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THE NATURE OF REACTOR ACCIDENTS

by

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PAPER

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ABSTRACT

Reactor accidents are events which result in the release of radioactive material from a nuclear power plant due to the failure of one or more critical components of that plant. The failures, depending on their number and type, can result in releases whose consequences range from negligible to catastrophic. By way of examples, this paper describes four specific accidents which cover this range of consequence and for each outlines a possible sequence of events and an estimate of the expected frequency.

RÉSUMÉ

Les accidents nucléaires sont des événements qui provoquent des dégagements de matières radioactives d'une centrale nucléaire à la suite de la défaillance d'une ou de plusieurs composantes critiques de cette centrale. Selon leur nombre et leur type, ces défaillances peuvent produire des dégagements dont les conséquences vont du négligeable au catastrophique. A titre d'exemples, le présent document décrit quatre accidents qui couvrent toute cette gamme de conséquences et donne, dans chaque cas, une suite éventuelle d'événements et une estimation de leur fréquence.

Preface

This paper is specifically written for an interdepartmental group charged with the responsibility of conducting a review of the federal government policies on nuclear matters. As one of several papers tabled with the group, it is intended for the use of the members and for their discussions with persons whose experience with nuclear power reactors is very limited.

This paper is not intended to be a comprehensive review of the potential for accidents in nuclear power plants. Rather it is intended to give examples only of some possible reactor accidents along with a discussion of their likelihood and possible consequences. By this means a lay person should get some understanding of the nature of the risk posed by nuclear power plants.

1. Introduction

As the regulatory agency for nuclear power plants, the Atomic Energy Control Board (AECB) is responsible for ensuring that nuclear power plants are sited, designed, constructed and operated in such a fashion as to avoid an undue risk to the workers and the public. In common with those of the rest of the world, the AECB safety requirements embody the principle of defence-in-depth which, as implemented in Canada, can be described in the most basic terms as follows:-

1. A nuclear power plant must be designed to minimize the frequency of serious failures.

2. Notwithstanding the good design, it is assumed that serious failures will occur and therefore defences must be erected to cope with these failures.

3. Further, it is a design requirement to show that should a serious failure occur, and simultaneously one of the defences should fail, the remaining defences would be capable of limiting the consequences to prescribed limits.

These general requirements are discussed in a variety of AECB publications (1)(2)(3). Very recently the AECB decided that it was necessary to clarify and expand, and in some respects modify, these general requirements (4). To implement this decision of the Board, AECB staff has prepared four licensing guides which address the requirements for

i) Safety Analysis (5)
ii) Containment (6)
iii) Shutdown Systems (7)
iv) Emergency Core Cooling (8)

These have been issued for public and industry comment.

From the point of view of accidents at CANDU nuclear power plants the experience has been good; there has not been an accident at a CANDU power reactor which has had an adverse effect on members of the public. Two independent groups have reviewed the design and operation of nuclear power plants in Ontario and concluded that they are acceptably safe (9)(10). Neither of these preceding statements implies that reactor accidents will not happen. On the contrary, they are made with the knowledge that reactor accidents have happened (see Appendix 1) and with the expectation that accidents will happen in the future.

The range of accidents which are possible extends from inconsequential to catastrophic. In the following sections four particular accidents will be described which cover this range:

i) Failure of a reactor control system;
ii) Loss of coolant;
iii) Loss of coolant combined with an impairment of the reactor containment building;
iv) Reactor core meltdown.

In each case a possible course of events, the likelihood of such an accident, and the potential consequences will be outlined. The discussion of consequences will address the release of radioactive materials and the resultant effects on members of the general public. It will not deal with economic consequences nor the hazard which workers would encounter in the event of an accident.

In describing these four examples, it is important to recognize that they are just that - examples. They cannot represent all possible accidents because there is such a wide variety of different initiating events each of which can have a variety of consequences, depending on the responses of the safety features and the operators. The accident which occurred at Three Mile Island was one in the spectrum of possible accidents and was perhaps more 'typical' in that it was more complex that the examples presented. The four examples do, however, cover a very wide range of probabilities and consequences.

