a.m., and 8 p.m. and 10 p.m., the proportion warned exceeds 80% in the initial periods of warning, resulting in approximately 88% being alerted in the first fifteen minutes. Given quite different broadcast alerting probabilities, reflecting the period of the day differences, the results at the end of fifteen minutes are remarkably similar, but the trajectory within the period is very different. Hence for technological hazards with various onset times, the broadcast system requirements will be somewhat different. For technologies either with long onset times or where warning systems can be activated early, systems may place greater dependence on the birth process in emergency warning.

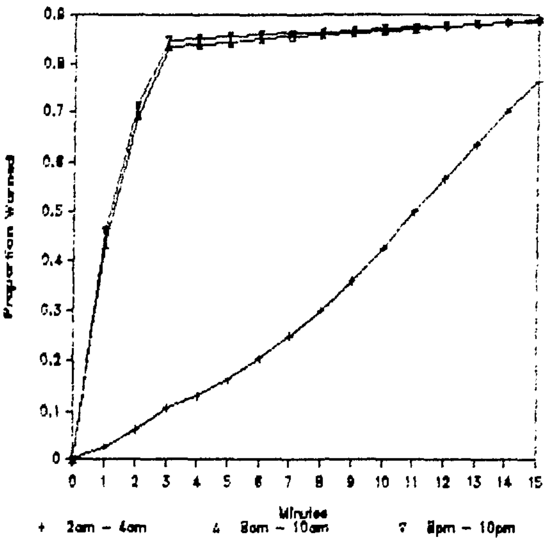

Figure 3 - Timing of Warning

V. POLICY IMPLICATIONS AND CONCLUSIONS

Social network alerting deserves full recognition as a valid aspect of the overall emergency warning process. It cannot be assumed that everyone, especially younger children, is able to properly interpret the meaning of the warning signal, even if it is "heard," and thereby recognize the impending threat. Therefore, the cascading effects of networking become an integral part of the system. The dissemination of warning provided by the cascading of the warning message through the social network significantly enhances coverage of the warning alert. Hence, emergency warning systems can effectively alert residents in adjacent areas by taking advantage of the social network dissemination of warning. This is particularly true for fixed-location technological hazards, such as nuclear power plants. However, it is incumbent upon risk managers of such facilities to increase public awareness of the potential for hazard, ability to recognize and interpret the alerting signal, and awareness of what actions to take.

All emergency warning systems take advantage of both an alert signal and a further dissemination of the warning through the social network. The trade-off between the two processes rests on considerations of cost and timing of adequate coverage. Because the birth process depends on alerted people to disseminate the warning message, the more expensive broadcast of emergency warning is inherently faster. For hazards with onset trajectories similar to hurricanes, the warning system can place more reliance on the networking process. Relatively slowly evolving emergencies not only provide time for the social networking process to be highly effective, but these hazards also allow people to become attuned to the impending hazard, which "pre-charges" the network for further alerting and notification as information about the hazard gets to be more intensive, and the danger becomes more acute. On the other end of the hazard spectrum, rapidly evolving hazards require greater reliance on a broadcast system, even though such a system can never be completely effective on its own. Hence, one key factor in determining the extent to which the less costly social network dissemination of warning can be employed concerns the technologies which permit an early detection of particular hazards. Can the hazard be detected with sufficient lead time to alert the public? Another factor in the selection of an efficient emergency warning system (i.e., obtaining coverage with an optimum mix of the broadcast and social network processes) is the reliability of early warning, and appropriate policy decisions to warn at early stages of a possible disaster. This involves another trade-off between the issuance of early warning and the probability of a false alarm.

To the extent that there is actable time, any warning system can be improved, in the sense of alerting more people with less time, both through improvements in the broadcast system and by enhancing the social network process. The broadcast system can alert more people by enhanced coverage (e.g., louder signals, or more complete distribution of warning devices among the population). While these system improvements are desirable, the social network process can also be enhanced by encouraging people to contact others when they are alerted. By encouraging people to become involved in the emergency warning process, emergency preparedness beyond better warning is improved, because people are more likely to develop an understanding of the potential hazards, the nature of potential threats, the kinds of available protective actions needed, and take an active role in assuring their own safety, and in enhancing the safety of their relatives, friends and neighbors.

[6]Minutes is perhaps better described as steps, inasmuch as we remain uncertain about the exact duration required for message passing. While given adequate saliency household contagion probably take less than a minute, message passing in the neighborhood process is less certain. Hence, what we label as minutes, reflects steps that probably range in duration from somewhere near 15 seconds upwards to 3 to 5 minutes The actual duration of these time intervals is almost certainly dependent on the nature of the hazard, its saliency, timing and action requirement.

