LIGHT WATER REACTOR (LWR) SAFETY

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In this paper, a historical review of the developments in the safety of LWR power plants is presented. The paper reviews the developments prior to the TMI-2 accident, i.e. the concept of the defense in depth, the design basis, the large LOCA technical controversies and the LWR safety research programs. The TMI-2 accident, which became a turning point in the history of the development of nuclear power is described briefly. The Chernobyl accident, which terrified the world and almost completely curtailed the development of nuclear power is also described briefly. The great international effort of research in the LWR design-base and severe accidents, which was, respectively, conducted prior to and following the TMI-2 and Chernobyl accidents is described next. We conclude that with the knowledge gained and the improvements in plant organisation/management and in the training of the staff at the presently-installed nuclear power stations, the LWR plants have achieved very high standards of safety and performance. The Generation 3 + LWR power plants, next to be installed, may claim to have reached the goal of assuring the safety of the public to a very large extent.

This review is based on the historical developments in LWR safety that occurred primarily in USA, however, they are valid for the rest of the Western World. This review can not do justice to the many many fine contributions that have been made over the last fifty years to the cause of LWR safety. We apologize if we have not mentioned them. We also apologize for not providing references to many of the fine investigations, which have contributed towards LWR safety earning the conclusions that we describe just above.

KEYWORDS : LWR Safety, TMI-2 Accident, Chernobyl Accident, Design Basis Accidents, Severe Accidents, Presently-Installed LWR Power Plants, New LWR Power Plants

1. INTRODUCTION

The LWR Safety that we are concerned with in this paper is, basically, about estimating the risks posed by an individual or a population of nuclear power plants to the public at large and the efforts to reduce these risks. The public of most concern is that which resides in the vicinity of a nuclear power plant but also at other locations, which could be affected by an accident in a nuclear power plant located anywhere.

The basic goal of LWR safety is to assure that a LWR will not contribute significantly to the individual and societal health risks. This basic goal translates to the prevention of the release of radioactivity into the environment from the power plant. A complementary aim is to prevent damage to the plant and to protect the personnel at the plant from injury or death in an accident.

Since LWR safety aims to protect public at large, it is heavily regulated. Each nuclear power country (and some without nuclear power plants) have regulatory commissions (bodies), which regulate every aspect of a nuclear power plant from design and construction to operation and any modifications. They require very extensive analyses, documentation and quality control. The reactor safety design has to follow definite rules and basis. Some of these requirements will be described in the text.

The reactor performance on the other hand, is concerned with long term steady state operations, since most LWR plants are base-loaded and strive to operate at full power, without interruption, between scheduled outages for maintenance. Reactor performance is concerned with efficiency, capacity factor, fuel cycle costs, maintenance costs and the radiation dose to the operating staff. Thus, it is not regulated. However, it has been found that a well-running LWR power plants is, generally, a safer plant with a much lower frequency of incidents, which, generally, are the precursors to more serious events.

2. THE EARLY DAYS

The nuclear era started with the natural-uranium-graphite pile built by Fermi and his associates at the Stagg field of the Chicago University [1]. It did not involve light water as a coolant since only natural Uranium was available and criticality could be achieved only with graphite or heavy water. The safety concepts developed there, however, were adopted by the LWR plants that developed several yeas later. It was recognized by Fermi and his associates that:

- nuclear fission reactions, which are the basis of nuclear power, emit high levels of radioactivity and thus could be a health hazard to any person exposed to it. This implied shielding, containment and remote siting.
- the safe operation of the reactor (or pile) would require protective and control measures, as evidenced by the provision of a control rod in the pile that Fermi and his associates built.

The shielding and remote siting were practised for the plants that were built for the production of plutonium in USA and other countries during the years before and after the end of the Second World War. Remote siting of these plants not only protected the public but also maintained the secrecy of the production of the material for nuclear weapons for a number of years.

The containment aspect for protecting the safety of the public from a nuclear accident was not considered or employed for the plants generating plutonium. Those were the years of above-ground nuclear weapons tests, which were releasing considerable amounts of radioactive fission products in the atmosphere in any case. Fortunately, there are no reported accidents of any great significance in the plutonium production plants in USA or in other Western countries.

The leak-tight containment as a safety system for a civilian nuclear power plant was not long in coming. It was proposed in 1947 [2] for a sodium-cooled fast reactor which was the focus of the power reactor development by the US Department of Energy at that time. Later, the leak-tight containment was adopted by the LWR power plant developers.

3. THE DEVELOPMENT OF THE CIVILIAN LWRS

The LWR development started as a military program in USA from the initiative of Admiral Rickover, the father of the US Nuclear Navy [3]. The pressurized water-cooled reactor (PWR) was conceived as the power plant for submarine propulsion by his team, since a sodium-cooled fast reactor, the focus of the US national program was considered as unsuitable for a nuclear submarine submerged in water. The funds and the considerable intellectual resource assembled by Admiral Rickover resulted in an extraordinarily rapid development of the PWR power plant for the US submarine fleet.

President Eisenhower issued the call for Atoms for Peace in 1954 [3] which became the signal for the adaptation of the military developments for civilian purposes. The construction of the Shipping port PWR [3], which was completed in 1957, provided the prototype for nuclear power plants generating a substantial amount of electrical power for public consumption. It should be remarked here that EBR-1, a fast reactor, was the first nuclear reactor in USA to demonstrate generation of electrical power. However, the quantity generated was insufficient to transmit for public consumption.

The development of the other civilian water-cooled nuclear power reactor, i.e. the Boiling Water Reactor (BWR) was started almost in parallel with that of the PWR and the construction of the Shipping port PWR power plant. The BWR development was spear headed by the General Electric (GE) Company, a private enterprise, which, in fact, invested their own funds to develop the BWR as a commercial power plant. In this they were aided by the work performed at National Laboratories in USA, e.g. Argonne National Laboratory, which built a 5 MW BWR system [3] and the Idaho Laboratories, where experiments were performed [4] to demonstrate the stability and safety of the BWR system. The first prototype commercial BWR power plant was designed and built, as a dual-cycle (i.e. it had a separate steam generator for the steam that went to the turbine) plant, already in 1960 by the General Electric Co.

In USA, the first truly commercial nuclear power plant was the Yankee-Rowe plant, a PWR, which was also constructed in 1960. This plant was conceived as a commercial venture and was specifically commissioned by a utility company supplying electricity to the public. The Yankee-Rowe plant was constructed with a leak-tight containment and it was approved for commercial operation by the regulatory authorities in the United States Atomic Energy Commission (USAEC). The plant designers at that point in time did not realise that their decision to employ a leak-tight, pressure-bearing, containment was the most important safety decision that they took.

The civilian use of nuclear energy was very popular with the public during 1960s. Claims were being made that nuclear energy could provide unlimited electric power, too cheap to meter. Projections were being made of constructing hundreds (or even a thousand) power reactors in USA alone. Some proposals involved the siting of the plants very close to the cities to provide generation sources near the large consumption centers, in order to become more economic in the total cost of the electricity to the consumers. The 1970s saw a large number of orders placed by the US utility companies with the US vendors of which the most prominent were [1] Westinghouse for the PWR plants, since it was the vendor for the naval PWRs, [2] General Electric Co for the BWRs, since they were the developers of this reactor type, [3] Babcock Wilcox for PWRs, since they had much experience in construction of conventional power-generation equipment. Later Combustion engineering, another vendor of the conventional power-generation equipment joined their ranks and constructed PWRs. There was a quick scale-up of reactor power from 300 to 600 to 1000 MWe.

LWR nuclear plant construction programs were started in Germany, France, Japan, Sweden, Soviet Union and some other countries. England chose to construct gas cooled nuclear power plants.

4. EARLY SAFETY ASSESSMENTS

Assessments of the hazards of a major accident in a nuclear plant were started in early 1950s. The 1955 Geneva Conference, which was the first gathering of nuclear reactor scientists from East and West, provided the first estimates of the possible hazards of a hypothetical accident in a LWR. The paper presented by the US investigators [5] estimated the consequences to be 200-500 fatalities and 3000 to 5000 high exposures. Even before these results were fully digested, the study WASH-740 [6] was published. This study was performed with the stated purpose of estimating the consequences for the 'worst-case' nuclear accident, in order to provide data for nuclear plant insurance legislation. The authors of WASH-740 assumed that 50% of the radioactive inventory of a 500 MWe reactor would be released in the atmosphere and at the same time the most unfavourable weather conditions would be prevailing. They estimated that up to 34000 fatalities, 43000 injuries and contamination of 240000 square kilometres of land could occur. A probability estimate of ~10% was quoted pertaining to these consequences. The WASH-740 authors stated categorically, that the estimates of deaths, injuries and land contamination were highly conservative because of the assumptions made in deriving these consequence estimates.

5. THE SITING CRITERIA

The consequences and the risks estimated in the WASH-740 study hastened the enactment of the site criteria by the US Atomic Energy Commission (USAEC). These criteria recommended in the report TID-14844 [7,8] published in 1959, are based on the recognition that the pressure-bearing containment provided on the projected LWRs would, most probably, survive in a hypothetical accident and that the release to the atmosphere will only be through the leakage of the fission products deposited in the containment (the Source-Term). The study assumed a certain fraction of the gaseous and solid fission products, contained in an irradiated core, to be deposited in the containment as it was done for the WASH-740 study. The difference, of course, between these two studies is that the authors of the WASH-740 study released it immediately to the atmosphere, while the authors of TID-14844 released it to the atmosphere only at the leak rate of the containment, i.e. 0.1%/day.

The TID-14844 required the establishment of an exclusion and a low-population zone (LPZ) on whose boundaries the limits of exposure that could be suffered by the thyroid and the whole body of a person, situated there, were prescribed.

The recommendations of TID-14844 were considered in the site criteria enacted by the USAEC in 1962. Those criteria provided the minimum distance that a nuclear plant should be situated away from a low population center as a function of its thermal (or electric) power capacity. These criteria are the first regulatory action towards recognizing the potential of using a nuclear reactor for generating electricity if it is sited correctly.

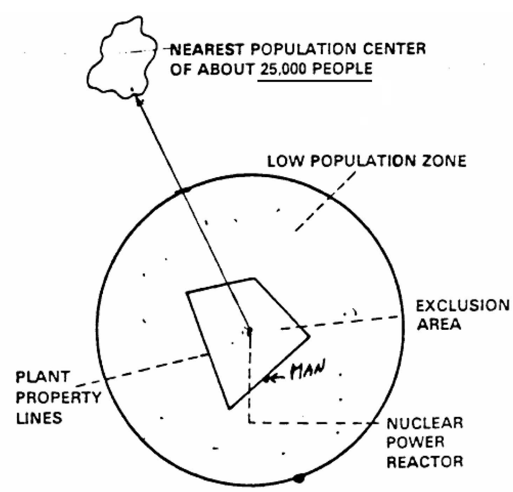
5.1 Assumptions and Requirements of TID-14844 and 10 CFR 100

The authors of the TID 14844 and the code of Federal Regulations that resulted: 10 CFR 100, made the following main assumptions about the source term, i.e. the fission products released into the containment during the nuclear accident:

- 100% of the noble gas inventory in the core
- 50% of the halogen inventory
- 1 % of the solid fission products
- 50% of the released halogens remain available for further release from the containment; spray, wash-down features and filtering devices could provide additional reduction. However, these were not credited.
- Containment leak rate of 0.1% per day.

The radioactive fission product transport in the atmosphere was assumed to be under the following conditions:

- atmospheric dispersion under inversion-type conditions; no shift in wind direction for the duration of the leakage.
- no ground deposition of particulates.



- At Boundary of Exclusion Area, No Member of Public May Exceed 25 rem Whole-Body Dose For Design Basis Accident.
- Timely Evacuation of Low Population Zone Must be Planned (Emergency Evacuation Plan)
- Population Center Must Be at Least 1 – 1/3 Times Farther Away than LPZ Boundrary

Fig. 1. Part 100 Distance Requirements, (Typical Plant)

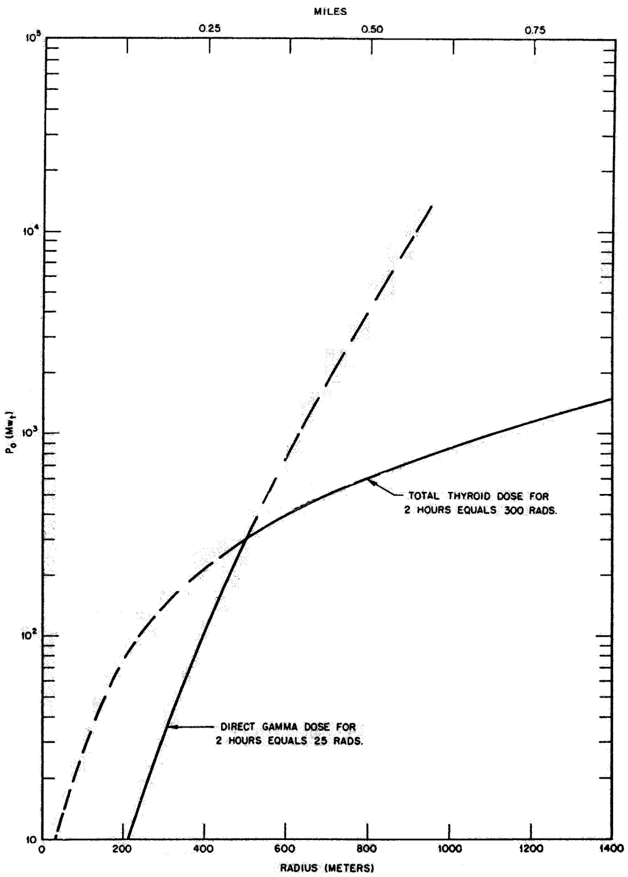


Fig. 2. Exclusion Radius Determination

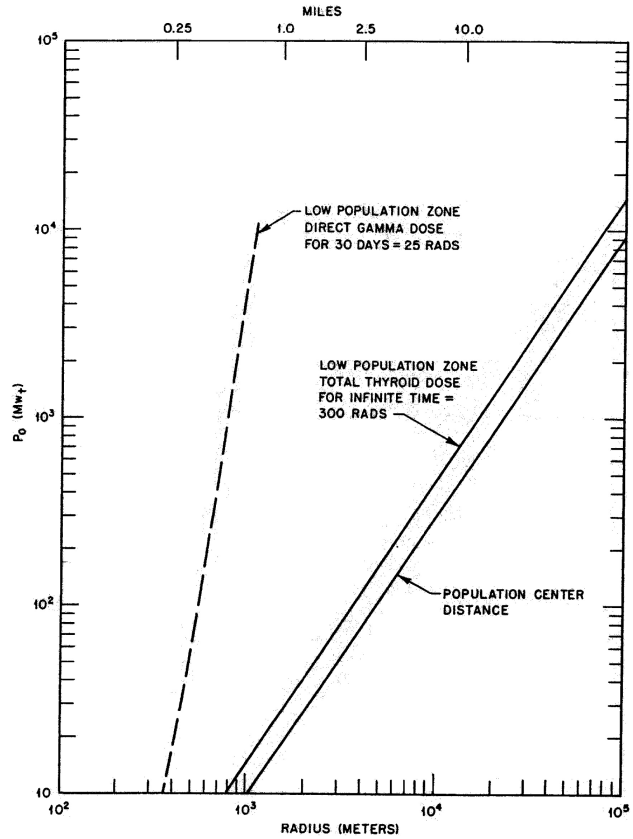


Fig. 3. Low Population Zone and Population Center Radius Determination

The doses to the population provided in these documents are as follows:

- for 2 hours exposure at the boundary of the exclusion area, maximum whole body dose of 25 rem and thyroid dose of 300 rem.
- For 30 days, or infinite exposure, at the outer boundary of the LPZ, the maximum whole body dose of 25 rem and thyroid dose of 300 rem.

The definitions of the areas around the site are also provided in 10 CFR 100, as follows (see Fig. 1):

- exclusion area is the fenced area around the plant where public is normally not allowed
- low population zone is the area around the exclusion area whose extent is determined by the dose rates established above
- the nearest population center containing about 25000 residents should be away from the reactor by at least 1.33 times the distance from the reactor to the outer boundary of the LPZ.

The other requirements on siting are, of course:

- regulations on land use for plant and transmission lines,
- regulations for water use and the temperature of the water body to which the plant is discharging water to,
- the access and corridors for evacuation of the residents of at least the LPZ,
- other environmental regulations,
- the attitudes of the local population and Government bodies.

Figures 2 and 3, taken from TID-14844 show the distances needed for the exclusion area, the LPZ and the population center as a function of the thermal power of the LWR to be sited at a particular location.

6. SAFETY PHILOSOPHY

Before a fleet of nuclear power reactors were constructed for the commercial market, it was important that a philosophy for the safety design and a safety design basis be developed [8]. These were developed as the orders for the reactors started coming in USA and the regulatory part of the USAEC started to function in earnest.

6.1 The Defense in Depth Approach

The defense in depth approach to the safety design followed intuitively from the configuration of a LWR, which provides 3 important physical barriers to the release of the fission products to the environment viz, the clad on the fuel element, where the fission products are generated, the reactor vessel, which contains all the fuel elements forming a reactor core and the leak-tight containment, which is supposed to keep any fission products inside the containment from escaping to the environment. Assuring the integrity of each of these physical barriers in any accident scenario becomes the defense in depth approach against the release of radioactivity to the public environment.

In practical design aspects, the defense in depth approach for safe design was refined as a set of preventive measures as follows:

- perform careful reactor design, reactor construction and reactor operation so that malfunctions, which could lead to major accidents will be highly improbable,
- provide systems and equipment, which would prevent such malfunctions, as do occur, from turning into major accidents. Examples are: Scram systems to shut down the fission reactions in the core and leak-before-break detection equipment to anticipate serious loss of coolant from the reactor primary system,
- provide systems to reduce and limit the consequences of the postulated major accidents, e.g. the emergency core cooling systems (ECCS)

There are at least 3 echelons for the defense in depth approach. The first echelon provides accident prevention through sound design, which:

- can be built and operated with very stringent quality standards,
- provides high degree of freedom from faults and errors,
- provides high tolerance for malfunctions should they occur,
- employs tested components and materials,
- employs considerable redundancy in instrumentation, control and mitigation systems.