2. Failure of Reactor Control System

2.1 Introduction

Control of the fission process in a reactor under normal conditions is an exacting process. It is achieved by a control system which adds or removes neutron absorbers (cadmium, boron, water) to or from the reactor core. If the control system fails in such a way that too much of a neutron absorber is added, the chain reactor process will no longer be self-sustaining and the reactor power will be reduced to near zero. If the failure causes too much neutron absorber to be removed there will be an increase, sometimes rapid, in the reactor power. One of the special defences, a reactor shutdown system, would need to act to stop this increase and thereby prevent potentially massive failures of reactor components. For safety analysis purposes, a failure of a reactor control system that would increase power at an uncontrolled rate is considered to be a reactor accident. Such failures of reactor control systems have occurred in CANDU nuclear power plants.

2.2 Sequence of Events

An event which occurred at the Pickering nuclear power station illustrates one way that such a failure can occur. With the reactor operating at 100% of full power, simple modifications were being made to the system which controls reactor power by adding water to or removing it from compartments in the reactor. A fault occurred which caused the compartments to fill with water and this reduced reactor power to near zero. Reactor power was then raised to about 50% of full power and the operators attempted to restore the condition of the compartments to normal. In the process, an error was made and the compartments drained rapidly. Reactor power increased rapidly from 50% to 100% of full power and was increasing at an accelerating rate when the independent shutdown system detected the problem and automatically reduced reactor power to effectively zero in a matter of seconds.

2.3 Frequency of Events

The frequency of unsafe failures of reactor control systems (those which result in an uncontrolled increase in reactor power) has been, on the average, one for each ten reactor years of operation (8 events in about 80 reactor-years of operation, 6 of which occurred in the early years of the Pickering station). With improvements which have been made in the systems, designers estimate that a failure frequency as low as one per 100 reactor-years may be achievable. Recent experience has indeed been good.

2.4 Consequence of Events

In each case of a control system failure, a shutdown system intervened to terminate the power increase without any damage to the reactor fuel or other components. The consequences to members of the public were exactly zero. Therefore, while recognizing that failures of a reactor control system are serious failures, it is perhaps inappropriate to call them accidents.

3. Loss-of-Coolant Accident

3.1 Introduction

Most accidents in nuclear power plants (of which failure of the reactor control system is only one example) will result in zero plant damage and zero radiological impact on members of the general public if the special defences built into a nuclear power plant function properly. However, for at least some loss of coolant accidents* the special defences (shutdown systems, emergency core cooling system (ECCS) and containment) can only limit the damage to the reactor fuel and the releases of radioactive material from the plant.

The postulated loss-of-coolant accidents which are considered in the safety analysis of a nuclear power plant range from failure of a small (2 inch diameter) to a large (20 inch diameter) pipe. Included in this range is failure of a pressure tube inside the reactor core. In doing the safety analysis the emphasis is on searching for the particular size and location of a failure which is predicted to cause the most damage to reactor fuel. In the remainder of this section, the discussion will center around this 'critical break'. It is important to recognize however that this is a simplification for purposes of this paper. In addition, but not discussed in this paper, the analysis identifies a different break which is critical from the point of view of required speed of response of a shutdown system, a different break which is critical from the point of view of required speed of response of an emergency core cooling system and another which is critical from the point of view of required capability of the pressure suppression system in the containment building.

3.2 Sequence of Events

For the purposes of safety analysis, a LOCA is assumed to be an instantaneous and complete failure of a pipe of the right size and at the right location to correspond to the critical break. For a plant such as the Bruce 'A' nuclear power plant, the critical break would be equivalent to the failure of about a 10 inch diameter pipe at the coolant inlet end of the reactor. The effects of such a failure would be:

a) rapid depressurization of the reactor cooling system from the normal operating pressure of about 1300 psi** to a pressure of less than 100 psi in about three minutes;

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* When the term 'loss-of-coolant accident' is used in this paper, it refers to an event where coolant is escaping from the heat transport system (the system which cools the nuclear fuel with water at high pressure) at a rate which is greater than the make-up capacity of the normal systems designed to keep the heat transport system full. An emergency core cooling system is then required to keep the reactor fuel cooled. Large leaks (tens of gallons per minute) can and have occurred but, being within the capacity of the normal make-up systems, they are not referred to as loss-of-coolant accidents.