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Emergency Preparedness Training for Local Communities

Michael J. Cooley and Karen K. Thompson

ABSTRACT

Detroit Edison, in cooperation with Monroe County, has developed a comprehensive training program for local emergency workers in the area surrounding the Fermi 2 Nuclear Power Plant. Using expertise from both organizations, a program consisting of two videotapes, two slide-tapes and nine narrated slide series was produced to address the worker-specific training needs of county emergency workers. In June of 1985, the program was approved by Detroit Edison and the Monroe County Board of Commissioners. To date, Monroe County has trained more than 1,000 emergency workers.

This program has been so well received that the county staff has developed and presented a modified version of this program to the general public. The result of this cooperative effort is increased public confidence in emergency preparedness at the state, local and utility level and a renewed spirit of cooperation and trust between the utility and local units of government.

Over the last few years, the issue of emergency planning has received increasing attention, particularly in communities which surround nuclear power plants. Although local governments in Michigan are responsible for developing an emergency plan, the responsibility for approving training programs for emergency workers in local communities rests with the state. This division of responsibility sometimes requires a high degree of cooperation by all concerned to develop a comprehensive, yet community-specific program.

According to Appendix E of 10CFR50, a radiological orientation training program must be made available to off-site emergency workers. In the State of Michigan, the Emergency Management Division of the State Police is the responsible agency for ensuring that emergency workers are trained and prepared to handle emergencies relating to an accident at a nuclear power plant. The State has developed a four-part training program using NUREG-0654 as a guide. The program encompasses basic information on the operation of a nuclear power plant, effects and detection of radiation, and the emergency plans at the state, local and utility level. This program was implemented in Monroe County in 1984. Monroe County is the location of Detroit Edison's Fermi 2 Power Plant, 30 miles south of Detroit. Fermi 2 is powered by a boiling water reactor capable of generating 1100 megawatts.

The state-developed training program was originally presented jointly by state police and radiological health personnel, the local emergency preparedness coordinator and utility personnel. Overall, the program length was nearly four hours--much too long for the purpose intended. Furthermore, the program did not address specific duties and responsibilities of the local emergency workers.

To improve the program, Detroit Edison and the Monroe County Office of Civil Preparedness began discussions to explore alternatives. The first, and perhaps the simplest, was to revise the existing program. The county wanted a program with a new focus, one which related directly to <u>their</u> emergency plan and addressed <u>their</u> needs. A second alternative was to engage an independent consultant to analyze the needs of the county and to develop and deliver a program. The local coordinators approached the consultant who had assisted in the preparation of their emergency plan. The consultant offered a package which included a stand-up instructor, supplemented by slides and overhead transparencies.

There were several disadvantages that developed in exploring the use of a consultant for this project. For one, the consultant proposed a two to eight hour presentation. This was considered again to to be much too long by the county coordinators. In addition, the instructors from the consulting firm were strangers to the community. The Office of Civil Preparedness wanted to increase their involvement with the emergency workers and the public by making the presentations themselves.

Another disadvantage in having instructors from a consulting firm was their limited availability. The county wanted an instructor who would be able to schedule and conduct presentations throughout the year.

Finally, the contract proposed by the consultant was for development and initial implementation of the program. Any revisions and presentations in subsequent years would be at an added cost. The county wanted to minimize their long-term reliance on outside assistance. Ideally, one of their own staff, an individual familiar with the emergency plan and with the local community, would be best suited to make the presentations.

The Monroe County emergency planners decided on another alternative--to ask Detroit Edison to work with them in developing a new program. The emergency planning group at Detroit Edison had a close working relationship with the staff at the County Office of Civil Preparedness. The utility met with the county staff and agreed to work cooperatively on the new program. The county staff provided experts on their emergency plan throughout the development phase of the project. The Fermi 2 staff provided technical expertise on plant operations and basic radiation protection. Detroit Edison also provided overall project scheduling and coordination of production activities including photography, videotaping and artwork.

In planning the program, Detroit Edison and Monroe County personnel first agreed upon some parameters. The County wanted a program no more than one hour and 20 minutes in length. They also wanted it tailored to specific groups of emergency workers, to appear realistic and be a high quality production. The realism was achieved through extensive use of local workers, equipment and facilities in producing the program. The quality was heightened through the use of a variety of audio-visual media and a professional actor. Because this new program was being developed for Monroe County, the Director of the Office of Civil Preparedness was responsible for review and approval of all program materials.

A schedule was developed so that completion of the training program coincided with preparation for the annual Fermi 2 emergency exercise. Six months were allowed to develop the program from start to finish.

The original four-part program developed by the State Police Emergency Management Division formed the basis for the new Monroe County Training Program. The subject matter was adapted to specifically meet the needs of Monroe County.

The new program evolved into 4 major segments. Part I, a 15 minute slide-tape program, deals with plant operations and emergency planning at Fermi 2. Initial review and approval of this segment was performed by the supervisor of emergency planning for Detroit Edison. Final approval was given by the Director of the Office of Civil Preparedness.

