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Health and Safety Aspects of Nuclear Power Plants

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HEALTH AND SAFETY ASPECTS OF NUCLEAR POWER PLANTS

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ABSTRACT

This paper summarizes the health and safety impacts of nuclear power plants. It reviews the basic principles of plant safety and describes engineering systems designed to prevent or mitigate nuclear accidents.

The report addresses the current state of the art with respect to protecting public health and safety and the current state of scientific understanding of the health effects of radiation originating from routine as well as accidental releases of radioactivity from power plant operations. The causes and consequences of the accidents at Three Mile Island and Chernobyl are reviewed.

New reactor designs, which are expected to improve both the safety and the economics of nuclear power, are also discussed. These include the "evolutionary" advanced light water designs and other innovative designs which claim substantial safety advantages over the current generation of reactors.

The report also reviews institutional aspects related to nuclear plant safety. It describes the regulatory, management, personnel-training, and quality-assurance provisions required in developing countries to ensure that nuclear power plants are operated safely. The pertinence of this aspect to Soviet-designed reactors in eastern Europe is also discussed.

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EXECUTIVE SUMMARY

The present report is the fourth and final of a series of reports undertaken by the Bank under the EMENA Regional Studies Program in the field of nuclear power. The objective of the studies was to provide a knowledge basis consisting of factual, up-to-date information on aspects of nuclear power that are needed in sector reviews of countries with existing or planned nuclear units. The method used was a systematic and selective collection, condensation, and presentation of existing information for the use by generalist as well as by technical power staff. The studies were not intended as policy papers but rather as background sources to support detailed or specialized analyses and reviews of nuclear power projects.

The scope of the studies included some of the topics and issues deemed of paramount importance. Prime among these issues is the safety of nuclear reactors, a subject of intense controversy and debate among technical experts, political groups, and the public at large. This issue, together with those covered in the other papers, has in fact caused a halt in the development of nuclear power in most countries. Although the Bank is not considering the financing of new nuclear power projects, it is nevertheless engaged in discussions with several borrowing countries that have a sizeable nuclear component in their power system on the efficient and economic operation of their present systems in a manner guaranteeing the environment as well as public health and safety. This issue has acquired increased urgency as the Bank has started a lending program to Central and Eastern European countries that possess a number of Soviet-designed reactors of an older generation that do not have adequate safety provisions and are plagued with a number of other operational and maintenance deficiencies. Prominent among them are Bulgaria with four older 440 MW and two newer 1000 MW units that provide about 36 percent of the country's electricity and Czechoslovakia with two plants each comprising 4 x 440 MW units providing 28 percent of total electricity and two plants under construction at Temelin (2 x 1000 MW) and Mochovce (4 x 440 MW).

The present report was undertaken to familiarize the Bank staff with the scientific facts on nuclear safety, summarize the major accidents to date, outline the design and operational provisions that have been made to ensure safety, and to provide guidance to Bank staff in identifying areas of weakness that may be in need of attention and improvement. These

may include technical aspects, operator training, safety procedures, and management streamlining and strengthening.

The report (Section 2) addresses the current state of scientific understanding of the health effects of radiation , both somatic and genetic, originating from routine as well as accidental releases of radioactivity from nuclear power plant operations.

The report (Section 3) addresses routine emissions of radioactive material that result from normal plant operations. Data are provided that show that the average radiation dose to the general public resulting from normal operation of nuclear power plants is a small fraction of the average individual dose received from natural sources of radiation in the environment, and that average doses resulting from the operation of nuclear power plants are a small fraction of the variation in natural background radiation dose received in different geographic regions of the world. Recent epidemiological studies have produced no statistically significant evidence of an excess rate of cancer incidence as a result of living near nuclear installations; nor have they produced statistically significant evidence of adverse health effects among nuclear power plant workers whose occupational doses (received in normal plant operations) can often exceed natural background radiation levels. It must be noted that although these statements are endorsed by the consensus opinion of the medical, scientific and technical communities and by official country and international bodies, there exist dissenting reports and views on the matter.

The report (section 4) reviews the basic principles of nuclear power plant safety, and describes engineering systems that are designed to prevent or mitigate nuclear accidents in typical pressurized water reactors in market economies. Within this context, the range of possible accidents that could occur in a power plant is discussed together with goals for power plant safety that have been established by regulatory authorities, particularly in the United States. Methodologies for estimating the risk of severe accidental releases of radiation from a plant are outlined, together with current estimates of such risk. The effectiveness of regulatory systems and management in achieving operating safety is discussed.

Draft guidance from the International Atomic Energy Agency (IAEA) proposes a fatality risk of no greater than one in a million per plant-year of nuclear power plant operation and a risk of a large off-site release of radioactivity of no greater than one in a million per plant-year. In a major assessment of accident risks recently conducted by the US Nuclear

Regulatory Commission, the risks of both prompt accidental and latent cancer fatality that actually exist at currently operating plants in the U.S. were found to be well below such safety goals.

Two major accidents have occurred in the history of commercial nuclear power, Three Mile Island (TMI) and Chernobyl. In the first accident which occurred in 1979, the reactor core suffered major damage and enormous loss of investment and cleanup cost ensued. However, the releases of radioactivity to the environment were minor (owing to the effectiveness of the containment structure) and no measurable direct health effects were observed other than psychologically induced illnesses. In the other accident which occurred at Chernobyl of the Soviet Union in April of 1986, 31 persons, principally from the firefighting team, died within a month from acute exposure of radiation, large scale evacuation of population was necessitated, and large land contamination resulted. The long-term effects of radiation exposure resulting in increased cancer incidence is a matter of intense debate. The number of cancer deaths in the Soviet Union due to such exposure is estimated to be in the range of 10,000 - 40,000 during the period of the next 30 years but may not be statistically observable among the 70,000,000 deaths from naturally occurring cancer expected in the same time period. The design of the Chernobyl reactor (15 of which were operational at the time of the accident) was not exported by the Soviet Union and is no longer used for future units. About twenty units are still being operated in the USSR.

The report (Section 5) reviews the causes and consequences of the TMI and Chernobyl accidents, as well as those of an accident that occurred at a weapons material production plant in the United Kingdom in 1957.

Many of the major Western international vendors of nuclear power plants are now offering new reactor designs which are expected to improve both the safety and economics of nuclear power. These technologies range from "evolutionary" advanced light water designs which provide incremental improvements over existing technology based on past operating experience while maintaining the same basic underlying technology, to more innovative designs which claim substantial safety advantages over the current generation of reactors by the use of new engineering approaches and extensive reliance on "passive safety" principles i.e. systems that rely on natural laws such as gravity and natural circulation of heated water rather than on engineered systems. The report (Section 6) describes these new designs with some emphasis on the passively safe reactors. The developers of such designs maintain that they may be simpler in construction and operation and less dependent on active operator intervention to ensure safety than current plants, and may, therefore, be more appropriate in developing countries which lack the technical infrastructure to run plants of great complexity.

With the less industrialized countries in mind, the report (Section 7) provides a description of the regulatory, management and quality assurance infrastructure that would be required in developing countries to ensure that nuclear power plants could be operated safely.

This aspect is particularly pertinent to Central and Eastern European countries where proposals are being made for the upgrading and retrofitting (and possible decommissioning) of existing nuclear units and where substantial improvements in the existing regulatory regimes, human factor and management structure and practices would be necessary so that continued operation of the units does not endanger public health and safety. This issue is of an importance that transcends the borders of any particular country and concerns all the countries of Europe.

1.0 INTRODUCTION

This report is one of a series of World Bank reports on various aspects of nuclear power.[1] The series provides background information to Bank staff members who are dealing with energy issues in countries with an existing or planned nuclear component in the electricity generation sectors. The Bank's interest in nuclear power has recently increased because of increased lending to Eastern European countries (Bulgaria, Czechoslovakia, Hungary and Yugoslavia) and involvement in energy sector reviews in those countries.

The present report analyzes the health and safety experience of nuclear power plants and addresses the current state-of-the-art with respect to protection of public health and safety. The report also considers safety issues concerning the introduction of nuclear power in less advanced nations.

Nuclear energy is now a mature industry. In 1989, nuclear power provided 16.8% of total worldwide electricity generation, and is expected to increase that contribution in the years to come. At the end of 1989, 426 commercial reactors were in operation worldwide for a total of about 318,000 MWe of generating capacity. Another 96 were under construction, with a total capacity of 79,000 MWe. Table 1-1 identifies the amount of nuclear power generating capacity in various countries and nuclear power's share of total power generating capacity in each country.[2]

The large majority of the nuclear power plants in the world today use light water as the coolant and are known generically as light water reactors (LWRs). They consist of two types - the pressurized water reactor (PWR) or the boiling water reactor (BWR). Other types use heavy water or gas as the coolant. This report focusses principally on the LWRs because of their prevalence in the world's electric power systems.

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[1] Other reports are:
The Status of Nuclear Power - An Update, Energy Series Paper No. 27, April 1990;
Decommissioning of Nuclear Power Facilities, Energy Series Paper No. 28, April 1990;
Radioactive Waste Management, Energy Series Paper No. 36.

[2] International Atomic Energy Agency, Nuclear Power, Nuclear Fuel Cycle and Waste Management: Status and Trends 1990, Part C of the IAEA Yearbook, IAEA, Vienna, August 1990.

TABLE 1-1

DISTRIBUTION OF INSTALLED NUCLEAR POWER GENERATING CAPACITY IN THE WORLD (31 DECEMBER 1989)

	MW(e)	%
States members of the OECD	**256,113**	**80.47**
USA	98,331	30.90
France	52,588	16.52
Japan	29,300	9.21
Germany, Fed. Rep.	22,716	7.14
Canada	12,185	3.83
UK	11,242	3.53
Sweden	9,817	3.08
Spain	7,544	2.37
Belgium	5,500	1.73
Switzerland	2,952	0.93
Finland	2,310	0.73
Italy	1,120	0.35
Netherlands	508	0.16
States members of the CMEA	**43,826**	**13.77**
USSR	34,230	10.75
Czechoslovakia	3,264	1.03
Bulgaria	2,585	0.81
German Dem. Rep.	2,102	0.66
Hungary	1,645	0.52
States members of neither organization	**13,408**	**4.21**
Korea, Rep.	7,220	2.27
South Africa	1,842	0.58
India	1,374	0.43
Argentina	935	0.29
Mexico	654	0.21
Yugoslavia	632	0.20
Brazil	626	0.20
Pakistan	125	0.04
TOTAL	**318,271**	**100.00**

Note: The total includes 4924 MW(e) of generating capacity in Taiwan, China, representing 1.54% of world capacity.

Source: International Atomic Energy Agency, Nuclear Power, Nuclear Fuel Cycle and Waste Management: Status and Trends 1990, Part C of the IAEA Yearbook, IAEA, Vienna, August 1990, p. C4.

In the past two or three years, there has been some renewed interest in the production of electricity by nuclear power because of heightened concern over global warming, to which fossil fuel plant emissions are believed to be a major contributor. Nuclear reactors do not give rise to such pollutants.

As discussed in greater detail in the report, the operation of nuclear power plants results in the production of radioactive materials. Since radioactive materials can cause injury to living organisms, it is essential that nuclear power plants be operated in a safe manner to ensure that the release of these materials to the biosphere is kept to very low levels. Preventing releases of radioactive materials to the biosphere entails limiting "routine" emissions as well as avoiding accidents that have the potential for releasing large amounts of radioactivity. In addition to preventing releases outside the plant, safe plant operation requires that radiation doses to nuclear power plant workers be kept to very low levels.

This report attempts to characterize the risks to health and safety posed by routine and accidental releases of radioactivity from nuclear power plants; summarize and discuss the principal safety issues relative to the current technology; assess the prospects for safety improvements in advanced reactor designs that offer the promise of passive or "inherent" safety; and address safety issues concerning deployment of nuclear power facilities in developing countries.

In Section 2.0 of the report, the current state of scientific understanding of the health effects of radiation is addressed. The health risks from normal nuclear power plant operations and from potential accidents are subsequently addressed in Sections 3.0 and 4.0, respectively. Section 5.0 discusses the three major nuclear power plant accidents that have occurred to date and their consequences. Section 6.0 describes safety improvements incorporated into advanced reactor designs currently under development. Finally, Section 7.0 provides a brief commentary on nuclear safety-related questions that should be addressed when considering nuclear power development in developing countries.

Public Concern About Nuclear Power

The safety of nuclear power stations is a matter of considerable public concern. Past accidents such as the 1986 explosion of the reactor at Chernobyl in the Soviet Union and the 1979 accident at Three Mile Island (TMI) in the United States have contributed greatly to public fears (while also providing numerous lessons to the nuclear industry and its regulators that should lead to safety improvements at nuclear facilities). Public consciousness of the risk from nuclear power and associated activities appears to be much higher than for similar activities that pose comparable risks. The reason for this is a subject of long-standing debate; one reason could be a mental association of nuclear power with nuclear weapons. (In this regard, it is important to note that a nuclear explosion such as that which occurs in an atomic bomb cannot occur in a nuclear power plant because it contains neither the necessary type nor configuration of nuclear materials.) Another reason could be fear of the invisible risk posed by radiation, and the involuntary nature of the risk.

Regardless of the specific reasons for public fears of nuclear power and for higher aversion of nuclear than other hazardous industrial activities, it is clear that continued use of nuclear power in developed and developing countries, and the prospects of its further development, requires not only firm assurance that technical and institutional measures will be effective in protecting public health and safety but also that public confidence and broad political support can be obtained. The technical complexity of nuclear power technology is one barrier to public understanding, making it difficult for many members of the public to evaluate safety questions for themselves. This report does not address the question of public acceptance of nuclear power but rather seeks to characterize its health and safety risks, describe safety technology and regulations, and place the risk in perspective.

It should be noted here that recently reported safety problems in a number of facilities in the USA which are dedicated to the production of nuclear weapons materials have led to further public apprehension about nuclear matters. These facilities should not be confused with civilian nuclear power plants which are the subject of this report. Civilian plants carry out a completely different function and are licensed in a totally different manner by a civil regulatory body specifically set up for that purpose. Nonetheless, it is likely that public concern over the problems at these facilities will serve to exacerbate concerns over civilian facilities.

The Nature of Radiation

To assist the reader in understanding portions of this report dealing with radiation exposure, risks and consequences, a brief discussion of the nature of radiation and some of the units of radiation exposure and radioactivity is presented here.[3]

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[3] Further discussion of radiation measurements and units may be found in Appendix 1 to this report, "Radiation Fundamentals."

It is important to clarify the difference between "radioactive materials" and "radiation." In this context, radiation refers to the energy emitted from radioactive materials in the form of waves or particles as such materials decay due to instabilities within the atom. Radioactive materials that are routinely emitted from nuclear power plants into air and water or that could be emitted in an accident may be ingested or inhaled by humans, after which they will continue to decay and thus cause a radiation dose to be delivered to the exposed person. Various pathways for such internal consumption of radioactive material are the principal concern with respect to limiting radiation doses from nuclear power plants. The actual direct emanation of radiation from a nuclear power plant -- radiation "shine" -- is limited to the close proximity of the reactor itself and is of little concern to the general public living in the vicinity of the plant, although it is the principal concern with respect to personnel working at the plant (occupational exposure).

The radiation dose received by a nuclear power plant worker or a member of the public in the vicinity of a plant is measured in terms of the amount of energy deposited by the radiation in live tissue. The "rem"[4] is one of the standard units for measuring radiation dose and is the unit used throughout this report. Another important measure of radiation dose is the collective dose, which expresses radiation dose integrated over the population. Collective dose may be expressed in units of person-rem. As an example of the relationship between individual dose and collective dose, an exposure which causes 10 individuals to receive a dose of 1 rem each would produce a collective dose of 10 person-rem.

Finally, radioactivity is measured in terms of the number of radioactive emissions (or disintegrations) from a given quantity of material per unit time. The standard unit for measuring quantities of radioactivity that is used in this report is the curie. Each radioisotope has its own characteristic rate at which it "decays" as a result of these emissions. The time required for any given radioisotope to decrease to one half of its initial quantity is a measure of the speed with which the radioisotope undergoes radioactive transformation. This period of time is known as the half-life and is characteristic of that particular radioisotope. Materials which decay slowly -- i.e., that have a long half-life -- are thus less radioactive than materials that have a short half-life. Half-lives vary tremendously, from microseconds to billions of years. For example, krypton-90 has a half-life of 33 seconds, whereas the half-life of potassium-40 (which is naturally present in food, e.g. orange juice) is 1.3 billion years.

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[4] Stands for "radiation equivalent man" (rem). Detailed definitions of the rem and other units are provided in Appendix 1.

Natural Background Radiation

The most significant sources of radiation to which humans are routinely exposed exist naturally in the environment. These include cosmic rays; radiation emanating from the ground and building materials and from radon gas and its decay products; and radioelements in food and in the body. As a benchmark, the United States National Council on Radiation Protection and Measurements (NCRP) estimates that the average American receives from these natural background sources of radiation a dose of roughly 300 millirem/year (i.e., 0.3 rem/year), 200 millirem of which is the result of radon gas and its decay products.[5]

Actual individual doses due to natural background sources of radiation vary over a wide range depending on each person's place of residence and activities, which determine the exposure to these sources. For example, a person living in Colorado might receive an additional 100 millirem/year mainly because of the high altitude and the resulting higher levels of natural cosmic radiation. Similarly, a person taking a coast-to-coast airplane flight in the US would receive a dose of about 5 millirem due to cosmic radiation; a person taking a one-week vacation in the Massif Central or Brittany regions of France would receive an additional 2 to 5 millirem. Natural background radiation levels depend not only on variations in cosmic radiation with altitude, but also on variations in terrestrial radiation levels from different rock types.

Medical and dental radiation exposure brings the average person's radiation dose up significantly. The NCRP estimates a US average annual exposure from these sources of 53 millirem.[6] Consumer products contribute on average an additional 10 millirem per year. As a result, the average total radiation dose to individuals in the US resulting from natural background, medical, dental and consumer product radiation is about 360 millirem. The percentage of total radiation dose resulting from each of several sources is identified in Figure 1-1.

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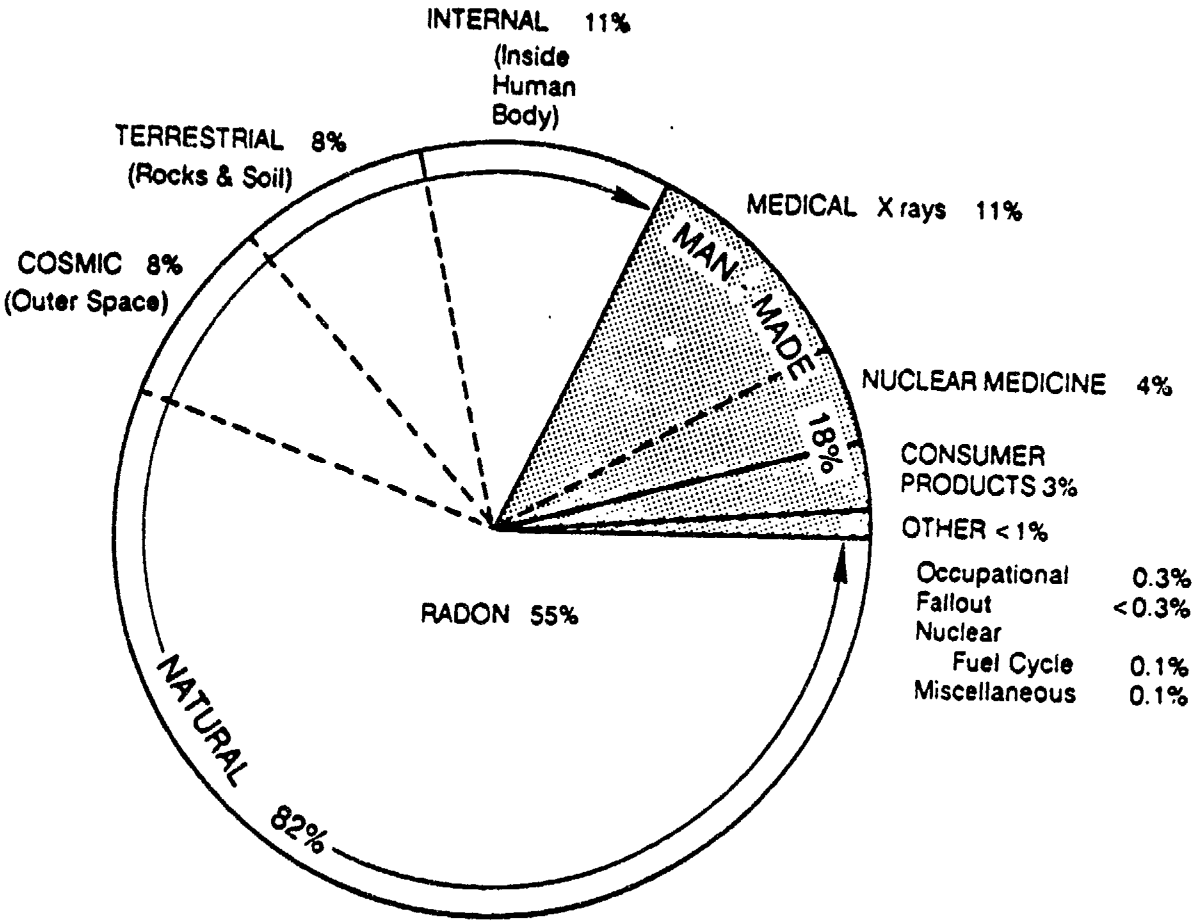
[5] National Council on Radiation Protection and Measurements, Ionizing Radiation Exposures of the Population of the United States, Report No. 93, 1987.

[6] Idem.

FIGURE 1-1

SOURCES OF RADIATION EXPOSURE



Source: National Research Council, Committee on the Biological Effects of Ionizing Radiations, Health Effects of Exposure to Low Levels of Ionizing Radiation (BEIR V), National Academy Press, 1990, p. 19.