** psi = pounds per square inch.

b) as the coolant in the core flashes to steam the density of the coolant in the core falls, thereby reducing the absorption of neutrons. The heat generation rate in the fuel would increase rapidly, reaching four times the normal rate in about one second after the failure, at which time the shutdown systems would halt the rise in power and reduce the power output from the fuel to about 10% of normal in 10 seconds and to about 2% in 10 minutes;

c) the pressure in the reactor building would rise rapidly to several pounds per square inch above normal before the action of the vacuum building and the dousing system*** would reduce the pressure to less than atmospheric pressure (in about 3 minutes);

d) any penetrations of containment which are normally open to the outside atmosphere (e.g., building ventilation system) would close automatically on indication of pressure rise in the reactor building to contain the radioactive materials;

e) the escaping coolant would remove much of the heat from the reactor fuel but eventually the deterioration of cooling conditions would cause fuel sheath temperatures to rise to high values (in excess of 1800 degrees F (1000 degrees C));

f) when the pressure in the heat transport system (HTS) falls to about 100 psi, there would not be much water left in the system and the emergency core cooling system would act to refill the HTS;

g) the combination of high fuel temperatures and low HTS pressure would result in fuel sheath failure and release of gaseous fission products into the heat transport system and then into the reactor containment building via the assumed break;

h) initially the containment building would prevent the release to the public of any significant amount of radioactive material. However, over a period of days the pressure in the containment would rise to atmospheric pressure or above;

*** This system reduces pressure within the containment buildings by condensing the steam with water sprays.

i) to limit any pressure build-up, gases from the reactor building would be released at a controlled rate through filters by the operators;

j) from the gases exhausted, the filters would remove almost all of the particulate matter, all but a few percent of the radioiodines, but very little of the tritium and none of the noble gases;

k) this outflow of radioactive gases which would occur over a period of days would result in radiation doses to members of the public living in the vicinity of the plant.

3.3 Frequency of Events

Because there have been no loss of coolant accidents in CANDU nuclear power plants, it is not possible to predict the frequency of such occurrences based on operating experience. Instead, it is necessary to base predictions of frequency on information from other types of plants. A survey of high pressure piping systems in non-nuclear plants suggests a frequency of less than one large pipe failure for each 1000 reactor-years of operation(11). (The critical break discussed above would be a large pipe failure). The frequency of small pipe failures would be expected to be higher (perhaps one every 10 to 100 reactor years) because there are many more small pipes in a CANDU reactor.

One should not attribute a high degree of accuracy to these estimates of frequencies. Clearly, with 80 reactor years of operation of CANDU nuclear power plants and no small pipe failures in a reactor cooling system (of sufficient size to be called a LOCA), an estimate of one every 10 reactor years seems too high. One every 100 reactor years appears more likely and such an estimate would not be inconsistent with experience elsewhere in the world. The estimate of less than one large pipe failure per 1000 reactor-years of operation cannot be proved on the basis of reactor experience in Canada and the rest of the world. One can only say that the estimate is not inconsistent with reactor experience.

If the estimate of less than one large failure per 1000 reactor years is correct, it would mean that with 10 operating CANDU reactors the average interval between large LOCA would be at least 100 years. With 25 operating CANDU reactors, the average interval would be at least 40 years.

3.4 Consequences of LOCA

The consequences of major concern in considering a LOCA are the radiation doses which would be received by an individual living near the plant boundary and the total radiation dose to all the people in the vicinity of the plant. The predicted consequences are subject to uncertainties in many areas. Three of the major areas of uncertainty are:

i) the quantity of radioactive gases and vapours which escape from the fuel and into the reactor containment building;

ii) the quantity of radioactive gases which escape containment;

iii) the weather conditions which prevail at the time of the accident.