Part II is a 15 minute videotape which presents information about radiation including definitions, discussion of types of radiation, how it is detected and measured, and how decontamination is performed. This part of the program was developed and approved cooperatively by the Fermi 2 emergency planning and health physics groups, the State Division of Radiological Health and the County Health Department.

Part III, a 20 minute slide-tape program, was developed by the State Police Emergency Management Division and describes the Michigan Emergency Preparedness Plan and responsibilities at the state level.

Part IV covers the Monroe County emergency plan and includes specific learning modules for each of the major areas of responsibility under the plan. These areas are: Emergency Operations Center (EOC) Staff, Public Information, Warning, Law Enforcement, Public Works, Fire Services, Health/Medical, Social Services, Emergency Medical Services/Transportation and School Services. EOC Staff Orientation, a 12-minute videotape, was developed for EOC staff officers. A 10-minute slide program was prepared on each of the other nine modules.

Parts I through III were presented as the basic package for all county emergency workers. In addition, one of the ten specific modules of Part IV were presented to the individual groups. For example, firemen received Parts I, II, III and the Fire Services module while law enforcement personnel received Parts I, II, III and the Law Enforcement module.

The program was completed within the prescribed six-month time frame and presented to the Monroe County Board of Commissioners. Upon their approval, presentations to county emergency workers began. To date, over 1,000 emergency workers have been trained. The vast majority of those who have seen the program have found it beneficial in outlining their emergency duties and in providing the needed background information.

Currently, a short review program is under development for emergency workers who completed the initial training. As a result of the high level of interest generated by this program, the county has independently developed an abbreviated version of the program for the general public. To date, they have presented it to over 200 county residents.

In addition to providing the needed information to emergency workers, this program has many other benefits. It has increased the credibility of the County Office of Civil Preparedness with the public. It has also greatly increased public awareness of emergency preparedness at the local and utility level. Finally, there is an increased spirit of cooperation and trust between Detroit Edison and Monroe County.

The cost to Detroit Edison for the entire program was significantly less than it might have been for some of the other alternatives--well worthwhile for an innovative approach to training and for the excellent relationships built with the surrounding community.

Emergency Planning, Public Information, and the Media: Some Recent Experiences

Steven B. Goldman

ABSTRACT Within the last two years, events at several nuclear plants generated national media attention. Most were classified as "UNUSUAL EVENTs" or "ALERTs". One event necessitated a "SITE AREA EMERGENCY" classification. None of these events posed a hazard to the general public, yet each posed a serious threat to the credibility of the utilities. These and other events show that (1) media response to a nuclear plant accident is not necessarily related to the technical severity of the accident and (2) utilities must have in place effective and coordinated emergency public information programs, particularly for "lower level" emergency classifications.

I. INTRODUCTION

American electric utilities with nuclear units have developed public information programs specifically designed for nuclear emergencies. Regulations for nuclear emergency public information (EPI) are few, but the consequences of a mismanaged nuclear EPI program are far reaching.

This paper will recount the public information problems of the March 1979 accident at the Three Mile Island Nuclear Station, followed by a brief review of resultant EPI regulations. Utility and media response during more recent emergencies will be presented and discussed; finally, lessons learned from these events will be identified.

To identify the role of emergency public information, it is helpful to divide nuclear emergency response into two broad efforts:

1) Placing the plant into a safe mode
2) Protecting the health and safety of the general public

The former consists of the various operational, technical and maintenance evolutions. The latter also consists of technical aspects (dose assessment, protective actions). Public response to an emergency -- and therefore the protection of the public's health and safety -- is a direct function of the public information effort. Lindell and Perry (1983) show that public response to an emergency -- any emergency -- depends upon:

- Pre-existing beliefs about the hazard
- Interpretation of information disseminated by official sources
- Credibility of those sources
- Distortion or addition of information by unofficial sources
- Evaluation of the threat
- Choice of available responses

Thus, in order to meet the goal of protecting the health and safety of the general public around a nuclear facility -- indeed, any facility -- an effective emergency public information program must be in place <u>before the emergency occurs</u>.

Media response to a nuclear plant accident may also be evaluated in relation to the above six items with perhaps some additions such as "amount of conflict present" and "other news stories occurring". A key lesson for emergency planners as well as utility managers is that media interest in a nuclear plant accident is not necessarily proportional to the severity of the accident.

II. REVIEW OF PUBLIC INFORMATION PROBLEMS DURING THE THREE MILE ISLAND ACCIDENT

It is instructive to review the technical aspects of the accident at the Three Mile Island (TMI) Nuclear Station Unit 2 here because, as George Santayana wrote, "Those who cannot remember the past are condemned to repeat it."