It should be noted that smokers are also exposed to the radionuclide polonium-210 which occurs naturally in tobacco, resulting in a radiation dose to the lungs of up to 20 rem, which is a very large dose compared to natural background radiation exposure.[7]

Data on radiation exposure resulting from routine nuclear power activities and comparison with natural background and other sources of radiation are given in Section 3.4 of the report.

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[7] National Research Council, Committee on the Biological Effects of Ionizing Radiations, Health Effects of Exposure to Low Levels of Ionizing Radiation ("BEIR V"), National Academy Press, 1990.

2.0 HEALTH EFFECTS FROM LOW-LEVEL RADIATION DOSES

2.1 HISTORICAL PERSPECTIVE AND BACKGROUND

The study of the effects of ionizing radiation on populations extends back to the end of the last century when medical scientists were beginning to realize that the use of radioactive materials (radium) and radiation (x-rays) in the diagnosis and treatment of patients might well lead to hitherto unknown side effects. Some side effects were already well known; for example, skin erythema and the loss of hair showed up rather promptly after exposure to the relatively large doses from therapeutic use of x-ray machines. However, the delayed effects of radiation, such as cancer, were unknown until they began to appear at a much later date in patients who had been treated with radiation or radioactive materials and in workers occupationally exposed.

One of the early examples of the effects of radionuclides taken into the body comes from the ingestion of radium by painters of fluorescent watch dials who consistently wetted and pointed their brush tips with their lips during the period 1915 to 1935 when the practice was stopped. High incidence of bone cancers and head carcinomas were observed among these workers and also among patients who had been treated internally with radium for tuberculosis of the bone.

During the 1920's and early 1930's a significant increase in mortality from leukemia among radiologists also began to be noticed. The actual doses received by such persons are very uncertain since at that time it was not common practice to monitor for radiation dose. It is, however, significant that these cancer excesses have not been observed in radiologists who entered practice after the 1930's when greater protective measures were taken.

One example of the latent effects of x rays in specific populations was the discovery later in life of an excessive likelihood of tumors in individuals who had been treated with x rays in their childhood for scalp ringworm or enlarged thymus glands.

These observations clearly demonstrated the delayed effects of internal and external radiation in fairly large doses and, together with other observations, led the medical scientists of the day to speculate on whether exposure to much lower doses of radiation might lead proportionally to similar latent effects particularly when exposure was prolonged over a period of time.

The most comprehensive information relating health effects to radiation exposure over a wide range of dose arose much later from the medical histories of the surviving populations who were exposed to the atomic bombing of Hiroshima and Nagasaki in 1945. The effect of high doses was immediately visible, but it was not until some time later that an excess of leukemias began to be observed among the survivors. In the ensuing years, medical follow-up of the survivors has revealed excesses of other cancers including cancers of the lung, stomach, thyroid, and breast.

There is little doubt that exposure to high doses of radiation increases the potential for cancer in humans as these experiences have demonstrated. However, extrapolating latent effects of radiation to demonstrate that an increased risk exists at lower radiation doses is quite another matter and much more problematic. Even in exposed populations such as those in Japan, the effects of exposure are not easy to quantify. For example, the medical records of the 80,000 atomic bomb survivors who were followed up from 1950 to 1978 showed that 23,500 persons had died, of which 4,750 had died of cancer. It has been estimated that only 250 of these were attributable to radiation exposure, i.e. in excess of the expected number of cancer deaths in a similar, but not exposed population.

To obtain an estimate of the effects of low doses, such as those experienced in and around nuclear installations, requires extrapolations from high dose observations based on either: 1) empirical evidence at high doses, i.e. epidemiology, and then extrapolation to low doses; 2) laboratory data on the effects of low radiation doses on animals; or 3) theoretical formulations which seek to quantify the relationship between dose and effect. Significant uncertainties are associated with these techniques as explained below.

2.2 THE EFFECTS OF HIGH-LEVEL VERSUS LOW-LEVEL RADIATION DOSES

While doses in the range of a few rads[1] are generally regarded by the scientific community as being low-level doses, and doses in excess of 100 rads are regarded as being high-level doses, the demarcation line between high and low level radiation is not a scientifically defined one. Nevertheless, the United Kingdom Radiological Board regard low doses as being less than 1 rad per year or low dose rates as less than 10 rad per day.[2]

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[1] A rad is a measure of the radiation dose and is defined in Appendix 1.

[2] Health Effects models developed for the 1988 UNSCEAR Report, NRPB-R 226.

Part of the reason for this is that health physics is concerned with two types of exposure: 1) a single accidental exposure to a high dose of radiation during a short period of time, which is commonly called acute exposure, and which may produce biological effects within a short time after exposure and a relatively higher probability of latent effects such as cancer and genetic damage; and 2) long-term, low-level exposure, commonly called continuous or chronic exposure, where the results of the exposure is of a much lower probability and will not apparent for years. Any such exposures are the result of improper or inadequate protective measures.[3]

Exposure of the whole body to an extremely high dose of radiation (of the order of 1000 rads) is almost certain to result in death within a matter of weeks but if a limited area of the body is briefly exposed to a very high dose, this may not be fatal. In fact hundreds of rads are used in many therapy regimes.

For example, an instantaneous whole body dose greater than 500 rads would probably be lethal, provided no treatment was given, as a result of damage to the bone marrow and gastrointestinal tract but if the same total dose is received over a period of weeks or months, there is more opportunity for cellular repair and there may be no early signs of injury although damage to tissues may have occurred and may be manifested later in life, or possibly in the irradiated person's descendants. However, if only a limited area of the body is briefly exposed to a high dose of this nature, it may not be lethal but early effects may occur such as reddening of the skin (erythema) in a week or so[4].

The most important long term effect of radiation is cancer but the fundamental processes by which it is induced are not fully understood and, moreover, there is no way at present of medically distinguishing cancers caused by radiation from those occurring naturally and those caused by other carcinogens.

The main source of information on the risk of cancer following whole-body exposure to radiation comes from studies on the survivors from the atomic bombings of the cities of Hiroshima and Nagasaki. The risks derived from studies of these populations are based

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[3] Introduction to Health Physics by Herman Cember, Pergamon Press, 1988.

[4] Living with Radiation, UK National Radiological Protection Board, 1989.

largely on exposure to high doses delivered over a short period of time (tens of rads or more), whereas most people are only exposed to low levels of radiation over long periods of time.

Because of the relatively low probability of effects occurring due to exposure to low-level radiation over long periods of time and, the risks of such exposure can only be calculated from the data available on exposure to high levels of radiation.

It is generally assumed for radiation protection purposes that there is a simple proportional relationship between dose and risk and for radiations from alpha particles this appears to be the case. However, for beta and gamma radiations and x rays there is considerable evidence that the risk is less at low doses and low dose rates than at high doses given at high dose rates.[5] In either case, there is no sound basis for assuming the existence of a threshold below which no cancers or other health effects are induced.

2.3 NATURE OF EPIDEMIOLOGICAL STUDIES

One of the major problems in conducting epidemiological studies to estimate the effects of low-level radiation is the difficulty of identifying the influences of the multiple factors which have to be taken into account in order to estimate the effects. These factors include: 1) the quality of the exposure and medical data being used; 2) the selection of appropriate controls; 3) the methodology and scientific design of the analyses; 4) occupational conditions and personal habits and, 5) the validity of the statistics for a given population size as discussed in the following sections.

2.3.1 Quality of Data

One of the more complicated problems is the quality of the data used as input into the study. Epidemiology studies usually employ mortality data (death rates from a disease) or incidence data (the occurrence rate of a disease). Mortality data is usually based on death certificate information where it is often not known for certain whether the primary cause of death is cancer and whether it has been accurately represented on the certificate. For

[5] Idem 3

example, the type of cancer reported on the certificate may be the result of metastasis and not the site of origin. A similar problem exists in incidence data where diagnoses are based on "registrations" of a disease occurring, and might not be accurate.

2.3.2 Availability of Realistic Exposure Measurements

A second problem is the limited availability of realistic exposure data. In most cases involving the deaths from, or diagnosis of, cancer in persons who are not occupationally exposed to radiation, valid exposure data almost never exists.

2.3.3 Validity and Significance

Third, for an epidemiological study to have a high degree of statistical significance, it is necessary to have a large enough data base. Detecting a subtle rise in cancer incidence above the "normal" and correctly attributing such an increase to low levels of radiation exposure requires the study of very large populations.

2.3.4 Age-Dependence

Another complication is that cancer is well known to be an age-dependent disease which is rarer in young people and much more prevalent in older populations. Consequently any analysis of cancer rates needs to account for the age distribution in the area in question and age-adjusted corrections made.

2.3.5 Other Factors

Other factors which affect cancer mortality and need to be considered in the analyses are duration and age when exposure begins; sex ratios; racial and ethnic factors; exposure to other environmental agents (viruses, carcinogens, smoking); and even social structure.

2.4 ESTIMATION OF RISK OF CANCER FROM EXPOSURE TO RADIATION

The derivation of the estimates of the risk of induction of cancer from exposure to radiation are carried out by various national and international bodies composed of internationally renowned experts in the fields of radiobiology, radiation epidemiology, health physics, medical radiology, statistics, genetics, etc. These bodies conduct comprehensive reviews of the evidence relating to risk of cancer, hereditary effects and other diseases as a result of radiation exposure.

Such bodies include the International Commission on Radiological Protection (ICRP), the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), the National Academy of Sciences Committee on the Biological Effects of Ionizing Radiation (BEIR) and the National Council on Radiation Protection and Measurements (NCRP) in the United States, and the National Radiological Protection Board (NRPB) in the U.K.

In 1977 the ICRP published its basic recommendations on the effects of exposure to radiation in which it assumed that there is no threshold below which radiation effects are not harmful and took the position that the probability of harmful effects increases with dose.

Also in 1977, UNSCEAR published a comprehensive review of animal and human exposure to radiation and the induction of cancers, from which it derived an estimate of the risk of the induction of leukemia as being in the range of 2 cases per million of population per 100 mrem for low doses.

This value was used by ICRP in its 1977 report which calculated the risk factor for the induction of all fatal cancers as being in the range of 10 cases per million of population per 100 mrem, or 5 times the leukemia risk.

On this basis if a population of 50-million people receive a dose of 100 mrem above natural background radiation as a result of exposure to medical and other sources, 500 additional cancer deaths due to this "extra" radiation can be expected some time in the future, 100 of which might be a leukemia, out of an annual total of about 125,000 deaths normally expected from cancer.

The UNSCEAR risk estimates can also be used to compare the risks from exposure to natural background radiation with the number that might be expected from the operation of a power station which, under normal circumstances and conservative assumptions, would deliver an increment of 4 mrem per year over natural background (see Section 3 for further discussion of typical doses received routinely by persons living near nuclear power plants). In this case, if the risk factor assessed by the ICRP in 1977 is used for the calculation of the number of deaths expected from the natural background radiation in a local population of 100,000 persons, the estimated number of deaths would be 0.2, and the additional number from the operation of the power plant would be 0.008 deaths (or 1 death in 125 years) which is virtually impossible to statistically detect even by a massive study covering the whole of the regional population for a very large number of years.

In 1990 the BEIR Committee published its fifth update of its report (BEIR V). This suggests that these earlier estimates are too low and that the risk factor should be increased by a factor of four, or even by a factor of 10 as some have suggested. Even if the higher figure was used, it would still require a very substantial period of follow-up or very large population groups to be able to state with any certainty that the additional deaths were due to the effects of radiation. Furthermore, it would still be extremely difficult to detect the additional number of deaths due to man-made radiation among those arising from natural background radiation.

2.5 STUDIES OF THE EFFECTS OF NATURAL BACKGROUND RADIATION

Since there are substantial variations in natural background radiation levels from one geographic region to another, as discussed in Section 1.0, it might be expected that some difference in the incremental health impact of this type of radiation exposure would be observable.

In an attempt to determine whether such difference exist, a number of studies have been carried out in the USA, the UK and several other countries to investigate cancer incidence in different regions where the population is exposed to different levels of natural background radiation. The results of these studies have been inconclusive.

Since the level of radiation that is routinely emitted from nuclear power plants (see Section 3) is substantially less than the variations in natural background radiation levels from one geographic region to another, it is not surprising that the incremental health impact of this type of radiation exposure, if any, is also very difficult to detect.

2.6 LEUKEMIA CLUSTERS

There have been persistent reports of leukemia in young persons living near nuclear facilities which have been described as "leukemia clusters". This issue is briefly addressed in this section.

In the context of a disease such as leukemia, the word "cluster" is generally used to describe an observation of an unusually high incidence of the disease in a small geographical area within a relatively short time period. It has also been used to refer to the persistent increased occurrence of the disease in a small area, such as might occur if the population of that area was permanently exposed to risk from a causative agent in the environment.

The actual rates and number of cases which occur in the locality being studied may be higher or lower than the national average. If the number is higher, then it may be described by some people as a "cluster". The definition of a leukemia cluster is complex but most specialists agree that it involves an unusually high incidence of leukemia in a small area for a limited time period. For example, it has been reported that a number of people living on the same road or in a small town developed leukemia within a few years of each other.

Reports of clusters of leukemia, due to unknown causes, have appeared in the medical literature for many years, one of the earliest being in the British Journal of Childhood Disease in 1917. Since the 1960's systematic searches, most not associated with nuclear energy, have been carried out in many parts of the world in an effort to determine whether leukemia cases tend to occur more closely together than would be expected by chance.

The results of these searches revealed the occurrence of leukemia "clusters" in the following places:

Leukemia	San Francisco	1948-55
Childhood leukemia	Buffalo, N.Y.	1943-56
Childhood cancer	Buffalo, N.Y.	1943-56
Childhood leukemia	Northumberland and Durham, U.K.	1951-60
Childhood leukemia	Liverpool, U.K.	1955-64
Childhood leukemia	Portland,Oregon	1950-61
Childhood leukemia	New Zealand	1953-64
Childhood leukemia	Atlanta, Georgia	1958-88
Leukemia/lymphoma	Bahrain	1966-76
Hodgkin's disease	King County, Washington, USA	1974-79

The clusters listed in the table above have been ascribed to a variety of causes including radiation, association with the dairy industry, and a flood disaster in New York.

It is still unclear whether leukemia occurs in clusters to a greater extent than would be expected by chance but it is clear that leukemia clusters can occur randomly without an apparent cause. This occurrence is consistent with statistical theory since a random distribution is not uniform and apparent clusters are the rule rather than the exception.[6]

2.7 GENETIC EFFECTS

Ionizing radiation can result in damage to the genetic material (DNA) in reproductive cells leading to mutations which may be transmitted to subsequent generations. Such mutations are not seen in irradiated individuals, but only in their immediate or generational offspring.

As the BEIR V report[7] points out, mutations in reproductive cells may occur spontaneously due to natural causes, including those which can be associated with exposure to natural background radiation. It is extremely difficult to estimate what small increments of mutations effects may be induced by man-made radiation above this spontaneous occurrence rate. Estimates of genetic effects in humans must rely more on results from experimental animal studies than on human epidemiology studies that are extremely sparse.

Genetic effects of ionizing radiation are detected through the study of certain endpoints such as chromosome abnormalities, spontaneous abortions, congenital malformations, or premature death.[8]

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[6] Dr. R.H. Taylor and Dr. G. Harte, Nuclear Electric plc. (United Kingdom), personal communication.

[7] "Health Effects of Exposure to Low Levels of Ionizing Radiation", BEIR V, 1989.

[8] It must be emphasized that mutations caused by radiation do not lead to the grossly deformed offspring as portrayed by popular science fiction. As the BEIR V report states: "Some mutations have drastic effects that are expressed immediately, and these are eliminated from the population quite rapidly."

Although it is generally believed by the scientific and medical community that there is a need to assess the genetic effects of radiation exposure, the perspective has changed since the 1950s. As the BEIR V report states:

> "...in regard to the induction of mutations, the greater current risk seems to result from exposure to chemical mutagens in the environment rather than from exposure of populations to radiation."

It is now clear that the more significant risk of health consequences in persons exposed to radiation is that of cancer, with genetic effects of lesser concern than earlier considered. As a consequence, substantial efforts have been made to limit personal exposure to reduce the risk of cancer and this in turn has limited genetically significant exposures.

2.8 CONCLUSIONS

Recent studies from the United Kingdom have reported increases in mortality from leukemia in young children, especially under the age of 10, living near certain nuclear installations. The reasons for these increases are not clear and there is no convincing evidence that they are connected with exposure to low-level radiation.

Nevertheless, because of the concerns raised by these observations, epidemiologic studies have been, and are continuing to be, carried out in the United Kingdom and a number of other countries with nuclear facilities in an attempt to determine whether there are any health effects on workers and populations living in the vicinity of those nuclear facilities which are explainable as a consequence of radioactive emissions from those facilities.

The most exhaustive of these studies that has been completed so far has been that carried out by the National Cancer Institute (NCI) in the United States.[9] This survey encompassed all 62 nuclear facilities that went into service in the United States prior to 1982 and evaluated over 900,000 cancer deaths occurring between 1950 and 1984 in 107 counties.

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[9] "Cancer in Populations Living near Nuclear Facilities", NIH Publication No. 90-874, July, 1990.

The results of this survey were evaluated by the Ad-Hoc Advisory Committee of medical and epidemiological experts set up by the NCI who concluded that the survey had:

> "produced no evidence that an excess occurrence of cancer has resulted from living near nuclear facilities. Further, measurements of radioactive releases from nuclear facilities indicate that the dose from routine operations is generally much below natural background radiation, and hence may be unlikely to produce observable effects on the health of surrounding populations."

The type of study undertaken by the NCI should help to provide the public with the reassurance that the normal operation of nuclear facilities does not pose undue public health risks.

3.0 ROUTINE RADIOACTIVE EMISSIONS

3.1 INTRODUCTION

Routine emissions of radioactive material in solid, liquid and gaseous forms result from the operation of nuclear power plants. These releases increase the amount of radioactivity in the biosphere; hence their impact on public health must be considered. Such emissions are an inevitable result of normal plant operations and must be clearly distinguished from non-routine, accidental releases from power plants (which are addressed in Section 4.0 of this report). This Section discusses the origins and quantities of such emissions, the doses involved, and their health impact. Furthermore, routine exposure of nuclear power plant employees to radiation must be taken into account and is reviewed in this Section.

In addition to nuclear power plants, nuclear fuel cycle facilities such as uranium mines and fuel reprocessing plants also routinely release small amounts of radioactivity to the environment which might be larger than the emissions from nuclear power plants. Such facilities also cause occupational radiation exposure. Since they are less likely to be used in developing countries, emissions from them are not addressed in this Section.

3.2 SOURCES OF ROUTINE EMISSIONS

There are three principal categories of radioactive materials produced as a result of the nuclear fission process in light water reactors: fission products, neutron activation products and tritium.

<u>Fission products</u> are produced when the uranium atoms in the nuclear fuel fission into two smaller atoms. They are produced in both solid and gaseous forms. <u>Neutron activation products</u>, in contrast with fission products, are produced outside of the fuel material in either the fuel cladding material, fuel assembly structural materials or the reactor structure itself. Neutron activation products result when neutrons emitted in a fission reaction are absorbed by these materials, thereby making them radioactive. Finally, <u>tritium</u>, which is the radioactive isotope of hydrogen, is produced in a variety of ways, including by neutron capture in the reactor's coolant water.

These radioactive materials find their way into nuclear power plant effluents in the following manner. During reactor operation, almost all fission products are retained within the uranium fuel material and the metal cladding within which such fuel is encased. (This fuel material is eventually removed from the reactor and either reprocessed into new fuel or disposed of in solid form as high-level nuclear waste.) However, a small percentage of the fission products may escape from the fuel rods through hairline cracks that may develop in the cladding material. Such cracks result either from welding defects or localized corrosion that occurs during reactor operation. As a result, the reactor's internal coolant water may become contaminated with gaseous and, to a lesser extent, solid fission products; similarly, small quantities of neutron activation products and tritium -- which are formed not in the fuel rods but in the coolant water or in structural materials that come into contact with the coolant water -- also contaminate the coolant water.

3.3 RADWASTE CONTROL SYSTEMS

Regulations require the application of "radwaste" systems (described below) whose purpose is to reduce radioactivity levels in plant effluents to what are believed to be safe levels, based on the current understanding of the effects of radiation, as discussed in Section 2.0 of this report. Regulations place numerical limits on such effluents, and require radioactive emissions to be reduced to levels that are "as low as reasonably achievable" (ALARA). The radiation doses to the general public that result from nuclear power plant operation after such reductions are made are discussed in this Section and are compared to natural background levels of radiation.

Radwaste systems consist of liquid and gaseous waste processing systems. Radioactive liquids are decontaminated in two ways: evaporation and demineralization. Both are methods of filtering the liquid effluents so as to separate the radionuclides from the water that will be discharged to the environment. Similarly, gaseous emissions are passed through a particulate filter to remove solid radioactive particles. Gaseous radioactivity is reduced by storing gaseous wastes in holdup tanks before discharge to air, thus allowing radioactivity levels to reduce by natural radioactive decay.

As a result of these decontamination procedures, the vast majority of the radioactivity that reaches the reactor coolant water is removed from that water, solidified by cementation or other methods, and shipped off-site for disposal as low-level radioactive waste. Effluents discharged from the plant into air and water contain only very small amounts of radioactivity that was not removed by these processes. The discharges must not exceed levels allowed under regulation. Tritium is particularly difficult to remove because it has chemical properties identical to hydrogen, and thus becomes an integral part of the reactor water.