The uncertainties in the quantity of radioactive gases and vapours which escape from the fuel include the uncertainties about the effectiveness of any ECCS. Will the ECCS be successful in rewetting and cooling the fuel in the reactor as predicted on the basis of extrapolations from laboratory tests? It is possible that the rewetting of some fuel channels will delay for an extended period of time the rewetting of others due to "short-circuiting" of the emergency coolant? Will fuel bundle and fuel channel distortions under accident conditions interfere with cooling by the ECCS to the point that additional gaseous fission products will be released from the uranium oxide fuel? There are no simple answers to these and other questions and therefore an analysis of the consequences of a LOCA involves a process of conservative assumptions in some cases and best engineering judgement based on extrapolations from available experimental information in others.

The uncertainties in the quantity of radioactive materials which escape containment are associated with the deposition of fission products in containment, the effeciency of filters, the correctness of operator actions associated with venting of the containment and the leak tightness of the containment.

Weather conditions can be a significant variable. For a given release of radioactive material from the plant, the predicted radiation dose to an individual at the plant exclusion boundary could easily vary by a factor of ten or more depending on the assumptions that are made with respect to the prevailing weather conditions and therefore the dilution of radioactive materials in the atmosphere.

There is a fourth factor which would affect the radiation dose to people; namely protective actions such as evacuation of the affected populace or distribution of tablets of a stable iodine compound. Use of such tablets can greatly reduce the radiation dose resulting from inhalation of radioactive elements of iodine. For the purpose of this paper the assumption is made that there would not be any evacuation of people or distribution of tablets despite the fact that the release of radioactive materials are assumed to continue for many days.

Bearing in mind the uncertainties, one can then examine the predicted radiation dose to an individual who lives at and remains at the plant exclusion boundary as well as to the general populace. For the Bruce 'A' plant the designers have estimated that the dose to an individual at the plant exclusion boundary could range from:

Thyroid dose	- 0.01 to 60 rem*
Whole body dose	- 0.02 to 0.3 rem

Although estimates of dose to thyroids as high as 60 rem have been made in extreme cases, a more realistic estimate would be less than 3 rem.

Using a higher value in each case and assuming that

a) there is a uniform population density of 640 persons/square mile (250 persons/km^2)** and

b) that unfavourable weather conditions will persist indefinitely,

the total dose to the population within 40 miles of the plant would be predicted to be no more than:

Thyroid dose	- 3.7×10^5 man-rem
Whole body dose	- 1300 man-rem

Based on widely accepted relationships between radiation dose and health effects, the consequences of such doses to an individual are predicted to be 0.6% chance (1 chance in 170) of developing thyroid cancer in his lifetime due to the 60 rem dose to the thyroid. The chance of a fatal cancer of the thyroid would be about 0.03% (1 chance in 3300). There would be a 0.004% chance (1 chance in 25,000) of fatal cancers due to the whole body dose of 0.3 rem.

The consequences of a population dose of 3.7×10^5 rem to thyroids are predicted to be two additional fatal thyroid cancers in the affected population. If the population dose was calculated on the basis of the more realistic estimate of 3 rem dose to the thyroid of the most exposed individual the population dose would be 19,000 man-rem and a 10% chance of one additional fatal cancer would be predicted.

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* Radioactive elements of iodine usually would be included in the releases under accident conditions. If inhaled, these radioiodines would be concentrated in the thyroid gland and would result in a significant dose to the thyroid without any significant dose to the rest of the body.

The uncertainties discussed above could increase the range of these predicted values.

** This density is approximately that of the town of Markham which has a population of 60,000 in an area of 210 square kilometers.

The consequences of the 1300 man-rem whole body dose to the populace are predicted to be a 16% chance (1 chance in 6) of one additional fatal cancer and a 10% chance (1 chance in 10) of producing one additional genetic defect in all succeeding generations.