The second echelon of the defense in depth approach assumes that there will be human or equipment failure. It provides detection and protection systems to maintain safe operation or shut the nuclear plant down safely when incidents occur due to the human or equipment failure. Examples of the detection and protection systems are:

- sensitive detection systems to warn of incipient failure of fuel cladding or the coolant systems,
- redundant sources of in-plant electricity,
- systems for automatic shut down on nuclear fission reactions in the core (SCRAM) or signals from the monitoring systems. This is generally achieved through insertion of control rods in the core.

The third echelon of the defense in depth approach is to provide additional margins to protect the public should severe failures occur despite the first two echelons. Examples of systems and equipment, which provide such additional margins are:

- the steel-lined concrete building containing the whole high pressure primary system of a LWR. This containment should be constructed with a pressure-bearing and leakage prevention rating,
- the ECCS to flood the core with water and to keep it covered if the high pressure coolant of a LWR is lost through a break in the piping of the primary system somewhere.

The above defense in depth approach is followed all over the world. It is quite comprehensive and has served the nuclear enterprise well over the ~50 years that the commercial plants have been in operation. There has not been a single catastrophic break in the large pipes of the primary systems in the LWRs installed so far.

7. SAFETY DESIGN BASIS

A basis for the design of the safety systems had to be provided to the LWR designers. In USA, this was provided by the USAEC, the precursor to the US Nuclear Regulatory Commission (USNRC), which was formed a few years later (1974) and was established as an independent civilian agency charged, specifically, to regulate the fast developing nuclear power industry in USA. The USNRC has other functions besides the regulation of the nuclear power industry: from radiation sources used in medical profession to waste management.

The safety design basis selected for the LWRs was, and is, the large break loss of coolant accident (LOCA). This is the two sided, guillotine break of the largest pipe in the primary system, i.e. the coolant discharge from such a break is supposed to occur from both sides of the break. The large LOCA is considered to be an enveloping accident removing water from the primary system at the largest rate. The consequence of a large LOCA is the uncovery of the reactor core in a very short time (~30 seconds) requiring in turn supply of water to the reactor at a commensurate rapid rate to fill the vessel and submerge the core in water, before the decay heat would raise the temperature of the Zircalloy clad above the threshold temperature for the exothermic Zircalloy – steam oxidation reaction, which can lead to the clad and the fuel melting.

It must be remarked here that the PWR and BWR have significantly lower probability of a power increase accident due to reactivity insertion. In these reactor designs, as the core heats up and increases the void fraction due to the boiling of the water coolant, there is a large negative reactivity and power feed back, which shuts down the fission reaction, even when the control rods are not inserted. The reactivity induced accident (RIA), thus, was not considered as the safety design basis accident.

Besides the large LOCA as the enveloping accident, other accident and/or incident events were specified and it was required that the specified events be analysed and documented for review in the Chapter 15 of the Safety analysis Report (SAR) that each plant owner has to submit before it could be granted a construction or operating license.

Some examples of transients specified for safety design basis are as follows:

- increase in heat removal by the secondary system
- decrease in heat removal by the secondary system
- decrease in reactor coolant system flow rate
- reactivity and power distribution anamolies
- increase in reactor coolant inventory
- decrease in reactor coolant inventory
- radioactive release from a subsystem or component
- anticipated transients without scram.

These transients were chosen since they affect the state of the reactor and can lead to additional complications in operations. For example, an increase in heat removal by the secondary system in a PWR would lead to low temperature for the primary coolant which would add reactivity to the core and increase power. A decrease in heat removal in the secondary system would lead to higher pressure in the vessel of a PWR.

A decrease in the core water inventory may be through a small break LOCA, which complicated the transient experienced by the TMI-2 (as described later) and lead to the accident. The small break LOCA, in particular, can go on for a considerable time to become a complex transient. The operator response and actions can change the course of the transient to a benign or a more demanding state for the reactor parameters.

Besides the chapter 15 of the SAR, the regulatory authorities require the submittal of comprehensive information on other related topics in the SAR. These include for example:

- site description
- functional performance
- description of all safety systems and the engineered safety features
- conformance with the General Design criteria for the design, construction and operation of the plant
- quality assurance program and pre-operational testing
- periodic testing requirements for operations
- failure mode analysis
- radiological monitoring and surveillance requirements
- possible R&D needed to confirm the design chosen.

7.1 LOCA and the ECCS Controversies

With the successful operation of the PWR and BWR demonstration reactors at Shippingport and Dresden, respectively, the US electric power utility industry wanted to construct plants of much greater power rating. Meanwhile, the large LOCA was approved as the design basis, which demanded the design and performance of a very robust emergency core cooling system (ECCS). The USAEC appointed a task force to study the various ECCS designs submitted by the vendors for the reactor plants that the utility companies wanted. The task force was also chartered to evaluate the consequences in case the ECCS did not function sufficiently well. Their findings [9] that this could lead to core melt-down and possible containment failure posing a great hazard to the public created much uncertainty all around. The USAEC responded to this uncertainty by requiring improvements in the new ECCS designs of the vendors, e.g. by providing greater capacity, redundancy, diversity and assurance of electrical supply. The other reactors were also asked to install or improve their ECCS.

A major recommendation of the ECCS task force to the USAEC national research program was to (a) perform experiments to observe the thermal hydraulic processes of the ECCS, (b) obtain data, (c) develop models for predictive analyses and (d) validate the models against the measured data. The USAEC started an experimental program by building a small scale thermal-hydraulic loop simulating a large-break LOCA with ECC injection. The first experimental results obtained lead to uncertainty about the efficacy of the ECCS. It was observed that the injected water bypassed the core and did not reach the hot rods in the electrically-heated core. The analysis models at that point in time had not recognized that the steam generated in the hot core would not let the water enter the core due to phenomenon of counter-current flooding (CCF). It was later observed that the CCF breaks down during an extended ECCS injection and water could reach the hot rods. The USAEC responded [10,11] by demanding additional margins in the ECCS calculation models. Quite detailed criteria were issued for the assumptions to make and the heat transfer correlations to use in the models for predictions of the thermal hydraulic behaviour of the plant during the large LOCA and the ECCS injection following the large LOCA. A limiting temperature for the Zircaloy cladding of the hot rods was proposed, which was kept below the temperature at which the exothermic Zircaloy-steam reaction accelerates. The specification of these criteria did not satisfy the critics and public hearings on the ECCS performance in LWRs were organised in January 1972.

The ECCS hearings [11] lasted for more than 18 months and the conclusions reached, pointed to the inadequacy of the knowledge and understanding of the phenomena defining the thermal hydraulic behaviour during the large LOCA; a very violent event. In addition, it was concluded that the calculational models available at that time could not be defended easily. Another study [12] of large LOCA and ECCS for LWRs was conducted by a group assembled by the American Physical society. This study also concluded that there was insufficient knowledge-base to make reliable quantitative predictions of the plant behaviour and consequences in reactor accidents. This American Physical society group recommended an intensive research effort for 10 years or more, to acquire sufficient knowledge about the very complex phenomena that prevail during the large LOCA accident. They emphasized the development of validated models, which could be used for LOCA with ECCS for prototypic plants. They also pointed to the need of quantifying the margins which may be available in the mitigation of a large LOCA by the ECCS.

The acrimonious debates during the ECCS hearings, the differences in the opinions of various experts and the recommendations made by the various independent groups prompted the US to start an ambitious research program on LOCA and ECCS. Simultaneously a code of Federal Regulation (10CFR 50) [13] was enacted, which had the force of a federal law, providing the safety design-basis and the general design criteria for the safe operation of a LWR plant. This design basis included a large LOCA and a set of operational transients for which analyses results had to be submitted. The large LOCA analyses had to be performed on a very conservative basis with prescribed assumptions and correlations for heat transfer. The clad temperature limit was specified to be 1200°C (2200°F) and the limit on clad oxidation was prescribed to be 17%. Several guideline documents were written, which for example, provide categories of accidents, classes for various levels of quality control, etc. As an example, the primary system had to be class 1, which required rigorous quality control and inspections on the materials, the manufacturing and the welding processes employed. The corner-stone for LWR safety was established as (i) remote siting, (ii) prevention of any radioactivity release in the design-basis accidents (DBAs), (iii) defense in depth, (iv) strong containment and (v) deterministic safety analyses. These corner stones are still the basis for safety design of the LWRs.

The countries in Western Europe and Japan watched the developments on the LWR safety in USA. They chose to follow the rules and regulations that were enacted in USA for the design, construction and safe operation of LWR plants. They may have added some more regulations but they did not subtract any of the important criteria or regulations in 10CFR50 and 10CFR100. These countries also followed the US ECCS Research Program and they collaborated with it, and supplemented it, by building several experimental facilities of their own.

The large LOCA and the ECCS research conducted in the USA and other countries was very comprehensive and very expensive, since several large scale integral effect and separate effect facilities were constructed. The largest of these was the LOFT (loss of fluid test) facility, which employed a nuclear core generating ~55MWth power. The scaling employed in all of these facilities was that the ratio of power/primary system volume was kept equal to the prototypic value from a 1000 MWe LWR power plant. This scaling was found to be appropriate for most of the thermal hydraulic processes that occur during the large break LOCA and the ECCS injection. Hundreds of large and small scale, integral effect, and separate-effect, experiments were performed in these facilities to understand the physics of the two-phase thermal hydraulic phenomena occurring and to obtain pertinent data for the validation of computational codes, e.g. the RELAP series of codes and the TRAC code, which was developed later on. Many of the separate-effect experiments illuminated the details of the phenomena, which helped in the formulation of the computational models that were later employed in the integral codes. For example, the reflooding process being so complex was modelled with representative models for which insight and data were obtained from the separate-effect experiments.

Most of these experimental facilities were closed down in 1990s. There are, however, a few large scale facilities left, e.g. ROSA in JAERI, Japan, PKL in Germany, where research on any new issue that may arise in LWR thermal hydraulics and safety would be performed. Presently, it is believed that the codes RELAP-5 and TRACE (successor to TRAC) are able to generate reasonable predictions of the thermal hydraulic behaviour of PWRs and BWRs in the large LOCA accident with ECCS injection. These codes without the large LOCA assumptions provide best-estimate analysis results for the large LOCA. The operational transients can also be analysed, since, recently, those codes have incorporated the control systems with their time lags, the secondary systems of PWRs and the actions of the safety and the relief valves.

After the TMI-2 accident in 1979 (described later in this paper), the integral and separate effect facilities built for the research on large LOCA were employed for the research on small break LOCA, which posed its own unique thermal hydraulic phenomena, e.g. phase separation (since more time is available), natural circulation, etc. Again hundreds of separate-effect and integral-effect experiments were performed to delineate the physics of the new phenomena and models were developed for incorporation in the codes. Later, the LOFT facility was also employed for a few tests in which severe accident conditions were simulated and indeed clad and fuel damage occurred and fission products were released. These were the terminal tests for the LOFT facility; data obtained in those tests has been employed for validation of core-degradation models in the LWR severe accident codes.

It must be remarked here that in all of the experiments conducted, over the many years, on the integral and separate-effect facilities for the LOCA and ECCS research, at no time the clad on the heater rods or on nuclear fuel rods (in LOFT) experienced temperatures exceeding 2200°F. It has been re-assuring to the reactor safety community that the ECCS, as designed for the PWRs and BWRs, will be able to protect the core (with perhaps some minimal damage) and prevent any significant release of radioactivity to the containment or to the environment. It should be added that containments are designed for the large LOCA thermal and pressure loadings and their integrity should not be in question for the large LOCA accident.

8. PUBLIC RISK OF NUCLEAR POWER

Late 1960s and the early 1970s were the glorious years for nuclear power in USA and the World. The promise of cheap nuclear power was still in full bloom and there were firm orders and many orders in the wings for nuclear power plants in USA. The power ratings were increasing and more and more companies were becoming nuclear power plant vendors. The prospect of a large number of nuclear plants dotting the landscape of USA and of other countries in a relatively few years made some persons quite apprehensive and questions arose about the risk posed to the general public by accidents in nuclear power plants. Since, there was no quantitative measure of public risk in 1960s, Farmer [14] of UK proposed such a measure through a curve of probability vs. consequences, with the risk defined as probability x consequences. The proposed curve was basically intuitive and recognised that as the consequences increase, the probabilities of occurrence for such consequences should decrease. The risk of a certain enterprise would be acceptable to the public if the probability of a certain consequence remained below the proposed curve. In contrast, the probability values above the curve, for specific consequences, would not be acceptable to this public.

Farmer also recognised that public may well accept accidents with low consequences at a reasonable frequency, however it may not accept accidents with very high consequences at an equal risk level. Thus, the high consequence accidents should pose a low overall societal risk.

Farmer proposed the curve shown in Figure 4 with the accident consequences represented by the release of curies of ^{131}I on the abscissa and the probability of occurrence on the ordinate. The risk level of 1 is chosen for the consequence level of 10^3 curies of radioactive ^{131}I released with a probability of 10^{-3}. The curve is flattened at the top so that the highest probability of some (10 curies) radioactive ^{131}I release is 10^{-2}. The curve can be given a slope of -1 for an equal risk for high consequence accidents, but more likely public acceptance would be for the line with a slope of -1.5, so that the very high consequence events occur with a relatively low public risk, e.g. a release of 10^6 curies of ^{131}I would be acceptable only with an occurence probability of 10^{-8}, i.e. with a risk level of 10^{-2}.

Farmer's curve and approach did not specify any risk values for accidents in nuclear plants, but it clarified societal acceptance of risk for a new technology and it provided a base for the quantification of the risk of nuclear power.

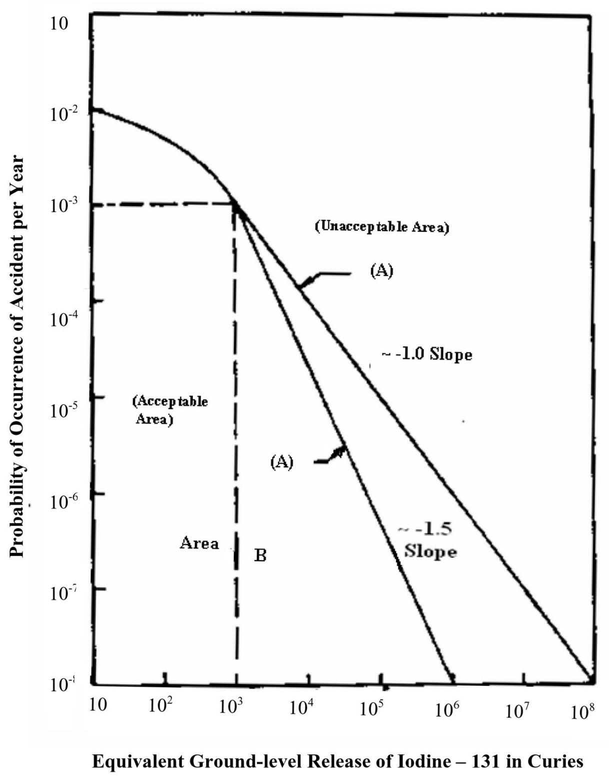


Fig. 4. Farmer's Curve

8.1 The Reactor Safety Study (WASH-1400)

The United States Nuclear Regulatory Commission (USNRC) was established in 1974. One of the early initiatives of the USNRC was to sponsor a study of the public risk of nuclear power under the leadership of Professor Norman Rasmussen of MIT with the very able assistance of Saul Levine of USNRC. This study, named the Reactor Safety Study (RSS), published as the report WASH-1400, [15] provided the first structured assessment of the public risks of accidents in the U.S. LWRS.

The RSS employed a comprehensive and detailed fault and event tree methodology to obtain the probabilities of faults and of the accident scenarios that could release radioactivity in the environment to damage the health of the public in the vicinity of the plant and also contaminate the land around the site of the nuclear plant. A typical PWR and a typical BWR were chosen for the Level 1, 2 and 3 probabilistic safety analysis (PSA). State of the art methodology was employed for the Level 2 and Level 3 consequence estimations. Clearly, the severe accident progression and consequence models employed were not as detailed and sophisticated as they became later; but, looking back, it is remarkable that the estimates made for the consequences in many of the beyond the design basis (BDBA) scenarios were reasonably good. This speaks for the good engineering judgment capability of the U.S. researchers working on the RSS. This study was published in 1975.

In the following paragraphs we will provide some snapshots of the methodology employed by the RSS researchers and then describe the principal results obtained by them, as reported in WASH-1400 [15].

The RSS researchers recognized quite early that the integrity of the containment, which is the last barrier to the release of fission products to the environment, is the key to the determination of the consequences of the severe accident. In this context the containment failure was categorized as shown in Figure 5. The α and γ modes of containment failure (rupture) were considered as catastrophic due to a fast-acting loading generated either by an in-vessel steam explosion or by a hydrogen detonation in the containment. The $\dot{\varepsilon}$ mode of failure applied, primarily to the Mark-1 BWRs in USA. The δ mode of failure was ascribed to the over pressure created in the containment due to the steam released from the primary system during a break, but more significantly due to the molten corium concrete interaction that occurs when the molten core is discharged on the containment basement in the event of the failure of the vessel. The energetic modes of containment failure would not provide any retention of the containment aerosol source term but the δ and β (containment leakage) modes of failure could be credited with retention due to (a) the natural processes of aerosol deposition on walls, floors etc. and (b) operator-action or automatic remedial actions e.g. spray actuation.