A. Prior to the Accident

From a public information point of view, the TMI accident probably could have happened to almost any U.S. nuclear utility. The Utility, and in fact the U.S. nuclear industry in general was not prepared for an <u>ongoing</u> accident, where time was of the essence. The only previous major U.S. nuclear accident was the fire at Browns Ferry in March 1975 which was essentially over by the time the situation was made public. The Windscale accident in Britain during October 1957 had far reaching offsite consequences, but with regard to the media and public perception of nuclear power at that time, it occurred in an entirely different era. TMI was the first of its kind.

Additionally, the Utility's Public Information staff had, on the average, less than one year's experience with nuclear power, and no one person on that staff had any technical background. Complicating the situation was the fact that, with rare exception, media personnel also had little technical background. Virtually no one was prepared for a major nuclear power plant accident because so few believed that one would happen.

B. During the Accident

The Public Information problems during the TMI accident were severe. The overall problem was that a chain of information flow was not established from the plant to the public. Information was not coordinated among the key players -- the Utility, the state, the federal government, local officials -- and was not presented to the media and public in a cohesive manner. Because information flow was not established, the Utility's Public Information staff was not kept informed by plant technical personnel; the Public Information staff was at times hours behind the accident sequence. Thus, initially it appeared that the Utility Public Information staff consciously minimized the significance of the accident. In actuality, they did not know how far the accident had progressed.

At a press conference during the first day of the accident the Utility failed to say that there had been small offsite radiation releases and that more could be expected. The state's Lieutenant Governor, acting as state spokesman, knew that radiation levels above background had been detected. He asked the Utility's spokesperson why the Utility had not told this to the media. The Utility's spokesperson replied, "Because it did not come up." From that point on, the state was suspicious of the Utility's information and motives, and relied upon its own staff. At a press conference later that day, the Lieutenant Governor told reporters, "Metropolitan Edison has given you and us conflicting information."

This statement -- particularly in the post-Watergate, post-Vietnam era -- was a red flag to the media. Reporters may not have known the difference between a neutron and a cooling tower, but they knew that "conflicting information" made a great story.

What followed for the next week or so was public information chaos. Official news sources were confused. Technical jargon was used, overused and misused. Sources of information were everywhere and nowhere. "Experts" appeared and were quoted extensively. A representative of the U.S. Nuclear Regulatory Commission acted as the official spokesperson for the federal government and the Utility stopped talking. However, this did not stop the "Hydrogen Bubble" story from occurring. That, you may recall, was when speculation led to perceived imminent disaster.

There are several other examples of TMI public information problems, but the basic problems can be summarized as follows:

1. No information flow path was established from the plant technical staff to the public information staff
2. There was little coordination of information among the organizations involved
3. Each organization did not have a principal, designated spokesperson with access to appropriate information in a timely manner
4. There was no single location from which the media and public could find official and timely information
5. Neither the Utility's Public Information staff nor the media had the technical background sufficient to understand or explain the accident.
6. There were no plans or procedures to deal with the Public Information aspects of a major nuclear plant emergency

C. After the Accident

Studies show no real evidence of the utility or others consciously conspiring to hide information. The basic problem was that no one was prepared for the accident. In the aftermath, however, several regulations were developed to ascertain that utility and governmental agency Public Information staffs would be prepared in the future.

III. U.S. NUCLEAR EMERGENCY PUBLIC INFOR-MATION REGULATORY REQUIREMENTS

As a result of TMI, several U.S. government regulations were issued. The basic planning document for U.S. nuclear utilities is called NUREG-0654/FEMA-REP-1 (1980). Other guidance, particularly from the Federal Emergency Management Agency, is being developed.

Although not regulatory, the Institute of Nuclear Power Operations (INPO) has included EPI as one of its emergency preparedness evaluation areas (INPO 85-001, 1985). As in other evaluation areas INPO developed performance criteria for EPI programs to serve as a basis for evaluation. And as with the NUREG requirements, the INPO criteria are general.

Thus, EPI regulations are not specific. This lack of specificity has proved to be a mixed blessing for the nuclear industry. On the one hand, it allows utilities the flexibility to provide quality, comprehensive public information programs that serve well during an emergency as well as during non-emergencies. On the other hand, the lack of specifics also allows utilities to meet only the letter of the law and go no further. Those utilities who choose the latter course will cause the nuclear industry problems, for they are not prepared to handle a major nuclear emergency. They have not learned from the past.

IV. RECENT UTILITIES EXPERIENCE

There has been much improvement in utility EPI efforts since TMI. While many agree that quality programs are necessary, few agree about what constitutes a quality program. A review of recent emergencies provides some clear indications of what utilties may expect should an emergency occur.

Within the last 2 years, events at U.S. nuclear plants generated national media attention. None of these events posed a hazard to the public yet they posed a serious threat to the credibility of the utilities. The following is a brief summary of those events:

A. Event A

At the Vermont Yankee Nuclear Power Plant in June, 1984, a source of radiation was accidentally withdrawn from its shielding; this resulted in abnormally high radiation levels in one area of the plant. An "ALERT" was declared at about 9:00 a.m. and within four hours the utility received over 250 telephone calls from the media, public, and financial organizations. Those calls were placed to the corporate headquarters where there were only four public information personnel. Although approximately 20 people were assigned to the EPI organization, this organization was not activated. Many people who tried to call could not get information because the phone lines were busy.