4. LOCA Combined with Impairment of the Reactor Building

4.1 Introduction

The LOCA sequence described in Section 3.2 is a very simplified sequence. Since neither machines nor human beings are perfect it is reasonable to expect that unforeseen problems will occur. One can think of a vast variety of different failures that may occur. In Canada it is an AECB requirement to assume that if a LOCA (or a variety of other accidents) occurs there will also be a failure in either a shutdown system, the ECCS or the reactor containment system. For the illustrative purpose of this paper, a LOCA combined with a failure in the containment system will be described. While recognizing that a number of different failures in the containment system could be postulated (e.g., failure of dousing, failure of building pressure relief valves* to open, or deflated seals on airlock doors**), a specific failure in the reactor building ventilation system, namely failure of ventilation dampers*** to close when a LOCA occurs, will be described.

4.2 Sequence of Events

The sequence of events would proceed as described in section 3.2 until step (g) in the sequence. The remainder of the sequence would be:

h) assuming that the exhaust dampers on the building ventilation system fail to close, the containment would be open to the outside atmosphere through a ventilation duct having a diameter of about 8 inches (in the case of a Bruce 'A' reactor);

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* These are the valves which, under accident conditions, would open automatically to connect the building housing the reactor to the vacuum building in a plant such as the Bruce nuclear power plant.

** To prevent the escape of radioactive material from a reactor building all doors are sealed by inflated rubber seals.

*** Under normal operating conditions there is a controlled flow of air into the reactor containment building to ensure a suitable environment for workers. This air is exhausted from the building and discharged through a ventilation stack. Should an accident occur, the ventilation lines into and out from the building are isolated by the automatic closure of dampers in the lines.

i) during about the first 3 hours after a LOCA, the containment system would be at sub-atmospheric pressure due to the action of the vacuum building;

j) during this 3 hour period, air would flow into the containment through the ventilation duct in which the dampers had failed to close;

k) after this 3 hour period there would be a continuous outflow of air from the containment carrying with it radioactive material;

l) the situation would be restored to "normal" when and if the dampers could be closed.

4.3 Frequency of Events

In section 3.3 estimates of the frequencies of LOCA were presented:

Small LOCA - once per 100 reactor-years

Large LOCA - less than once per 1000 reactor-years (i.e., less than 10^{-3}/year).

Tests of the dampers which isolate the ventilation duct (along with tests of all the instrumentation and logic associated with this isolation feature) show that isolation is unavailable about 0.3×10^{-3} year/year; that is, about 2.6 hours per year. Therefore the probability of a large LOCA occurring with isolation unavailable would be the product of these two numbers: $10^{-3} \times 0.3 \times 10^{-3} = 0.3 \times 10^{-6}$ per reactor-year. With twenty-five operating reactors the probability would be about one in 130,000 per year.

Even this simple example requires some qualification. The probability of one in 130,000 per year is based on the expectation that a large LOCA will have no effect on the availability of the isolation system. While the isolation system is designed to be unaffected by a large LOCA, one cannot rule out some unforeseen mechanism which could render the isolation system ineffective.

The above discussion has been limited to a LOCA combined with one particular mode of containment impairment. There are other possible failure modes for containment. The example discussed is clearly only one member of the family of postulated accidents called "LOCA with Impaired Containment".

4.4 Consequences of a LOCA Combined with Failure of Reactor Building Dampers to Close

Bearing in mind the uncertainties of the type discussed in Section 3.4, maximum doses to an individual standing at the plant exclusion boundary for an indefinite period after a large LOCA would be:

Thyroid dose - 155 rem
Whole body dose - 2 rem

Again assuming a uniform population density of 640 persons/square mile (250 persons/km^2), the dose to the population within 40 miles of a plant would be:

Thyroid dose - 5.9×10^5 man-rem
Whole body dose - 7100 man-rem

Again, based on widely accepted relationships between radiation dose and health effects, the consequences of such doses to an invididual are predicted to be a 0.08% chance of developing a fatal cancer of the thyroid in his lifetime due to the 155 rem dose to the thyroid and a 0.025% chance of fatal cancers due to the whole body dose of 2 rem. The consequences of a population dose of 5.9×10^5 rem to thyroids are predicted to be three additional fatal thyroid cancers in the affected population. The consequences of the whole body doses of 7100 man-rem to the populace are predicted to be one additional cancer death and a 50% chance of one additional genetic defect in all succeeding generations.