α	β	γ	δ	ε
CONTAINMENT RUPTURE DUE TO VESSEL STEAM EXPLOSION	CONTAINMENT LAKAGE	CONTAINMENT RUPTURE DUE TO HYDROGEN BURNING	CONTAINMENT RUPTURE DUE TO OVERPRESSURE	CONTAINMENT RUPTURE DUE TO MELT-THRU

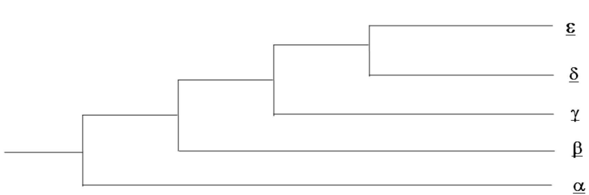


Fig. 5. PWR Containment Event Tree

Figure 6, shows the event tree for the BDBA large break, LOCA scenario. An event tree is inductive and it looks forward. Its logic is very similar to that of a decision tree, as employed for decision-making in business, economics etc. An event tree is generally drawn from left to right and begins with an initiator. This initiator is an event that could lead to shutdown or failure of a system or a component. In the event tree, the initiatiors are connected to other possible events by branches; a scenario is a path of these branches. The tree in Fig. 6 shows the probabilities of success and failure for each of the systems or processes listed at the top. These, in turn, determine the final probability of the consequences represented by each branch of the tree on the right.

The event tree shown in Fig. 6 can be reduced to that shown in Fig. 7 to concentrate on the major consequences

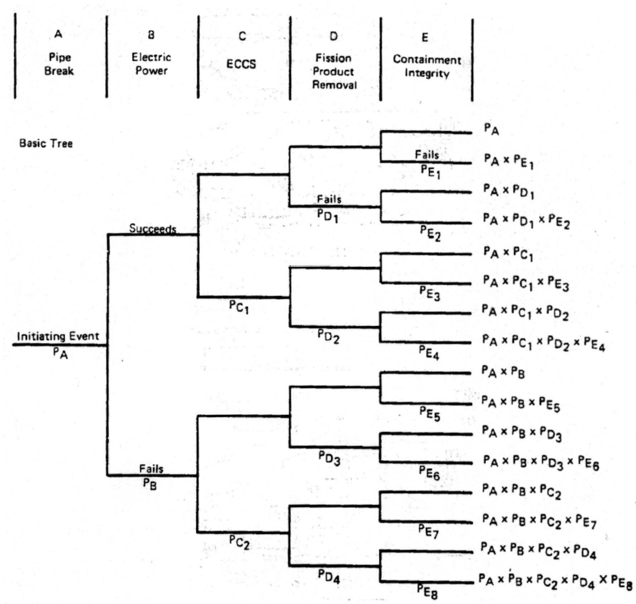


Fig. 6. Simplified Event Tree for a Large LOCA

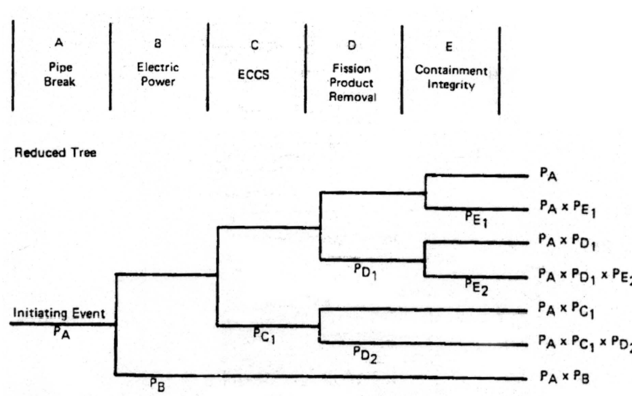


Fig. 7. Reduced Event Tree for a Large LOCA

and their probabilities. This tree shows, for example, that with the non-availability of electric power both the ECCS and fission product removal systems are unable to function. The containment failure probability is same as the probability of the failure of electric power to function. The tree in Fig 6 or 7 could be combined with the containment failure mode tree in Fig. 5.

The fault tree, an example for which is not shown here, is a construction to determine the probability of the initiating fault or failure. Thus for each of the tree branches shown in Fig. 6, the probability shown, e.g. P_B for the failure of the electric power to function or P_{C1} the failure of ECCS to function, in spite of the success of having electric power functioning, are determined by the fault tree analysis in which the plant electrical and mechanical systems and components are examined for the probability of their failure, which leads to the probability P_B or P_{C1} described above. The fault trees can be huge with many branches since many components e.g. electric relays, switches, pumps etc. may be involved in the functioning of a safety system.

The fault trees and event trees provide the probability of the events occurring; however, the consequences of the events have to be determined by developing models for the physical process that occur in the BDBA. The researchers of the RSS developed a code for estimating the source term (fission products resident in the containment as a function of time) and, then, the release and transport in the environment for the various modes of failure of the containment and the various metreologies assumed for the areas around the location of the PWR and the BWR that were considered in the study. The fission products released, being, so many were combined into certain number of categories and their biological damage, in terms of early fatalities, or exposures that would lead to early illnesses, was calculated. In addition, estimates were made of the total property damage that would occur and the land area that could be subject to contamination due to the release. In this respect it was found that a higher-consequence accident scenario would have a lower probability of occurrence. For example catastrophic containment rupture which would release fission products without

Table 1. Consequences of Reactor Accidents for Various Probabilities for One Reactor

Chance per Reactor-Year	Early Fatalities	Early Illness	Total Property Damage $\$10^9$	Decontamination Area ~Square Miles	Relocation Area Square Miles
One in 20,000*	<1.0	<1.0	<0.1	<0.1	<0.1
One in 1,000,000	<1.0	300	0.9	2000	130
One in 10,000,000	110	3000	3	3200	250
One in 100,000,000	900	14,000	8	-	290
One in 1,000,000,000	3300	45,000	14	-	-

* This is the predicted chance of core melt per reactor year

Table 2. Consequences of Reactor Accidents for Various Probabilities for One Reactor

Chance per Reactor-Year	Latent Cancer** Fatalities (per year)	Thyroid Modules** (per year)	Genetic Effects *** (per year)
One in 20,000*	<1.0	<1.0	<1.0
One in 1,000,000	170	1400	25
One in 10,000,000	460	3500	60
One in 100,000,000	860	6000	110
One in 1,000,000,000	1500	8000	170
Normal Incidence	17,000	8000	8000

* This is the predicted chance of core melt per reactor year.
** This rate would occur approximately in the 10 to 40 year period following a potential accident.
*** This rate would apply to the first generation born after a potential accident. Subsequent generations would experience effects at a lower rate.

retention in the containment, as could occur with an in-vessel steam explosion, or with a hydrogen explosion in the containment, is of much lower probability since these energetic events have much lower probabilities of occurrence.

Tables 1 and 2, extracted from the report WASH-1400, provide the main results of the RSS. Table 1 shows the consequences of early fatalities, early illness, total property damage in billions of dollars, decontamination area in square miles and the relocation area in square miles as a function of probabilities which vary from $5x10^{-5}$ (1 in 20,000) to $1x10^{-9}$ (1 in a billion). The highest probability selected was $5x10^{-5}$, which was the calculated probability of core melt occurring in a LWR/year. Table 2 presents the long term consequences of latent cancer fatalities, thyroid nodule cancers and genetic effects in the population close to the PWR and the BWR plants considered in the RSS.

What is immediately clear is that the threshold for the public hazard is the occurrence of a core melt accident in a LWR. There is no hazard to the public if such an accident does not occur, since no fission products are released until there is a core heat up and clad failure. The probability of core melt occurring is quite small. Certainly, an individual reactor would be decommissioned long before, and that is the hope and prayer of each of the owners of the LWR nuclear plants. Another conclusion from these two tables is that the early consequences are indeed very small. Only when the probability values are very small (10^{-7} to 10^{-9}) that, the early fatalities and illness values can be called significant. The longer term effects appear to be significant at the levels of probability equal to 10^{-6}. However, here the latent cancer fatalities due to the postulated core meltdown accident in a LWR are competing with the cancer fatalities caused by cigarette smoking and the environmental hazards that a public regularly, and by their own volition, accepts. In fact the latent cancer incidence of 170 is less than the statistical uncertainty in the normal cancer fatalities/year.

The thyroid nodules are generally not associated with the other environmental hazards. Thyroid is caused by the deposition in the thyroid of a person of the radioactive Iodine released as a fission product in the core-melt accident. Children are more susceptible to the thyroid malignancy. This is the reason for the distribution of iodine tablets to the population around a nuclear plant, so that the thyroids are already saturated with non-radioactive iodine. The genetic effects supposedly are caused by mutations in the

Table 3. Incidence Per Year of Latent Health Effects Following a Potential Reactor Accident

Health Effect (per year)	Chance per Reactor per year		Normal** Incidence Rate in Exposed Population (per year)
	One in 20,000*	One in 1,000,000*	
Latent Cancers	<1	170	17,000
Thyroid Illness	<1	1400	8000
Genetic Effects	<1	25	8000

* The rates due to reactor accidents are temporary and would decrease with time. The bulk of the cancers and thyroid modules would occur over a few decades and the genetic effects would be significantly reduced in five generations.

** This is the normal incidence that would be expected for a population of 10,000,000 people who might receive some exposure in a very large accident over the time period that the potential reactor accident effects might occur.

Table 4. Consequences of Reactor Accidents for Various Probabilities for 100 Reactors

Chance per Reactor-Year	Latent Cancer** Fatalities (per year)	Thyroid Modules** (per year)	Genetic Effects*** (per year)
One in 200*	<1.0	<1.0	<1.0
One in 10,000	170	1400	25
One in 100,000	460	3500	60
One in 1,000,000	860	6000	110
One in 10,000,000	1500	8000	170
Normal Incidence	17,000	8000	8000

* This is the predicted chance per year of core melt for 100 reactors.

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

*** This rate would apply to the first generation born after the accident. Subsequent generations would experience effects at a decreasing rate.

Table 5. Average Risk of Fatality by Various Causes

Accident Type	Total Number	Individual Chance per Year
Motor Vehicle	55,791	1 in 4,000
Falls	17,827	1 in 10,000
Fires and Hot Substances	7,451	1 in 25,000
Drowning	6,181	1 in 30,000
Firearms	2,309	1 in 100,000
Air Travel	1,778	1 in 100,000
Falling objects	1,271	1 in 160,000
Electrocution	1,148	1 in 160,000
Lightening	160	1 in 2,000,000
Tornadoes	91	1 in 2,500,000
Hurricanes	93	1 in 2,500,000
All Accidents	111,992	1 in 1,600
Nuclear Reactor Accidents (100 plants)	-	1 in 5,000,000,000

Table 6. Average Probability of Major-man-caused and Natural Events

Type of Event	Probability of 100 or More Fatalities	Probability of 1000 or More Fatalities
Man-Caused		
Airplane Crash	1 in 2 years	1 in 2000 years
Fire	1 in 7 years	1 in 200 years
Explosion	1 in 16 years	1 in 120 years
Toxic Gas	1 in 100 years	1 in 1000 years
Natural		
Tornado	1 in 5 years	Very small
Hurricane	1 in 5 years	1 in 25 years
Earthquake	1 in 20 years	1 in 50 years
Meteorite Impact	1 in 100,000 years	1 in 1,000,000 years
Reactors		
100 plants	1 in 100,000 years	1 in 1,000,000 years

cells in the body. Again the numbers are too small to be statistically significant, since such effects are also caused by other environmental substances e.g. chemicals or even airplane rides. This is illustrated in Table 3, also extracted from WASH-1400, in which comparison is made for the latent (or long-term) health effects caused by the core melt accident in one reactor against those that occur normally in the population that was exposed to the core-melt accident. It is seen that even for the very low probability of 10^{-6}, i.e. a rather severe accident in which containment failure did take place and a large fission product release occurred, still the latent health effects are of the order of 1/10th of those of normal incidence.

The consequence - probability estimates derived by the authors of WASH-1400 for one reactor can be extrapolated to a population of reactors in a country or the World. Table 4 shows such an extrapolation for 100 reactors, which is approximately the current population of the LWR power plants in the USA. The calculated incidence of one core-melt accident in such a population is 1 in 200 years or $5x10^{-3}$/ year. The consequences remain the same i.e., insignificant to statistically insignificant.

Interesting data are shown in Table 5, also obtained from WASH-1400, which shows the average risk of fatality by various man-caused and nature-caused events per year. These statistics are for USA in late 1960's and early 70s. The highest number of fatalities are self and man-caused by the operation of motor vehicles, followed by falls, fires, drowning, firearms, air travel etc. It is seen that the most risky enterprise that we engage in, is that of operation, and being in the vicinity, of motor vehicles. The numbers have improved since late 1960s and early 1970s because of the many improvements (airbags, seatbelts, etc) in modern cars, but the traffic has worsened and speeds have increased.

The nature-caused events are also listed in Table 5, which include lightning, tornadoes and hurricanes. The fatalities caused by hurricanes and tornadoes shown in this Table may be less than what they are in more recent years. The probabilities for an individual suffering a fatal nature-caused accident may, also, have increased recently as these events have been of greater strength lately. The probability of an individual suffering a nuclear-accident-caused fatality has been estimated to be 1 in $5x10^{9}$ or $2x10^{-10}$/ year, which is really insignificant.

Another comparison of man-caused, nature-caused and one reactor severe accident in a population of 100 reactors is shown in Table 6, which shows probabilities for large consequence accidents, which may result in fatalities of 100 or 1000 persons. It is seen that the most frequent cause is the airplane crash for a 100 fatality accident and a hurricane for a 1000 fatality accident. The recent Tsunami in which 200,000 persons died or the earthquake in Pakistan-India in which more than 50,000 persons died are, basically, very high consequence unique events. The probability of 100 fatalities occurring in a nuclear accident for a country with

100 nuclear plants was estimated by WASH-1400 to be 1 in 100,000 years and for 1000 fatalities occurring the probability was estimated to be 1 in a million years.

The most famous and most quoted results from WASH-1400 are shown in the Figs. 8, 9 and 10 which compare, in turn, the fatalities and the property damage caused by a nuclear accident in USA, with its 100 nuclear plants, against the man-caused and nature-caused events. Clearly, the probabilities at any consequence level for the 100 nuclear plants are many orders of magnitude smaller than those for the other man-caused or for the nature-caused events. The close comparison of the public risk from the 100 nuclear plants to the nature-caused event of a meteorite hitting the earth is apt but it was ridiculed by some of the vocal critics of WASH-1400.

The other significant results from the WASH-1400 were a comparison of the probabilities for the various consequences for a PWR vs. those for a BWR. It was found that the risks were quite the same for those two types of LWRs. An example is shown in Fig. 11 for the consequence of early fatalities/year from a severe accident in either reactor.

A startling finding of the RSS was that the operator errors could be a significant contribution to the probability of a core-melt accident occurring. There was no quantification of this contribution; however, it was clear that in complex events, operator actions could aggravate the situation, which could progress into causing damage to the core. A case in particular, was of the small break LOCA, which may continue for 1 to 2 hours during which wrong actions of the operator could result in a core-melt accident. This is exactly what happened during the TMI-2 accident. It must be noted here that large scale two phase phenomena were not as well known at that time and there were surprises, which were later understood and recognised.

WASH-1400 received an exhaustive review from a diverse group of scientists, including a panel set up by the USNRC under the leadership of Prof. Lewis [16]. The reviewers liked the methodology employed but questioned the estimation of the uncertainties and the final values for the