The United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) has compiled data on the quantities of various radioactive materials that are discharged from reactors worldwide, which vary depending on the reactor type.[1] UNSCEAR's figures for average releases during the period 1980 to 1984, normalized per gigawatt-year of electricity production, are provided in Appendix 2. The principal categories of effluents are:

- Fission product noble gases (such as krypton and xenon), which have short half-lives and high radioactivity levels but produce low radiation doses because they are chemically inert;

- Activation gases produced in gas-cooled reactor operation, especially argon-41 and sulfur-35, which also have high activity levels but cause low doses;

- Tritium, which, as mentioned above, has chemical properties identical to hydrogen;

- Carbon-14, which has a long half life (5730 years) and therefore is of concern in terms of long-term dose commitment. It is produced mainly from reactions with nitrogen and oxygen in the fuel and moderator; heavy water reactors (HWRs) produce relatively high levels of C-14 due to presence of oxygen in the moderator;

- Iodine-131, which has a half-life of 8 days, is mobile in the environment, and selectively migrates to and irradiates the thyroid;

- Particulates, which either arise directly, as decay products of fission product noble gases, or from corrosion of materials in the primary coolant circuit; and

- Other liquid effluents.

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[1] United Nations Scientific Committee on the Effects of Atomic Radiation, Sources, Effects and Risks of Ionizing Radiation, 1988 Report to the General Assembly, United Nations, New York, 1988.

It should be noted that the figures presented in Appendix 2 are based on available data which are limited by the fact that releases of certain low quantity streams are not reported in all countries. However, they provide a general indication of the relative quantities of each category of effluent emitted by the various reactor types.

3.4 DOSE LEVELS TO THE GENERAL PUBLIC AND HEALTH IMPACT

The principal pathways for human exposure to radioactive effluents are inhalation; ingestion of food crops and animal products; ingestion of drinking water; ingestion of fish and invertebrates; air submersion; and ground irradiation. Other pathways which have been found to cause generally much smaller doses include direct exposure from waterborne activities (swimming, boating, shoreline recreation) and ingestion of crops that were irrigated with contaminated water.

Based on known airborne releases from nuclear power plants in the United States, estimates of the doses received by persons residing near those plants have been calculated by the US Nuclear Regulatory Commission (NRC)[2]. Table 3-1 gives the average distribution of doses in 1987 among the estimated population of 140-million living within 2 to 80 km around each site for 70 nuclear power plants in the US. About 84% of the population at risk from airborne releases has been estimated as receiving a dose commitment of between 0.000003 and 0.001 mrem[3]. About 0.4% of the population at risk received a dose of between 0.003 and 0.01 mrem.

The study did not estimate the maximum dose received by an individual, but licensee calculations at sites with the highest emissions indicated values of up to approximately 100 times the average individual doses, i.e., of the order of a few millirem per year.

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[2] Radioactive Materials Released from Nuclear Power Plants, Annual Report 1987, by J. Tichler et al, Brookhaven National Laboratory, for US Nuclear Regulatory Commission, NUREG/CR-2907, October 1989; and
Population Dose Commitments Due to Radioactive Releases from Nuclear Power Plant Sites in 1987, by D.A. Baker, Pacific Northwest Laboratory, for US NRC, NUREG/CR-2850, August 1990.

[3] For background information on radiation units and measurements, the reader is referred to Section 1.0 and Appendix 1.

TABLE 3-1

FRACTION OF TOTAL POPULATION RECEIVING VARIOUS AVERAGE INDIVIDUAL TOTAL-BODY DOSE COMMITMENTS FOR ALL SITES

Dose Range (millirem)	Percentage of Population
$< 1 \times 10^{-6}$	10
$1 \times 10^{-6} - 3 \times 10^{-6}$	3
$3 \times 10^{-6} - 1 \times 10^{-5}$	11
$1 \times 10^{-5} - 3 \times 10^{-5}$	18
$3 \times 10^{-5} - 1 \times 10^{-4}$	23
$1 \times 10^{-4} - 3 \times 10^{-4}$	21
$3 \times 10^{-4} - 1 \times 10^{-3}$	12
$1 \times 10^{-3} - 3 \times 10^{-3}$	1
$3 \times 10^{-3} - 1 \times 10^{-2}$	0.4

Source: Population Dose Commitments Due to Radioactive Releases from Nuclear Power Plant Sites in 1987, D. A. Baker, Pacific Northwest Laboratory, for USNRC, NUREG/CR-2850, August 1990, p. 1.8.

Similarly, using the international effluent data given in Appendix 2, UNSCEAR calculated doses to populations living near nuclear power plants resulting from each category of emissions and for each type of reactor. Their results are presented in Table 3-2. It should be noted that whereas the NRC figures given above are stated in terms of dose to the average individual, the UNSCEAR figures are given in terms of total population dose, which is equal to the average individual dose multiplied by the size of the exposed population. UNSCEAR also estimates that when these figures for nuclear power plants are combined with comparable figures for other nuclear fuel cycle facilities, the total population dose is 400 person-rem per gigawatt-year, so that with current nuclear power generation at about 190 gigawatt-years per year, the annual dose to the world's population resulting from nuclear power plants and nuclear fuel cycle facilities is about 76,000 person-rem. For comparison purposes, it should also be noted that UNSCEAR has estimated that radioactive material emitted to the atmosphere from the burning of coal in coal-fired electricity generating plants results in a population dose of about 400 person-rem per gigawatt-year also.[4]

For purposes of comparison with the individual and population doses due to nuclear power given above, it should be noted, as discussed in Section 1.0, that the average person receives a dose due to natural background radiation of roughly 300 mrem/year, including the dose from cosmic rays, naturally occurring radioactive materials in the ground and building materials (including radon gas), and radioelements in the body. As also noted in Section 1.0, medical and dental radiation exposure, plus exposure to radiation from consumer products, bring a person's radiation dose up significantly; the average total radiation dose to individuals in the US resulting from natural background radiation and these other sources is about 360 millirem. (In addition, smokers are also exposed to the radionuclide polonium-210 which occurs naturally in tobacco, resulting in a radiation dose to the lungs of up to 20 rem.)

Furthermore, by multiplying the average individual background dose figure of 300 millirem/year by the world population of approximately 5 billion, the total worldwide population dose due to natural background radiation can be determined to be about 1.5 billion person-rem/year, which can be compared with the figure of 76,000 person-rem/year for world population dose due to nuclear power and nuclear fuel cycle facilities.

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[4] Idem 1

TABLE 3-2

AVERAGE POPULATION DOSE FROM WORLDWIDE NUCLEAR POWER PLANTS, 1980-1984, PER GIGAWATT-YEAR OF ELECTRICITY PRODUCED (person-rem)

Reactor Type	Effluent Category						
	Airborne Noble Gases	Tritium	Carbon-14	Iodine	Particulates	Liquids Discharged to River	Liquids Discharged to Coastal Waters
PWR	2.57	2.2	62	.058	2.4	0.16	0.36
BWR	.56	0.2	59	0.48	23	.066	2.8
HWR	N/Av.	23	1,140	.012	.022	N/Av.	N/Av.
GCR	N/Ap.	8.8	200	.072	0.68	N/Av.	N/Av.
LWGR	N/Av.	0.1	230	4.0	9.0	N/Av.	N/Av.

<u>Source</u>: United Nations Scientific Committee on the Effects of Atomic Radiation, <u>Sources, Effects and Risks of Ionizing Radiation</u>, 1988.

This comparison of the radiation doses routinely received by the population as a result of nuclear power and those received from natural and medical sources strongly suggests that routine emissions from nuclear power plants should have little or no impact on public health. To corroborate this assumption, a number of studies have been done to assess whether or not the incidence of cancer is greater in populations living near nuclear power plants than populations living in other locations. A report published in 1987 by the British Office of Population Censuses and Surveys[5] on cancer risk in the vicinity of nuclear facilities in England and Wales found that, overall, there was no evidence to conclude that cancer mortality near UK nuclear installations was higher than elsewhere in the U.K.. It did, however, note an increase in deaths from leukemia in young persons under age ten in the vicinity of the Sellafield fuel reprocessing plant but the reasons for this were not clear.

Subsequently, the National Cancer Institute in the US initiated a large-scale survey of the incidence of cancer in persons living near nuclear facilities in the United States, published in July 1990.[6] The authors conclude that the survey produced no evidence that an excess occurrence of cancer has resulted from living near nuclear facilities. They also conclude that measurements of radioactive releases from such facilities indicate that doses due to routine emissions are generally much lower than doses from natural background radiation and therefore may be unlikely to produce observable effects on the health of surrounding populations.

3.5 OCCUPATIONAL EXPOSURE

While the doses to the general public from nuclear power plants have been generally quite low and the health impact thus far undetectable, the doses to workers are naturally higher and often exceed natural background radiation dose levels. UNSCEAR has tabulated data on worldwide occupational radiation exposure at nuclear power plants during the period 1980 to 1984.

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[6] Cook-Mozaffari, PJ et al, "Cancer incidence and mortality in the vicinity of nuclear installations in England and Wales 1959-1980," Office of Population Censuses and Surveys, London, HMSO, 1987.

[6] Jablon S. et al, Cancer in Populations Living Near Nuclear Facilities, National Cancer Institute of the National Institutes of Health, NIH Publication No. 90-874, July 1990.

Table 3-3 gives the annual average individual dose levels broken down by country and reactor type. (Note: data for all years were not available from certain countries).

For the same period, UNSCEAR also determined the following average collective occupational dose levels on a per-gigawatt-year basis, grouped by reactor type:

Reactor Type	Collective Dose Per Unit Energy Generated (person-rem per gigawatt-year)
Light Water Reactor (LWR)	1300
Heavy Water Reactor (HWR)	400
Gas-Cooled Reactor (GCR)	500
High Temperature Gas-Cooled Reactor (HTGR)	10
Light-Water Cooled, Graphite-Moderated Reactor (LWGR)	200

Among light water reactors, it was found that the collective dose in BWRs can be up to a factor of two higher than in PWRs, possibly because more maintenance work in radiation areas is necessary in BWRs.

According to figures recently published by the US Nuclear Regulatory Commission, in 1989 nuclear power plant workers in the US received an average dose of 340 mrem that year (somewhat higher than the average natural background radiation dose level).[7] The average collective dose per reactor was 344 person-rems. These figures are based on exposure data at 108 PWRs and BWRs, and represent a 14% decline from 1988 levels. Broken down further, the average PWR work force received a collective dose of 296 person-rems and the average BWR work force received a collective dose of 439 person-rems.

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[7] US Nuclear Regulatory Commission, Internal Memorandum, LWR Occupational Dose Data for 1989, June 28, 1990.

TABLE 3-3

AVERAGE ANNUAL OCCUPATIONAL RADIATION DOSE AT
WORLDWIDE NUCLEAR POWER PLANTS, 1980-1984
(millirem per worker)

Country and Reactor Type	1980	1981	1982	1983	1984	1985
Argentina (HWRs)				500	400	800
Canada (HWRs)	280	240	180	300	260	180
Finland (2 PWRs, 2 BWRs)			170	90	180	140
Japan						
- LWRs (ratio PWRs:BWRs ~ 1:1)	190	160	140	130	120	100
- GCR	30	20	10	20	20	30
Netherlands (1 PWR, 1 BWR)	250	470	570	560	500	
Sweden (3 PWRs, 9 BWRs in 1985)		320	250	310	250	210
Switzerland (2 PWRs, 2 BWRs)	460	440				
United Kingdom (GCRs)	210	130	110	90	90	
United States						
- LWRs (ratio PWRs:BWRs ~ 2:1)	670	660	620	660	560	

Source: United Nations Scientific Committee on the Effects of Atomic Radiation, Sources, Effects and Risks of Ionizing Radiation, 1988.

4.0 THE SAFETY OF NUCLEAR POWER REACTORS

4.1 INTRODUCTION

The energy produced in the core of a nuclear reactor comes about as a result of the fissioning of the nuclei of uranium[1] atoms in the nuclear fuel. To convert the heat released through nuclear fission into electricity, a coolant flows through the reactor core to absorb the heat and make steam, which spins turbines that power electrical generators.

The materials produced from the fissioning of uranium (the "fission products") are highly radioactive and have to be strictly contained to prevent them from being released to the environment, as do other categories of radioactive materials produced in a nuclear power plant. As discussed in Section 3.0, small quantities of fission products and other radioactive materials contaminate reactor coolant water under normal operating conditions. This contamination is largely removed using specially designed systems which ensure that routine releases of contaminated water to the environment remain strictly within regulatory limits.

More importantly, accidental releases of radioactive material to the environment must be prevented. There are many possible events which could lead to accidental releases; for example, in a pressurized water reactor (PWR), the rupture of one of the many tubes in the heat exchangers would allow radioactive water to escape from the primary coolant into the secondary coolant. The large pipes of the secondary cooling circuit penetrate the containment structure of a PWR. Consequently, once outside the containment, radioactive material leaked from the primary system through the ruptured tube could then potentially escape into the environment.

However, the principal event which could lead to large quantities of radioactive material being released to the environment is the loss of coolant accident (LOCA). A LOCA could be initiated by a number of means such as steam line breaks, sudden expulsion of control rods, loss of offsite power, and severe natural phenomena such as earthquakes, tornadoes

[1] In addition to uranium, plutonium that is produced as the result of the absorption of neutrons by uranium-238 -- i.e., mainly plutonium-239 -- is also fissionable and contributes to the total power generated in the reactor core.

and hurricanes. The most serious event which has been postulated would be a break of a major pipe in the circuit that provides coolant to the reactor core, which could lead to the loss of substantial quantities of coolant. It is essential that the core be kept completely covered with coolant at all times. Otherwise the core, or parts of it, could overheat and the fuel elements could degrade and possibly melt, resulting in the release of large quantities of radioactive materials into the reactor vessel.

It is primarily to the prevention of such LOCAs and their potential consequences that nuclear reactor safety is addressed. To prevent LOCAs from evolving into serious accidents with offsite releases of radioactive material, reactors must be designed to shut down quickly and reliably when necessary, and redundant cooling systems must be available to remove the heat that remains in the reactor core after the nuclear chain reaction has been shut down (see Figure 4-1). In the event that such systems fail and fuel melting does occur, possibly allowing radioactive materials to breach the reactor vessel, most nuclear power plant designs include a large concrete containment building that would limit the release of radioactive materials to the environment.

This Section reviews the general principles behind nuclear power plant safety and describes the basic systems in a nuclear power plant designed to prevent or mitigate nuclear accidents. Subsequently, a discussion of the range of possible accidents that could occur is provided, the methodology for estimating the risk of severe nuclear power plant accidents is explained, and current estimates of such risk are given. Finally, the effectiveness of regulatory systems and management on safety, including the impact of human factors, are addressed.

Most of this discussion focuses on light water reactors (LWRs) -- comprising PWRs and boiling water reactors (BWRs) -- which account for about 78% of the nuclear power plants in operation worldwide today (or 86% in terms of net MWe). The other major reactor types currently in use are gas-cooled reactors (9%), heavy water reactors (7%), graphite-moderated light water reactors (5%) and liquid metal fast breeder reactors (1%).[2] Safety aspects of new, advanced reactor designs are discussed in Section 6.0.

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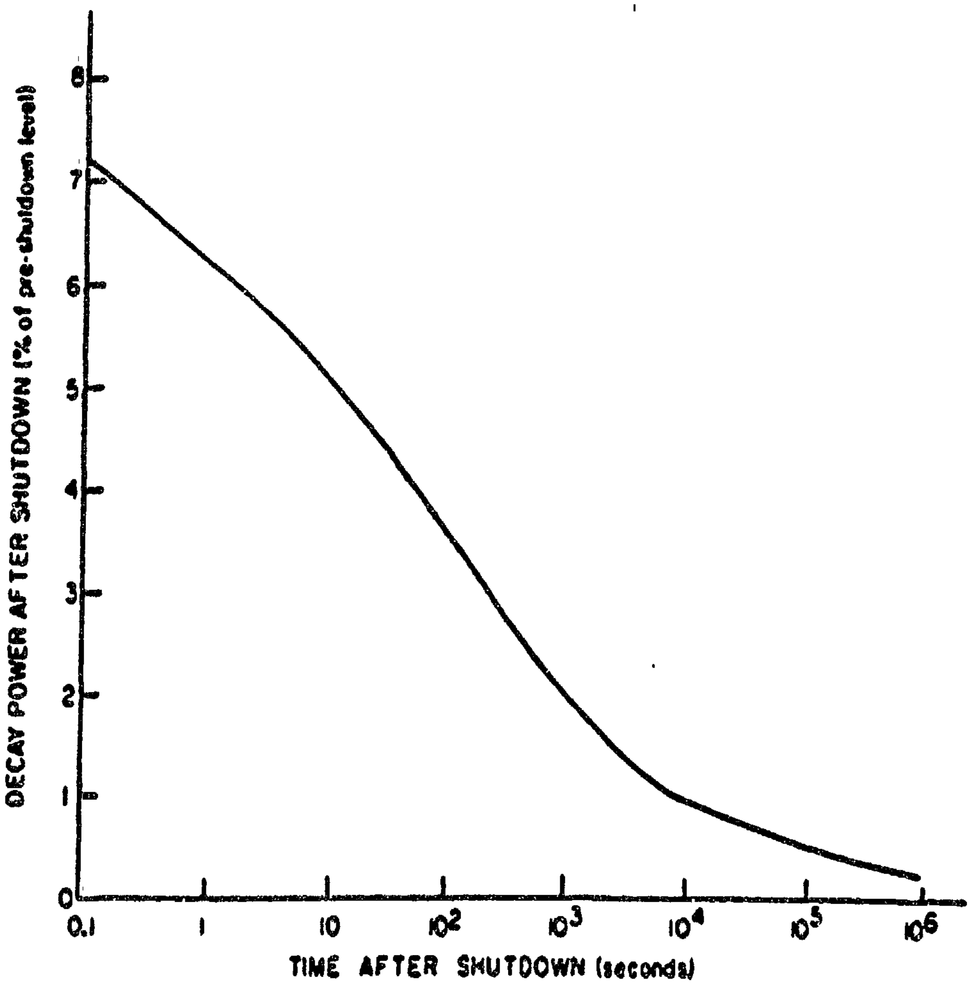
[2] "World List of Nuclear Power Plants," Nuclear News, Vol. 34 No. 2, February 1991, p. 72.

FIGURE 4-1

THERMAL POWER AFTER REACTOR SHUTDOWN



After the nuclear chain reaction ceases, radioactivity remaining in the fuel will generate heat as a result of radioactive decay. Assuming that the reactor had been operating for a substantial period, the power generated immediately after shutdown will be approximate 7% of the level before shutdown. For a 3000 MWth reactor, with 1000 MWe capacity, this implies and initial decay power level of about 200 MWth. Due to the rapid decay of short-lived species, this decay heat level decreases rapidly, but it is this heat that imposes the requirement that, in a light-water reactor, cooling water remain available to prevent damage to the fuel.

From A.V. Nero, Jr., <u>A Guidebook to Nuclear Reactors</u>, University of California Press, 1979, p. 54.

4.2 SAFETY PHILOSOPHY: DEFENSE IN DEPTH

The concept of "defense in depth" is the most fundamental principle underlying the safety of today's nuclear reactors. As stated in the International Nuclear Safety Advisory Group's basic safety principles, published in 1988, it centers on having "several levels of protection including successive barriers preventing the release of radioactive material to the environment."[3] Defense in depth includes 1) avoiding accident "precursors" that could lead to physical damage to the plant and to the various barriers to the release of radioactive material (accident prevention); and 2) measures to a) prevent accident precursors from evolving into accidents and b) protect the public and the environment from harm in the event that accidents do occur and barriers to the release of radioactive material are not completely effective (accident mitigation).

There are five levels of protection under the defense in depth philosophy:

1. Conservative design, quality assurance, surveillance activities and a general safety culture. This combination is intended to ensure that the reactor and various plant components will operate with a high degree of reliability with only a small chance of malfunctioning.

2. Control of operations, including the ability to respond to abnormal, anticipated events or to any indication of system failure. Redundant instruments monitor the various operational process variables (such as the temperature of water as it leaves the reactor) and trigger automatic responses, such as shutting down the reactor, when necessary. An example of an abnormal event would be the loss of off-site power to operate critical safety systems that could be needed in an emergency, which is compensated by having several backup electricity generators at the plant.

3. Engineered safety features (ESFs) that halt the progress of accidents that are considered in the design and, when necessary, mitigate their consequences. The most extreme among the range of accidents considered in the design are

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[3] Basic Safety Principles for Nuclear Power Plants, International Nuclear Safety Advisory Group, International Atomic Energy Agency, Safety Series No. 75-INSAG-3, 1988.

termed *design basis accidents*, such as a major break in the primary coolant system that leads to a LOCA. *Severe accidents* that are beyond the design basis have a low probability of occurrence, but backup safety systems offer protection even if an accident progresses beyond the design basis assumptions.

ESFs work in parallel or as backup to normal operating systems to safeguard essential safety functions: controlling reactor power, cooling the fuel, and confining the radioactivity within the reactor. An example of this is a system which automatically moves the control rods into the core to shut down the reactor in the event of an equipment malfunction in the core. Since the coolant water must still continue to circulate so as to prevent a heat build-up and possible fuel melting, another ESF provides emergency core cooling in the event that the primary coolant circuit has also been damaged. The containment structure around the reactor is another ESF, serving to prevent or limit the release of any radioactivity that is released from the reactor in an accident situation. (Further detail on ESFs is provided in Section 4.3.)

4. *Accident management strategies* to help operators decide quickly on appropriate actions. The aim of these measures is to prevent or limit damage to the core, preserve the integrity of the containment, and maintain the functions of design features such as vents and filters installed in the containment intended to preserve containment integrity in the event of a serious accident.

5. *Emergency planning*, which is intended to mitigate the radiological health consequences should an accidental release of radioactive material occur. Emergency planning includes early notification of an accident, radiation monitoring, decontamination, sheltering and/or evacuation of nearby residents within a specified radius of the plant, and possibly the administration of protective measures. (One such protective measure is distribution of potassium iodide tablets, which would block radioactive iodine that could be emitted in a nuclear accident from accumulating in the thyroid and increasing the risk of thyroid cancer.)

Another way of looking at defense-in-depth is in terms of the various physical barriers present in a nuclear power plant that serve to prevent the release of radioactive material.