5. Core Meltdown Accidents

5.1 Introduction

Core meltdown accidents of the type to be described here have never occurred in any commercial power reactor, although the sequence of events at Three Mile Island went part-way along the path. Nor has any study on core meltdown accidents been done for the CANDU reactor (although an initial examination of possible sequences is being sponsored as part of the AECB's research program) In the absence of relevant Canadian information, this report presents results of work done by N.C. Rasmussen, et al., as described in the Reactor Safety Study (WASH-1400) issued in 1975 by the US Nuclear Regulatory Commission (12). The following information borrows extensively from that document and, although not strictly applicable to CANDU reactors, does give useful illustrative information on very serious potential accidents. The differences in the design of CANDU and USA light water reactors can significantly alter the sequence of events, and can reduce or increase the probability and the consequences of an accident.

5.2 Sequence of Events

The Reactor Safety Study has defined two broad types of situation that might potentially lead to melting of the reactor core: a loss-of-coolant accident (LOCA), and transients (see below).

In the event of a LOCA, the normal cooling water would be lost from the main cooling system but core melting would normally be prevented by the action of the emergency core cooling system(ECCS). However, if the ECCS failed to act, melting of metallic components of the core and eventually of the uranium oxide fuel itself would probably occur.

The term "transient" refers to those situations where there is an uncontrolled increase in reactor power or a loss of normal cooling flow, both of which require the reactor to be shut down. Following shutdown, the decay heat removal systems act to keep the core from overheating. However, if the reactor fails to shut down or the decay heat removal systems fail, melting of the core would ensue.

The Rasmussen study conservatively assumed that if any fuel melting occurred, then complete core melting would occur. It was then predicted that the molten core, consisting of a mixture of molten uranium oxide, stainless steel, zirconium and other core structural materials, could melt through the bottom of the 8 inch thick steel reactor vessel and through the 12 foot thick concrete base slab of the containment structure. The study estimated the time for going through the reactor vessel to be 1 to 1½ hours and through the base slab to be an additional 13 to 28 hours. The molten mass was then predicted to sink into the ground an additional 10-50 feet before coming to rest. (This sequence is often portrayed as the so-called "China Syndrome".) However, much of the core's radioactive material, which strictly speaking has escaped from containment, is prevented from reaching the environment because the ground in this case acts as an effective filter. Further, considerable radioactive decay would have occurred by the time ground water leaching could contribute to the spread of contamination. Thus, while this was considered the most likely sequence following a major core meltdown, it would not necessarily produce a dispersal of the bulk of the core's radioactive material into the environment.

Much larger consequences could be associated with core meltdowns which also cause failures in the containment structure above ground. If the containment sprays malfunction or are damaged by flying debris (generated by a LOCA or transient) the steam being released from the reactor core would not be condensed. This steam, along with various vapours and noncondensible gases could cause failure of the containment structure due to overpressurization. Also, hot Zircaloy from the fuel sheaths and steel would react with water to produce large volumes of hydrogen. Detonation of this hydrogen (reacting with oxygen) might damage the containment or, if not, the heat of combustion combined with high steam pressures would at least add to the pressure loads on the structure. A further contributor to containment pressurization would be the large quantities of carbon dioxide generated as the molten core melts through the concrete base slab. Another possibility is one in which the molten fuel falls into the pool of water in the bottom of the reactor vessel; this could result in a steam explosion which could rupture the reactor vessel with the formation of flying debris which could, in turn, damage the containment structure. All post-meltdown occurrences which threaten to damage or breach the containment structure can result in the release of substantial amounts of radioactive material to the environment.