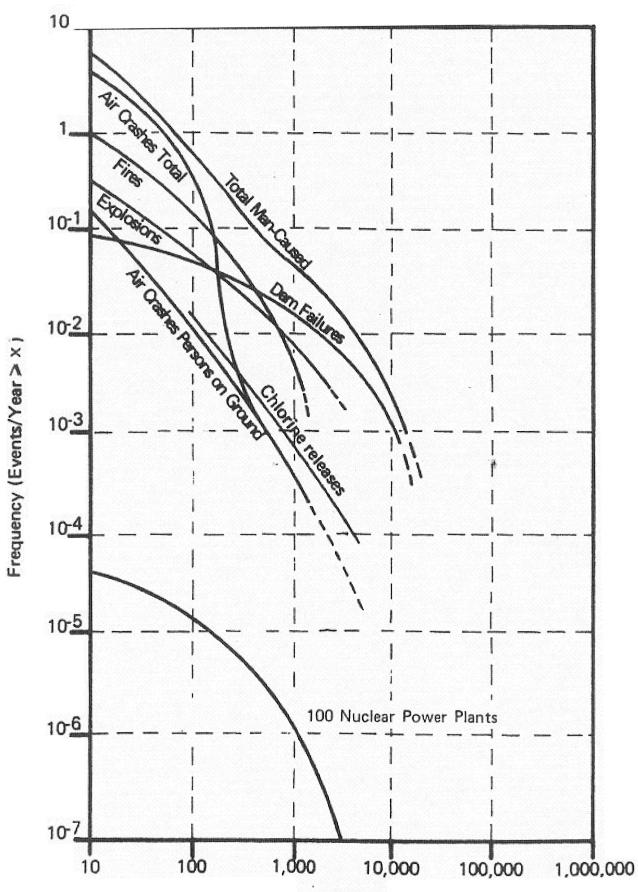


Fig. 8. Frequency of Fatalities due to Man-Caused Events

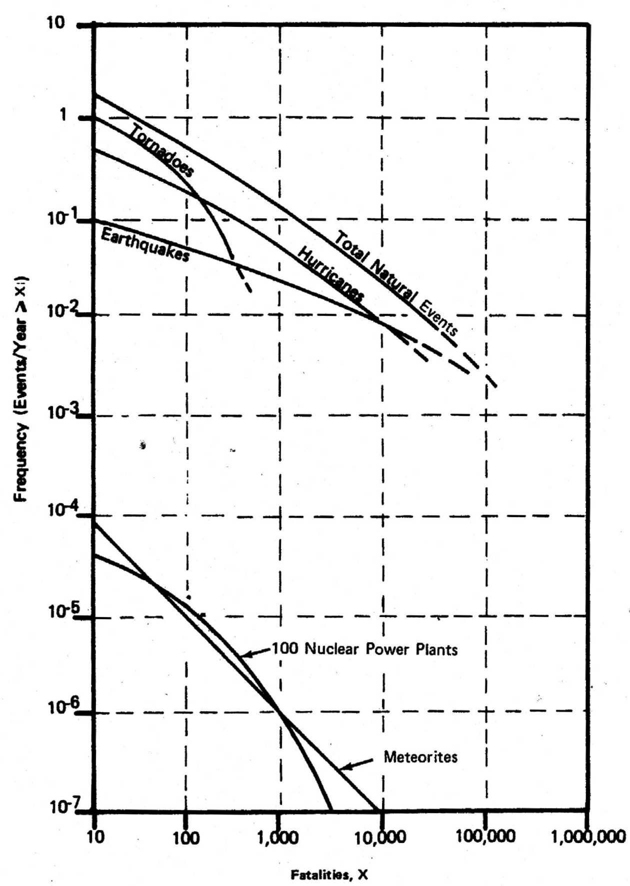


Fig. 9. Frequency of Fatalities due to Natural Events

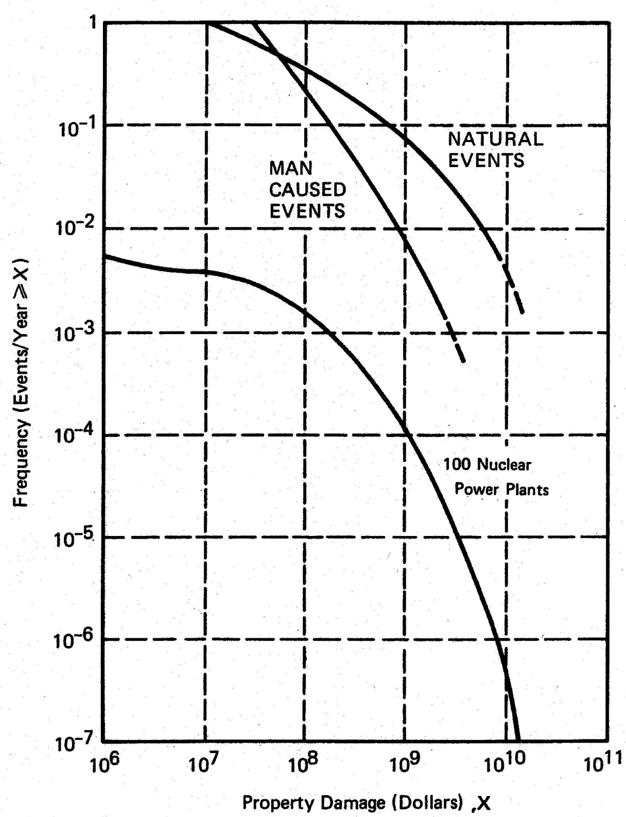


Fig. 10. Frequency of Property Damage due to Natural and Man-Caused Event

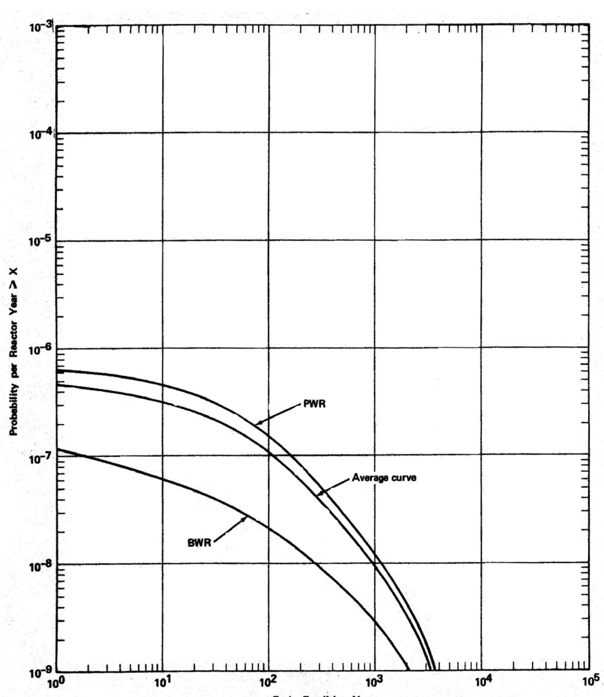


Fig. 11. Probability Distribution for Early Fatalities per Reactor Year

core-damage frequency. No major discrepancies, however, were discovered. Germany followed the same procedures as in WASH-1400 for their plants and published the German Risk study in 1980 [17].

The WASH-1400 prediction of the risk to the public from a population of say 100 nuclear plants to be so much less than any other risk that the public faces, was a great vindication for supporters of nuclear power. The order stream for new nuclear plants continued to grow and actually was overwhelming the capacity of the vendors. These were the years of optimism. It was still not clear whether the cost of nuclear electricity will be less than that from coal-fired plants, since the design criteria, the strict quality control and its documentation and the submission of safety analysis report and other documents was adding tremendously to the capital costs of LWRs. In addition, the court challenges by interveners were delaying the construction of several nuclear plants in USA which were also adding very significant sums to their capital costs.

9. THE ACCIDENTS

9.1 The TMI-2 Accident

The core melt accident in the Three Mile Island – 2 (TMI-2) reactor near Harrisburg Pennsylvania occurred in March 1979, i.e. less than four years after the publication of the RSS (WASH-1400). This accident was entirely unexpected and it was a shock to the nuclear establishments all over the world. The detailed results of the WASH-1400 were not known to a large part of the nuclear community and suddenly there was a general realization that we missed something vital in our perceptions. That a core can melt and melt so fast was never in our thought process. For a number of years, it was thought that, perhaps, only a small part of the core was damaged. Only after the removal of the upper internals, it became clear that, at least, half of the core had melted. Later, it was found that some (20 tonnes) of the melted core had reached the lower head. If the operators had not filled the TMI-2 vessel with water, or if a much larger quantity of melt had dropped into the lower head, it is not clear whether the lower head would have survived and that all the melt would be retained in the vessel. Release of melt to the containment, and the possible melt-concrete interaction, would have created much greater uncertainty and untold additional issues in 1979.

The accident started with the loss of feed water to the steam generators, which resulted in the dry-out of the secondary side of the steam generators within 10-15 minutes. The dry-out of the secondary side stopped the heat removal from the core and the reactor pressure started increasing. Meanwhile, this fault automatically tripped the turbine and scrammed the reactor. The reactor vessel pressure increase opened the pilot-operated relief valve (PORV), as it should have, to decrease the pressure in the vessel. As the pressure in the vessel decreased, the PORV should have closed,

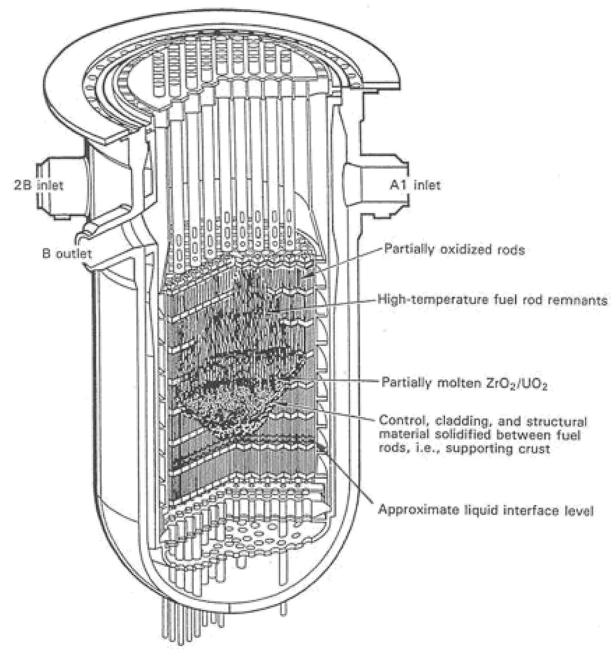


Fig. 12. Hypothesized TMI-2 Core Damage Configuration at 173 Minutes after Scram

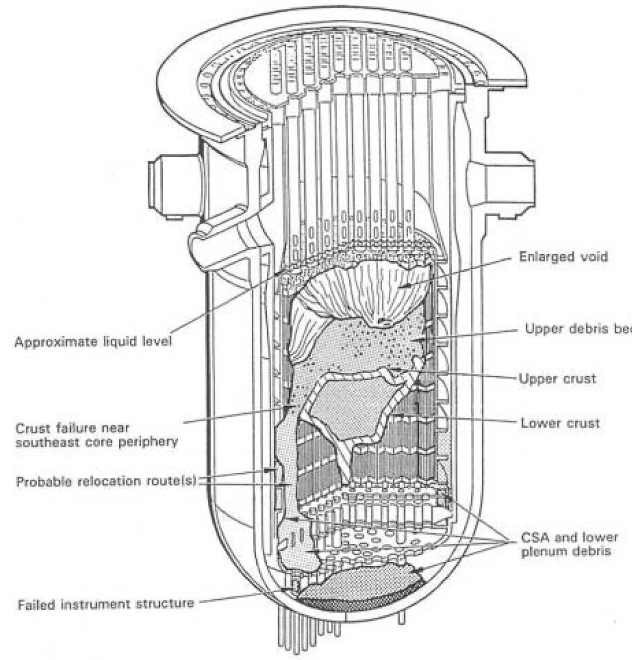


Fig. 13. Hypothesized TMI-2 Core Damage Configuration at 226 Minutes after Scram

however, it did not, and the coolant kept discharging from the vessel and the vessel pressure started decreasing. This led to the start of the high pressure ECCS, as it should, and some water started to be added to the vessel. Meanwhile, the pressure decrease in the vessel created much steam, a phase separation and the formation of a steam bubble. The water in the pressurizer, which could have come to the vessel, was blocked due to the steam bubble or by the flow of steam (CCFL phenomenon) and the pressurizer indicated full. The operators reacted too slowly to the fact that PORV was open, which they closed after a considerable amount of water had been lost from the vessel. Another error made by the operators was that they closed the ECCS injection to the core, following their instructions in case of the indication of a full pressurizer. They also stopped the pumps since they had started cavitating due to the passage of steam along with water in the primary system.

The above equipment failure, coupled with faulty operator actions resulted in loss of much water from the vessel, no water addition to the vessel, boil-off of water in the vessel and finally the uncovering of the core at approximately 130 minutes after the first malfunction. No primary feed water was being added, since the pumps were stopped and the ECCS was shut off; the continuing boil-off resulted in core uncovering almost completely. The decay heat and the absence of heat removal raised the clad temperature

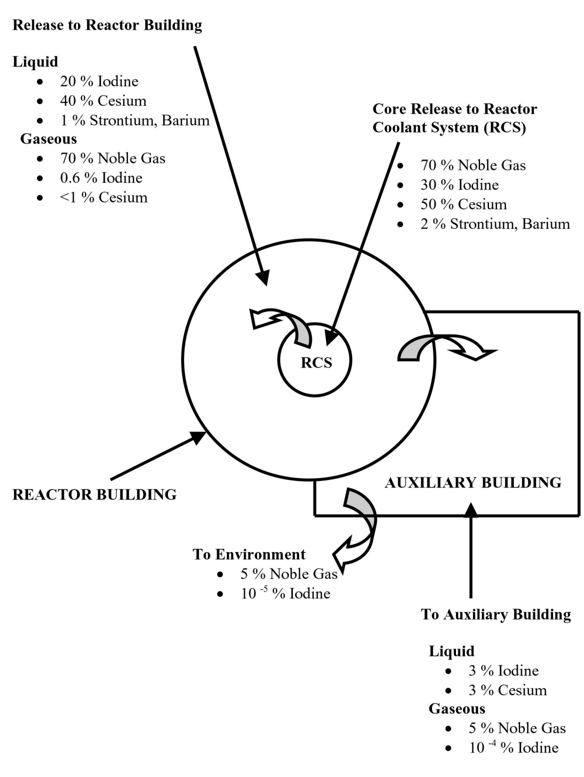


Fig. 14. Release Fractions at TMI-2

to values where the exothermic steam – Zircaloy reaction started. From then on, there was no turning back from a core-melt scenario! About 50% of core was melted and flowed down to form blockages near the bottom; see Figures 12 and 13.

Luckily, the operators started the cavitating pumps back on and filled the vessel with water. This action stopped further melting of the fuel elements but it fragmented much of the already molten fuel in the core region to form a debris bed, which could not be fully cooled, even when submerged in water. The fragmented debris in the core region reheated and about 20 tonnes of the molten material broke through the crust at the side and flowed down to the lower head, while ablating the core cylinder. The water present in the lower head and in other parts of the vessel, finally quenched the melt; although it took a few hours. The vessel survived and the next day, the pumps were stopped. The natural circulation flow between the vessel and the steam generators was sufficient to remove the decay heat from the core.

The Zircaloy-steam exothermic reaction produced hydrogen which was released to the containment through the open PORV. It accumulated in the containment and several hours after the start of the accident burned; producing a pressure spike of 2 bars in the containment. The containment, designed for the pressure rating of 5 bars, had no problem with the hydrogen burn and did not sustain any damage.

The volatile fission products released during the core heat up, melting of cladding and melting of the fuel also accumulated in the containment and radioactivity was detected in the containment. At that time a door was open from the containment to the auxiliary building and some fission products were released to the auxiliary building, before the door was closed by the operators. Despite the fact that the auxiliary building was not built like a leak-tight containment only ~ 0.01 % of fission products escaped from it to the environment. In total, less that 10^{-5} % of ^{131}I inventory of the core was released to the environment. During the first 16 hours, after the accident, only approximately 10 Ci was released to the environment and ~70 curies of I was released over the next 30 days (see Fig. 14). Radioactive material found within the exclusion area surrounding the reactor included ~0.5 Ci of ^{137}Cs and ~0.1 Ci of ^{90}Sr. In this context, it is perhaps, instructive to know that the inventories of ^{131}I, ^{137}Cs and ^{90}Sr in a prototypic LWR core, near the end of a cycle, are ~91, 5 and 4 million curies. The TMI-2 core was only 90 days old and its inventory of fission products would have been somewhat less than these values.

The TMI-2 accident did not cause any injuries or deaths or property damage. It also did not release sufficient fission products to contaminate the soil around, except perhaps the exclusion area slightly. It, however, caused serious psychological harm on the whole USA and, in particular, on the population of Harrisburg and Pennsylvania state. The major cause was the duration of the perceived threat: for almost one week, there were news that there may be a hydrogen bubble inside the vessel, which could explode, fail the vessel and the containment resulting in a devastating release of radioactivity at any moment. This was faulty science since a hydrogen explosion could not occur due to the lack of oxygen. But it was not contradicted vigorously by the USNRC. This lead to panic-driven evacuation from a very large area around the plant, even though the authorities never called for an evacuation. This publicity, the interviews with so many so called 'experts' and the many 'what if' projections soured the public completely on nuclear power. It is only very recently that the public view of nuclear power is changing.

9.2 The Aftermath of TMI-2 Accident

The occurrence of the TMI-2 accident, in spite of the very small public physical damage, was taken very seriously by the authorities and the nuclear industry. The President appointed a special commission to enquire into the causes and the circumstances of the accident. They [18] identified eighteen faults and errors: five in design, two in regulation and eleven in operation. An error identified for the USNRC was their failure to inform TMI-2 plant personnel of a similar event that happened earlier at the Davis-Bessie plant (designed and built by the same vendor) which was successfully terminated. The commission also faulted the USNRC for not making the effort to admit that it had made an error about the possibility of a hydrogen explosion occurring and informing the public forcefully about it.

The TMI-2 accident was a wake-up call for the whole nuclear enterprise in USA. It was realized that this accident was unlike the design-base accidents but more like the accidents postulated in WASH-1400. The equipment errors (valve malfunctioning) and operator errors that initiated it and the circumstances of the accident called for much better equipment testing, control room instrumentation, operating procedures, lines of authority, lines of communication, technical support to the operator, emergency planning, public evacuation etc. It identified that non-technical aspects e.g. operator training, emergency procedures, organization, management, etc. are as important as the technical aspects, e.g. equipment design, construction, equipment qualification and safety analysis. The industry responded to these new challenges immediately by forming a new organization named INPO and the industry research arm EPRI started the Industry Degraded Core (IDCOR) research program. The focus of safety research was put on the beyond design base accidents. Further research was initiated both by USNRC and EPRI. The USNRC worked with the utility industry to require TMI-2 back fits for the plants. These back fits included hydrogen control measures, since the TMI-2 containment was subjected to a hydrogen burn generating a 2 bar pressure spike during the accident and containments of several plants had either lower design pressure or smaller volume than the TMI-2 containment. Hydrogen combustion research was initiated to test igniter systems for control of hydrogen in the containment.

Several good results observed from the TMI-2 accident were also noted. These included the performance of the leak tight containment, which did not suffer any damage when hydrogen burned. The heat sink was established with natural circulation flow between the core and the steam generators and a safe stable state was reached without failure of both the vessel and the containment.

Another good result from the TMI-2 accident was the absence of significant airborne iodine and cesium radioactivity in the TMI-2 containment. Most of the radioactivity was found to be in the sump water. Analysis activity [19] initiated soon after the accident pointed out that almost all of the iodine and cesium fission products released in the accident, converted to the highly-water-soluble compounds, CsI and CS OH, which were removed from the containment atmosphere either by the water spray activated, or by the aerosol agglomeration and deposition on the floors and the walls of the containment and eventually transported to the sump. This reduced, by orders of magnitude, release that could occur through the containment leakage to the environment.

The LWR safety research became the LWR Severe Accident Safety Research, starting from 1980, even though the LOCA experiments and research did not terminate. But the whole focus of the LWR safety research shifted to the beyond the design-base accidents. We shall address this topic after we describe the other major accident which affected the history of nuclear power safety.

9.3 The Chernobyl Accident

A core melt accident happened in one of the four RBMK reactors situated in a complex called the Chernobyl nuclear power plant (NPP) in Ukraine, Soviet Union. The RBMK are water- cooled channel type reactors in which the water boils as in a BWR, however they are moderated by graphite. The core configuration is that of a large graphite block in which about 2000 channels are drilled each of which contains a pressure tube and a large fraction of them contains a fuel bundle through which the cooling water flows from the core bottom to the top. The channels are connected at the bottom through several headers, to water inlet. The channels at the top are connected to a multitude of pipes which bring the steam (two phase mixture) to the steam drum, from which the separated steam is taken to the turbine and the water flows back to the inlet piping at the bottom of the channels. Fresh feed water is admitted to the steam drum. A picture of the RBMK configuration is shown in Figures 15 and 16. The RBMK core is physically much larger than that of a LWR since it is moderated by graphite, which has a much larger diffusion and slowing down length for neutrons. The core is fuelled, at power, by a fuel machine resting on the shielding above the core. The fuel machine takes a bundle out and replaces it, in general, with a fresh fuel bundle. This it does almost every day.