These physical barriers include the fuel pellets themselves; the fuel cladding, which seals the fuel pellets into the fuel rods; the boundary of the primary coolant system; and, in most reactor designs, the containment structure, which is a hermetically-sealed building designed to confine radioactivity that might otherwise escape because of the failure of other safety barriers.

The 1979 accident at Three Mile Island (TMI), in the United States, which is discussed in detail in Section 5.0, is an example of a major accident in a reactor leading to the breach of several barriers in which a serious release of radioactive material to the environment was prevented because the containment structure maintained its integrity.

4.3 ENGINEERED SAFETY FEATURES

Current light water reactors and most other kinds of nuclear power plants are designed to withstand rare but potentially very serious events. This includes the rupture of a major coolant pipe (which could, under circumstances in which additional safety systems fail, result in the complete loss of coolant from the reactor core) as well as other initiating events such as those identified above. Protection against such events is provided by engineered safety features. The principal ESFs in PWRs are the emergency core cooling system; the containment building; systems that spray, clean and cool the containment atmosphere; auxiliary feedwater systems which ensure continuous heat removal in the plant's steam generators; and emergency electric power sources. Similar ESFs are used for BWRs.

The function of the emergency core cooling system and the containment building are reviewed in this section. For the sake of brevity, descriptions of these ESFs are provided for only the case of PWRs, which is illustrative of the basic safety principles that also apply to BWRs as well as other designs. (For discussion of the specific ESFs for the various reactor types, see Rahn, Adamantiades et al.[4])

It should be noted that in LWRs, water serves as both the moderator and the coolant. In BWRs, the heat generated in the reactor core turns the coolant water directly to steam. In contrast, in PWRs the coolant water that passes through the core is under such high pressure that it cannot boil, but rather transfers its heat in a "steam generator" to a

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[4] A Guide to Nuclear Power Technology: A Resource for Decision Making, F. Rahn, A. Adamantiades et al, Wiley, New York, 1984.

secondary coolant circuit that is maintained at a lower pressure, thus allowing the secondary coolant water to boil into steam.

4.3.1 Emergency Core Cooling System

Although early designs of LWRs included systems for core cooling, they did not include the type of emergency cooling systems that could be shown to prevent fuel melting in the event of a loss of coolant accident. This was raised as a major concern in the 1960s by nuclear safety researchers in the US, who felt that such systems must be included in reactor designs to prevent what was labelled "The China Syndrome."[5] Extensive hearings were held on this subject by the Atomic Energy Commission and its successor agency the Nuclear Regulatory Commission (NRC) in the United States. The critics of nuclear power participated in these hearings (and challenged the adequacy of the requirements for emergency core cooling that were imposed as a result of these hearings). Today the EmergencyCore Cooling System (ECCS) is seen as being critical to preventing the reactor core from overheating in the event of a LOCA. However, ECCS adequacy continues to be questioned by certain nuclear power critics.

If substantial melting were to take place in the core, large amounts of fuel would be deposited at the bottom of the reactor vessel, and, in the absence of restored cooling, could eventually melt through the reactor vessel. Even if such an extreme state is not reached, high temperatures and breach of the fuel cladding could drive off volatile fission products such as iodine, cesium and noble gases, which could then escape into the environment if there was also a breach of containment.

PWRs usually have three independent ECCS subsystems that operate at different system pressures. Each subsystem has multiple backups in terms of both equipment and flow path. If a small break in a PWR's primary coolant circuit occurred, causing a moderate pressure drop, the ECCS's high-pressureinjection systemwould be activated to replenish the primary coolant lost through the break.

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[5] This expression takes its name from a concept, initially spoken of in jest, which assumed that if the fuel in a reactor melted down it could burn its way through the bottom of the pressure vessel and concrete substrate and continue burning "all the way to China".

A larger break in the coolant circuit would cause a rapid large pressure drop and would actuate the *accumulator injection system*. This subsystem ensures that large tanks of water containing boron are available to flood the reactor core. (The element boron is a neutron absorber and would ensure the cessation of fission reactions in the reactor core.)

Finally, the *low-pressure injection system* would be actuated if pressure continues to drop below a preset level. This subsystem would continue to pump borated water from a large storage tank into the reactor long after the accumulators are empty, after which the subsystem automatically would switch to pumping borated water from the containment sump (into which excess water would have overflown). The accumulator injection system is a passive system; it does not require the operation of pumps and valves, which are dependent on a power supply. The high- and low-pressure injection systems, on the other hand, require active operation, i.e. the presence of a power supply.

4.3.2 Containment Systems

Containment systems are intended to hold essentially all the steam and radioactivity that might be released from the reactor vessel in a LOCA, and are provided for all LWRs as well as most other reactor types. The exception is a number of earlier design reactors in the Soviet Union, discussed later in this chapter, which have a less effective containment system.

A typical PWR containment structure is made of reinforced concrete, over one meter in thickness, with an internal steel liner. The entire primary coolant circuit, including the reactor vessel, is enclosed in the containment structure. The containment structure is capable of withstanding the maximum temperature and pressure that could be expected if all the water in the primary coolant circuit was expelled into the containment building as steam. Various systems inside the containment building would be available to cool and clean up the containment atmosphere. One of these involves spraying water on to the steam, or passing the steam over ice beds, to cause it to condense. In addition, sodium hydroxide may be introduced to the containment atmosphere through the containment spray system in order to remove radioactive iodine; similarly, the ice inventory may include chemical substances that would remove fission products from the containment atmosphere. Filters are also used to remove iodine and particulate matter. Radioactive noble gases, krypton and xenon, cannot be removed, but holding them in the containment building for a certain period allows them to decay substantially into non-radioactive species before they are released to the environment.

Unfortunately, when a serious accident occurred at the Chernobyl-4 reactor in the Soviet Union in 1986, the containment provided at the plant was incapable of containing the accident that occurred. A graphite-moderated light water reactor of the Soviet RBMK design, Chernobyl-4 was equipped with a concrete structure capable only of partial confinement of radioactive gases. Thus, the structure was unable to effectively contain the accident and massive graphite fire at the reactor. There are approximately 20 RBMK units currently operating in the Soviet Union. Details of the RBMK reactor and the Chernobyl-4 accident are described in Section 5.0.

A number of the Soviet PWRs (VVERs) do not have containment either. These Soviet PWRs do not comply with the guidelines established by the International Atomic Energy Agency (IAEA) and the regulatory agencies in Western countries and would not be licensed in most countries in the world. There are, moreover, several such reactors outside the Soviet Union: four in former East Germany (which have been shut down due to safety concerns), four in Bulgaria and two in Czechoslovakia. Evaluations are currently taking place on how to improve the safety of the Bulgarian and Czech plants to bring them up to Western safety standards. The four German plants are not expected to restart.

4.4 TYPE AND SEVERITY OF NUCLEAR ACCIDENTS

A wide range of incidents and accidents can be postulated in nuclear power plants, from minor incidents, such as when a specific operating procedure is not followed or a non-safety related piece of equipment malfunctions, to a variety of serious accident scenarios that are conceivably possible.

National regulatory authorities as well as the IAEA and the Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OECD/NEA) have, within the past few years, developed accident severity scales in order to facilitate communications and understanding among nuclear personnel, the media and the general public through the use of a simple classification system.

The French adopted an accident severity scale in April 1988 which includes six progressive categories, with the least important incidents designated level 1 and the most serious accidents being designated level 6. The levels are distinguished by the risk of radioactive discharge outside the installation where an accident has occurred. Lesser incidents are those in which radioactive discharges are below the allowed annual limits, or in which operating difficulties have occurred that do not impose directly a radiation risk but which may reveal weaknesses that need to be remedied.

An accident severity scale has also been developed and used in Japan. It is graduated into nine levels, from 0 to 8, and assesses event severity in terms of three criteria: effect on the public; effect on personnel; and effect on reactor safety (impact on integrity of "defense in depth," etc.). All reactor events that are reportable to the regulatory authority can be evaluated on this scale, including minor events having no effect on plant safety.

Drawing from the experience in using severity scales in France and Japan and proposals to use such scales in other countries, the IAEA and the NEA recently convened a series of meetings of experts and developed the International Nuclear Event Scale (INES). The scale will be used for a trial period of about one year in countries that choose to participate, although the French and Japanese continue to use their own scales for the time being. The INES is presented in three forms. First, the underlying logic of the scale is explained in a matrix, provided in Figure 4-2. As shown, events are considered in terms of three broad criteria: off-site impact, on-site impact, and degradation of defense in depth. Second, for public information purposes, it is presented by combining the three criteria and defining each level, as shown in Figure 4-3. Third, detailed guidance is provided to enable those assessing incidents to determine the appropriate level according to the international standard.[6]

The scale ranges from Level 0, for very minor events of no safety significance, to Level 7, for major accidents. It is divided into two sections, where Levels 1 to 3 relate to *incidents* and Levels 4 to 7 related to *accidents*. The scale is intended to be more or less logarithmic, so that each successively higher level on the scale should correspond to about a tenfold drop in the number of events. As examples of how past events have been classified in INES, the 1986 Chernobyl accident, which had widespread environmental and health effects, has been classified as Level 7. The 1979 TMI accident severely damaged the reactor core but had very limited offsite consequences; thus, it was classified as Level 5. The 1957 Windscale accident, in which there was a significant external release of fission products, has been classified as Level 5.

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[6] INES: The International Nuclear Event Scale, User's Manual, jointly prepared by IAEA and OECD/NEA, IAEA-INES-90/1, August 1990.

FIGURE 4-2

UNDERLYING LOGIC OF THE INTERNATIONAL NUCLEAR EVENT SCALE
(Criteria given in matrix are broad indicators only)

	CRITERIA		
Level/ Descriptor	Off-site impact	On-site impact	Defense-in-depth degradation
7 Major accident	Major release: Widespread health and environmental effects		
6 Serious accident	Significant release: Full implementation of local emergency plan		
5 Accident with off-site risks	Limited release: Partial implementation of local emergency actions	Severe core damage	
4 Accident mainly in installation	Minor release: Public exposure of the order of prescribed limits	Partial core damage Acute health effects to workers	
3 Serious incident	Very small release: Public exposure at a fraction of prescribed limits	Major contamination Overexposure of workers	Near accident Loss of defence-in-depth provisions
2 Incident			Incidents with potential safety consequences
1 Anomaly			Deviations from authorized functional domains
0 /Below scale			No safety significance

Source: IAEA and OECD/NEA, <u>INES: The International Nuclear Event Scale, User's Manual</u>, IAEA-INES-90/1, Vienna, August 1990, p. 2.

FIGURE 4-3

THE INTERNATIONAL NUCLEAR FUEL CYCLE
For Prompt Communication of Safety Significance

LEVEL	DESCRIPTION	CRITERIA	EXAMPLES
ACCIDENTS			
7	Major accident	. External release of a large fraction of the reactor core inventory typically involving a mixture of short and long lived radioactive fission products (in quantities radiolo radiogically equivalent to more than tens of thousands of terabecquerels of iodine-131). . Possibility of acute health effects. Delayed health effects over a wide area, possibly involving more than one country. Long term environmental consequences.	Chernobyl, USSR (1986)
6	Serious accident	. External release of fission products (in quantities radiologically equivalent to the order to tens thousands of terabecquerels of iodine-131). Full implementation of local emergency plans probably needed to limit serious health effects.	
5	Accident with off-site risks	. External release of fission products (in quantities radiologically equivalent to the order of hundreds to thousands of terabecquerels of iodine-131). Partial implementation of emergency plans (e.g. local sheltering and/or evacuation) required in some cases to lessen the likelihood of health effects. . Severe damage to a large fraction of the core as a result of mechanical effects and/or melting.	Windscale, UK (1957) Three Mile Island, USA (1979)
4	Accident mainly in installation	. External release of radioactivity resulting in a dose to the most exposed individual off-site of the order of a few millisieverts. Need for off-site protective actions generally unlikely except possibly for local food control. . Some damage to reactor core as a result of mechanical effects and/or melting. . Worker doses that can lead to acute health effects (of the order of 1 Sv).	Saint-Laurent, France (1980)

Source: IAEA and OECD/NEA, <u>INES: The International Nuclear Event Scale, User's Manual</u>, IAEA-INES-90/1, Vienna, August 1990, p. 4-5.

FIGURE 4-3 cont.

LEVEL	DESCRIPTION	CRITERIA	EXAMPLES
INCIDENTS			
3	Serious incident	. External release of radioactivity resulting in a dose to the most exposed individual off-site of the order of tenths of a millisievert. Off-site protective measures not needed. . High radiation levels and/or contamination on-site as a result of equipment failures or operational incidents. Overexposure of workers (individual doses exceeding 50 mSv). . Incidents in which a further failure of safety systems could lead to accident conditions, or a situation in which safety systems would be unable to prevent an accident if certain initiators were to occur.	Vandellos, Spain (1989)
2	Incident	. Technical incidents or anomalies which, although not directly or immediately affecting plant safety, are liable to lead to subsequent re-evaluation of safety problems.	
1	Anomaly	. Functional or operational anomalies which do not pose a risk but which indicate a lack of safety provisions. This may be due to equipment failure, human error or procedural inadequacies. (Such anomalies should be distinguished from situations where operational limits and conditions are not exceeded and which are properly managed in accordance with adequate procedures. These are typically 'below scale'.)	
Below scale /zero	No Safety significance		

4.5 PROBABILISTIC RISK ASSESSMENT

Whether consciously or unconsciously, individuals and institutions use probability and risk assessment in everyday decision making. Financial institutions, for example, use probability in decision making to determine whether a borrower will be able to repay a loan. In mathematical terms this can be written as $p(R\backslash H)$ where p is "the probability of" and R is "the loan will be repaid" and H is "the borrower's past financial history, assets, liabilities etc". The symbol \ denotes "given."

Risk is a commonly used word that conveys a variety of meanings to different people but is defined in the dictionary as "the possibility of loss or injury." In the example quoted above, risk is also an element in the decision process and for a bank represents the probability of a given loan not being repaid. Risk is defined mathematically as the product of the probability of an outcome times the consequences of this outcome. Thus by combining probability (or likelihood) assessments with risk analysis, a financial institution can minimize the adverse outcomes from loan non-repayment.

The same principles of risk analysis (probability and consequence) can be used to evaluate an individual's estimated lifespan, airline accidents, nuclear and chemical plant accidents, etc. Regardless of the evaluation being performed, the mathematical laws of probability and the utility assessment of risk are the same.

In recent years, scientists and engineers working on the development and construction of new nuclear reactors have turned increasingly to the use of probabilistic risk assessment (PRA) as a tool to help estimate the likelihood and consequence of accidents in the facility that could lead to financial losses and personal injury on- and off-site. The first major application of PRA to nuclear reactor safety was made by Professor Norman Rasmussen of the Massachusetts Institute of Technology (MIT) and colleagues who performed the well-known Reactor Safety Study.[7] The study pointed out that a major element in the characterization of the radioactive releases associated with potential nuclear power plant accidents is the identification of the accident sequences that can potentially lead to risks to the public.

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[7] US Nuclear Regulatory Commission, Reactor Safety Study: An Assessment of Accident Risks in US Commercial Nuclear Power Plants, WASH-1400, NUREG-75/014, 1975.

Current PRA analyses of nuclear power plants utilize what is called "event tree methodology." An event tree is a logic method for identifying the various possible outcomes of a given event called the initiating event. (This technique, which is known in business circles as the application of decision trees, is widely used in many business applications where the initiating event is a particular business decision and the various outcomes depend on subsequent decisions.) In nuclear reactor safety the initiating event is generally a system failure and the subsequent events are determined by the characteristics of the reactor system and the engineering.

Use of PRA enables nuclear engineers to identify and, as a consequence, rectify prior to construction any weaknesses in a system which could lead to system failure and possible release of radioactive materials into the biosphere. It also facilitates prioritizing possible safety improvements that could be made in terms of their degree of safety significance. PRA also enables an estimate to be made of the probability of a serious accident occurring when a number of reactors are operating over a long period of time.

PRA also helps demonstrate whether a plant meets "safety goals" which the US Nuclear Regulatory Commission and other national regulatory bodies have developed in recent years, as discussed in the next section.

4.6 SAFETY GOALS

After the TMI accident in 1979, the US Nuclear Regulatory Commission decided to develop an explicit policy statement on safety philosophy and the consideration of costs in NRC safety decisions. The agency proceeded to develop a policy statement on safety goals for nuclear power plants, issuing interim safety goals in 1983 and final goals in 1986. The final goals provided both qualitative and quantitative goals:

1. Nuclear power plant operation should not impose significant additional risk to an individual's life and health;

2. Societal risks should be comparable to or less than the risks of generating electricity by viable competing technologies;

3. The risk of prompt fatalities to the average individual in the vicinity of a power plant should not exceed 0.1% of all prompt fatality risks from other accidents in the US; and

4. The societal risk of cancer fatalities to the population near a power plant should not exceed 0.1% of all cancer fatality risks from other causes.

In addition, NRC stated that, as a guideline for implementing these goals, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 10^{-6} per reactor-year of operation.[8] Subsequently NRC decided that a core damage probability of less than 10^{-4} per reactor-year as a subsidiary goal could be useful in evaluating regulations related to accident prevention.[9]

Other countries have since developed safety goals comparable to the NRC's, and the IAEA is now developing international guidance on the establishment of national safety goals. For example, in its report on the tolerability of risk from nuclear power plants, the UK Health and Safety Executive estimated in 1988 that most people living in the vicinity of a nuclear power plant face a risk of $\leq 10^{-6}$ per year of dying from cancer caused by a nuclear accident, but that a few are nearer to 10^{-5} and a handful may exceed that level. (If 20 reactors operated in the UK at a risk level of 10^{-6} per reactor year, the risk of an accident anywhere in the UK would be 2×10^{-5} per year.) The same report proposes that a figure of 10^{-4} for uncontrolled releases anywhere in the UK per year might be accepted as tolerable.[10]

A draft report of the IAEA compares safety goals that have been established to date by certain countries and offers guidance on the setting of such safety goals.[11] In addition to

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[8] Safety Goals for the Operation of Nuclear Power Plants: Policy Statement, US Nuclear Regulatory Commission, 51 Federal Register 149, August 4, 1986, p. 28044.

[9] US NRC, Staff Requirements Memorandum Re. SECY-89-102 - Implementation of the Safety Goals, June 15, 1990.

[10] The Tolerability of Risk From Nuclear Power Stations, Health and Safety Executive, London, December 1987.

[11] The Role of Probabilistic Safety Assessment and Probabilistic Safety Criteria in Nuclear Power Plant Safety. A Safety Guide, Draft 6, International Atomic Energy Agency, August 1990.

the US and UK safety goals identified above, the report identifies the following French safety objective:

> The design of a PWR unit should be such that the overall probability of unacceptable consequences for this unit does not exceed 10^{-6} per reactor-year. ("Unacceptable consequences" is defined in terms of families of events, such as containment failure in aircraft crashes and prolonged uncovering of the reactor core in the event of a total station blackout that disables safety systems.)

The Netherlands Energy Research Foundation has also established goals that the chance of core damage should be less than 10^{-5} per reactor year; that the chance of a relatively small release (0.1% of fission products) should be less than 10^{-6} per reactor year; that the chance of a medium release (1% of fission products) should be less than 10^{-7} to 10^{-8} per reactor year; and that the chance of a large release should be excluded (ie., less than 10^{-9} per reactor year).[12]

The IAEA report states that although there is no general international consensus on appropriate levels to be adopted for all measures of risk from nuclear accidents, values for three measures -- individual risk of fatality, large off-site release and core damage -- should be established. The IAEA proposes the following goals for these three safety measures:

- A lower level on the risk of individual fatality should be set at a target frequency of 10^{-6} prompt fatalities per site per year; and national authorities may wish to set different criteria for latent fatalities.

- Until an international consensus is reached on the most appropriate measure for a large off-site release, a target frequency of a large off-site release should be 10^{-6} per reactor year. "Large off-site release" is defined as one with severe social implications.

- Core damage frequency should not exceed 10^{-5} per reactor year, with no single accident sequence contributing a significant percentage of the target.

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[12] Rapporten Herbezinning Kernenergie, Summary of the Nuclear Energy Review Project, Stuurgroep Project Herbeezinning, May 1988.

4.6.1 Determination of Actual Plant Risk and Comparison With Safety Goals

In April 1985, in response to a question from a subcommittee of the US Congress, the NRC estimated that there was a 45 percent probability of a core meltdown accident at a US nuclear power plant within the next 20 years, based on an average estimated frequency of 3×10^{-4} per reactor-year (compared with NRC's subsidiary goal noted above of 10^{-4} core meltdown accidents per reactor-year). When recently asked to update this estimate, NRC declined, citing "large uncertainties associated with quantitative estimates of core damage frequency," but suggested that "there is reasonable evidence that the ensemble of operating US nuclear power plants meets the NRC safety goals and that there is reasonable assurance that the health and safety of the public are adequately protected."

More recently, the NRC has conducted a major assessment of accident risks at five US plants of different designs, using PRA to estimate the probability and consequences of accidents. A draft report on the findings of this assessment, published in June 1989,[13] compared the estimated accident risk at these plants with the NRC safety goals, as illustrated in Figure 4-4. As indicated, these PRAs determined that the risks of both prompt accidental fatality and cancer fatality resulting from internally-initiated accidents were well within the NRC safety goals.