Many media reports were inaccurate. A major television network reported that there had been an explosion, which was untrue. A number of reports stated that plant workers were overexposed to radiation; this was also untrue. A post-accident review of media reports indicated that local media were generally accurate in reporting information. National media, however, were often inaccurate and tended to sensationalize the stories.

B. Event B

An "ALERT" was declared at a boiling water reactor because of an equipment malfunction in a component associated with the reactor coolant system. The plant was shut down with no hazard to the public. However, the media began to telephone the corporate headquarters soon after the "ALERT" was declared. Although the utility did not count the number of calls, the volume was so large that additional personnel were assigned to answer those calls. One major problem was an inability to get information to the local media in a timely manner. EPI procedures called for EPI personnel to first telephone wire services, and then notify media with offices close to the plant. However, since so many phone calls were received at the corporate headquarters, the staff did not have time to call the local media. Thus, many of the local media went to the plant for information. A public information representative was on duty at the plant, but was assigned to the plant Technical Support Center. As a result, no one at the plant was available to talk to the local media.

A second problem resulted when local government officials notified industries near the plant to be prepared to evacuate. When the media became aware of this, they became skeptical of the utility's statements saying there was no hazard. The utility was telling the truth, but this conflicting information resulted in a loss of utility credibility and created an environment to cause the public to be concerned about its safety.

C. Event C

The two previous events occurred during normal working hours. At Sacramento Municipal Utility District's Rancho Seco Nuclear Power Station, a hydrogen explosion occurred in the electrical generator at night. This event was classified as an "UNUSUAL EVENT". This was followed later by an unrelated loss of all non-nuclear instrumentation, which necessitated an "ALERT" declaration for under 10 minutes. Again, as with the other two events, the public was not endangered. In a yet further unrelated event offsite, a pipe bomb was set off at a school near Sacramento. The explosion was heard throughout the county, and the media believed that the sound of the explosion came from Rancho Seco. When the media began to call the utility headquarters for information, the telephone switchboard called public information personnel at their homes. The utility did not activate their EPI staff; two public information personnel attempted to answer media inquiries from their homes.

Responding to the explosion rumor, a number of reporters went to the plant for information. The plant had a public information officer position, however, no one had been assigned to fill that position. Thus, there was no one for the media to talk to at the plant.

The following day a number of media reports sharply criticized the utility for not providing information. The local government held a hearing where officials similarly criticized the company public information efforts. One State congressman criticized the utility and compared the accident to the one at Three Mile Island. There was, of course, no relationship between the two events but the congressman made headlines implying there was. The key here is that the utility's response - not the accident - became the story.

D. Event D

Late in January, 1986, Cleveland Electric Illuminating Company's Perry Nuclear Power Plant Unit 1 was undergoing several test and pre-fuel loading evolutions. Unit 2 was in a deferred construction status. Fresh nuclear fuel was onsite in the Fuel Building; it was expected that the low power license would be granted within a month.

At 1148, on Friday, January 31, the Perry Nuclear Power Plant, (PNPP) experienced an earthquake measuring approximately 5.0 on the Richter Scale. As a result of the earthquake, the PNPP Emergency Plan was implemented. Emergency response actions went into effect at approximately 1206 when the emergency was classified as a "SITE AREA EMERGENCY". As part of the Site Area Emergency response, all non-essential personnel were evacuated from the site for accountability purposes. The situation was downgraded to an "ALERT" at 1300 and subsequently terminated at 1430. Recovery went into effect and continued until Saturday at 1100 when PNPP and NRC conducted a meeting and agreed on an Action Plan. There were no injuries to workers. There was also never any risk to the public.

The Perry Plant has a Community Relations Section which handles site media relations; the Cleveland Electric Illuminating corporate office has its own Public Information Department. The two work together during PNPP events.

By 1215, Community Relations Section personnel had learned of the event and established contact with both the Control Room and Site Security. By 1230, a Public Information Emergency Response Team had been activated and was answering media calls. An Information Liaison had reported to the Technical Support Center to establish an information flow from the TSC to the Community Relations staff. Within 3 hours of the earthquake, 2 media statements were issued. That afternoon eight media outlets (2 television, 3 radio, 3 newspaper) arrived onsite. Hundreds of media and industry calls were logged onsite. Over the next two days, two more media statements were released with resultant inquiries. By Monday, things were back to "normal".

During the first afternoon, there was some friction between the site and corporate Public Information staffs. Site thought Corporate was too demanding, Corporate thought Site unresponsive. As it turned out, during the site evacuation the telephone system was secured and all incoming telephone calls went through one Security Officer. Thus several people including the media and the state could not get through to the Community Relations Staff. They in turn called the corporate offices. This sounds like a typical post-accident critique, but here is the key point: corporate logged over 2000 phone calls - this for a plant not yet in operation.