Besides the differences of moderator between the RBMK and a PWR or a BWR and that of multiple pressure tubes (as in a CANDU reactor), instead of a pressure vessel, there is a major difference (deficiency) in the RBMK that it has a containment only on the bottom part of the core and on the piping underneath the core; see Fig. 17. The outlet piping and the top of the shielded core are enclosed in a confinement building which is not pressure-bearing or leak tight. This building is accessible to the plant personnel, even when the reactor is operating. A water pool is situated below the containment at the bottom of the core to serve as a condenser pool for any steam release from a break in the inlet piping under the reactor core.

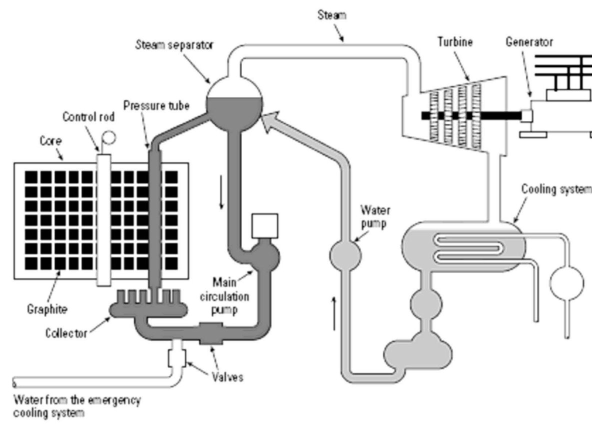


Fig. 15. Schematic Diagram of the RBMK-1000

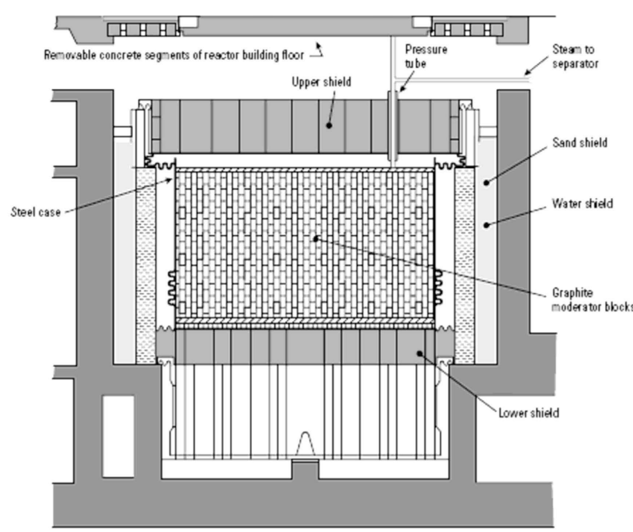


Fig. 16. Cross Sectional View of Reactor Vault

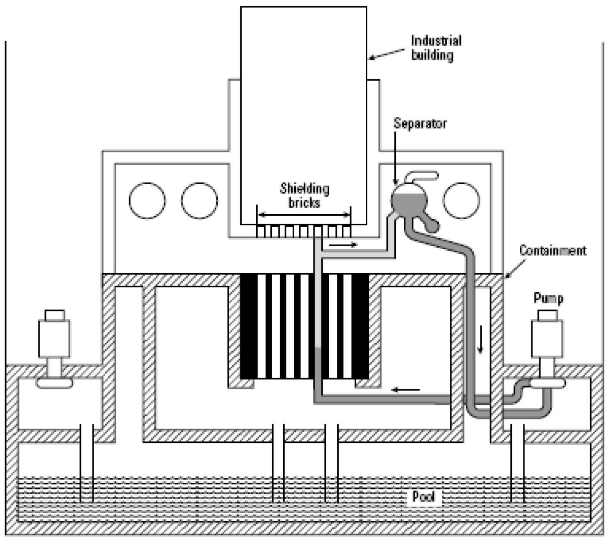


Fig. 17. Chernobyl Containment

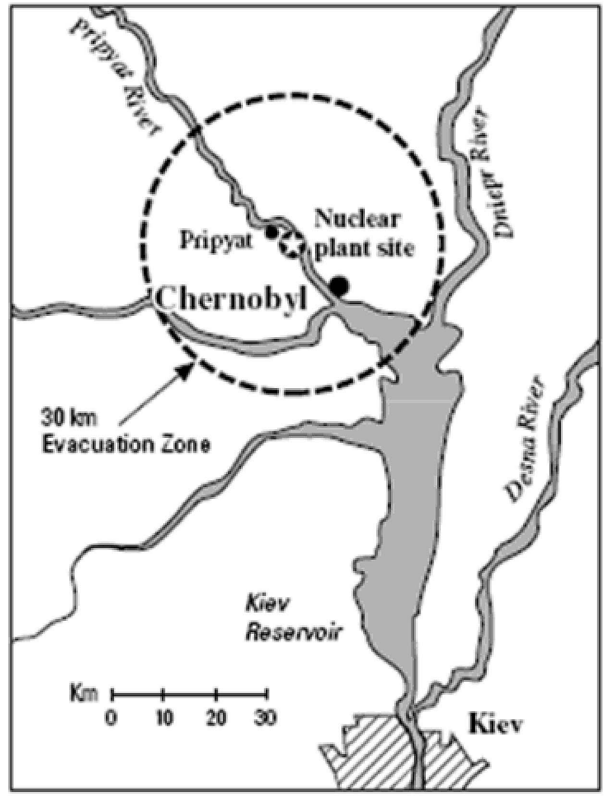


Fig. 18. Area Nearby the Chernobyl Reactor Site

The reactor is controlled by the control rods that enter the core from the top. These control rods are different from those in a LWR, since they are follower type rods in which a graphite region is inserted in the core followed by an absorber. This, as we shall see, acted badly during the accident.

The RBMK is also different from a LWR in two other very important aspects viz., its reactivity feed back characteristics and its stability behavior at low power level(s). The LWR being optimally moderated by the water loses reactivity as the water density decreases e.g. by boiling of the water coolant. Thus a BWR, for example, will experience a large negative reactivity, if for some reason (e.g. rise in power, depressurization, etc.) the void fraction (quality) increases. This will decrease the power level and shut the fission reaction down even if the control rods are not inserted. The RBMK, on the other hand, being optimally moderated by graphite will have a positive reactivity feed back, if the water density decreases from that in regular operation, since that reduces the absorption in the coolant. This behavior: positive void coefficient and the rather unstable operation at low power required the reactor operation to be limited by certain constraints, the principle ones being that:

- the reactor should be operated only when a certain number of control rods are in the core.
- operation of power levels below 20% of full power should be avoided.

The former is required, since the control rods do not have to travel as much to insert the absorber in the core. The latter is to avoid the large instabilities at low power levels which are hard to control and adjust. The operation of an RBMK without the presence of the required number of control rods in the core was strictly forbidden.

Fig. 19. Chernobyl Reactor Location

It should be remarked at the outset that the Chernobyl accident was a reactivity increase accident (RIA) rather than a heat removal degradation accident as was the TMI-2. The accident happened on the night of April 26, 1986.

Most of the information including the figures in the following paragraphs is extracted from a Canadian report [20] which was published in Sept. 1987 after the post-accident briefing that the Soviet scientists conducted at the IAEA offices in Vienna. We believe this is the most cogent description of how and why the Chernobyl accident happened.

The four reactors of Chernobyl are located near the small town of Chernobyl about 105 km north of Kiev in Ukraine. The nearest city is Kiev and the plant personnel lived in the specially constructed town of Pripyat, 3 km from the NPP with a population of 45,000. Pripyat River flows next to the town of Pripyat on its way to the reservoir near Kiev. Fig. 18 shows the geography of the area around the NPP. Fig. 19 shows the location of Chernobyl with respect to the neighboring countries. Normally such a figure will not be required for the description of a nuclear accident, since all studies for the atmospheric transport of radioactive releases from a LWR containment had predicted that fission products would not be found beyond a 20-30 km zone around the NPP. In the case of the Chernobyl accident, however, radioactivity was first detected at the Forsmark plant on the east coast of Sweden at a distance of more than a 1,000 km from Chernobyl. Forsmark plant personnel initially thought that something had gone wrong with their plant but soon confirmed that the radioactivity found came from somewhere in the East. The source was confirmed only after 2-3 days when the Soviet authorities announced the accident in the Chernobyl reactor no. 4.

The following description of the accident is taken verbatim from the Canadian Report [20].

9.4 How and Why It Happened

9.4.1 A Test for Safety Sets It Off

It is one of history's ironies that the worst nuclear accident in the world began as a test to improve safety. The events of April 26 started as an experiment to see how long a spinning turbine could provide electrical power to certain systems in the plant. The reason for the test? Well, the Soviets, in common with most of the rest of the world, design their reactors not only to withstand an accident, but also to cope simultaneously with a loss of electric power. This may seem a little strange—to run out of power at a generating station—but in an accident the reactor is shut down right away, so can't generate its own power directly. It would normally get power from the electrical supply *to* the station or from the other reactors at the same site. To ensure an extra layer of defence, it is considered that there is a possibility that these sources have also failed. The normal backup is to provide diesel engines at the site to drive emergency generators, just as hospitals do in case of a power failure. These diesels usually start up in 30 seconds, and in Western plants this is a short enough interruption to keep important systems going. For the Chernobyl reactor, the Soviets felt this was *not* short enough, and they had to have almost an uninterrupted supply. Now even with the reactor shut down, the spinning turbine is so heavy, it takes a while to slow down, and the Soviets decided to tap the energy of the spinning turbine to generate electricity for the few seconds before their diesels started. The experiment was to see how long this electricity would power the main pumps which keep the cooling water flowing over the fuel.

The test had been done before on unit no. 3, with no particular ill-effects on the reactor. However, the electrical voltage had fallen off too quickly, so that the test was to be redone on unit no. 4 with improved electrical equipment. The idea was to reduce reactor power to less than half of its normal output, so all the steam could be put into one turbine; this remaining turbine was then to be disconnected, and its spinning energy used to run the main pumps for a short while. At the meeting in Vienna the Soviets were at some pains to point out that the atmosphere was not conducive to the operators performing a cautious test:

- The test was scheduled to be done just before a planned reactor shutdown for routine maintenance. If the test could not be done successfully **this** time, then the people would have to wait another year for the next shutdown. Thus, they felt under pressure to complete the test this time.
- Chernobyl unit #4 was a model plant – of all the RBMK-1000 type plants, it ran the best. Its operators felt they were an elite crew and they had become overconfident.
- The test was perceived as en electrical test only, and had been done uneventfully before. Thus, the operators did not think carefully enough about the effects on the reactor. There is some suggestion that in fact the test was being supervised by representatives of the turbine manufacturer instead of the normal operators.

9.4.2 How the Trap Was Set

The accident really began 24 hours earlier, since the mistakes made then slowly set the scene that culminated in the explosion on April 26. The 'Event Sequence', attached shows a summary of all the things the operators did and how the plant responded; here we describe the key events.

At 1 a.m. on April 25, the reactor was at full power, operating normally with steam going to both turbines. Permission was given to start reducing power for the test, and this was done slowly, with the reactor reaching 50 % power twelve hours later at 1:05 in the afternoon. At this point only one of the two turbines was needed to take the steam from the reactor, and the second turbine was switched off.

Normally, the test would then have proceeded, with the next step being to reduce power still further to about 30%. However, the people in charge of the distribution of electricity in the USSR refused to allow this, as apparently the electricity was needed, so the reactor stayed at 50% power for another 9 hours. At 11:10 p.m. on April 25, the Chernobyl staff got permission to continue with the power reduction. Unfortunately, the operator made a mistake, and instead of holding power at about 30%, he forgot to reset a controller and the power fell to about 1 % - the reactor was almost shut off. This was too low for the test. Now in all reactors, a sudden power reduction causes a quick build-up of a material called Xenon in the uranium fuel. Xenon is a radioactive gas, but more important it sucks up neutrons like a sponge, and tends to hasten the reactor down the slope to complete shutdown. As well, the core was at such a low power that the water in the pressure tubes was not boiling, as it normally does, but was liquid instead. Liquid water has the same absorbing effect as Xenon. To try to offset these two effects, the operator pulled out almost all the control rods, and managed to struggle back up to about 7% power – still well below the level he was supposed to test at, but as high as he could go because of xenon and water.

It was as if you were trying to drive a car with the

Event Sequence

Time		Event	Comments
April 25	01:00	Reactor at full power Power reduction began	As planned
	13:05	Reactor power 50% All steam switched to one turbine	As planned
	14:00	Reactor power stayed at 50% for 9 hours because of unexpected electrical demand	
April 26	00:28	In continuing the power rundown, the operator made an error which caused the power to drop to 30 MW (th), almost shutting the reactor off.	This caused the core to fill with water & allowed Xenon (a neutron absorber) to build up, making it impossible to reach the planned test power
	01:00 -01:20	The operator managed to raise power to 200 MW (th). He attempted to control the reactor manually, causing fluctuations in flow and temperature.	The RBMK design is unstable with the core filled with water – i.e., small changes in flow or temperature can cause large power changes, and the capability of the emergency shutdown is badly weakened.
	01:20	The operator blocked automatic reactor shutdown first on low water level, then on the loss of both turbines.	He was afraid that a shutdown would abort the test. Repeat tests were planned, if necessary, and he wanted to keep the reactor running to perform these also.
	01:23	The operator tripped the remaining turbine to start the test	
	01:23:40	Power began to rise rapidly. The operator pushed the manual shut downbutton.	The reduction in flow as the voltage dropped caused a large and fast increase in boiling leading to a fast power rise. Too late. The damage was done in the next four seconds. The emergency shutdown would have taken six seconds to be effective
	01:23:44	The reactor power reached about 100 times full power, fuel disintegrated, and excess steam pressure broke the pressure tubes	The pressure in the reactor core blew the top shield off and broke all the remaining pressure tubes.

accelerator floored and the brakes on- it's abnormal and unstable.

Indeed it is a very serious error in this reactor design to try to run with all the control rods out. The main reason is that some of these same rods are used for emergency shutdown, and if they are all pulled out well above the core, it takes too long for them to fall back into the high power part of the reactor in an emergency, and the shutdown is very slow. The Soviets said that their procedures were very emphatic on that point, and that "Not even the Premier of the Soviet Union is authorized to run with less than 30 rods!"

Nevertheless, at the time of the accident, there were probably only 6 to 8 rods in the core. At any rate, the operator had struggled up to 7 % power by 1.a.m. on April 26, by violating the procedure on the control rods. He had other problems as well – all stemming from the fact that the plant was never intended to operate at such a low power. He had to take over manual control of the flow of water returning from the turbine, as the automatic controllers were not operating well at the low power. This is a complex task to do manually, and he never did succeed in getting the flow correct. The reactor was so unstable that it was close to being shutdown by the emergency rods. But since a shutdown would abort the test, the operator disabled a umber of the emergency shutdown signals.

After about half an hour trying to stabilize the reactor, by 1:22 a.m. the operators felt that things were as steady as they were going to be, and decided to start the test. But first they disabled one more signal for automatic shutdown, and this sealed the reactor's fate. Normally the reactor would shut down automatically if the remaining turbine were disconnected, as would occur in the test, but because the staff wanted the chance to *repeat* the test, they disabled this shutdown signal also. The remaining automatic shutdown signal would go off on abnormal power levels, but would not react immediately to the test. The staff were now in the worst possible situation for a rise in power which could not be caught in time by the shutdown systems. And this is what happened.

9.4.3 The Test Begins

At 1:23.04, the turbine was disconnected and its energy fed to 4 of the 8 main pumps. As it slowed down, so did the pumps, and the water in the core, now moving more slowly over the hot fuel, began to boil. Twenty seconds later the power started rising slowly, then faster, and at 1:23:40 an operator pushed the button to drive in the emergency rods and shut down the reactor. We do not know for sure why de did it – the individual was one of the early casualties – but likely he saw either the power begin to rise or the control rods start to move too slowly in to overcome the power rise. But it was too late. The shutdown rods were so far removed from the core they would have taken six seconds to begin to shut the reactor down. Actually, the insertion of the graphite region of the control rods into the core added more reactivity initially, since it displaced water in the channel. Within four seconds, the power had risen to perhaps 100 times full power and had destroyed the reactor.

The destruction of the whole reactor and the release of the radioactivity to the environment was exacerbated by a common-cause fault in the Chernobyl design: the concrete lid on top of the reactor is lifted by the pressure due to steam entry from the failure of a small number of pressure tubes. In this accident with the large power increase the fuel disintegrated in small particles, failed the clad, mixed with water to generate steam very rapidly (explosively) which failed a number of pressure tubes, which in turn lifted the concrete lid, breaking all 1660 exit pipes, making the steam, the fission products and fuel fragments available for release. The pressure and energy generated were sufficient to hurl the concrete lid, topple the fuel machine, blow the roof off the confinement building and form a high pressure plume which rose to heights where the winds transported the radioactive products to the neighboring countries. The deposition pattern was dictated by the weather patterns over the neighboring countries. Most of the radioactivity deposited in Ukraine, Belarus and Russia, however, radio-

Fig. 20. Excellent Photo of Chernobyl 4 and 3, Right After the Explosion

active particles were detected in many neighboring European countries and even in countries far away from Chernobyl e.g. Japan.

The heavy land contamination, which persists to this day occurs in Ukraine and Belarus. Much cleanup was performed by thousands of liquidators, who came from the scientific, civilian and military infrastructures of Soviet Union. The number of immediate deaths is quite small [31] for such a disastrous accident. It is possible that there may be shortening of life for part of the liquidator population which absorbed substantial doses during the clean-up process of the site and environs. A recent UN report has stated that the effects of radioactivity released on the public health and the environment are not as large as were predicted earlier.