NRC has observed from past experience with PRAs that plant designs often include plant-specific vulnerabilities which contribute heavily to risk, and concluded that a systematic search should be conducted to identify and remedy such vulnerabilities at each facility. A program of Individual Plant Examinations (IPEs) is now under way to improve the understanding of severe accident risks at each US plant. To a limited extent, IPE results will also be compared with the subsidiary safety guidelines on the risk of core meltdown and the risk of an accident causing a large release of radioactivity.[14]

Recent results of PRAs performed in France also give an indication of accident risk that may be compared with the safety goals identified above. PRAs conducted by the utility

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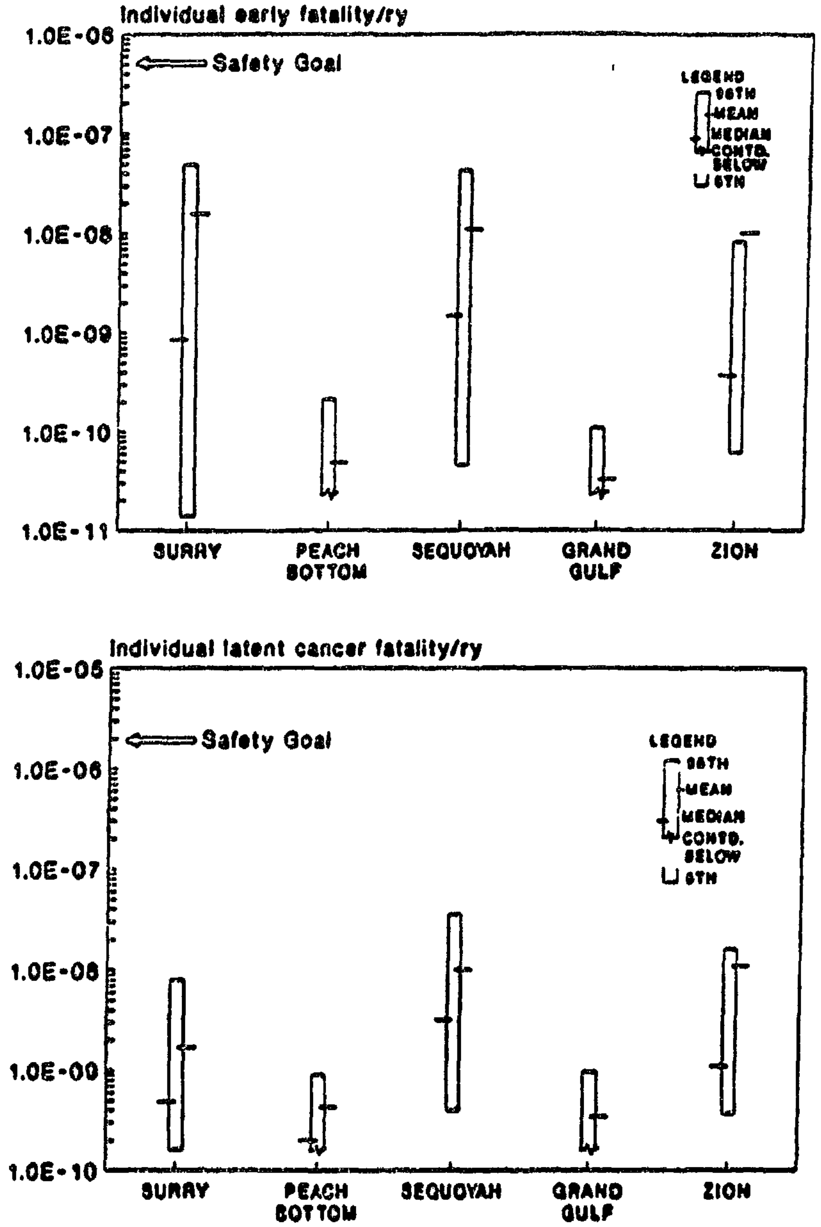
[13] US Nuclear Regulatory Commission, Severe Accident Risks: An Assessment for Five US Nuclear Power Plants, NUREG-1150, Second Draft for Peer Review, June 1989.

[14] US NRC, SECY-90-104, Role of Individual Plant Examinations (IPE) in Assessing Industry Status With Respect to the Commission's Safety Goal Policy, March 20, 1990.

FIGURE 4-4

Comparison of individual early and latent cancer facility risks at all plants - internal initiators.



Source: US NRC, Severe Accident Risks: An Assessment for Five US Nuclear Power Power Plants, NUREG-1150, Second Draft for Peer Review, June 1989, p. 12-7.

Note: According to NRC, the safety goal for early (accidental) fatality shown here is based on actual fatality rate from all accidents of approximately 50 per 100,000 persons per year, multiplied by 0.1%. Similarly, the goal for latent cancer fatality is based on acutal US cancer fatality rate of approximately 200 per 100,000 persons per year, multiplied by 0.1%.

Electricité de France (EdF) and the Institut de Protection et de Sûreté Nucléaire (IPSN) estimated that France's 900 MWe PWRs have an overall core melt risk of 4.95 x 10^{-5} per reactor-year and that the newer 1,300 MWe PWRs have an overall core melt risk of 1.08 x 10^{-5} per reactor year. It should be noted that these results are only comparable with those of other PRAs to a limited extent due to differences in the scope of accident sequences considered. The French PRAs did not consider external initiators but did consider low-power and shutdown modes.[15]

A comprehensive PRA on PWR accident risks was also recently performed in Germany by the Gesellschaft für Reaktorsicherheit (GRS), using the Biblis-B facility as the reference plant. The study calculated a probability of sequences leading to severe core damage of 3 x 10^{-5} per reactor-year.[16]

When risk assessments indicate that plants are not in compliance with safety goals, various backfits may be ordered to correct the situation. For example, installation of filtered vents[17] has been considered for BWRs with small containment volumes and implemented in several cases.

4.7 REGULATION OF NUCLEAR POWER AND ITS EFFECTIVENESS

Regulatory authorities serve on behalf of the government to license, regulate and oversee the safe operation of nuclear facilities and the safe use of nuclear materials. Such organizations are responsible for developing safety standards and regulations; conducting reviews of license applications against those standards and regulations, and taking licensing actions; monitoring the operations of licensed facilities to ensure their continued safety; taking necessary enforcement measures where safety levels are not met; conducting safety research programs; and providing safety and licensing-related information to the public. Regulatory organizations need to have sufficient resources and technical expertise to carry

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[15] "Paris Seminar Discusses PRA Results for France's Two Main Reactor Types," Inside NRC, May 21, 1990, p. 6.

[16] "Biblis-B PWR Three Times Safer Than Thought, German Study Concludes," Inside NRC, July 17, 1989, p. 4.

[17] A filtered vent containment is designed so that any overpressurization of the containment can be released to the atmosphere through a filter which traps radioactive particulate matter.

out their responsibilities, as well as the necessary legal authority and free access to facilities and information.

The approach to fulfilling this regulatory responsibility varies substantially among countries who have licensed nuclear power plants. Some countries -- most notably the United States -- follow a highly prescriptive approach, under which detailed technical regulations have been developed as well as detailed guidance on complying with these regulations. Demonstration of compliance with the regulations is accepted as a demonstration that overall standards of safety will be achieved. Other countries are less prescriptive and require the licensee to demonstrate only that broad safety requirements will be met. In either case, the regulatory authority ultimately requires plants to comply with detailed operating specifications -- or "technical specifications" -- limiting the plant's conditions of operation. Also, in either case, the burden of proof that a proposed facility will not have an adverse impact on public health, safety and the environment falls on the licensee.

While regulators should maintain an arm's length from the industry they regulate, experience suggests that regulatory organizations should also work cooperatively with industry rather than in an adversarial manner. The US NRC has been accused in the past of taking too adversarial an approach towards license applicants and unnecessarily causing cost increases.

At the same time, the NRC has been accused of not maintaining sufficient independence from industry. For example, the Union of Concerned Scientists, a leading nuclear power critic in the US, wrote in 1987 that "nuclear power is an inherently dangerous technology requiring the highest standards of care and performance." They fault the NRC for "indifference and shortsightedness [which] have allowed so many generic technical problems to persist for so long," and state that "NRC's primary and instinctive allegiance is still to the industry it regulates... Congress must assume a more assertive oversight role to see that the NRC lives up to its safety-first mandate."[18]

The organizational relationships between nuclear power proponents and regulators differ from country to country. In the United States, the Atomic Energy Commission was responsible for both promotion of nuclear power and regulation thereof until 1974, when the regulatory functions were separated out and given to the new Nuclear Regulatory

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[18] M. Adato et al, Safety Second, Union of Concerned Scientist, 1987.

Commission. NRC is organizationally completely separate from its licensees. In contrast, the French Service Central de Sûreté des Installations Nucléaires (SCSIN) and its sole reactor licensee Electricité de France (EdF) are both part of the Ministry of Industry. SCSIN is now moving in the direction of independence from EdF. However, there is some perception that this move is intended primarily to please the general public and was not necessary to ensure regulatory independence, as well as apprehension that this could lead to power struggles and adversarial relations as in the US.[19] Another example is the UK, where the Nuclear Installations Inspectorate (NII) is part of a larger agency which regulates industry in general, the Health and Safety Executive (HSE), but is completely separate from the industries it regulates (including utilities).

The International Nuclear Safety Advisory Group's (INSAG) basic safety principles recommend a clear separation between the responsibilities of the regulatory authority and other organizations, so that the regulators retain independence as a safety authority and are protected from undue pressure. INSAG also believes this will ensure that safety is the only mission of the regulatory personnel.

With respect to future applications to construct power plants of advanced designs, there is increasing support in the US and elsewhere for modifications in the nuclear plant licensing procedure to introduce greater licensing efficiency as well as preserve or possibly improve safety levels. In the US, one of the key proposals is to streamline the licensing process by issuing a combined construction and operating license in a single step rather than the approach followed in the past in which these licenses were issued separately and as the result of separate licensing proceedings. As discussed in Section 6.0, standardized designs for certain advanced reactors are now being reviewed by the NRC, which will decide whether to certify that these designs are acceptable for referencing in subsequent utility license applications. This pre-approval of designs could make the licensing process more predictable by removing most design questions from the process. Pre-approval of possible power plant sites is also gaining support as it would expedite the licensing schedule.

As discussed further in Section 7.0, regulatory considerations are very relevant to the question of building nuclear power plants in developing countries. Government authorities in developing countries would require adequate resources and capabilities to review and

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[19] Electricité de France, Sûreté Nucléaire 1989, Rapport de l'Inspecteur Général Pour la Sûreté Nucléaire.

evaluate nuclear power plant license applications and to oversee safety during plant operations. As noted above, there are no hard and fast rules regarding the organization of national regulatory authorities, except that the authority should be organizationally independent of the regulated industry and competent.

4.8 EFFECT OF MANAGEMENT ON SAFETY

One recent study by MIT's Nuclear Engineering Department compared nuclear operating experience in the major nuclear power countries to understand why US plants have been consistently outperformed (i.e., in terms of plant availability) by their foreign counterparts.[20] The study found that managerial reforms are the key to improving US plant performance, rather than changes in the environment within which these plants are operated including US regulatory zeal, diverse plant ownership patterns, and financial regulation by the states, factors which have been widely blamed for the poor performance of US plants. Industry-wide cooperation between utilities, suppliers and regulators has started late in the US and there is still deep distrust between utilities and regulators as well as competition among suppliers, the authors found. They also suggest that utilities that show consistently good results operate with a large degree of managerial involvement in day-to-day activities. Investing in a plant's intellectual resources through training programs and staff exchanges with other organizations can foster an "esprit de corps" that benefits plant operations.

Good management can be expected to benefit plant safety as much as plant performance. INSAG's 1988 basic safety principles include guidelines with respect to safety culture and responsibility of the operating organization. With respect to safety culture, they state that:

> The starting point for the necessary full attention to safety matters is with the senior management of all organizations concerned. Policies are established and implemented which ensure correct practices, with the recognition that their importance lies not just in the practices themselves but also in the environment of safety consciousness which they create... These matters are especially important for operating organizations and the staff directly engaged in plant operation. For the latter, at all levels, training emphasizes the significance of their individual tasks from the standpoint of basic understanding and knowledge of the plant and the equipment

[20] K. Hansen et al, "Making Nuclear Power Work: Lessons from Around the World," Technology Review, February/March 1989, p. 31.

> at their command, with special emphasis on the reasons underlying safety limits and the safety consequences of violations...[21]

With respect to responsibility of the operating organization, the INSAG principles state that:

> Once the operating organization accepts possession, it is in complete charge of the plant, with full responsibility and commensurate authority for approved activities in the production of electric power. Since these activities also affect the safety of the plant, the operating organization establishes policy for adherence to safety requirements, establishes procedures for safe control of the plant under all conditions, including maintenance and surveillance, and retains a competent, fit and fully trained staff...[22]

4.8.1 Human Factors

One factor believed to contribute to anxiety about the safety of nuclear power plants is the possibility that accidents might be caused or aggravated by human error.

Human error can occur at many stages in the design, manufacture or construction of a nuclear power plant. It can also be crucial in the operation and maintenance of power plants. Human error has contributed to many past events and was chiefly responsible for the accident at TMI, in which an operator shut down the ECCS even though the reactor core was being uncovered, because he had faulty information (design flaws also contributed to the accident). Human error combined with the application of inappropriate operating procedures caused a chain of uncontrollable events which led to the disaster at the Chernobyl Power Station. (Both accidents are described in detail in Section 5.0.)

Judgments involved in the subject known as "human factors" are extremely complex. For example, the role of the operators of a nuclear power plant is far more than a mechanical one. Plant operators must have knowledge and understanding of the plant which they will

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[21] See Note 3 supra, p. 10.

[22] See Note 3 supra, p. 11.

be required to apply, in conjunction with the plant's automatic control system, to ensure that the plant operates reliably and safely.

Human factors has attracted a great deal of attention in the nuclear industry since the TMI accident. Improvement in control room designs is one major benefit that has resulted from application of this science to nuclear power plants. However, it is clear that in addition to human factors, good management is a key element of accident prevention. To the extent that accidents caused by human error reflect the shortcomings of the management system, efforts to correct defects in organization, training, or procedures will lead to commensurate gains in plant safety level.

It is clearly essential that utility managers at the highest level make a high priority of nuclear safety and allocate sufficient funds for safety-related activities, including human factors. As discussed in Section 7.0, developing countries investing in nuclear power must have adequate funds and technical infrastructure for maintaining nuclear plants and assuring continued safety.

5.0 MAJOR NUCLEAR ACCIDENTS

Although there have been many minor incidents at commercial nuclear energy facilities, there have in fact been only three major accidents to nuclear power plants since the development of nuclear energy began in the late 1940's. The three accidents, which occurred at Windscale in England, Three Mile Island in the USA and Chernobyl in the Soviet Union, were so fundamentally different in kind from each other that a description of each is considered to be worthwhile.

5.1 THE WINDSCALE ACCIDENT

In the early 1950's, the United Kingdom decided to go ahead with the development of nuclear weapons for defense purposes. In order to proceed on this route the UK required a supply of weapons-grade plutonium which it decided to manufacture in a number of weapons-material production "piles" (the original name for nuclear reactors) which it built at Windscale on the Northwest coast of England.

The Windscale Pile (as it is known) was one of the first reactors ever to be built and was, by today's standards, a very primitive type of system. The fuel utilized was natural uranium in the form of metal rods clad in a special alloy made from magnesium and aluminum, chosen because of its low neutron absorption characteristics. The moderator was high purity graphite and the whole system was cooled by air. The reactor ran at a relatively low temperature.

5.1.1 Wigner Stored Energy

At the time these reactors were built very little was known about the effects of bombardment of graphite at low temperatures resulting in the production of so-called "defects" in the internal structure of the graphite due to the carbon atoms being knocked out of their normal positions in the graphite lattice. These displaced atoms are capable of returning to their normal positions again at which time their stored energy (known as Wigner stored energy after the name of Eugene Wigner who discovered it) is released in the form of heat.

Although the phenomenon of Wigner stored energy had been known for some time when the reactors were built, there was very little knowledge about how such stored energy might be released. It was known that stored energy builds up progressively with irradiation and the rate at which it accumulates is temperature dependent, virtually no energy being stored above 400°C, and that a spontaneous release of stored energy could occur as happened in the Windscale No. 1 Pile in 1952, while the pile was shut down, but without any harmful effects.

As a result of this experience procedures were instituted for the controlled release of this stored energy by allowing the chain reaction in the reactor core to commence without coolant airflow thus raising the graphite and uranium temperatures and starting the so-called Wigner energy release in the graphite. Under these conditions the release becomes self-sustaining. Eight such releases of stored energy had taken place by the end of 1956 but it had been found difficult to release energy in all the graphite in the pile and on three occasions a second heating was found necessary.

5.1.2 The Windscale Fire

In October 1957 the No. 1 pile was shut down and a Wigner energy release was started. After some hours the nuclear heating was stopped as planned but the temperature of the graphite appeared to the plant operator to be dropping rather than increasing. Consequently the pile operator decided to boost the release with a second nuclear heating.

During this second heating a rapid rise in temperature of the uranium cartridges was observed at which time the control rods were again inserted to reduce the power.

As a result of this second heating the graphite temperatures rose rapidly, leading to oxidation of the uranium which had been exposed by the overheating. This gradually led to the failure and combustion of other uranium cartridges and subsequently to combustion of the graphite itself, all of which was exacerbated by the introduction of air into the pile in attempts to cool it.

Over the next day a number attempts were made to cool the pile but without effect and eventually it became necessary to couple water hoses to the top of the pile and to flood the affected channels. This technique proved successful and after 24 hours the pile was cold.

5.1.3 Health Consequences of the Accident

Examination of the workers revealed that fourteen had received exposures higher than normal during the accident but even the highest exposure was only 50% above the ICRP safe continuous level. Moreover, the highest level of Iodine-131 measured in the thyroid gland was very small and well below the level at which harm could be done.

The exhaust cooling gases from the Windscale piles were normally fed to the atmosphere through tall stacks equipped with filters to trap radioactive particulate matter. During the fire these filters worked adequately and it was subsequently found that no harmful amounts of plutonium or any other elements had been released with the exception of Iodine-131, a radioactive isotope of iodine.

The risk from inhalation of radioactive materials was found to be as insignificant outside the factory as it was inside and no restrictions were placed on the consumption of vegetables, eggs, meat and water in the area as a consequence.

However a problem arose due to the deposition of the radioactive iodine on the grass under the area over which the plume from the chimney passed and which was eaten by the cows from the local farms. Since iodine tends to concentrate in the milk, the potential danger to young children and others drinking milk from cows which had eaten the grass on which the radioactive iodine had deposited becomes apparent.

As a precaution, therefore, milk deliveries from twelve milk producers within a two-mile radius of Windscale were stopped for a time until the levels of radioactive iodine had reduced to an acceptably low level.

The special Committee from the Medical Research Council which was set up after the fire to investigate the health consequences of the accident concluded:

> "After examining the various possibilities, we are satisfied that it is in the highest degree unlikely that any harm has been done to the health of anybody, whether a worker in the Windscale plant or a member of the general public".

5.1.4 Cause of the Accident

Following the report by the Committee of Inquiry, the Chairman of the Atomic Energy Authority (then Sir Edwin Plowden) wrote in a Memorandum that the cause of the accident was attributable to inadequacies in the instrumentation provided for the operation of the Wigner energy release and to faults of judgement by the operating staff, which themselves were attributable to weakness of organization (the Atomic Energy Authority).

5.1.5 Commentary

This particular accident occurred in a type of reactor which no longer exists since the Windscale piles which were used exclusively for the production of military materials were, in fact, never restarted after the 1957 fire. The gas-cooled, graphite moderated power reactors (Magnox and AGR) in operation in the United Kingdom today run at much higher temperatures than the Windscale piles and do not require the periodic release of Wigner stored energy. Moreover, they utilize carbon dioxide as a coolant, which is inert and would not lead to increased combustion in the event of an overheating incident.

The accident itself led to a number of organizational changes in reactor management and in particular to the realization of the need for close liaison between the management of the reactor site and the local interests. These changes have led to the tight controls which are in place today in the United Kingdom.

5.2 THREE MILE ISLAND ACCIDENT

In March 1979, the No. 2 unit at the Three Mile Island (TMI) nuclear power station in Pennsylvania, USA, suffered a severe core degradation accident, the only one of its kind to happen to any pressurized water power reactor (PWR) in the world todate.

In order to understand what happened during the accident, a brief description of the plant is necessary.

TMI-2 is a pressurized water reactor with three interrelated cooling circuits. Heat generated by the reactor core is transferred to water circulating in the primary circuit which is under high pressure (about 2200 psi) to keep it from boiling. The heat from the primary circuit is transferred to a secondary circuit by means of two steam generators which produce the steam for the steam turbine which drives the electricity generator. After passing through the turbine the steam is condensed back to water by a third circuit which circulates water between the condenser and the cooling towers.

Under normal operating conditions, essentially all radioactivity is contained within the uranium oxide fuel pellets, and the fuel cladding tubes which are made from zirconium alloy that resists corrosion and high temperatures.

In the event that fission products escape through the fuel cladding, such as through defects, these are trapped in the primary coolant from which they can be removed in the reactor purification system. However, krypton and xenon do not readily dissolve in water, particularly at high temperatures and collect as a gas above the coolant when the system is depressurized.

The core of the reactor is encased in a pressure vessel which is a 36-foot high tank with steel walls about nine inches thick. The reactor pressure vessel and the remainder of the primary coolant system, which includes the pressurizer, steam generators and associated piping, are contained in the reactor (or containment) building. The containment building has steel-lined thick concrete walls and is the final barrier to the outside environment.

The auxiliary building is located close to, but external to, the containment building. During the TMI-2 accident, radioactivity was released to the environment when radioactive liquids were pumped from the reactor building to this auxiliary building.

5.2.1 Events Contributing to the Accident

On the day of the accident, a malfunction occurred to components that maintain the flow of coolant water to the steam generators in the secondary loop. This resulted in a loss of ability to remove heat from the primary loop with the result that most of the heat generated by the reactor remained in the reactor vessel and primary loop. This caused the coolant water temperature and pressure to increase rapidly which, in turn, caused a relief valve on the pressurizer to open allowing steam and water to discharge to the reactor coolant drain tank located in the basement in accordance with design procedures. The drain tank is equipped with a pressure-limiting rupture disc.