Overall the Perry Public Information response was good. At no time did the PNPP response become the story.

E. Event E

At approximately 3:50 a.m. on July 29, 1986 a secondary side steam line ruptured near the condenser of the Rochester Gas and Electric (RG&E) Company's R. E. Ginna Nuclear Station. In accordance with procedures, the incident was classified as an "UNUSUAL EVENT" at 4:00 a.m. The unit was manually shutdown and there was no release of radiation, nor were there any injuries. The UNUSUAL EVENT was terminated at 5:14 a.m.

RG&E's public information people went into action. At 8:30 a.m. that morning RG&E and the two area counties held a media briefing. Approximately 15 reporters covered the briefing. Additionally, RG&E received over one hundred phone calls over two days. This incident had all the makings of a major news story - morning accident, pipe failure, steam release, plant shutdown, same state as Shoreham, same reactor vendor as Seabrook, major accident there is 1982, etc - but it was not. Although one T.V. station gave RG&E its usual hard time, the other media outlets were fair. This story did not make national news.

F. Event F

The tragedy at Chernobyl carries several public information lessons. It is clear that there are philosophical differences in nuclear accident response and public information practices between the Soviet Union and the United States. Still, western media coverage was extensive. It is of interest to note that the lack of official information about the accident did not deter the press in gathering its stories. As seen with American nuclear plant accidents, the media will go around the system to find information. The quality of that information may be suspect at times, but the information will be found.

One Chernobyl lesson is that when official information is scant or not available, unofficial information increases in value and credibility. Recall the incident where a Dutch amateur radio operator allegedly monitored a Russian transmission telling of hundreds of casualties and appealing for help. This story so far has proved untrue, but it was a major story of the accident.

Another lesson concerns information withholding or "processing". The Soviets are masters of information processing, whether it be full disclosure, partial disclosure, no disclosure, filtering, embellishment or euphamising. But when the efforts of the media are concentrated as they were with Chernobyl, the official subterfuges did not work. Yet there are U.S. utility managers today who think they can succeed where the Soviets could not; they prefer not to release information to the public or even to their Public Information people. Event after event shows that this philosophy, although once in a while lucky, harms the utility in the long run.

V. LESSONS TO BE LEARNED

A. What You Should Expect

I have discussed media response to several nuclear incidents. There were two Unusual Events, two Alerts, a Site Area Emergency and TMI and Chernobyl, which clearly would have been General Emergencies. Of the American plants, two were located in the north, one each in the south, west, east and midwest. Despite all these differences, there are common threads to the media's response to all of these accidents. Based upon these and other experiences, this is what you should expect. First, some groundrules:

- Media interest in a nuclear plant incident is not necessarily proportional to the accident's technical severity.
- A utility's technical perception of a nuclear plant incident is not related to nor will it influence the media's perception of that incident.
- Media interest in nuclear events is high.
- The general public has a fear of radiation - any amount of radiation. Microcuries and millirems mean nothing compared to cancer.
- The media's coverage of nuclear issues and events is in a class by itself.
- Simplistically, all nuclear power topics for the general public and media may be categorized into two issues: radiation and money.
- When politicians become involved, the media interest and perceived seriousness of the problem doubles.
- Even a minor event can generate considerable media interest.

This is what you should expect if you have a nuclear incident:

- There will be much interest in the event.
- You will receive dozens if not hundreds of phone calls.
- The media will most likely show up at your site or corporate headquarters or both. The use of helicopters and satellite transmission facilitates this.
- The media will want a spokesperson; if you don't supply one, they will find one.
- If the media are not satisfied with your press operation, they will go around your system.
- Inaccuracies will occur; rumors will be generated.
- Local media will be more accurate than national media; national media will be very demanding.
- All media want a steady stream of information. If they don't get it from you, they will get it elsewhere.
- The media may not be fair. Reporters are human, they too have pressures; "bad news sells"; and the media - particularly TV - want to beat the competition. Objectivity is a worthy goal but in nuclear coverage it is not always attained; some reporters and editors do have axes to grind.
- If your company is privately owned, the financial community will be extremely interested in the accident.
- Your company's image will rest upon how well your Public Information staff does.
- "Low level" events can cause a utility as many informational problems as a major accident.
- Media response will be a function of your response. The better you do, the better they do. If you shoot yourself in the corporate foot, the media will duly record it; they will not call a doctor. If so, your response - and not the accident - will become the story.

B. What You Should Do

If you could prevent accidents at your plants, this conference and your jobs would not be necessary. It should be clear as to what this means to nuclear emergency planners. First, you will have accidents and incidents. Second, your emergency public information effort should be top quality. Just like your onsite organization, your Public Information Organization should be able to transfer from normal operation to emergency operations smoothly and efficiently.