5.3 Frequency of Occurrence and Consequences

The Reactor Safety Study calculated the health effects and the probability of occurrence for many possible combinations of radioactive material release magnitude, weather conditions, and the population exposed. The results were as follows:

APPROXIMATE VALUES OF EARLY EFFECTS AND LATENT EFFECTS

	Consequences			
Chance per year[e] per reactor	Early Illness	Early Fatalities	Latent[b] Cancer Fatilities (per year)	Genetic[c] Effects (per yr)
1 in 20,000[a]	<1.0	<<< 1[d]	< 1.0	< 1.0
1 in 1,000,000	300	5[d]	170	25
1 in 10,000,000	3,000	100	460	60
1 in 100,000,000	14,000	1,000	860	110
1 in 1,000,000,000	45,000	3,300	1,500	170
Normal incidence per year[f]	4×10^5		17,000	8,000

(a) This is the predicted chance per year of core melt.

(b) This rate would occur approximately in the 10 to 40 year period after a potential accident.

(c) This rate would apply to the first generation born after the accident. Subsequent generations would experience effects at decreasing rates.

(d) Not given in study. Values have been interpolated from the other values in the table.

(e) Chance per year that a combination of events will occur which will have the consequences given.

(f) Normal incidence in the 10 million people who might be exposed in a very large accident over the time period that the potential reactor effects might occur.

In addition to these health effects, a nuclear accident may contaminate the surrounding area and require relocation of the populace. For the most likely core melt accident,having a probability of occurrence of one in 20,000 per year, little or no contamination would be expected. Consequences associated with other probabilities have been estimated in the Study and are as follows:

LAND AREA AFFECTED BY POTENTIAL NUCLEAR POWER PLANT ACCIDENTS

Chance Per Year	Consequences: Decontamination Area (Sq Mile)	Consequences: Relocation Area (Sq Mile)[b]
1 in 20,000	< 0.1	< 0.1
1 in 1,000,000	2,000	130
1 in 10,000,000	3,200	250
1 in 100,000,000	(a)	290
1 in 1,000,000,000	(a)	(a)

(a) No change from previously listed value.

(b) Area from which people would need to be removed until decontamination was complete.

In connection with the estimates of probability one should keep in mind the comments of The Risk Assessment Review Group under the chairmanship of H.W. Lewis, as recorded in its report published in September,1978.[13]

"We are unable to define whether the overall probability of a core melt given in WASH-1400 is high or low, but we are certain that the error bands are understated. We cannot say by how much. Reasons for this include an inadequate data base, a poor statistical treatment, an inconsistenty propagation of uncertainties through the calculations, etc."

"We do find that the methodology, which was an important advance over earlier methodologies applied to reactor risks, is sound, and should be developed and used more widely under circumstances in which there is an adequate data base or sufficient technical expertise to insert credible subjective probabilities into the calculations".

6. Conclusion

The foregoing illustrative description of the nature of reactor accidents has necessarily been simplified. The four accidents described were chosen from a large set of potential failures and failure combinations. While they are typical of that set and while they cover the potential range of probability and consequence, there are many other possible accident sequences which are not really characterized by these four.

The values given for the probabilities of specific accidents and their associated consequences are open, no doubt, to argument. This paper identifies the many uncertainties associated with these estimates and these uncertainties should be kept in mind when interpreting these values. In general, the quoted consequences bound or over-estimate the likely consequences by using conservative assumptions where uncertainties are felt to exist. Thus,the likely effect of this approach would be to produce consequences estimates which are too large.

Finally, this paper has confirmed itself to a discussion of reactor accidents and has not described the elements of reactor safety - the means by which a high level of safety is achieved and maintained in nuclear power plants. As such, the paper is somewhat one-sided.

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APPENDIX 1

REACTOR ACCIDENTS HAVING SIGNIFICANT CONSEQUENCES

ACCIDENT	DATE AND LOCATION	SUGGESTED READING (see numbered reference below)
NRX	1952; Chalk River, Ontario	1, 2, 3, 4
WINDSCALE	1957; Windscale, England	5, 6
NRU	1958; Chalk River, Ontario	7, 8, 9
SL-1	1961; Nuclear Reactor Test Station, Idaho, USA	10, 11, 12, 13
FERMI	1966; Chicago, USA	14, 15
LUCENS	1969; Lucens, Switzerland	16, 17
BROWN'S FERRY	1975; Decatur, Alabama USA	18, 19
TMI	1979; Harrisburg, Pennsylvania, USA	20, 21, 22

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