As there was no valve position indicator for the pressure relief valve clearly visible in the control room, the fact that the pressure relief valve was open was not deduced by the operators for more than two hours during which time water continued to be discharged through the valve into the drain tank.

As the reactor pressure continued to fall due to the open pressure relief valve and resultant loss of primary coolant, the high pressure safety injection system (which is part of the emergency core cooling system) began automatic operation as intended. This system was twice cut off or reduced in flow manually by operators who interpreted instrument readings to indicate that water level in the reactor was adequate. As coolant inventory declined due to continued loss of coolant through the open relief valve and the cutback of the high pressure injection system, a number of flow anomalies developed to which the operators responded in varying ways.

During this period, significant fractions of the core became uncovered for extended periods and core damage resulted. Voids were created in the system, preventing natural circulation cooling and interfering with forced circulation, which was finally reestablished in one of the two loops of the system.

So much water and steam were discharged through the relief valve that the storage capacity of the drain tank was quickly exceeded, causing the rupture disc to burst, allowing some 250,000 gallons of radioactive coolant to be discharged into the reactor building sump and basement.

Radioactive coolant water in the reactor building sump was then automatically pumped into the sump tank in the auxiliary building which was already about half full. Consequently, much of the water spilled into the auxiliary building, which was not designed to contain radioactive material. This liquid did not contain significant amounts of radioactivity, however, because major fuel damage did not occur until about two hours later.

After fuel damage occurred, radioactive materials were transported through the primary coolant system via the letdown line to the makeup and purification system in the auxiliary building. Because this liquid was a stream of primary coolant directly from the reactor, it contained significant amounts of radioactivity. As a result of liquid leaks in the makeup and purification system, large amounts of radioactive material were released into the auxiliary building. No longer held under pressure, krypton, xeron and other volatile radionuclides evolved from the water into the auxiliary building atmosphere.

In one of the least expected and most highly publicized facets of the incident, the upper section of the reactor pressure vessel became occupied by hydrogen formed by reaction of primary coolant water with overheated zircaloy cladding when the core was partially exposed. A portion of this hydrogen escaped into the reactor containment building with the water vented through the pressure relief valve, and this hydrogen ignited at 10 hours after initiation of the incident, resulting in a containment pressure spike of 28 psi. This hydrogen ignition or explosion was unreported for some time to both the NRC and the press, but for several days, wide and sensational publicity was given to the presence of the hydrogen "bubble" in the reactor, and to the possibility of its explosion, the risk of which was nonexistent, since no free oxygen could be present in the gas under the conditions in the reactor.

Removal of the hydrogen from the system became, in press reporting, one of the most dramatic and risky aspects of the incident, with attention focused on the possibility of explosion in the containment building as hydrogen released to the containment atmosphere built up. This was avoided by activation of a catalytic recombiner which kept the hydrogen concentration below combustible limits.

5.2.2 Radioactivity Released to the Environment

During the accident, approximately 50 percent of the noble gases and particulate cesium, 30 percent of the iodine and small quantities of other fission products normally present in irradiated fuel were released from the damaged fuel into the primary coolant water. Before being released into the environment, the small amount of the airborne radioactivity released to the reactor building was filtered and monitored. The high efficiency filtration system in the auxiliary and fuel handling buildings was designed to remove more than 99 percent of radioactive cesium, strontium and alpha-emitting radionuclides. In addition to mechanical filtration, ventilated air in these buildings was also passed through multiple charcoal filters, which chemically removed 90 to 95 percent of the radioactive iodine. However, neither the mechanical filter nor the charcoal absorbers were designed to prevent the discharge of the chemically inactive krypton and xenon gases which escaped to the environment.

While offsite radiation levels never exceeded about 35 mrem/hr, and total exposure to any individual in the areas of highest activity are estimated not to have exceeded 80 mrem, compared to a typical background exposure of around 300 mrem/yr per person, the presence of offsite radioactivity an the exaggerated threat of its massive release from a possible breach of the containment due to possible hydrogen explosion, coupled with sensational media reporting, generated intense nationwide concern for over a week.

The prospect of a large-scale evacuation of the surrounding population and the planning for that contingency further aroused public concern. On the third day, evacuation of pregnant women and children under six years of age residing within five miles of the plant site was officially recommended by the governor of Pennsylvania. General evacuation was never ordered (although the possibility of such evacuation was openly discussed by the press and officials at every level, including President Carter) but an estimated 80,000-200,000 residents of the area voluntarily left their homes. Most returned shortly after Governor Thornburgh advised some days later that it was safe to do so.

5.2.3 Potential Health Effects from the Accident

It was the release of radioactivity from the plant and the appearance of detectable amounts of radiation beyond the plant boundaries which, as would be expected, led to the most serious public and media reactions even though radiation exposure and contamination never reached significant levels from the standpoint of health.

Traces of radioactive iodine were detected by some public health authorities in some milk samples, but at levels so low that none was ever removed from the market.

It has since been estimated that cumulative exposure from the incident could result in one additional cancer death, one added nonfatal cancer, and one additional birth defect over the next 25 years among the two million people within 50 miles of the facility. These two million people are statistically expected to suffer 325,000 cancer deaths from natural causes other than the TMI accident.

Although there have been many allegations of increased leukemia and other cancers in people living in the area, particularly young people, studies by the Pennsylvania Department of Health have not revealed incidences greater than normal. These findings have, moreover, been confirmed by a team of independent epidemiologists who have been studying the allegations for the TMI Health Fund and whose results were published in September 1990.

Nevertheless, one of the major concerns emerging during the period of the accident was the psycho-behavioral impact on local residents. The Pennsylvania Department of Health has reported that for some months after the accident many local residents suffered from severe distress and had to turn to tranquilizers, sleeping pills, alcohol and tobacco for remediation.

Fortunately these problems have not persisted, nor has the use of these substances.

5.2.4 Commentary

The Three Mile Island accident was caused by a combination of human error and system malfunction. It resulted in the degradation and partial melting of the reactor core and the total loss of the reactor which is still being decontaminated.

Nevertheless, in spite of the enormous media attention given to it at the time, the safety features engineered into the system prevented the release of all but trivial amounts of radioactivity into the biosphere.

The accident had a big effect on the industry and also on the regulators since it clearly pointed up the defects in operator training and in some of the engineered safety features of the reactor system which had hitherto been considered adequate.

As a result, the industry set up it own "watchdog", the Institute for Nuclear Power Operations (INPO), which is independent of the Nuclear Regulatory Commission. INPO oversees the operations of the industry and will take action as and when required to ensure industry compliance with good practice.

For its part, the NRC required a number of backfits to be carried out on existing reactors to ensure that future incidents of this kind have a very low probability of happening. The NRC also instituted a program for reactor operator training which is designed to improve the quality of performance of future operators.
Most of these improvement have since been implemented by overseas operators of light water reactors as part of the overall safety improvements of reactors all over the world.

5.3 THE CHERNOBYL ACCIDENT

On April 26, 1986, the worst accident in the history of commercial nuclear power generation occurred at the Chernobyl Nuclear Power Station some 60 miles north of Kiev in the Ukraine. The accident caused extensive damage to the reactor and the building which housed it; some 31 people died as a result of the fire and explosion, or as a result of receiving lethal radiation doses. A significant release of fission products occurred, contaminating the land around the station and requiring the evacuation of around 135,000 people from their homes. The radioactive cloud generated by the accident over many days was carried by winds all over Europe and led to restrictions on the consumption of meat and vegetables which became contaminated from it.

Although the latent health effects may not be statistically significant when viewed against the normal mortality rate over the next 40 years, nevertheless the accident has had a big impact on public concern about nuclear safety. It is, therefore, desirable to provide a brief description of the reactor and the events which contributed to the accident.

5.3.1 Design of the RBMK

The RBMK (Reaktor Bolshoi Moschnosti-Kanalye) reactor is a direct cycle boiling water pressure tube graphite moderated reactor, developed from the first nuclear power plant commissioned in 1954 at Obninsk. The design is unique to the Soviet Union. At the time of the accident, the Chernobyl nuclear power station had four 1000 MWe RBMK reactors operational and two more under construction.

The reactor core is 12 m. in diameter and 7 m. high, and is built up from graphite blocks penetrated by vertical channels each containing a zirconium niobium alloy pressure tube in which two fuel assemblies are located end to end.

The fuel is in the form of pellets of 2% enriched uranium oxide encased in zirconium alloy tubes. The maximum power from each of these channels is 3.25 MW. The fuel is cooled by boiling pressurized water which passes into one of the two identical coolant loops which incorporate two steam drums in each. The dry steam from the steam drums passes to the turbogenerators.

The RBMK reactor is equipped with an emergency core cooling system which feeds both coolant loops. The primary coolant system is housed in a series of compartments which act as the containment in the event of an accident and are designed to withstand a pressure of 4.5 bar.

5.3.2 Events leading to the Accident

Ironically the immediate cause of the accident was an experiment designed to improve the safety of the plant. The objective of the experiment was to test the turbogenerator's ability to provide in-house power after shutting off its steam supply for the short time needed for the emergency diesels to start and come online, nominally 40 to 50 seconds, and required the reactor to be at about 25% full power.

This test had been attempted twice before in 1982 and 1984, on which occasion it was found that the voltage output decreased faster than desired and the purpose of the test was to verify proper operation of a new voltage regulator design for the generator.

In the subsequent inquiry into the causes of the accident, it became clear that the experimental test had been badly planned, that the safety case had been inadequate, and that the operators had departed from laid down operating procedures and had violated several operating rules.

The test procedure itself called for turning off the emergency core cooling system. Operator actions included disconnecting the signal that automatically shuts down the reactor when two turbogenerators are disconnected, operating the main coolant pumps in a regime where cavitation might occur, turning off various protection system signals, and operating with less than the minimum required number of inserted control rods.

Power reduction to the test power level of 700-1000 MWt began but was halted while the operators disconnected one of the two turbogenerators from the reactor. Four main cooling pumps, and two feedwater pumps were connected to the turbogenerator to be run down. The operators also disabled the signal which results in automatic reactor shutdown when both turbogenerators are disconnected. This action was intended to permit rerunning the test if needed - but the test procedure did not call for disabling this emergency system. In addition, the operators disabled the emergency core cooling system, but this was done in accordance with test procedure.

Before power reduction could continue, the grid controller requested the operators to hold power and not continue with the test. In complying with this request a further violation of normal plant operating procedures occurred since continuous operation at power with the emergency core cooling system disabled is a violation.

Following the delay, the operators continued the power descent and disengaged the local automatic power regulation system. A further operator error was made at this point when the operators failed to set the backup automatic controller to its proper "hold power" setpoint. This resulted in the operators being unable to control the reactor power which began a rapid unplanned power reduction, falling to as low as 30 MWt before they were able to stabilize power at about 200 MWt.

This unplanned power reduction allowed the build-up of xenon (which is a strong neutron absorber) to a sufficiently high level that it reduced core reactivity which had to be compensated by withdrawal of control rods. Attempts were then made to increase power to the required level of 700-1000 MWt but were unsuccessful due to low core reactivity. This was made more difficult by the fact that the control rods had been mostly withdrawn to compensate for the buildup of xenon.

Consequently, with the reactor only at 200 MWt, the decision was made to proceed with the test and two of the eight main circulation pumps, which up to then had not been in operation were started up and the flow rate of the water to the core was thereby increased. The result was a reduction in steam formation and a fall in water level in the steam drums which the operators tried to increase by using feedwater pumps.

The immediate effect was to reduce core reactivity because of the reduction in steam voids and the operators responded by removing manual control rods from the core. Conditions were thus produced with a potential for a large increase in steam voids and core reactivity.

Nevertheless the experiment was started at which time events were only about one minute away from disaster.

At a lower operating regime the RBMK reactor is fundamentally unstable due to its design. The reason for this involves the concepts of the positive void coefficient[1] and the positive power coefficient. At the time of the accident the Chernobyl reactor was being operated at less than 20% full power and thus in the unstable region.

Immediately the experiment commenced, steam supply to one of the operating turbo-generators was shut off, which should have automatically caused a shutdown of the reactor. However, the operators had deliberately disabled the protection system to keep the reactor running so that the experiment could be repeated if the first attempt was unsuccessful.

The turbo-generator rapidly decelerated and the four main circulating pumps connected to it started to run down. The water in the core started to boil increasing the volume of steam and creating voids in the core.

As a result of the positive void coefficient the power of the reactor started to rise and a positive feedback ensued (i.e. the power increased by itself). Although the operators tried to stop the reactor from "running away" by inserting the control rods as rapidly as possible, it was far too late. The rate of increase of power was such that the power rose in an uncontrolled manner to some 100 times full power in a matter of a few seconds causing severe fuel damage and fuel channel disruption.

A violent steam explosion occurred due to the interaction of water with the molten fuel, and blew off the 1000 tonne reactor cap and ejected burning material into the air, some of which landed on the roof of the joint turbine hall and put the adjacent undamaged reactor at risk.

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[1] The term "void coefficient" means that if the power from the fuel increases, or the flow of coolant water decreases, or a combination of both, the amount of steam in the fuel channel increases. This causes the density of the coolant to decrease because of the steam voids (or holes) which have been created in the water. In most reactor designs, such as the LWR, the production of voids causes the number of neutrons to decrease and thus reduce the power. In this case the void coefficient is said to be "negative". In the case of the RBMK, however, the design of the core is such that it has a "positive void coefficient" so that when the coolant density decreases the number of neutrons increases and thus the reactor power increases.

Although the initial fires were started by the burning debris ejected from the reactor, the main fire was due to the graphite also catching fire. This raging fire acted as a chimney to loft particulates of the fuel and fission products very high into the air.

5.3.3 Health Effects of the Accident

The Soviets have calculated that about 10% of the graphite (about 250 tons) was burned and some 3-4% of the fuel was expelled from the core. The release of radioactive material from the core did not occur in a single massive event. Only 25% of the materials released escaped during the first day of the accident as a result of the explosion. The rest escaped over a nine-day period as a result of the fire before it was contained.

It has been estimated that the proportions of the core inventory deposited at various distance from Chernobyl was as follows:

On-site	0.3-0.5%
0-20 km	1.5-2%
Beyond 20 km	1.0-1.5%

The immediate health impact was, of course, on the plant personnel and rescue workers, a number of whom received massive doses of radiation from which they died. Others had to be hospitalized for treatment of radiation burns.

Outside the immediate area of the reactor accident the doses were too small to cause "acute" radiation effects. Initially the local population was instructed to remain indoors and to close their windows, but as the levels of radiation began to increase, evacuation commenced and arrangements were made for decontamination of the skin and clothing where necessary.

Outside the Soviet Union the doses received from the fallout were large enough, particularly in Western Europe, to cause an appreciable increase above the average natural radiation exposure. In the first year, for example, the estimated increase over natural background was about 20% in the countries of the European Community. Nevertheless, an International Panel of Experts, convened by the Commission of the European

Communities[2] to advise on the feasibility of studies on health effects in western Europe from the Chernobyl accident, concluded that the levels of exposure were so low as to preclude any effects being detected in the exposed populations of the EEC.

Although there was no immediate health impact on the local population from the fallout from the accident, the doses received might be expected to result in an increased incidence of radiation-induced cancers in later years. At the present time, however, adverse health effects due to radiation exposure have not been observed. A study carried out by the International Chernobyl Project[3] has pointed out that many of the local clinical investigations of health effects were poorly done and produced confusing , and often contradictory results. Nevertheless, the International Project stated in its recent report that "....adverse health effects have not been substantiated by those local studies which were adequately performed or by the studies under the Project". The Project report points out, however, that there were significant non-radiation health disorders in the populations of both survey contaminated and surveyed control settlements studies by the Project, but no health disorders that could be attributed directly to radiation exposure.

However, as the Soviet Union has a population around 275-million and over the next 40 years the number of deaths can be expected to be approximately 30-million, of which some 7.6-million will be from cancer. Consequently, deaths due to Chernobyl, which appear to lie in the range 4,000/38,000, may be impossible to detect with any certainty. In fact, the International Chernobyl Project states in its recent report that "On the basis of the doses estimated by the Project and currently accepted radiation risk estimates, future increases over the natural incidence of cancers or hereditary effects would be difficult to discern, even with large and well designed long term epidemiological studies".

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[2] "Commission of the European Communities - Radiation Protection: Feasibility of Studies on Health Effects in Western Europe due to the Reactor Accident at Chernobyl and Recommendations for Research", EUR 12551, 1991.

[3] "The International Chernobyl Project - The Radiological Consequences in the USSR of the Chernobyl Accident: Assessment of Health and Environmental Effects and Evaluation of Protective Measures", presented at the International Conference on the International Chernobyl Project, Vienna, May 21-24, 1991.

5.3.4 Commentary

A number of factors combined together to bring about this accident. It could not have occurred had the RBMK not had certain design features, notably its positive void coefficient at low power operation, or if the design had included safety systems to cope with such a characteristic. Equally the accident could not have occurred had there not been the application of inappropriate operating rules by the operators.

From the standpoint of the developing countries the lessons from the Chernobyl accident are of great importance. In the first instance it stresses the need for ensuring that the reactors being built have adequate safety features, particularly those which prevent an accident from happening in the event of operator error. Such systems will, presumably, be purchased from exporting countries and will have been certified by the regulatory bodies in those countries.

More importantly, however, the Chernobyl accident stresses the importance of good operator training and supervision. It is quite clear from the Soviet report on the accident that the operators did not understand their plant and were not sufficiently aware of its special technological features. It is also clear from the same report that the operators had become complacent (as they had at TMI) to the extent that they had slipped into the dangerous attitude that an accident could never happen.

These features clearly indicate that if nuclear plants are to be built in developing countries, it will be essential for such countries to set up arrangements for operator training and to develop adequate arrangements for supervision of plant operations by good management.

6.0 PROSPECTS FOR SAFETY IMPROVEMENTS IN ADVANCED REACTOR DESIGNS

6.1 INTRODUCTION

Although nuclear power plants have, for the most part, operated safely for over thirty years and today provide roughly 16% of the world's electricity supply, concerns over both safety and cost have discouraged further commitments to nuclear energy in several countries. In the United States for example, not a single new nuclear power plant has been ordered since 1978, even before the accident at Three Mile Island. While this can be attributed to a variety of factors including a slow-down in electricity demand growth and the unfavorable economics of several reactor construction efforts, concerns about safety have contributed significantly to the problem. Of the facilities that have gone forward, construction and licensing have been delayed significantly in many instances due to local public opposition to nuclear power expressed on safety grounds.

As a result of public and political concern over the safety of nuclear facilities as well as negative past experience with project economics, utilities and investors in some countries have been reluctant to commit to further nuclear plants, even though electricity demand forecasts in some countries point to shortfalls occurring beginning in the 1990s. At the same time, some countries have clearly formed the judgment that nuclear power is important to their energy security and that nuclear plants can be operated safely. Overall, however, there appears to be a widely shared view in several countries that the acceptability of and confidence in nuclear power can be improved if safety advances are made in reactor design. This view also obviously affects the prospects for utilization of nuclear energy in developing countries.

With the benefit of a substantial base of experience both in building and operating nuclear power plants and in seeking public acceptance of these facilities, many of the major vendors of nuclear equipment are now offering a range of new reactor designs which have the potential to alleviate concerns over safety and economics and to recapture utility interest. These new concepts, which are at varying stages of development, range from large "evolutionary" advanced designs that provide incremental improvements to existing LWR designs based on past operating experience but maintain the same basic underlying technology, to "innovative" designs which claim substantial safety advantages over the current generation of reactors by the use of new engineering approaches and extensive reliance on passive processes (e.g. gravity, natural circulation etc). However, these designs will require the construction of appropriate demonstration projects.

Proponents of the so-called "evolutionary advanced light water reactor" (evolutionary ALWR) designs -- which would be large plants rated at ≥1300 MWe -- argue that LWRs have proven their worth and reliability over many years and that there is no basis for making radical changes in the fundamental approach employed for producing nuclear power. Instead, they assert that the evolutionary ALWR designs offer significant economic and safety benefits because they are modifications based on extensive operating experience with a known technology, including incident and accident experience. They point to the Three Mile Island accident and the absence of off-site health consequences there as a testament to the safety of LWRs. While they favor improvements and simplifications in the design of LWRs, including the incorporation of passive safety elements to the extent possible, they also believe it is necessary to retain the "defense-in-depth" safety characteristics of existing LWRs, as discussed in Section 4.0.

The principal proponents of designs which make evolutionary (incremental) improvements over existing reactors are vendors in the US, France, Germany and Japan. These reactor concepts are described in Section 6.2.1. It should also be noted that Europe's two largest nuclear vendors, Framatome of France and KWU of Germany, are already building and operating modern large PWRs based on designs which have evolved from earlier plants: the N4 and the Convoy A designs, respectively.