The accident was terminated by adding with helicopter almost 5000 tons of sand, clay and boron bearing material on top of the core region. It is not clear whether this was the best thing to do, since it acted as a heat shield: inside the graphite burned for at least 7 days during which time all the remaining fuel bundles melted releasing volatile fission products to the environment. The molten material spread through the space under the reactor and flowed to the basement; see Fig. 20 to 23. The water in the pool had been emptied to prevent any steam explosion at the cost of the lives of the two volunteers from the NPP staff. A sarcophagus was built on top of the destroyed reactor with haste. This building is still standing but it is not in good shape.

The G-7 countries are currently funding the construction of another confinement (sarcaphogus) building on top of the present one. The material inside is in the form of very fine particles (dust) still containing much radioactivity. The collapse of the roof of the present sarcophagus could generate a dust cloud which could again contaminate the areas surrounding the NPP.

Chernobyl accident was the worst blow to nuclear power. The confidence of the public already shaken by the TMI-2 accident was lost to the nuclear power. This was also further eroded by the many so called scientists who greatly exaggerated the long term health and genetic effects of radiation.

The Chernobyl accident cannot be considered as a LWR accident, since the reactors are so different and the Chernobyl reactor, basically, had some design flaws and no containment (where needed). The public, however, does not know the difference between a RBMK and a LWR. To them our assertion that a nuclear power plant cannot blow up like a bomb was found to be, at least, wanting if not wrong. The magnitude of fission products released during the Chernobyl accident exceeded those released in the Hiroshima bomb and their transport to many neighboring countries and their effect on the life style of several sets of populations was destructive to the safety case of nuclear power as was expounded with the results of the WASH-1400. The costs of clean up in the evacuation area around Chernobyl is estimated to be 7 billion rubles

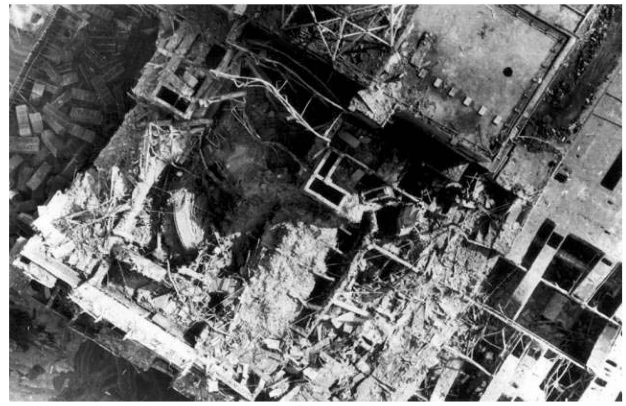


Fig. 21. Looking into Debris-filled Reactor (look for the inverted lid on the left)

Fig. 22. The Photo is Made from the Helicopter on May 3, 1986. The Smoke May Be From the Graphite Fire

(at that time ruble was more expensive than a USD) and these costs were with the use of personnel from the Soviet Army and nuclear laboratories. Chernobyl accident exacted

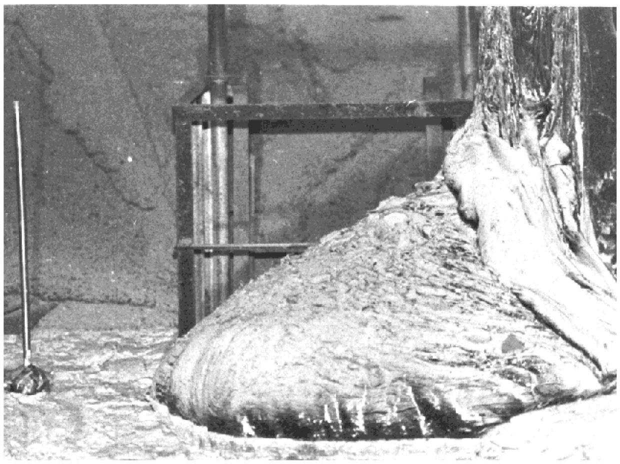


Fig. 23. The "Elephant's Foot". Once Molten Fuel/debris Mixture that Dripped Down Through the Floors of the Damaged RBMK-1000 Reactor at Chernobyl

very large economic, social, psychological and political costs from the Soviet Union and the nuclear enterprises all over the world. It stopped many nuclear power programs.

Recently a UN report stated that the effects of Chernobyl on the populations around have not been as bad as they were originally envisioned. Many evacuees have returned to Pripyat and many of the liquidators are in reasonably good health 20 year after the accident. In spite of these benign news, the effects of Chernobyl cannot be belittled.

10. THE DIFFICULT YEARS

The TMI-2 accident and later the Chernobyl accident created the difficult years for the nuclear industry. The 'nuclear opposition' in USA, which was already quite active before 1979 gathered tremendous force and opposed the completion of the plants under construction and ordering of any new plants. Their tactic of delaying the construction and completion through legal challenges increased the capital costs of the plants. The plants were also subjected to inspections by INPO and the post TMI-2 requirements that USNRC placed on the plants. The latter included, e.g. valve function, hydrogen management specific plant improvements, operator training, man-machine interface, instrumentation, safety culture, etc., etc. Very soon, the nuclear electricity, which was hailed as too cheap to meter became too expensive to generate and bring on line. The plants also being new did not operate too well: their capacity factors were in the range of 50 to 70%, which were too low to break even. The utility companies lost money, and did not have sufficient funds to invest in plant improvements. The vendors lost many orders and a number of plants were left incomplete with great losses to the utility companies. Nothing went well for the nuclear industry during those years, which extended till, at least, year 2000.

11. SEVERE ACCIDENT RESEARCH

The wake-up call from the TMI-2 accident to the nuclear industry and the regulatory authorities was the realization that in spite of many years of earnest efforts at prevention of a core-melt accident, such an accident can occur. It was also realized that only a core-melt accident will provide the public hazard of LWRs. The Chernobyl accident provided a vivid demonstration of the hazard of the core-melt accident to the public, if the containment fails. It was clear that such accidents had to be prevented and mitigated; and towards that purpose a knowledge-base on those accidents had to be acquired.

The core melt accidents were initially called degraded core accidents, lately they have been called Severe Accidents. They are also called at times – beyond the design basis accidents (BDBA) and/or the Design Extension accidents (DEA). The most-generally accepted designation these days is severe Accidents (SA).

The knowledge-base about SA was very poor in 1979. Except for the work performed by the WASH-1400 team, which itself was quite preliminary, there was no organised on-going effort anywhere in the West or East. After the TMI-2 accident, immediately, a resolution was made to acquire knowledge about the progression and consequences of severe accidents. This was done not only in USA, but also in European countries and both experimental and analysis development research programs were initiated [21]. A research center: NSAC (nuclear safety analysis center) was formed at Electric Power Research Institute (EPRI), the research arm of the utility industry. The USNRC laboratories geared up for experiments in which fuel elements would be subjected to the kind of heat transfer degradation scenario that TMI-2 experienced. The EPRI effort also included development of the MAAP code [22] for the determination of consequences in various severe accident scenarios for PWRs and BWRs. The objectives of the research were to determine if the present LWR plants were sufficiently safe or they required substantial backfits, both for the prevention of the initiation of severe accidents and to mitigate their consequences if they do start. Simultaneous with the initiative of the research and development, the tools developed by WASH-1400, namely the fault tree and the event tree analyses were formalized into the Probabilistic Safety (Risk) analysis – 1 (PS(R)A-1) and the USNRC required the plants to perform the PRA-1 studies in the so-called individual plant examination (IPE) program to discover any vulnerabilities in plant equipment, instrumentation, procedures, etc., which could lead to a severe accident.

The severe accident research effort in USA was almost as long as that for ECCS, but it was not as extensive. Lately, the programs started by European Union, and the European

National governments have supplemented the US severe accident programs admirably. In particular, the PHEBUS program pursued by CEA, France, with international collaboration has performed core melt experiments on a prototypic rod bundle with (fission products) through a representative primary system and the containment of a PWR.

Clearly, the phenomena involved in a severe accident are extremely complicated, since the main characteristics of the severe accident scenarios are the interactions of the core melt with the reactor structures and water; and the release, transport and deposition of the fission product carrying vapors. The interactions of core melt may lead to (i) ablation of structures, (ii) steam explosions, (iii) vessel failure, (iv) concrete melting and gas generation, (v) spreading/dispersion of heat generating core melt (debris). These phenomena involve the disciplines of thermal hydraulics, high temperature chemistry, high temperature material interactions, aerosol physics, among others. Predictions of the consequences of a severe accident have to be based on experimentation and models whose veracity may be limited by the scale at which the information about phenomenology is derived. Scaling considerations become very important since large scale experiments with prototypic melts are very expensive and very difficult to perform.

The emphasis in severe accident research was placed on the integrity of the containment. That, this is the correct choice follows from the consequences of the TMI-2 and the Chernobyl accidents. The TMI-2 containment was full of fission products released from the core during the core heat-up and melting process but the fission products were retained in the containment. As mentioned earlier the containment by pass due to the open door to the auxiliary building and the leakage over time were the contributors to the very small release of iodine. More recently emphasis has also been placed on the survivability of the reactor vessel and in retaining the melt inside the vessel by flooding the containment with water and cooling the outer surface of the vessel. This has been adopted for the AP-600 [23], AP-1000 and the Korean APR-1400.

The loadings which can fail the containment were identified and they were classified into two groups: (1) which could fail the containment early and (2) which could fail the containment much later. This early versus late distinction arose due to the natural processes that control the concentration of the fission product aerosols in the containment atmosphere. It was found that almost all of the fission product aerosols agglomerate and deposit on the walls and floor of the containment in approximately 4 hours and from there they are transported to the sump. Thus, they are not available for release to environment on the failure of the containment. In this process, the more toxic fission products, i.e., ^{131} I and ^{137} Cs which had formed the highly water-soluble compounds CSI and CSOH are removed from the containment as soon as a spray action is activated. In this context, it should be mentioned that the USNRC requires maintenance of containment integrity in a severe accident of at least 24 hours with a conditional (on the occurrence of a severe accident) probability of 10%.

Hydrogen combustion-detonation, steam explosion, direct containment heating (DCH) and melt attack on the BWR Mark-1 containment liner were identified as the energetic processes which could fail the containment early. The longer-term gas producing molten corium concrete interaction (MCCI), which would pressurize the containment and the lack of coolability were identified as the processes, which could fail the containment later. It should be noted that a release during the later (after 24 hours) failure of the containment may be 4 to 5 orders of magnitude smaller, than that for the early failure of the containment.

The in-vessel accident progression determines the containment loadings of fission products (the source term), the hydrogen and the mass and composition of the melt delivered to the containment. Thus, any meaningful evaluation of the energetic processes, mentioned above, and their loads on the containment requires a good description of in-vessel accident progression and estimates of its products: fission products, hydrogen and melt characteristics. Accurate description of the in-vessel accident progression is also essential for evaluating the success of the accident management strategy of retaining and cooling the melt within the vessel by cooling the external surface of the vessel. This, of course, requires the flooding of the containment prior to the arrival of the melt to the lower head. Thus, in the following paragraphs, we will provide a short review of the phenomenology of in-vessel progression for a PWR and a BWR.

11.1 In-Vessel Progression for a PWR

The TMI-2 accident provides a vivid example of the in-vessel progression for a PWR. This, accident was stopped by the cooling of the 20 tonnes of melt in the lower head and of what remained in the original confines of the core. The morphology of the melt disposition in TMI-2 is what is expected in a PWR, except that in the hypothetical severe accident, the water is not supplied to the vessel. Thus, the melting process in the core would be more prolonged and the outer rows of the core will also melt and contribute to the melt volume that would transfer to the lower head. The melting process will follow the route of candling, blockage at the lower edge of the core, melt pool formation in the core, its break through either on the side or at the bottom to pour into the lower head. We believe that sufficient research results have become available to describe the early part of the in-vessel accident progression quite accurately.

The late phase of the in-vessel accident progression begins with the transfer of core melt from the core region to the lower head. This, most probably would occur as a jet, which may break up or/and impinge on the wall of the bottom head. The issues for PWR have been. (1) the possibility of a sufficiently energetic steam explosion to rupture the lower head or rupture the bolts on the upper head of vessel, producing a missile, which would rupture the containment and (2) the failure of the lower head due to the thermal loads imposed by the melt jet. Both of these issues invoked much research, analysis and evaluation. The lower head was found to be strong enough to withstand a strong explosion [24] and the failure of the vessel upper head and its subsequent flight to the containment and impact-failure of the containment was found to be [25] of very low (10^{-3}) conditional probability. The jet thermal loads were also found to be insufficient to make a hole in the vessel or fail a penetration,

In the absence of the immediate failure of the lower head, the melt interacts with the water in the lower head and most probably will form a debris, which may not be coolable. Most probably the water will be evaporated by the sensible heat delivered from the melt, resulting in a dried out particulate debris bed, which is generating decay heat. This leads to formation of a circulating melt pool with a metal layer on top, which would fail the lower head, not at the bottom but on the side of the lower head.

There are still some issues outstanding on the late phase in-vessel accident progression of PWRs, in particular, on the effects of melt composition and chemistry on the melt pool stratification, which affects the thermal loading on the vessel wall. More research work is anticipated. Besides that, we believe sufficient knowledge has been gained to make reliable predictions about the consequences of the late phase in-vessel progression of the accident in a PWR.

11.2 In-Vessel Accident Progression for a BWR

The in-vessel accident progression for a BWR is not as well known as for a PWR, simply because, there is no data on a BWR like that obtained from TMI-2 for a PWR. It appears that the early melting of the cruciform control rods and accumulation of their melt on the lower core plate may lead to the failure of the lower core plate, particularly in the higher probability dry core scenario. The melt resident in the core may discharge through the bottom of the core into the lower head. The channel boxes on the fuel bundles in a BWR core do not promote core-wide blockages and melt from individual bundles may dribble down to lower head, obviating the issues of melt jet impingement and steam explosion induced failure of the lower head. The melt jets will break up and form a debris.

The quenching of melt debris and its subsequent re-melting are part of the late-in-vessel scenario for a BWR. The lower head of a BWR contains hundreds of in-vessel control rod guide tubes (CRGTs), which could have a small water supply. These tubes may provide some capability for cooling the melt debris and certainly can prolong the late phase in-vessel accident progression. The failure of a BWR lower head would most probably, occur at one or more of the many penetrations.

The BWR in-vessel accident progression may lead to greater hydrogen concentrations than for a PWR due to the presence of much more Zirconium coming from the channel boxes. The BWR also contains much steel. Thus, the composition of the melt pool and of the discharged melt to the containment will include much more metal (stainless steel and Zr). The late phase in-vessel accident progression for a BWR could last longer than for a PWR, since the BWR lower head contains much more water than does the PWR lower head. However, the possibility of the early failure of a penetration could shorten this phase.

The BWR would also form a circulating melt pool with a metal layer on top. This metal layer may be quite thick due to the large mass of the metals in the core and in the lower head of a BWR. The BWR lower head could also be cooled from the outside and the melt retained in the lower head. Such a scheme has been proposed by Siemens/Areva for their BWR design.

11.3 Fission Product Release and Transport During the In-Vessel Accident Progression

The core heat-up and melting release 60 to 80% of the volatile fission products, e.g. Iodine and Cesium isotopes. Tellurium is sequestered by the unoxidized Zirconium in the core and is released when the Zr is oxidized. The volatile fission products form compounds with each other and with the steam carrier. The predominant compounds are CsI and CsOH, which are highly soluble in water. Some small fraction of the Iodine released may be in gaseous form.

The vapour compounds form aerosols as they encounter separators and driers in a BWR. The aerosol transport process from the core to the containment can result in deposition of a sizeable fraction of the released fission products on the surfaces of the primary system piping. This source term is not immediately available in the contain in the fission products heats up the primary system to revaporise the deposited fission product compounds. This late production can be a significant source term in the containment available for release to the environment. Again, if the containment does not fail at least 4 hours after the revaporized fission products reach the containment; there is no significant increase in the environmental or public impact.

11.4 Ex-Vessel Accident Progression

The study of the ex-vessel accident progression of a severe accident is basically on the processes that may fail the containment. Direct containment heating (DCH), hydrogen detonation, steam explosion and melt attack on the BWR Mark-1 containment liner were identified as the loads, which could cause early containment failure. We shall provide very brief discussions on these topics in the following paragraphs:

11.4.1 Direct Containment Heating (DCH)

Failure of the containment by the rapid pressure increase caused by the heating of its atmosphere by the fragmented core melt, as it is released at high pressure on vessel failure is the process. A focussed program for experiments and analysis was performed in USA in 1980s and early 1990s to show that for the Westinghouse reactor containments, the pressure reached will be below the containment failure pressure and the containment failure for these plants is highly unlikely. This good result for the Westinghouse containments is due to the configuration of the lower part, which directs most of the fragmented melt particles to a dead-end room where they are trapped. Only a small fraction of the particles from the discharged mass will reach to the main containment volume. This may not be so for the containments of other reactor types, e.g. the German and French PWRs. These containments are currently under study.

The DCH is prevented through unintended depressurization due to the failure of the surge line caused by the natural circulation flow of very hot steam from the core to the steam generator. It is also prevented by the intentional depressurization that can be affected by the opening of the relief valves by the operator in case of a high pressure severe accident scenario or by the cooling of the primary water by increasing the flow in the secondary side of the steam generator. The BWRs generally depressurize for most accident sequences through the Automatic Depressurization System (ADS).

A model has been constructed [26] for the DCH, which can be employed for calculating the containment pressure response.