Space limitations prevent a listing here of key elements of a successful emergency public information program. However, just as we can learn from mistakes, we also should learn from successes. The Perry Plant successfully handled the media aspects of their earthquake because of the following reasons:

- A detailed emergency public information plan was in place.
- The Community Relations Section established a chain of information flow, outlined responsibilities, coordinated information, and provided spokespersons. The Public Information staff controlled the situation.

- The Public Information staff not only partcipated in exercises, they held separate Public Information drills. These proved extremely valuable.
- CEI and PNPP Management support the Public Information effort. They devoted time and resources to Public Information - a wise investment which has reaped dividends.

A successful and credible emergency public information operation is the culmination of much planning, training and coordination of efforts among those affected: the utility; federal, state and local officials; special interest groups; the media and the public. Within the utility alone, coordination is required among emergency planning, plant information staff, operations, plant management and corporate management. Nuclear Emergency Public Information is much more than being able to write news releases.

An effective emergency public information program is an integral part of a site emergency operation as well as being the cornerstone of a total corporate emergency communication effort. Emergencies at nuclear plants have proven that the media, governmental officials, the financial community, employees, stockholders and special interest groups will actively seek information about emergency activities. To respond to this interest and to service the company's own interests, a company must develop a comprehensive corporate emergency communication program. This requires the involvement of the total company, not just public information personnel.

The challenge to emergency planners is to make emergency public information programs effective. Each event that occurs reinforces more strongly the necessity of a good Emergency Public Information program. In the minds of the public, the Emergency Public Information efforts are the face, soul, and conscience of the utility. The performance of a utility's Emergency Public Information program will determine how well public safety is protected and will determine the public's verdict on the utility. In a larger sense, Emergency Public Information has and will impact the future of nuclear power in the United States. This is one of Three Mile Island's lessons learned.

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Factors Influencing Media Coverage of a Radiological Incident

Roy K. Bernhardt and Layton J. O'Neill

ABSTRACT Most organizations have an existing policy for interactions with the media. This policy often requires that interactions be with or through a professional group of public information officers or the Office of Public Affairs. This policy tends to give individual members of an organization the belief that they are not responsible or in some instances, even allowed to interact with the media. To achieve good media relationships and/or coverage, individual interactions are necessary and required. The guidelines for media interactions provided in the Federal Emergency Management Agency (FEMA) sponsored Radiological Emergency Response course are relatively straight forward and simple to adopt.

The Nevada Operations Office (NV) Radiological Assistance TEAM (RAT) is Team No. 4 under the Department of Energy Region VII Coordinating Office. In addition to the normal U.S. Department of Energy (DOE) RAT standby radiological instrumentation and health physics capabilities, the NV RAT has a number of unique features.

Beginning in 1970, NV was given the responsibility by the Regional Coordination Office for the Radiological Assistance Program in the state of Nevada due to the mountain barrier separating the California teams from Nevada.

From 1972 to the present, the NV has had a formalized agreement with the state of Nevada, which identifies the NV 24-hour emergency phone number as the primary phone number to call in the state of Nevada for radiological assistance. The agreement provides for the NV team to perform the initial response and control at the incident scene until a state representative takes charge. NV provides a supply of small billfold cards (Illustration 1) to the State Office of Preparedness for their distribution throughout the communities of Nevada. Approximately 2,000 have been distributed to date.

A comprehensive NV RAT Notification Procedure was developed, which calls for routine notification of both the State Office of Emergency Management and the State Division of Health, Radiological Section, if a response is made to a state controlled location.

For an expeditious response, the NV maintains a RAT Captain Duty Officer roster with beeper call-out capability. To further enhance the team captain's response capabilities, the team captain has a dedicated vehicle having the customary radiological equipment along with minor clean-up equipment, such as a High Efficiency Particulate Air (HEPA) filtered industrial vacuum cleaner, A/C generator, brooms, shovels, flood lights and air sampling

NV-155 (10/85) FRONT

FOR RADIOLOGICAL ASSISTANCE CONTACT

NEVADA OPERATIONS OFFICE
U. S. DEPARTMENT OF ENERGY
Las Vegas, Nevada
24 HOUR TELEPHONE
(702) 295-3343
OR — Region 7 Coordinating Office:
San Francisco Operations Office, U.S.D.O.E.
Telephone: day or night (415) 273-4237

BACK

IT IS ESSENTIAL TO INFORM THE PARTIES CONTACTED OF THE EMERGENCY NATURE OF THE CALL

The Radiological Assistance Plan is a service provided to the public by the U.S. Department of Energy. Its function is to advise and assist local authorities in dealing with accidents involving radioactive materials.

Radiological Assistance Teams are available to local authorities at all times. Collect calls will be accepted if necessary.