Work is also underway for passive midsize and other innovative advanced reactors designs. These would, generally, be smaller than current reactors, allowing the expansion of electrical generating capacity in smaller increments. These advanced reactors are expected by their proponents to be simpler to build, operate and maintain and to have an inherently higher level of safety than the current generation of large, complex light water and other reactors. Large segments of the reactor and plant would be amenable to shop fabrication, thus reducing the probability of errors, assuring higher standards of quality control and reducing cost. Claims of improved safety are based on the greater reliance on passive safety principles than engineered safety features (although some concepts combine both characteristics). For example, whereas in the event of a loss of coolant accident, current light water reactors would rely on the active intervention of backup emergency cooling systems to prevent temperatures in the reactor fuel from exceeding the melting point of the fuel cladding (as discussed in Section 4.0), some of these new designs are intended to tolerate off-normal conditions for longer durations without damage to the core thus avoiding dependence on active cooling systems. They rely on physical principles such as gravity and natural circulation to shut down the reactor and prevent core damage, rather than safety systems that are vulnerable to interruptions in external electricity supply and may require active operator action. One factor that contributes to the alleged greater "inherent" safety of these designs is the downsizing, which allows natural circulation to have a greater impact on reactor cooling requirements.[1]

As discussed in Sections 6.2.2 and 6.2.3, a wide range of advanced reactor technologies based on passive safety principles are now being developed, spanning from passive mid-size ALWRs to other more revolutionary designs. The principal proponents of advanced reactors with passive safety features are vendors in the US, Canada, Sweden, the UK and Italy. There have also been recent indications that utilities in Japan are considering the need for R&D on passively safe reactor concepts that are "safer, easier to operate, and more economical."[2]

Clearly, two distinct schools of thought have emerged regarding the technologies to be employed for future nuclear power plants. A major element of the debate between the two groups is the question of whether the more innovative designs will in fact be safer, as claimed, than existing plants or evolutionary technology. A segment of the technical community (as expressed by Paul Gray, MIT President) maintains that "all of the advanced second-generation nuclear power plants under consideration are safer" than the 100+ plants currently operating in the United States.[3] This claim is being studied by the UK Atomic Energy Authority, among others, which has thus far drawn some general conclusions. One of them is that reliance on inherent properties instead of engineered systems does not

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[1] It should be noted that many in the nuclear power industry object strongly to the phrase "inherently safe" because it gives the impression that reactors currently in operation are unsafe and implies that the new designs are absolutely risk-proof. The industry feels this terminology could raise false hopes and in any event may not be necessary for the development and deployment of a new generation of reactors that possess safety advantages not found in the earlier reactors which have been built in several countries.

[2] "Japan's Nuclear 'Vision' Seeks to Reconcile Public With Nuclear," Nucleonics Week, October 4, 1990, p. 7.

[3] Gray, PE: "We Need Safer Smaller Simpler Reactors," Popular Science, April 1990, p. 76.

necessarily raise the safety level, but could allow the same safety level to be achieved by a simpler and cheaper system. Another line of thinking cautions that the importance of passive safety features may be overestimated, and that all reactor concepts rely to some extent on both active and passive safety systems; it is the combination of these that allows high safety standards to be achieved.[4]

Although the debate over the comparative safety of these categories of reactors continues, the order in which these concepts would likely be deployed has become clear. Because the large, evolutionary ALWRs have progressed the furthest of the various advanced reactors under development, it is likely that they would be ordered first, possibly by the mid- to late-1990s, followed by passive, mid-size ALWRs and in the longer-term the more innovative designs. It should be noted that construction of full-scale demonstration reactors would not be needed for the large evolutionary ALWRs, whereas such demonstrations would likely be needed for passive, mid-size ALWRs and the innovative designs to establish their technical and economic viability. The US DOE is supporting development of ALWRs to make them commercially available by the mid-1990s and development of two of the other innovative designs -- the MHTGR and the LMR (see discussion below) -- to make them commercially available early in the next century.[5]

The Electric Power Research Institute (EPRI) in the US has produced a "Utility Requirements Document," guidelines compiled by utilities, vendors and architectural engineering firms which will be used as the generic basis for certification of standardized ALWR designs by the US Nuclear Regulatory Commission.[6] Among EPRI's requirements is a goal that ALWRs should have a probability of core damage of less than 10^{-5} per year (i.e., a factor of 10 lower than NRC's core damage guideline). NRC has recently indicated that they expect advanced reactor designs to achieve higher standards of safety performance

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[4] "Outlook on Advanced Reactors," Nucleonics Week, March 30, 1989, pp. 8-10.

[5] Griffith, JD, US Department of Energy, "Overview of Safety and Environmental Characteristics of Advanced Commercial Nuclear Reactors," presented at Workshop on the Future of Nuclear Power: Responding to the Challenges of Public Concerns, The MITRE Corporation, McLean Virginia, September 5 and 6, 1990, p. 3.

[6] Advanced Light Water Reactor, Utility Requirements Document, Electric Power Research Institute, Palo Alto, California, March 1990.

than earlier designs and encouraged industry's commitment to attaining a higher safety goal.[7]

Relevance to Developing Countries

The question of unit size may be of particular relevance to the appropriateness of these technologies in developing countries. Whereas large reactor units like the evolutionary ALWRs are well-suited to industrialized countries facing shortages of power plant sites, including Japan, France and Germany, small units offer the advantage that they can be added in smaller, less costly increments with shorter construction lead times. Smaller units could be manufactured at the factory, in modules, and shipped to the plant site. This could be expected to help overcome the loss of economies of scale which can result (but in past experience often have not resulted) from building larger units. Some developing countries have already experienced delays in bringing large nuclear units on line because of high cost. Furthermore, power grids in developing countries would have difficulty tolerating removal of a large unit from the grid for regular maintenance outages; multiple small units could be more appropriate. It should be noted that US and other utilities anticipate requiring capacity additions in smaller increments than in the past.

In general, developing countries have not acquired nuclear power stations because of their large size and, more importantly, because they have lacked the technical infrastructure to run plants of such complexity. Some proponents of innovative reactor concepts, such as the MHTGR, have suggested that the deployment of such concepts should, in time, help introduce nuclear power in a wider range of nations. This would derive from the fact that such units would be smaller, passively safe, simpler to operate and less dependent from a safety perspective on active operator intervention. Experience suggests, however, that such concepts will first have to be successfully deployed in their countries of origin before they will be amenable to export.

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7 US Nuclear Regulatory Commission, Staff Requirements Memorandum Re. SECY-89-102 - Implementation of the Safety Goals, June 15, 1990.

6.2 TECHNOLOGY CHOICES

This section provides an overview of the various advanced reactor concepts currently under development, divided into three categories: large evolutionary ALWRs; passive mid-size ALWRs; and other innovative designs.

6.2.1 Large Evolutionary ALWRs

Large Evolutionary ALWRs make incremental improvements over existing LWRs and have simplified designs. Because they are based on existing plants, no demonstration plants would need to be built in order to gain regulatory approval. The US Department of Energy estimates that the probability of severe accidents at evolutionary ALWRs will be a factor of 10 lower than at current LWRs, due in part to the application of passive safety features, to the extent possible.[8] There are two principal US designs in this category, offered by General Electric and ABB-Combustion Engineering. In addition there are two European concepts, being developed by Electricité de France (EdF) and Siemens/KWU.

6.2.1.1 General Electric ABWR

The Advanced Boiling Water Reactor (ABWR) has been developed by General Electric and an international team of BWR manufacturers, incorporating features from BWRs in Europe, Japan and the US. It is a 1350 MWe reactor featuring internal recirculation pumps (adopted from German and Swedish BWRs) and other innovations. Emergency Core Cooling and Residual Heat Removal systems will be triple redundant. The ABWR is the furthest developed of all advanced reactor concepts; two ABWR units are now being built for Tokyo Electric Power Co. in Japan, for 1996 and 1997 start-up. They will be supplied by a joint venture of General Electric, Hitachi Ltd. and the Toshiba Corp. In addition, the ABWR design is currently under review by the US Nuclear Regulatory Commission for design certification as a pre-approved US standard design. NRC's best-case estimate is that it would grant the Final Design Approval (FDA) to the ABWR in July 1991.

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[8] See Note 5 supra.

6.2.1.2 ABB Combustion Engineering APWR

ABB Combustion Engineering's System 80+ standard design evolves from its System 80 standard PWR design, three units of which have already been built at the Palo Verde station in Arizona. The new design is a 1350 MWe unit employing a steel sphere containment enclosed in a secondary cylindrical concrete containment building. Increased redundancy of safety systems is provided, including four trains of emergency coolant water. Steam generator improvements increase the volume of water available to remove decay heat from the reactor. Control room design has been dramatically altered to simplify the presentation of information to the operator. ABB-CE is teamed with Duke Power Co., the US utility, feeding operating and construction experience into the effort. Like the General Electric ABWR, this design is currently under review by the US NRC who project granting the FDA to the System 80+ in August 1993 or thereafter.

6.2.1.3 Siemens/Kraftwerk Union Convoy B

The Kraftwerk Union division of the German company Siemens is currently designing the so-called Convoy B Concept as an evolutionary follow-on to its Convoy A PWR design, three of which were built in the 1980s. The Convoy B design will similarly be in the 1300-1400 MWe range. The German utilities believe that the next generation designs should be evolutionary so as to capture the lessons learned from the current generation of reactors, and do not find acceptable a concept relying totally on passive safety features. The safety philosophy is to increase safety margins in the plant, provide a longer "grace period" before operator action is needed, and reduce to very low levels the probability of radioactive releases outside the plant during a core melt accident, but not to eliminate the role of the operator in responding to an accident. The Convoy B would more than meet the EPRI ALWR targets for core melt frequency, and would nevertheless include several measures to mitigate the consequences of a core melt.

6.2.1.4 Electricité de France REP 2000 Project (including the "N-plus" and "X-plus" concepts)

Electricité de France (EdF) launched in 1989 its REP 2000 project to define the next LWR design for future EdF orders after the year 2000. EdF has also regarded REP 2000 as a possible vehicle for defining a common European PWR design via interaction with other

European utilities, the two leading European vendors Framatome and Siemens/KWU (and their joint venture called Nuclear Power International), and national safety authorities. The objective is to develop a European PWR concept that could be built anywhere in Europe while satisfying both utility requirements and safety standards jointly agreed to by national regulatory authorities. Although REP 2000 will look at all PWR concepts, the primary focus is at present the "N-plus" concept, which would be an evolution of the 1400 MWe N4 design EdF is now building, and "X-plus," which would be a more clearly European PWR that possibly could be based on the so-called "common product" Nuclear Power International will design. The expectation is that N-plus and X-plus will eventually merge together to define a single set of specifications.

EdF strongly believes that the next design should be large in size (1300-1400 MWe) and evolutionary in concept in order to build on the experience base in the current reactor generation. EdF has a limited number of 4-unit sites and thus wishes to maximize total capacity at those sites by building large units. There is little or no interest in systems that depend totally upon passive safety approaches. REP 2000 will rely upon the defense-in-depth approach to reactor safety, and containment design will receive considerable attention. The fundamental design objective is a severe accident probability of about 10^{-6} per reactor-year.

The outcome of REP 2000 promises to materially shape the design of EdF's next generation PWR, and, in turn, what EdF would call upon the French vendor Framatome to build. However, because the European Community is promulgating procurement rules to open up public sector procurements, presumably any vendor could be asked to bid on the design EdF adopts. There is considerable pressure now to develop a common "European" PWR design, and though the REP 2000 project is just beginning, it appears to be the primary pathway to achieving that objective.

6.2.2 Passive Mid-Size ALWRs

US vendors Westinghouse and General Electric have each developed mid-sized ALWR designs that are based on proven LWR technology but at the same time offer some of the passive safety features that have contributed to the attractiveness of the more innovative designs. As specified in EPRI's ALWR Utility Requirements Document, in the event of a design basis accident, including loss of all AC power, these designs should provide that no operator intervention will be necessary to ensure that core protection regulatory limits will not be exceeded for at least 72 hours following initiation of the accident.[9] One common feature is lower power density, which runs counter to past design philosophy which sought to maximize the available power from a reactor vessel to improve plant economics. The new thinking underlying the low density approach is that the plants will be simpler, will require far less backup safety equipment, will have fewer unplanned outages and can be manufactured in factories, and will therefore actually be cheaper to build and operate.

The US Department of Energy decided in September 1989 to support the continued design development and testing of both the Westinghouse and GE designs, at a funding level of $50 million apiece. Conceptual designs for both reactors have been submitted to the US NRC for review. DOE anticipates that Safety Analysis Reports for the AP-600 and SBWR will be submitted to NRC in July 1992 and October 1992, respectively. The NRC expects that final design approvals (FDAs) for these two designs could be issued by as early as May 1995. It should be noted that neither of these reactors is expected to require construction of a demonstration unit as a prerequisite to NRC certification.

6.2.2.1 Westinghouse AP-600

The Westinghouse AP-600, an advanced PWR with a 600 MWe power rating, uses experience gained from past PWR operation combined with certain innovations intended to improve safety and performance and minimize cost. The power density of the fuel core is reduced which makes it easier to control this reactor than existing LWRs; larger heat-transfer surfaces would allow heat to be conducted away from the core more quickly. Passive safety systems are incorporated into the design to provide emergency cooling water, such as the use of the natural gravity feed from large tanks holding as much as 400,000 gallons of water which are placed above the reactor vessel within the containment building to provide an abundant supply of water. Another feature is the use of natural circulation instead of the use of pumps. Furthermore, the steel containment building is surrounded by a concrete shield building, with an airflow space between the two allowing cooling by natural circulation.

By introducing this passivity, the designers avoid the need for large amounts of piping, safety-related pumps and other complex components and systems that may themselves be subject to failure. Furthermore, large sections of the plant could be factory prefabricated

[9] See Note 6 supra, (Volume 1, p. 26 in reference).

in modules, which could be shipped to the plant site by rail and inserted into preconstructed areas of the plant. The smaller number of parts would also make the plants easier to maintain than existing LWRs.

6.2.2.2 General Electric SBWR

GE's Simplified BWR (SBWR) employs some of the same passive safety concepts as the Westinghouse AP-600. The core of the reactor, which also has a 600 MWe rating, has low power density and natural circulation core flow. Natural circulation requires more water and therefore a larger reactor vessel than a forced-circulation BWR of similar size. The large reactor vessel is surrounded by a passive pressure suppression containment system, including large pools of water that can be injected into the vessel under the force of gravity to ensure that the core stays covered with water. A passive containment cooling system can remove residual heat from the containment for several days without any operator action. These measures allow significant plant-wide reductions in the need for safety-grade systems and equipment. The simplified design allows the use of more factory fabricated components, which should allow faster construction, enhanced quality control, improved maintenance and cost savings.

6.2.3 More Innovative Designs

Aside from the advanced versions of the LWR discussed above, other reactor technologies under development are, for the most part, fundamentally different from existing reactors and, therefore, generally have not been demonstrated in an actual plant. All of the designs discussed in the following sections are designed to optimize safety. Proponents of the ALWR designs discussed above maintain that those improved designs can achieve perfectly adequate safety performance and that people can have greater confidence in them because of past experience with the technology, whereas some of the more innovative designs have never been built and could still encounter unforeseen operational difficulties. Proponents of the more innovative designs counter that their alleged "inherent" safety allows much simpler designs, without the need for extensive backup safety systems based on the defense-in-depth philosophy. The simpler designs are expected to avoid operational difficulties that some large complex LWRs have encountered, offering competitive cost in spite of certain design concepts such as their lower power densities and smaller unit size.

As noted above, the US DOE is providing funding support for the development of the MHTGR and the LMR. These designs are discussed in the following two sections, followed by summaries of three other innovative designs: the PIUS, SIR and CANDU-3 reactors.

6.2.3.1 MHTGR

The Modular High Temperature Gas-Cooled Reactor (MHTGR) design, as conceived by a consortium comprised of General Atomics, Siemens and Bechtel, consists of four separate reactor modules, built in below-grade silos, yielding a total power output of 550 MWe. The design is based on past experience with gas-cooled reactors mainly in the UK, US and Germany, including the 15 MWe AVR reactor in Germany which has been running successfully since the 1960s and has been subjected to extreme tests proving its inherent safety. The design employs the inert gas helium as the coolant. General Atomics suggests that the MHTGR is ideally suited to developing countries, many of which may experience continually rising demand for electricity, because it is smaller and simpler to operate and maintain and does not require a large, expensive support infrastructure. It should be noted that the design is highly amenable to factory prefabrication, aiding project economics.

The MHTGR's inherent safety results mainly from the capability of the fuel and the graphite core and support structures to withstand high temperatures without melting or damage. The refractory coated fuel takes the form of tiny uranium fuel kernels coated in several layers of ceramic materials to a diameter of less than one millimeter. The coatings are highly heat resistant and hold in radioactive fission products, serving as miniature "containments." (In fact, the designers of the MHTGR propose that the reactor does not require a containment building.) These fuel particles are mixed with a graphitic material and formed into fuel elements, which can be either packed into graphite fuel element blocks or -- in the German design -- into billiard ball-sized pebbles. Test data show that no fuel particles fail if the fuel is maintained below 1800 degrees C. If the normal primary cooling systems become disabled, decay heat is dissipated passively by natural conduction and radiation to the Reactor Cavity Cooling System (RCCS). Because of the small size of each reactor module, cooling to the reactor's silo structure and surrounding earth alone would limit fuel temperatures to about 1600 degrees C.

Whereas other passively safe reactors assure safety by guaranteeing that coolant can reach the fuel by passive means, the MHTGR is designed to go beyond this and to assure that even in the absence of coolant the fuel will not melt. MHTGR proponents state that, unlike LWRs, the reactor does not require active systems to respond to subsystem malfunctions, and is immune to major structural failure and operator error.[10] There is a strong consensus on the MHTGR's inherent safety, although this must still be demonstrated and there are some remaining questions about reliability, performance and economics. At the 330-MWe Fort St. Vrain helium-cooled reactor, in Colorado, which went into service in 1974, the fuel performed well as a containment device but a non-safety related design flaw plagued operations and the plant has been retired. A newer plant in Germany, the 300-MWe THTR, faced long and expensive construction delays and finally went on line in 1986, but was removed from service in 1989 due to financial difficulties of the operating utility that were unrelated to the specific type of reactor.

Remaining doubts about the MHTGR's viability could be allayed if the US DOE decides to build an MHTGR as a replacement for aging reactors used for the production of tritium for the US nuclear weapons program (a decision on whether to build an MHTGR, a light water reactor or a heavy water reactor for this purpose is scheduled to be announced by DOE in November 1991). Such MHTGR would include a containment structure. DOE will subsequently submit more detailed design documents for NRC review, including information on the adequacy of containment. On the other hand, General Atomics and its partners have argued to-date that containment is not needed for the proposed civilian reactor design; the NRC has nearly completed the review of the conceptual design that excludes a containment structure, and has issued a draft Safety Evaluation Report[11] on it in spite of the fact that the NRC and its Advisory Committee on Reactor Safeguards (ACRS) have thus far favored the use of a containment with the MHTGR.

As for the German HTGR program, which is being pursued jointly by ABB and Siemens/KWU, the shutdown in 1989 of the THTR-300 demonstration plant and the collapse of a proposed German-Soviet HTGR project in the USSR have delivered a serious blow to the prospects of the HTGR in the German utility market. German utilities are interested principally in large LWRs. A smaller HTGR concept developed by Siemens could offer process heat applications in the industrial sector, but that market in all likelihood will not materialize in the absence of a resurgence of reactor orders by German

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[10] Lidsky, LM, "Nuclear Power: Levels of Safety," Radiation Research, 113, 1988, p. 217.

[11] US Nuclear Regulatory Commission, Draft Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor, NUREG-1338, March 1989.

utilities and an improvement in the public acceptance climate for nuclear power in Germany. Electricity demand will not require the German utilities to place orders for new reactors until 1995 at the very earliest. However, the recent unification of Germany, the shutting down of the Greifswald Soviet-designed nuclear units and the poor pollution control in existing fossil plants may change this outlook.

6.2.3.2 Liquid Metal Reactor

Nuclear industry leaders in several industrialized nations held the view for many years that the long-term viability of nuclear power would depend on the successful development and deployment of the liquid metal fast breeder reactor (LMFBR). This derived from the assumption that there was only a finite quantity of natural uranium available in the world and that only the fissionable isotope, uranium-235, which constitutes about 0.7% of all natural uranium, was useful for producing power. The fast breeder reactor, which would be fuelled with the plutonium which is produced in thermal reactors (like the LWRs), could, if appropriately designed to incorporate depleted uranium (the otherwise unusable uranium) in a blanket around the core, essentially "breed" plutonium from the depleted uranium blanket, possibly even more than it consumed; thus the fast breeder reactor could be essentially a renewable resource.

Since the pressure on the uranium market has abated in recent years due to a slowdown in nuclear programs, the sense of urgency in developing a commercially viable breeder reactor has also abated. Nevertheless, several of the industrialized countries still include the development and commercialization of the breeder in their longer-range nuclear plans with the expectation that capital cost reductions might, in time, make the breeder competitive with LWRs in the first part of the next century. Such programs have been under way for many years in France, West Germany, the United Kingdom, Japan, the Soviet Union and the United States, most of which have been associated with relatively large-size breeder designs.

Considerable emphasis is being given in current FBR design programs to improving the safety features of the proposed future units. The West Europeans are developing a common design which would satisfy the national licensing requirements of each of the participants. Japan is now building a 300 MWe prototype breeder reactor -- known as "MONJU" -- that will include some new passive safety characteristics.

The US in recent years has primarily devoted its energies to exploring a smaller concept with passive safety features, known as the Integral Fast Reactor Concept (or IFR). An innovative design now being studied is the IFR combined with the PRISM[12] concept, being pursued jointly by the Argonne National Laboratory and General Electric under US DOE sponsorship. The IFR/PRISM concept would involve the deployment of small modular reactors combined with a pyroprocessing plant that would separate the uranium, plutonium and long-lived actinide wastes that are produced in the reactor from the shorter-lived fission product wastes. The separated uranium, plutonium and actinides would be fabricated into fuel and returned to the reactor, in a closed, self-contained fuel cycle. To accomplish this, the fuel would be different from that of most currently operating reactors in that it would be made of uranium metal rather than uranium oxide.

The GE PRISM concept is designed as a 1395 MWe plant consisting of nine 155 MWe LMR modules arranged in three identical 465 MWe power blocks. The modules are located below grade and are thus highly resistant to the effects of possible sodium fires and other phenomena. Active and passive safety systems also act to prevent or mitigate sodium fires. PRISM provides normal shutdown heat removal through its intermediate heat transport system (IHTS). In the unlikely event that the IHTS became inoperative, the reactor vessel auxiliary cooling system (RVACS) is intended to provide backup heat removal in a fully passive manner, maintaining sodium temperatures well below structural limits. This feature, combined with the fact that the reactor core is designed to provide strong negative feedback with rising temperature, is intended to enable PRISM to withstand severe undercooling or overpowering accidents.