11.4.2 Hydrogen Detonation

The hydrogen combustion loads were the first to be addressed by the USNRC right after the TMI-2 accident, since there was a 2 bar pressure spike in its containment due to the combustion of the hydrogen produced by the Zircaloy oxidation during the core heat-up. The hydrogen rule required management of the hydrogen combustion loads. The small containments of BWRs have all been inerted by replacing most of the air with nitrogen. The PWR ice condenser and the BWR Mark-3 containments have installed igniters to burn the hydrogen as it is produced and not allow it to reach >10% concentration. The PWRs in USA have not installed any special devices for hydrogen control. However, the PWRs in Europe have installed passive systems for hydrogen recombination. It should be mentioned here that high concentration of steam (generally present) in the containment atmosphere effectively acts as an inert gas.

Hydrogen detonation can also occur on turbulence generation due to flow or other sources. The phenomenon of transition to detonation is under investigation. It has been found to occur only for quite high hydrogen concentration mixtures [27]. Thus, hydrogen detonation can be prevented through management of hydrogen concentration in the PWR containments.

11.4.3 Ex-Vessel Steam Explosion

Ex-vessel steam explosion can be postulated for (Westinghouse) PWRs that flood their cavities and for the Swedish BWRs that flood their dry walls. These actions are intended in both of these reactors as an accident management strategy for cooling the melt discharged from the vessel and preventing the MCCI to occur on their basemats.

Steam explosion has a long history of research which appears to be increasingly hopeful in excluding the occurrence of a strong steam explosion. However, the results are not conclusive. The available experimental evidence from the various tests conducted [28] so far, is that the oxidic melt: (non eutectic UO_2+ZrO_2) is difficult to explode and when it does it has a very low energy yield. The non-eutectic corium melt mixture has a small separation between the solidus and the liquidus temperature lines. The cooling transient that a melt droplet suffers in water, tends to make a mushy layer at the boundary, which prevents its fine fragmentation needed for a steam explosion. There are other limiting mechanisms, e.g. the vapour formation in the vicinity of the droplet due to the thermal radiation from the droplet. A quantification of these effects is under development but not at hand yet. Elaborate three-field analysis codes, [29] have been developed, however, their validation is questionable. Fundamental single drop experiments are being performed to discern the role of physical properties on the explosivity of melt and the energy yield.

11.4.4 Melt Attack on BWR Mark-1 Containment Liner

This mode of early containment failure is particular to the Mark-1 BWR due to the short distance between the vessel and the containment liner. The contention was that the melt discharged from the vessel, on its failure, would traverse that distance readily and attack the liner to fail it. Experiments and analysis performed [30] showed that if the dry well would be filled with water, the thermal load on the liner would be insufficient to fail it. Thus, the preventive action for the Mark-1 is to add water to the dry well before vessel failure in order to prevent early containment failure. Such provisions have been made for the BWR Mark-1 plants.

11.4.5 Molten Corium Concrete Interactions (MCCI) and Basemat Failure

The MCCI occurs for the dry containments in which no water has been added or present. It also occurs in wet containments if the melt pool can not be cooled. The MCCI leads to concrete melting and gas (H_2O, CO, CO_2) generation. This results in (a) the pressurization of the containment by the non-condensible gases CO and CO_2 and the gradual sinking of the melt pool into the basemat. The melt pool forms crusts at its top surface, therby, the heat loss to the containment environment is quite low. It is feared, although not proven, that the heat generating melt could, in time, melt through the basemat and attack the ground underneath. It would, however, probably pressurize the containment sufficiently to fail it before the complete melt through of the basemat. This, of course, depends on the thickness of the basemat provided in a particular plant.

Another concern of MCCI is the ratio of the radial/axial ablation. It appears from the most recent MCCI experiments [31] that the radial ablation may be a factor of two greater than the axial ablation for the siliceous concrete employed in Europe. It is almost the same ablation for the limestone-common sand concrete, employed in U.S. plants, in radial and axial direction. The difference between the two concretes, primarily, is that the limestone-common sand concrete generates much more gas than the siliceous concrete does.

The codes describing the MCCI are not describing the recent experiments as well as needed. Thus, there is more research needed on the MCCI process. The MCCI can be avoided by cooling the melt pool to temperatures below the concrete ablation temperature of ~1000°C. The coolability of ex-vessel corium particulate beds and melt pools is discussed next.

11.4.6 Melt Debris Stabilization and Coolability

11.4.6.1 Cooling of Ex-Vessel Particulate Beds

Melt coolability is perhaps the most vexing issue impacting severe accident containment performance in the long term [32]. As mentioned earlier, melt coolability is essential to prevent both the basemat melt-through and the continued containment pressurization, thereby, to stabilize and to terminate the accident, without the fear of radioactivity release from the containment.

Provision of deep (or shallow) water pools under the vessel may not assure long term coolability (quenchability) of the melt discharged from the vessel. Interaction of the melt jet with water may lead to a particulate bed, which may be difficult to cool if it has low porsosity. Incomplete fragmentation will lead to a melt layer on the concrete basemat under a particulate debris layer and a water layer.

The coolability of the ex-vessel particulate debris beds in the BWR dry well and in the Westinghouse PWR vessel cavity is determined primarily by the dry out heat flux, since the bed will be water-logged. The bed will, most probably, be radially and axially stratified. It could also have very low porosity and a small mean particle size if a steam explosion occurs, which produces very small size particles. It has been observed in the POMECO-facility at the Royal Institute of Technology (KTH) that beds with porosity ~40% and particle size ~2 mm are coolable with top flooding. Lower porosity and smaller particle beds are not easily coolable, except when water is injected from the bottom [33].

We believe that the ex-vessel debris beds will be three dimensional, and stratified. There should be transverse paths available for water to penetrate the bed and cool the regions where dry out may occur. Most of the debris bed experiments performed so far have provided one - dimensional addition of water, either from top or bottom. We certainly expect that the 3-D cooling will be much more efficient than top flooding.

11.4.6.2 Cooling of Ex-Vessel Melt Pools

Coolability of a melt pool interacting with a concrete basemat by a water overlayer was under intense investigation in the MACE Project. Three experiments were performed successfully in which melt pools of $30 \times 30 \times 15$ cm depth, $50 \times 50 \times 25$ cm depth and $120 \times 120 \times 20$ cm depth were generated on top of limestone common sand (LCS) concrete basemats and water added on top. The melt material contained Uranium oxide, Zirconium oxide, Zirconium and some concrete products. The decay heat generation in the melt was simulated through electrical heating. It was found that for these three tests, the effect of the sidewalls dominated the phenomena, since the insulating crust formed on top of the melt pool attached itself to the sidewalls. The crust prevented intimate melt-water contact and the heat transfer rate slowly decreased from approximately 2 to 0.1 MW m^{-2}, which is less than the decay heat input to the melt.

Four modes of heat removal from the melt pool were identified. These are, (1) the bulk boiling during the initial melt-water contact; (2) water ingression into the crust and conduction; and (3) melt eruptions into water, when the heat generated in the melt is greater than that removed by conduction or water ingression through the crust, and (4) local crust break-up or fractures leading to renewal of melt-water contact, which may form another crust underneath. In the large test ($120 \times 120 \times 20$ cm), it appears that significant water ingression occurred, and/or water entry through local holes, since after the test the crust (or cooled melt) was 10 cm thick, i.e. about half the melt was cooled. Continued concrete ablation led to the separation of the melt pool from the suspended crust, and the conduction heat transfer decreased substantially.

The integral test program was modified to investigate the modes of heat transfer through separate-effect tests with the intent of developing validated models which could be employed for the evaluation of prototypic coolability configurations.

A new project named MCCI was completed recently under the sponsorship of OECD/NEA. The objective was to continue the separate effect tests. Tests were performed to study the water ingression mechanism. These tests appear to find that the water ingression mechanism is melt material dependent and, in particular, it was found that the addition of concrete products to the oxidic melt pool decreases the water ingression rate markedly. The strength of the crust formed during the water ingression tests was measured and was found to be rather small, indicating that large span crusts, probably, will fracture under the water loading imposed during the flooding process. The other mechanism of heat removal from the melt pool: melt eruption into water depends on the gas generation rate from concrete ablation. This mechanism will not be as active in the ablation of the siliceous concrete found in Europe since its gas content is quite low. The ablation of the limestone-common sand concrete may be able to support melt eruptions due to the larger gas generation, however, it is not clear what fraction of the melt pool could be cooled with this mode of heat transfer.

Currently, it is not evident that coolability of a corium melt pool by a water overlayer can be certified. Perhaps, at plant scale, with spans of several meters, the top crust will be unstable, and there would be periodic contact between the melt and water to eventually cool and quench the melt. It is clear that some basemat ablation will occur during the coolability process. One benefit of the water overlayer should be mentioned: the water will scrub most of the fission products that are produced during the molten corium concrete interaction (MCCI).

Since melt coolability with a water overlayer may be hard to achieve, alternative and innovative means have been explored to cool and quench the melt. Experiments have been performed at the COMET facility in Forschung Zentrum Karlsruhe (FZK) in which water is introduced at the bottom of a melt pool with a slight overpressure, either through nozzles or through a porous concrete substrate. It has been found [33] that melt cooling and quenching is quite readily accomplished and that no steam explosion occurred even with the Al_2O_3 melt. The COMET design is currently being optimized through a series of experiments at different scales. This concept has merit since it uses the same principle as in the coolability of a particulate debris bed with water injection at the bottom. The co-current water and steam flow are much more efficient in cooling and quenching a melt pool than the counter-current flow that occurs when the melt pool is flooded at the top. It is advisable to inject water into the melt bottom boundary before a large quantity of siliceous concrete mixes with the corium melt, and imparts greater viscosity and a glass-like structure to the melt.

The COMET concept can only be accomplished in the current plants with extensive modifications in their containments. Another concept which is under study at KTH is that of downcomers built into the containment which channel the water from top of the melt pool to the bottom of the melt pool, thereby utilizing the already proven high cooling efficiency of the bottom water injection. A loop will be established in which water goes down in the downcomer and the rising steam (after the evaporation of water entering at the bottom through the debris bed or the melt pool) provides the buoyancy head. Thus, the quenching of the debris bed or melt pool with top-flooding, which has to fight the CCFL is enhanced by the much more efficient co-current cooling process brought on by the availability of water at the bottom. We believe that such an innovative cooling system can be installed in existing PWRs and BWRs, quite easily, without jeopardizing the regular functioning of the plant, or the periodic shutdowns or inspections.

Experiments performed in the POMECO facility [34,35] have demonstrated the benefits of the downcomers through a several fold enhancement of the dry out heat flux and the quenching rate. A similar experiment performed in the COMECO (COrium MElt COolability) facility with a melt pool, around a downcomer, flooded from top indicated [36] substantial benefit of the downcomer, since a quench front progressed from both top and bottom of the melt pool.

11.4.7 Containment Bypass

Containment bypass, as the term clearly states would effectively negate all the beneficial effects of the containment. In the bypass scenario, a path is found for the fission product source term to escape from the containment without its failure. The PWR has two possible such paths. (i) the path from the containment to the auxiliary building caused by an interfacing LOCA and (ii) the steam generator tube rupture (SGTR) providing a path to the environment through the dump valves on the secondary side of the damaged steam generator.

We believe that the interfacing LOCA path was identified and the PWR plants have closed that probability or have made it highly improbable to occur. The SGTR containment bypass path can not be closed, since the secondary side of the steam generator is a pressure vessel, which normally requires a safety (dump) valve. Some new designs have proposed the discharge of the dump valves into the containment rather than into the environment. However, dump valves in all of the current PWRs communicate with the environment.

The scenario of concern is that of the high pressure accident as was the TMI-2. The high priority accident management action for the high pressure accident scenario is to depressurize the vessel to obviate the risk of DCH and containment failure. The vessel pressure is reduced to much below that of the secondary side of the steam generator so that the flow is from the secondary side to the primary side if any tube rupturs. Second accident management action is to fill the secondary side of the damaged steam generator with water immediately so that any fission products that may escape to the secondary side from the primary side get absorbed in the water. The secondary side being a very tall vessel full of water will have a very high value decontamination factor.

There is on-going research on the fission product transport in the empty secondary side of a steam generator. Decontamination factors (DFs) are being measured for aerosol flows with different aerosol size distributions and flow rates. It appears that water filling of the secondary side will be needed to obtain a comfortable margin for DFs.

11.4.8 Fission Product Release and Transport During Ex-Vessel Accident Progression

The fission products and the core materials during the core heat-up process arrive into the containment, as aerosols. Their transport in the containment is governed by aerosol physics, which determines the fission product concentration in the containment atmosphere as a function of time. As mentioned earlier, if there is steam atmosphere in the containment (as it should be for a severe accident), the fission product aerosol concentration in the containment atmosphere decreases exponentially with time, largely due to the process of aerosol particle size growth (due to steam condensation), agglomeration and deposition. Another aerosol deposition process active is that of Stefan flow carrying aerosols to the walls of the containment where the steam is condensing. As mentioned earlier, typically, fission product concentration in the containment atmosphere can decrease by a factor of 10^{-4} in about four hours.

The release of fission products during the ex-vessel accident progression can occur during the MCCI due to the gas sparging and the high temperatures in the melt. The releases of interest are those of the less-volatile fission products, e.g. Ba, Sr, Ce, Ru, MO, since the volatile fission products have already been released.

The ACE experiments [37] provided systematic data on the release of the above-mentioned fission products. The measured values for releases were less than 1% of the inventory for all of the less-volatile fission products. These values were much smaller than what were previously calculated.

Management of the iodine concentration in the containment immediately after the accident and for the long term is essential in order to reduce the potential of harmful releases due to containment leakage or other events. In this respect, the processes of concern are (i) the interaction of iodine with paints on containment surfaces to form organic iodine, which is difficult to remove and (ii) the radiolytic formation of iodine. Thus, iodine chemistry in the containment is important and the use of p-H control to reduce the iodine concentration is needed for the long term management of the iodine concentration.

12. SEVERE ACCIDENT MANAGEMENT

After the TMI-2 and the Chernobyl accidents, it was clear that (1) the great environmental and human costs of Chernobyl will be entirely unacceptable; (2) the public consequences of the TMI-2 accident were minor, in spite of the great turmoil caused and that was due to the TMI-2 containment preserving its integrity. Severe accident research since 1980 showed that indeed there could be dynamic loads imposed on the containment, which could fail the containment early to cause a very large release of radioactivity to the public. Such a large release would pose a greater public risk than that prescribed in TID14844 or even that calculated by WASH-1400. It became clear that the accidents, which may impose large dynamic loads on the containment had to be (a) prevented and/or (b) mitigated completely. This was the birth of severe accident management (SAM) as an active tool for minimising the public risk of a severe accident. SAM may be defined as follows "SAM is the use of existing and alternative resources, systems and actions to arrest and mitigate accidents that exceed the design basis of nuclear power plants".

The earliest SAM action in the LWR plants was that of the management of the hydrogen concentration in the containment. This was required by the USNRC in view of the hydrogen combustion event in TMI-2 containment. Other actions followed, some requiring backfits in the plants, some requiring operator actions for which training schedules had to be devised. Severe accident management guidelines (SAMGs) were produced for the Westinghouse PWRs and the General Electric's BWRs, which were appropriately modified for each specific plant. Most of the utilities in USA have already implemented the SAMGs for their individual plants. Some of the European plants also have implemented their individual SAMGs. They have closely followed the guidelines produced by the appropriate owner groups in USA. They have adapted the set points, curves and computing aids that were produced by the US Owner Groups for their specific plants. The French, German and Swedish plants have a rather open approach for SAMGs, since no generic standard guidelines are employed. Each plant or each set of plants (as in France) are performing work on their individual plants to implement SAMGs. In general, each plant is using some equipment backfits, specially designed to deal with a severe accident. They have also made many procedural and operational changes.

In the following paragraphs, the structure produced in the OECD Report [38] is employed to group the SAM actions under four main functions viz: (1) cooling a degraded-core, (2) managing combustible gases, (3) managing the containment temperature, pressure and integrity and (4) managing the release of radioactivity.

12.1 Cooling a Degraded Core

Adding water to the reactor (RPV) is an action that is very similarly implemented in many countries. There is a general agreement that the hazards posed by increased hydrogen generation; possible recriticality and increased steam production do not outweigh the benefits of retaining the degraded core inside the vessel. The criteria generally followed for this action is to supply to the reactor vessel with water as soon as injection capability is available. Westinghouse Owners Group (WOG) standard guidelines contain warnings about the side effects of increased hydrogen production, and their computing aids take into account, in a simplified way, the additional risk of hydrogen combustion in the containment. The issue of recriticality is generally considered to affect more the BWR, where borated water sources are less available and early control rod material meltdown and relocation is a possibility. General Electric (GE) standard guidelines specify the use of the liquid control system in case of core melt criticality, but no criteria are given on the water flooding rate.

RCS depressurization is also a generic SAM action that can be accomplished in a variety of ways. The preferred way for PWR is the “feed and bleed” system, adding water to the steam generators and depressurizing the secondary sides thereby cooling down the primary side and reducing its pressure. If this action is ineffective, depressurization can be accomplished by direct opening of pressurizer valves. There are numerous benefits to intentional depressurization, i.e. alternate means of cooling become available, and high pressure melt ejection is avoided, although there are also possible drawbacks, like increased hydrogen production and higher probability of in-vessel energetic fuel-coolant interaction. All PWRs have pressurizer valves that can be used, although sometimes pressurizer spray is a possibility. All BWR are designed to be easily depressurized in case of ADS failure.

The action of containment initial flooding in order to delay vessel failure by means of cooling through the vessel wall, is one where there is considerable variation among countries, it is recognized that the action can not by itself guarantee vessel integrity, especially for reactors with high power, but the action may delay vessel failure.

Containment flooding to several levels is recommended in the standard Westinghouse Owners Group (WOG) Severe Accident Management Guidelines, although specific implementation will depend on the design of the reactor cavity. Generic GE standard guidelines recommend drywell or primary containment flooding as an integral SAM action that could provide a means of core cooling through the vessel wall, and also a possibility of alternative vessel flooding through the relief valve tail-pipes. German plants do not consider cavity flooding and continue the concept of “dry cavity.” Finland has implemented the strategy in Loviisa plant. Swedish and Finnish BWRs have also implemented the strategy that a water pool is created under the vessel as soon as the water level may fall below the top of the core. However, the level of water does not reach the vessel and the vessel wall is not cooled. This action is for ex-vessel cooling of debris/melt which deposits into the water pool on the failure of the vessel. This action is not for cooling of the vessel from outside.