For specific instructions regarding initial procedures in the handling of radioactive incidents refer to:

EMERGENCY ACTION GUIDELINES FOR INCIDENTS INVOLVING RADIOACTIVE MATERIAL: "This instruction is found in the NV Radiological Assistance Team Notification Procedure Manual, which is issued by the Nevada Operations Office, USDOE, Las Vegas, NV."

BILLFOLD CARD

capability.

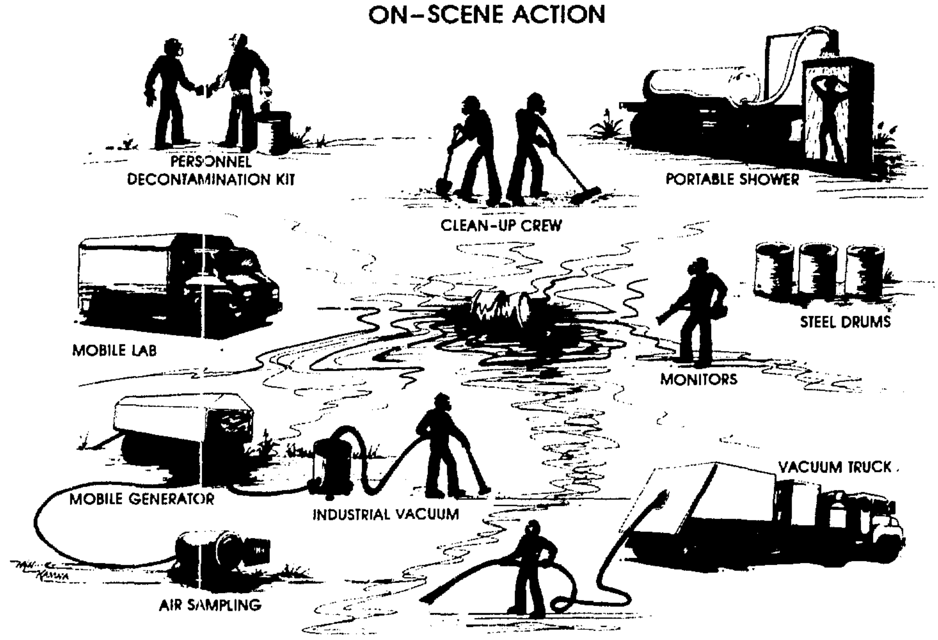
The DOE Aerial Measurements Systems (AMS) is operated by an NV contractor. The NV RAT has been able to rely upon the readily available AMS capabilities. These capabilities include the transportation of the team members and equipment, plus aerial monitoring of the airborne releases, surface deposition and lost radioactive sources.

In 1985, NV RAT again enhanced it's response capability to include the clean-up of transportation spills of defense low-level waste shipments, by developing a plan and equipment for a contractor operated Clean-up Team. (Illustration 2). The cornerstone of this capability being the specially designed HAZVAC truck. It was the first major exercise of the equipment used during the exercise.

The education of the news media and local authorities is another of the FEMA guidelines addressed in the Radiological Emergency Response Course offered at the Nevada Test Site. By being present at the exercise, the media had an opportunity to view first hand, and thus report on its use and operation, equipment and materials that can be made available to state and local emergency response organizations to assist in an actual emergency if one should occur. The capabilities of a complex piece of equipment, the "HAZVAC", was explained to the media. In the resulting media coverage, one story notes the equipment could also be used in the event of toxic spills.

NV RADIOLOGICAL RESPONSE CLEAN-UP TEAM

ON-SCENE ACTION



Clean-up Team that created the situation for stimulating the news media involvement.

The fact that the news media was invited to view the exercise caused some concern among team members and observers. The concern seemed to be caused by the "bad press" usually given to radiation and situations involving radioactive material. The decision to invite the media is in keeping with the first FEMA guideline concerning public relations and media interactions. The news media treatment of an accident and public reaction to that accident depend to a considerable extent upon what was done before the accident to educate the public about the specific hazards of the particular operation and what safeguards have been provided.

The media made the public more aware of the equipment and trained personnel available in the event of a transportation accident by reporting the exercise, the response team's actions and

One of the "lessons learned" from conducting this exercise was the need for an on-scene media spokesperson with the authority to issue statements to the media without first checking with headquarters. Provisions and guidance for this media representative should be included in emergency response plans. A designated spokesperson as part of the emergency response team will be in a position to ensure that information released is provided as quickly as possible, in the proper perspective, factual and consistent. A trained spokesperson will be able to safeguard classified information, yet still release facts about the incident. These facts should be released even if they prove embarrassing or admit to errors.

This attitude and commitment will develop an understanding and rapport between the media representatives and the information spokespersons. However, each individual member of the emergency response organization should receive training to be able to function as the media interface, should the need arise. A program of periodic exercises to test not only the emergency response team's abilities and equipment, but also the interface and liaison with the media is necessary and prudent.

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