Aside from safety considerations, the IFR/PRISM approach has gained substantial recent attention because of its potential advantages for the management of spent fuel. The fact that it can consume long-lived actinides present in spent fuel and prevent their accumulation means that the quantity of waste requiring deep geologic disposal potentially could be substantially less than for LWRs and other reactors. Actinides from LWRs can also theoretically be burned in an IFR, and the US DOE is now starting to seriously study this possible option.

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[12] Stands for "Power Reactor Innovative Small Module."

The US NRC has nearly completed the review of PRISM's conceptual design, and has issued a draft Safety Evaluation Report.[13] The draft SER identified two issues "of fundamental concern" with the PRISM design: potential accident scenarios that could lead to significant fuel damage, sodium boiling and subsequent positive reactivity, with the potential for a large early release of radioactive material; and the potential for an in-core flow blockage to escape detection and lead to fuel melting and a reactivity accident. Nonetheless, the SER concludes that the NRC staff "considers the PRISM design to have many favorable attributes and to have the potential to achieve a level of safety at least equivalent to that of current generation LWRs, with enhanced safety in certain areas."

6.2.3.3 PIUS

The PIUS reactor, which stands for Process Inherent Ultimate Safety, is a completely passive design developed in Sweden by ABB in the 1970s, although never built. PIUS is a PWR with two design options: one with steam generators outside the reactor vessel, rated at 300-400 MWe, the other with internal steam generators rated at 640 MWe.

In PIUS, the reactor core sits near the bottom of a large prestressed concrete reactor vessel in a large pool of cold water containing boron, which can absorb neutrons. The thermal power of the reactor cannot exceed the cooling capability of this pool water. Under normal operating conditions, this cold, borated pool water would stay below the hot, higher density water in the reactor core as a result of thermal layering. In the event of a failure of the primary coolant pumps, this hydraulic balance would break down, allowing pool water to flow into and cool the core, and -- due to its dissolved boron -- shut down the chain reaction. The amount of water inside the reactor vessel is sufficient to remove residual heat for one week by evaporation. This emergency cooling system would operate entirely by natural circulation, without the use of any electrically powered equipment.

The simplicity and inherent safety of this design concept have been praised by many, but at the same time there is concern (e.g., by the US DOE) over the plant's operability since it has never been built. ABB has submitted preliminary design information to the US NRC, and claims that the design features have already been tested extensively and a

[13] US Nuclear Regulatory Commission, Draft Preapplication Safety Evaluation Report for Power Reactor Inherently Safe Module Liquid Metal Reactor, NUREG-1368, September 1989.

demonstration plant is not needed. NRC has not yet determined its schedule for certification review of PIUS, but it is clear that it will be done subsequent to the large evolutionary and passive mid-size ALWRs.

6.2.3.4 SIR

ABB Combustion Engineering, Rolls-Royce and Associates, Stone & Webster and the UK Atomic Energy Authority are collaborating on the design of a 320 MWe PWR called the Safe Integral Reactor (SIR), so called because the core, major primary circuit components, steam generators and reactor coolant pumps are all built integrally into a single large reactor vessel. The reactor would be located in a below-ground cavity, with the core near the bottom of the vessel under 30 feet of water. Operating safety margins are high as a result of a very low core power density, about 60% that of current PWRs. Since primary circuit flow is all contained within the vessel, there are no large primary system pipes external to the vessel, which minimizes the risk to the core from large-break LOCAs. Protection against small LOCAs is also provided by the large amount of coolant present within the reactor vessel. The emergency coolant injection systems have a 72 hour capability without the need for AC power.

SIR's developers state that its design allows a short on-site construction time and minimal capital expenditure, and consider it particularly appropriate for smaller utilities. They have recognized that a demonstration of the design will be needed and are seeking UK support for a demonstration project.

6.2.3.5 CANDU-3

Atomic Energy of Canada Limited (AECL) has developed a new reactor design based on extensive past experience with its CANDU Heavy Water Reactor (HWR), which, unlike PWRs, employs heavy water (D_2O as opposed to H_2O) as the coolant and moderator. Over 14,000 MWe of CANDUs are now in operation worldwide. The CANDU-3, with a power rating of 450 MWe, is designed in 400-tonne modules that can be fabricated at or away from the construction site. As with other small reactor designs, the CANDU-3 is expected to be well-suited to utilities with uncertain load growth, small grid size, and/or limited financial resources, such as utilities in developing countries.

AECL has submitted preliminary design information to the US NRC. NRC has not yet determined its schedule for certification review of CANDU-3, but, like ABB's PIUS reactor, it is expected that it will be done subsequent to the large evolutionary and passive mid-size ALWRs.

6.3 VIEWS OF NUCLEAR POWER CRITICS ON ADVANCED REACTORS

In July 1990 the Union of Concerned Scientists (UCS) released a report evaluating three of the advanced reactor designs described above: the Westinghouse AP-600; the General Atomics MHTGR; and the General Electric PRISM.[14] The purpose of the study was to review issues surrounding licensing and deployment of advance reactors in the US, and survey key features of the three designs to identify issues requiring resolution before such reactors are deployed. The three concepts were chosen based on the following criteria: US domestic designs only; one design per vendor; one ALWR and two advanced non-LWRs; and no evolutionary ALWRs because they are already well understood and are an "unlikely choice for US utilities."

The UCS is a leading, technically-qualified group in the US that acts as a nuclear energy "watchdog" and critic. It should be noted that other groups who are dedicated to the anti-nuclear cause generally have different views on advanced reactors from UCS, and would oppose nuclear power regardless of advances in safety. For example, Joan Claybrook, President of Public Citizen, has stated that "there is no substance to nuclear industry claims that a 'new generation' of nuclear power plants could be designed to correct the economic, safety, and environmental problems that have plagued the current generation." On the other hand, Jan Beyea of the National Audubon Society states that "we remain opposed to the existing generation of nuclear plants for the obvious reasons: safety, nuclear waste, proliferation, economics. But we do take the [global warming] problem very seriously, and think there is some sense in finding out whether these so-called passive, second-generation reactors will work."[15]

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[14] Advanced Reactor Study, prepared by MHB Technical Associates, San Jose, CA, for the Union of Concerned Scientists, July 1990.

[15] Fisher, A: "Next Generation Nuclear Reactors: Dare We Build Them?," Popular Science, April 1990, p. 69.

The UCS report concluded that the three plants considered:

> "appear to have been designed with explicit attention to severe accident issues. This is a welcome and, we believe, completely necessary change in the approach to nuclear power plant design. Advanced reactor designs are also including consideration of operator performance, application of modular construction techniques to hold down capital costs, application of factory quality assurance controls to a greater scope of the plant design, and improvements in operation and maintenance. These developments are also welcome.
>
> Advanced reactor designs appear to hold promise for resulting in improved operations safety compared with existing plants... Notwithstanding the potential promise of advanced reactor designs, there remain some thorny issues to be addressed before construction and deployment of a large number of new reactors should be considered... The issues raised in this report are not intended to represent impediments to the deployment of advanced reactors. Rather, we believe that it is important to identify and resolve the principal safety, environmental, and economic issues inherent in the advanced reactor proposals before widespread deployment becomes a reality... If the issues we have raised (and that others will raise based on additional and, we trust, more detailed evaluations of advanced reactors) are not satisfactorily resolved, the second generation of nuclear power plants will be small in number and short-lived."

The UCS report also criticized claims of "inherent" and "passive" safety made by the reactor designers. They conclude that claims of inherent safety are not credible because one can always envisage an extreme scenario that would result in a large radiological release, though they acknowledge that the degree to which this is true varies from design to design. UCS suggests that claims of inherent safety confuse rather than clarify the safety debate, and that the proponents of advanced reactors should instead compare the safety of advanced reactors with the more familiar existing reactors, which would be much more meaningful than simply claiming inherent safety. With respect to the claim of passive safety, UCS stated that

> "The MHTGR Reactor Cavity Cooling System (RCCS) appears, in our view, to qualify as a "*passive*" system -- it always operates (absent some catastrophic failure), it requires no power to operate (relying on natural circulation of air), it requires no actuation signals to perform its safety function, it is not dependent on actuation of valves to perform its safety function, etc.
>
> In contrast, Westinghouse claims that its [AP-600] emergency coolant injection systems are "*passive*." It is our opinion that this claim does not pass muster... What has been eliminated (at least to a large extent) is injection function unavailability due to failure of pumps. While this may be a good thing, it does not make the

> emergency coolant injection safety function "*passive*." Again, we urge caution in the use of the word "*passive*" in the discussion of advance reactor safety characteristics."

The reactor manufacturers named in the UCS report, together with the nuclear industry and several government agencies around the world, will undoubtedly have strong rejoinders to many of the points made by the UCS and other groups and can be expected to give substantial emphasis to the advances that have been made or are being explored in reactor design from a safety perspective.

Nevertheless, it seems clear that the current emphasis on passive safety is unlikely, in itself, to remove the issue of nuclear power from the realms of public debate. It can be expected that the nuclear opponents will still seek to challenge proposed new nuclear projects and will continue to press for the utilization of alternate energy sources as well as greater emphasis on conservation. However, only time will tell whether the growing public concerns about the environment, specifically relating to acid rain and the "Greenhouse Effect", will lead to greater public acceptance of nuclear power in the future.

7.0 SAFETY-RELATED QUESTIONS IN DEVELOPING COUNTRIES

Numerous requirements would need to be met by developing countries desiring nuclear power programs before nuclear power plants could be built and operated. Among them are several considerations that would be necessary to ensure that deployment of nuclear power plants in these countries would meet generally-accepted international standards of reactor safety. These include:

- the existence of an independent regulatory authority in the host state to establish safety regulations, conduct licensing proceedings and enforce regulations;

- a quality assurance system that can meet regulatory and safety requirements to ensure safe reactor operation;

- well-trained personnel and a general level of technical and scientific development which can support continued safe operation and maintenance of nuclear power plants, including appropriate responses to unusual occurrences; and

- the existence of an adequate technical infrastructure and emergency response capability.

These topics are discussed in Section 7.1. In Section 7.2, the role of advanced reactors in developing countries desiring nuclear power programs is addressed.

7.1 SAFETY-RELATED PREREQUISITES TO NUCLEAR POWER DEPLOYMENT IN DEVELOPING COUNTRIES

As in the world's industrialized countries which already employ nuclear power, operation of nuclear power plants in a safe manner in developing countries would be extremely important not only for the purpose of preventing adverse consequences to public health and safety, but also to prevent damage to reactors that would render expensive facilities useless and cost hundreds of millions of dollars to safely clean up. The following attributes would likely help ensure that such facilities would be operated in a safe manner in developing countries.

7.1.1 Regulatory Organization

Irrespective of the type of technology that might be selected as being the most appropriate for the developing country, it is vital that an organizational structure with clearly defined responsibilities is set up to oversee and regulate the construction and operation of nuclear plants of any type being built in the country. National legislation should exist in the radiation protection and nuclear safety areas, establishing a radiation protection authority and a reactor safety authority (which could be combined in one agency).

In particular it is essential that a tough regulatory regime with effective review, inspection and enforcement capabilities is established. Such an organization is required right from the start to review and approve the safety characteristics of each type of reactor proposed, irrespective of whether it is supplied by an international supplier or whether it is designed and built by the developing country itself (which would be unlikely in most countries). The regulatory body must be sufficiently independent of nuclear power plant owners/operators that its primary objective of ensuring safety will not be compromised.

7.1.2 Quality Assurance System

Experience in the advanced nuclear countries has shown the necessity for the establishment at the outset of a quality assurance system designed to ensure that the quality of materials being used meets regulatory specifications and that methods of construction assure safe plant operation. Quality assurance standards must be achieved by various industrial suppliers of the nuclear industry as well as the plant operators themselves, and suppliers must be regularly audited to ensure that QA standards continue to be met.

7.1.3 Qualified Personnel and Sufficient Scientific/Technological Development

Once a plant has been constructed and brought into operation, its continued safe operation and maintenance can only be assured if it is operated by highly trained personnel having a general level of technical/scientific background which enables them to maintain plants in safe condition, as well as understand and react properly to any unusual occurrences that could pose a safety risk. Scientists and engineers of several different disciplines are needed at a plant to ensure the safety function. Some of the needed personnel could be brought in on a contract basis.

Among the various positions at a nuclear power plant for which the staff needs to have adequate training, the importance of training of nuclear power plant operators cannot be overemphasized. The accidents at Three Mile Island and Chernobyl clearly demonstrate the need for this. In both cases it has now become clear that human factors were a major, if not the major, contributor to the accidents. In the case of Chernobyl particularly, the application of inappropriate procedures by the plant operators led to the calamity. In the case of Three Mile Island, there were operator errors but the design of the reactor was such that the built-in safety features prevented a large release of radioactivity.

Beyond the need for adequately trained personnel, it is also necessary to have research institutes (or at least access to leading institutes abroad) which can advise nuclear operators, as well as the regulatory authorities, on safety and radiation protection issues and which can help in various types of problem-solving.

7.1.4 Technical Infrastructure

Related to the points raised above, countries desiring to begin a nuclear power program will also require a supporting industrial infrastructure to a certain extent. For example, existing industries must either be already sufficiently developed or must be strengthened to the point where they can achieve QA requirements, as discussed above. Also, an adequate emergency planning and response capability, including the participation of local officials, must be in place. Relatedly, relatively unpopulated areas are preferable for nuclear power plant siting.

7.2 ROLE OF ADVANCED REACTORS

As noted in Section 6.0 of this report, advocates of some advanced technologies with passive safety characteristics, including the MHTGR, have stressed the merits of their designs for use in developing countries, citing the smaller unit size and lower capital cost among the factors that could render them more appropriate for a less developed electric grid. The new reactors with passive safety features which are currently being developed are designed to reduce the likelihood of accidents and prevent serious consequences in cases where accidents do occur. It is anticipated that developing countries interested in nuclear power will be focusing particular attention on improved or advanced reactors in the coming years for safety and other reasons.

If developing countries do in fact choose technologies with greater passive safety characteristics, there could be some benefit in terms of meeting the safety-related prerequisites identified above. (However, it should be kept in mind that accidents which do occur could still damage these expensive facilities, and it would still be necessary to have in hand qualified manpower and an adequate technical infrastructure to prevent the loss of investment.)

It is worthy of note that technical assistance and training programs are offered to developing countries by both the IAEA and the individual countries who supply nuclear technology.

7.3 NUCLEAR REACTORS IN EXISTING POWER SYSTEMS

A number of nuclear power plants which are based on Soviet designs are currently operating in Eastern European countries. The safety-related systems of these plants differ from those required of nuclear power plants in other parts of the world, and are attracting growing attention. Nevertheless, they provide a substantial proportion of the electricity in the countries in which they are located which would make it virtually impossible to shut them down.

The Bank is proposing to lend to the electricity sectors in these countries and Bank staff involved in these programs may, therefore, find it appropriate to review the past operating experience of these reactors to identify where there are weaknesses which should be addressed.

Safety is clearly of overriding importance and indications of weakness in the safety of such plants can be observed by an examination of the records of the number of incidents/ accidents and unplanned shutdowns that have occurred. Other areas where weakness may be observed is in plant management and the training of the plant operators.

It must, however, be recognized that some weaknesses (e.g. pressure vessel embrittlement) are of a highly technical nature and can only be determined by qualified experts such as metallurgists or specialized engineers. Bank staff may, therefore, wish to consider involvement of such experts in their analyses.

Appendix 1

Radiation Fundamentals

The introduction to the main report provides a basic explanation of such terms as radiation, radiation dose, radioactivity and half life. This appendix provides further background discussion relevant to understanding radiation and radioactivity measurements.[1]

The interaction of radiation with matter causes the ionization of the absorbing atoms; i.e., affected atoms take on a positive or negative charge. Therefore, one fundamental way of measuring quantities of radiation is in terms of the amount of electric charge produced in matter. The roentgen unit is defined as the quantity of X-ray or gamma radiation that produces ions carrying one statcoulomb of electric charge (either positive or negative) in a cubic centimeter of air at 0° C and 760 mm Hg (standard temperature and pressure).

Since the degree of biological damage caused by radiation depends on the amount of energy absorbed from the radiation, the basic unit of radiation dose expresses the amount of energy absorbed per unit mass of tissue. One rad is defined as the amount of radiation that deposits 100 ergs[2] of energy per gram of tissue (or other matter). A quantity of radiation expressed in roentgens (electric charge per unit volume) can be converted into rads (energy deposited per unit mass) if the necessary conversion factors are taken into account: one ion carries a charge of 4.8×10^{-10} statcoulombs; the average energy dissipated in the production of ions is 34 electron volts (eV) (which is equal to 5.4×10^{-11} ergs); and air has a density of .001293 g/cm^3. Thus:

$$\begin{aligned} 1 \text{ roentgen} = {} & (1 \text{ statcoulomb/cm}^3) \times (1 \text{ ion}/4.8 \times 10^{-10} \text{ statcoulombs}) \\ & \times (5.4 \times 10^{-11} \text{ ergs/ion}) \times (1 \text{ cm}^3/.001293 \text{ g air}) \\ & \times (1 \text{ rad}/100 \text{ ergs/g}) = \mathbf{0.88 \text{ rads (in air)}}. \end{aligned}$$

Since tissue has a higher electron density than air, a roentgen produces a higher energy absorption in tissue -- approximately 0.95 rads. One roentgen of radiation exposure to

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[1] Detailed discussion of this subject may be found in H. Cember, Introduction to Health Physics, Pergamon Press, New York, 1988.

[2] One erg = 9.5×10^{-11} BTUs.

tissue is therefore considered to be approximately equivalent to one rad of absorbed dose. Since different types of radiation (e.g., alpha, beta, gamma) have different toxicity levels per unit of energy deposited, due to differences in the rate at which they deposit their energy as they pass through tissue, an adjustment must be made to doses expressed in rads if the toxicity level of one dose of radiation is to be comparable to that of another. Each type of radiation therefore has an assigned *quality factor* to account for its biological effectiveness relative to an arbitrarily chosen reference radiation. Radiation dose in rads multiplied by the quality factor for a given type of radiation yields the *radiation dose equivalent*, in *rems* (which is the unit used throughout this report). The value for dose equivalent expresses the toxicity level of a radiation dose regardless of the type of radiation, whereas values for doses in rads are not directly comparable as to their biological effectiveness for different radiation types.

In international SI units, the *gray*, which equals 100 rads, and the *sievert*, which equals 100 rems, are used to measure absorbed dose and radiation dose equivalent, respectively.

The *curie*, the unit which is used in the report for measuring quantities of radioactivity, is that amount of a radioisotope which produces 3.7×10^{10} disintegrations (e.g., alpha, beta or gamma emissions) per second. In international SI units the becquerel is used, which equals one disintegration per second. Therefore, 1 curie corresponds to 3.7×10^{10} Bq or .037 terabecquerel (TBq). It is important to note that a curie of one radioisotope is equivalent to a curie of another radioisotope only in the sense that the number of disintegrations per second is the same. They are not equivalent in mass. For example, one curie of the isotope Strontium-90 weighs about 7 milligrams while a curie of the isotope carbon-14 weighs about 220 milligrams. This difference is due to the fact that each isotope has a unique rate of radioactive decay; whereas strontium-90 is highly radioactive, with a half-life of 28 years, and it only takes 7 milligrams to produce one curie of radioactivity, carbon-14 is much less radioactive, with a half-life of 5,730 years, and it takes 220 milligrams to produce one curie (Note: the term "half-life" denotes the time required for a given radioisotope to decrease to one half of its original quantity).

Furthermore, a curie of one radioisotope does not produce the same radiation dose equivalent as a curie of another. One reason is that, as noted above, radiation dose equivalent depends on the type of radiation and the corresponding quality factor. In addition, even for similar types of radiation, different radioisotopes will produce radiations of different energy levels, which will produce different doses.

APPENDIX 2

AVERAGE RADIOACTIVE EMISSIONS FROM WORLDWIDE NUCLEAR POWER PLANTS, 1980-1984, PER GIGAWATT[1]-YEAR OF ELECTRICITY PRODUCED

Effluent Category

Reactor Type[2]	Airborne Noble Gases (TBq)[3]	Activation Gases (TBq)	Airborne Tritium (TBq)	Liquid Tritium (TBq)	Carbon-14 (GBq)[4]	Airborne Iodine-131 (GBq)	Airborne Particulates (GBq)	Liquids Excluding Tritium (GBq)
PWR	218 ± 40	N/Ap.	5.9 ± 2.4	27.0 ± 1.8	345 ± 80	1.75 ± 0.33	4.5 ± 2.9	132.4 ± 49.5
BWR	2150 ± 523	N/Ap.	3.4 ± 1.6	2.1 ± 0.5	330 ± 110	9.33 ± 4.9	43.3 ± 24.4	115 ± 47
HWR	212 ± 48	N/Ap.	670 ± 190	290 ± 68	6336 ± 3333	0.23 ± 0.08	0.040 ± 0.016	25.7 ± 8.7
GCR	N/Ap.	2318 ± 224	5.4 ± 0.9	96 ± 13	N/Av.	1.4 ± 1.1	1.39 ± 0.79	4520 ± 1790
LWGR	5466 ± 1365	N/Ap.	N/Av.	1.7	1300	80 ± 40	15.69 ± 16.19	N/Av.

1. One gigawatt is the approximate power rating of one reactor unit at a large nuclear power plant.

2. PWR = pressurized water reactor
   BWR = boiling water reactor
   HWR = heavy water reactor
   GCR = gas-cooled reactor
   LWGR = light-water cooled, graphite-moderated reactor

3. One terabecquerel (TBq) is the quantity of radioactive material that produces one trillion radioactive decays per second. One terabecquerel is approximately equal to 27 curies.

4. One gigabecquerel (GBq) is the quantity of radioactive material that produces one billion radioactive decays per second. One gigabecquerel is approximately equal to .027 curies.

Source: United Nations Scientific Committee on the Effects of Atomic Radiation, Sources, Effects and Risks of Ionizing Radiation, 1988.

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