12.2 Management of Combustible Gases

There are considerable variations in the strategies followed to reduce H_2 and CO inventory in the containment, because of the differences in existing equipment and the status of implementations. Many countries have decided on the use of catalytic recombiners in PWR containments, which can reduce H_2 and CO concentrations while keeping containment pressure low. Some BWRs and some PWRs use igniters, to produce intentional H_2 or CO burns. Venting of the containment is a strategy considered also for the reduction of combustible gas inventory.

Catalytic recombiners have demonstrated their capability of reducing H_2 concentration under steam-inerted atmospheres, very low H_2 concentrations, and presence of aerosols [38]. Installation of recombiners has been decided in Belgium, Germany, France and the Netherlands and in some Eastern European countries. Finland has decided on the installation of a new H_2 management system using catalytic recombiners, although currently igniters are being used. G.E. BWRs with Mark I and II containments and KWU German BWRs of old design are inerted and do not use ignition devices. G.E. BWRs with the larger Mark III containments have ignition systems. The PWRs in USA do not have any hydrogen management system. Their containment volume is supposed to be large enough to not create a high (>10%) concentration hydrogen mixture. The PWRs in Europe are being fitted with passive catalytic recombiners.

12.3 Management of Containment Temperature, Pressure, and Integrity

Automatic or manual initiation of containment sprays to condense steam released exists in most BWRs and PWRs although there is a significant variation in the equipment dedicated to the implementation of this action. Sprays are also used, in the longer term, in conjunction with heat exchangers, which can extract heat from the containment to avoid pressurization. German plants have spray systems. Swedish plants have an independent dedicated spray system. Loviisa in Finland, Zorita in Spain and two Belgian plants have external spray systems for their steel containments.

Fan cooler systems in PWRs can extract heat and avoid late pressurization due to release of non-condensable gases during MCCI, but not all plants have fan coolers as qualified safety grade equipment. The initiation of fan coolers for SAM in PWR containments is considered in Belgian, Spanish and UK plants, and it is included as a standard action in WOG SAMG.

Containment flooding is considered both in PWRs and BWRs. Also, a consensus is developing that initial containment flooding will improve the chances of ex-vessel melt coolability in case of vessel breach, in spite of the higher risk of energetic ex-vessel melt water interactions, and will reduce ex-vessel radioactive releases. Here we have to distinguish between PWRs with their larger and relatively strong containment and BWRs with their small containments and perhaps vulnerable vessel support structures, whose integrity may be threatened by a highly energetic steam explosion. Containment flooding before the discharge of the melt is not practised in the French and German PWRs and BWRs. There the water is added only after the melt has been deposited on the basemat. Coolability of the melt is not expected. There will be ablation of the concrete basemat and possibly a basemat melt-through..

Many European plants include the strategy of containment venting, to avoid late failure due to over-pressurization. Scenarios like complete loss of containment heat removal capability, or full power ATWS in BWR, are typical examples where containment venting becomes essential. This accident management action can avoid late failure due to pressurization by non-condensable gases released during MCCI, for which containment heat removal systems are ineffective. Venting can be used also to ease containment flooding, and to reduce the inventory of combustible gases. Considerable variation exists in the implementation of this SAM feature. The standard WOG SAMG do not include containment venting as a SAM strategy. Venting of containment with specially designed filtered vents systems is implemented in all PWRs and BWRs in France, Germany, Sweden, Netherlands and Switzerland. The U.K., Belgium and Spanish PWRs do not have venting. Spanish BWR have a dedicated manually operated venting system, which connects the suppression pool airspace to the off-gas stack, without filtering. The U.S. plants do not employ containment venting, except that some U.S. BWRs are installing the same venting systems as the Spanish BWRs.

12.4 Management of Radioactivity Releases

Standard strategies for mitigating the rate of radioactivity release through opening in the containment boundary include reducing the containment pressure, by means of available containment heat removal systems and through the venting systems. At later times in a severe accident revolatalization releases from the deposited aerosols in the RCS become a concern. Mitigation of those releases will involve cooling of the RCS walls.

A common strategy, for reducing the inventory available for release in the containment, is the initiation of containment sprays in PWR and BWR. Sprays were designed for early operation and steam condensation after LOCA; and not for long term operation during severe accidents. However, sprays can produce effective aerosol deposition due to interception of droplets. Also, sprays can remove some of the gaseous molecular Iodine as long as they do not become saturated with I. The effectiveness of sprays will depend on the availability of AC power and the extent of the areas covered by the spray system. Iodine volatility in many PWRs is reduced by means of additives that are included in the design of containment sumps, or the containment spray system.

Engineered filtering systems are installed in most PWR and BWR, with HEPA filters generally designed for conditions of normal operation. Use of engineered filtering sys-tems during severe accident environmental conditions is possible, but the efficiency of the filter may be reduced if additional technical features have not been provided (i.e. emergency filtering systems). Swedish plant containments have a venturi filtered venting system specially designed to deal with sever accident situations, which have a very high value of DF. The French plants have a sand-based filtered venting system for the severe accidents.

Removal of radioactive aerosol, by means of scrubbing in BWR suppression pools, is a beneficial side effect of the suppression pool functional design. Aerosol scrubbing by means of a water pool overlying the core debris is also considered in standard WOG and GE standard SAMG, as a strategy to reduce ex-vessel releases to the containment.

Secondary side flooding is a standard strategy, included in WOG SAMG, for mitigation of releases to the environment due to SGTR accidents, and protection of SG tubes from creep ruptures.

13. NEW LWR PLANTS

The presently-installed LWR plants in Western countries have been addressing their safety performance from the day they were installed and operating. Prior to the TMI-2 accident the safety design-base issues, e.g. the functioning of ECCS for various breaks, were of most concern. The plant concerns were also with the integrity of the primary system, e.g. the G.E's BWRs plants needed replacement of some piping, the vessel weld material was of concern, etc.

After the TMI-2 accident, the safety performance concerns were with the severe accident safety, i.e. the prevention and mitigation of these accidents. This has been formalized into the programs of Severe Accident Management (SAM) at most of the LWR plants. Severe accident research results have lead to backfits and accident management actions and procedures, which have enhanced the safety of the plants, or provided the rationale for deliberate decisions of not requiring any backfits or SAM measures. A representative list is as follows:

- hydrogen control with igniters and catalytic recombiners
- improved safety valves on PWRs
- no inerting of MARK-3 BWRs
- water addition to the MARK-1 drywell to prevent liner failure
- vessel depressurization for DCH protection
- no backfits for protection against alpha mode failure
- use of BWR suppression (condensation) pools for fission product removal
- hard vents for BWRs from the suppression pool
- flooding of PWR vessel cavity for Westinghouse PWRs
- flooding of drywell for Swedish BWRs
- additional water delivery sources for accident termination
- reinforcement of containment penetrations
- realistic ex-vessel source term specification
- pressurized thermal shock prevention procedure
- filtered venting
- long term management of Iodine in the containment.

Clearly, not all the severe accident issues have been resolved for the presently-installed plants. The most important of the unresolved issues is the coolability of the melt/debris produced during the postulated severe accident in order to stabilize and terminate the accident. This will assure that the containment remains intact and that there is no significant radioactive release, precluding either the evacuation of the nearby population, or their speedy return to their homes if any evacuation did occur. The issues of (a) ex-vessel steam explosion-induced containment failure, which is of concern for reactors that establish a deep water pool in their containments; (b) hydrogen detonation-induced containment failure, (c) DCH induced containment failure, (d) MCCI induced basemat failure or containment pressurization failure, can be addressed of through operational or accident management actions, respectively, by (i) not establishing a water pool in the containment, (ii) depressurizing the vessel in time and providing valves, which will bring the vessel pressure below 20 bars, (iii) availability of the hydrogen igniter and/or recombination systems and (iv)assuring the cooling of the melt/debris below the concrete ablation temperature of 1000°C.

One can reach the conclusion that if the melt/debris can be cooled and kept cool to stabilize and terminate the accident without having a pre-existing pool in the containment, all the remaining concerns about the danger of severe accidents in the LWRs may be addressed adequately. Alternatively if the melt can be cooled and retained in the vessel, thereby assuring containment integrity, the same conclusion may be reached. Recent concerns about the production of the fission product Ruthenium or the release of some small fraction of iodine as gaseous iodine are also addressed, since an intact and low leakage containment will protect the public against the hazard of these releases to the containment.

We believe that the Generation 3+ LWRs or the near-future new LWRs have focussed on the issue of the long term coolability, stabilization and termination of the severe accident as their goal. Two lines of design measures have been developed in these new LWRs; (a) in-vessel coolability and melt retention and (b) ex-vessel coolability and melt retention. We shall describe these very briefly in the following paragraphs.

13.1 The In-Vessel Melt Retention (IVMR) Strategy

The in-vessel coolability and retention is based on the idea of flooding the PWR vessel cavity or the BWR drywell with water to either submerge the vessel completely or at least submerge the lower head. The PWR or BWR lower head containing the melt pool is cooled from outside, which keeps the outer surface of the vessel wall cool enough to prevent vessel failure. This concept is employed in the Loviisa VVER-440 in Finland, where it has been approved by the regulatory authority STUK. The concept is also employed in the PWR designs: AP-600, AP-1000, Korea's Advanced PWR-1400 and in the 1000 MWe BWR design of AREVA.

The AP-600 design was analysed [23] with a bounding accident assumption of the lower head full of convecting melt pool. They found that the heat flux varied with angle, peaking near the equator. Fortunately, the heat removal by the water outside also varied with angle reaching highest value also near the equator. It was found that for a uniform corium pool for the 600 MWe AP-600 reactor, there was sufficient margin between the critical heat flux (CHF) on the water side and the incident heat flux from the corium pool. This margin of safety, however, may be reduced

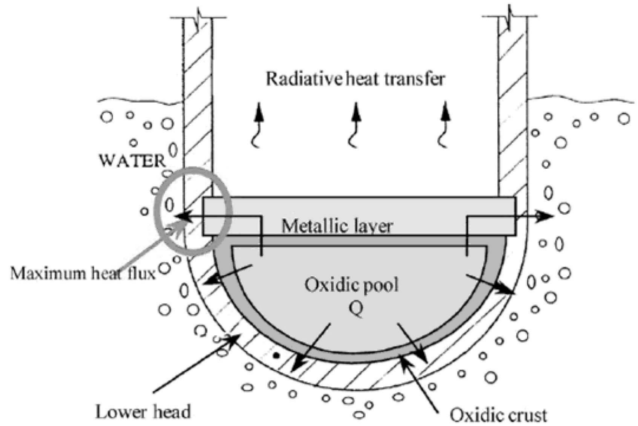


Fig. 24. In-vessel Melt Retention

substantially in case there is a metal layer present on top of the oxidic corium pool. The metal layer results from the steel present in the PWR and the BWR lower heads which is melted by the corium pool and since it is lighter it rises to the top of the corium pool; see Fig. 24. The metal layer receives heat from the corium pool and performs Rayleigh-Benard convection transferring heat transversely to the vessel wall, which is then subject to a highly elevated heat flux. This heat flux focusing is most intense for a thin metal layer since the transverse area for heat transfer is smaller. It was found [39] that for metal layers of < 30 cm depth the focused heat flux could overwhelm the critical heat flux near the equator. For the AP-600, it was found [23] that the metal layer would be thick and there was sufficient margin available between the focused heat flux and the CHF outside.

The power of the AP-1000 is 60% larger than that of AP-600 and that of Korea Advanced PWR by 230%. For the 1400 MWe reactor, the focused heat flux, most probably, would be greater than the CHF on the water side. The strategy of the Korean plant is to simultaneously flood the metal layer with water inside the vessel, which, hopefully, could remove sufficient heat from the upper face of the metal layer to reduce the focused heat flux to values less than the CHF. A dedicated water system has been installed in the plant for water injection to reach the melt pool in the lower head at the appropriate time.

Further complications have been introduced recently by the findings in the OECD sponsored RASPLAV and the MASCA Projects [40,41] of chemical reactions between the melt constituents which may create different layer configurations in the melt pool. For example it was found in the RASPLAV Project that the presence of even small amounts (<0.3%) of carbon in the system promotes the stratification of the melt pool by separating the oxides from the metals in the melt, thereby forming a light melt layer, rich in metals, and carbides residing on top of the oxide-rich melt pool. A finding from the MASCA Project is that of the combination of the steel components with Uranium to form a metal compound which being heavier than the oxidic pool sinks to the bottom of the oxide-rich melt pool. It is not clear whether all the steel will combine with the Uranium metal. The initiator of this steel-Uranium combination is the unoxidized Zr present in the melt. The worst situation would be in which some of the steel is taken by Uranium metal to the bottom of the pool, while some remains at the top to form a thin metal layer which can provide a strong focussed heat flux on the vessel wall. The melt pool composition and configuration situation is quite confused presently, since the more recent data obtained in the oxidizing atmosphere (steam) have shown that after Zr oxidation is completed the steel is released from Uranium and rises back to the top of the pool. More research on the pool stratification issue is anticipated.

13.2. The Ex-Vessel Melt Retention Strategy

This strategy has been adopted by the European Pressurized Water Reactor (EPR) design currently in realization in Finland and by the new Russian VVER-1000 designs for China and India. The EPR design [42] spreads the discharged corium mixed with sacrificial concrete, on a flat steel surface coated with a high temperature inert material, cools it from bottom with water flowing in channels, and floods it with water from the top. The idea behind this design is that with spreading the depth of the melt pool will be reduced to the extent that it can be cooled by a water overlayer with some assistance from a cooling system at the bottom. Sacrificial concrete is mixed with corium discharged from the reactor pressure vessel (RPV) in a concrete-lined pit to reduce its temperature, and more so, its solidus temperature. Thus the mixture remains as liquid over a much larger temperature range, and, in fact, will spread more easily over a large floor area. Figure 25 shows the configuration of the EPR melt retention enclosure.

Much research was performed on the efficiency of spreading of the melt at various European laboratories including that of the Nuclear Power Safety Division at KTH. We developed a very innovative scaling theory for spreading [43] which has been able to predict most of the spreading data obtained with simulant and prototypic melt materials. The EPR melt spreading analysis was also performed with this model and it was found that even with conservative assumptions, uniform spreading of the discharged melt and concrete mixture can be obtained in the EPR design. The depth of the melt (~40 cm.) unfortunately is greater than that can assure melt coolability with water flooding alone. The cooling coils built in the base of the spreading chamber will be needed to cool the melt. It appears, however that it will take considerable time before the center part of the spread melt pool will solidify.

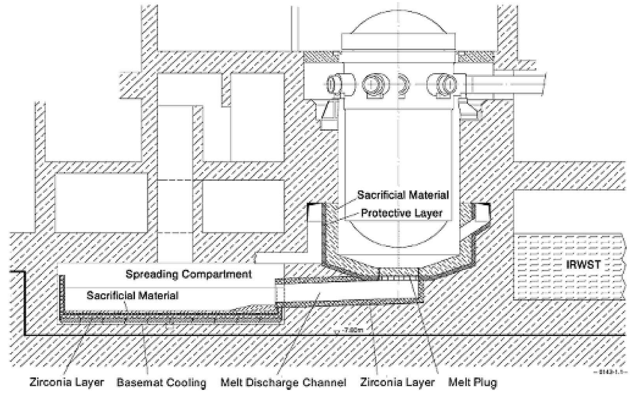
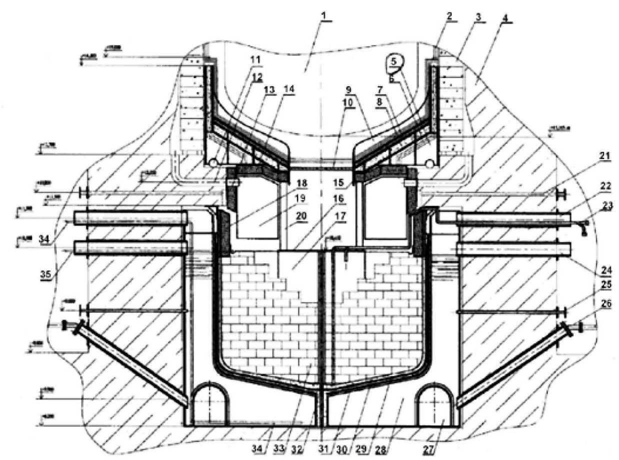


Fig. 25. EPR Core Catcher



1 Reactor pressure vessel
2 Thermal insulation of dry protection
3 Dry protection
4 Concrete cavity
5 Lower plate
6 Channel of lower plate cooling
7 Corium-liquefying substratum
8 Heat-resistant concrete
9 Bearing slab
10 Protective shield
11 Damper
12 Channel of dry protection cooling
13 Thermal protection of concrete cantilever end
14 Thermal protection of lower plate
15 Deflector baffle
16 Maintenance floor
17 Support truss
18 Thermal protection of concrete cantilever base
19 Ventilation duct
20 Support column
21 Channel of cooling water supply to concrete cavity
22 Channel of steam removal from heat exchanger
23 Blowdown channels
24 Heat exchanger drain channel
25 Level indicator channel
26 Channel of cooling water supply to heat exchanger
27 Arch-shaped header
28 Heat exchanger
29 Thermal insulation of heat exchanger
30 Basket
31 Honeycomb-like thermal shock shield
32 Support truss slab
33 Filling compound
34 Channel for thermocouples location
35 Basket drain channel

Fig. 26. Tian Wan VVER Core Catcher

The Russian VVER-1000 design employs a core catcher in the traditional sense. This core catcher shown in Figure

26 is a separate vessel installed under the RPV with an intake designed to cover almost the whole surface of the bottom head so that the melt discharged from the RPV is deposited in the core catcher even if the RPV failure occurs at an angular position close to the equator (which it probably will). The core catcher is like a lower head but of much larger volume and it is cooled from outside by a water pool as in the IVMR concept. The mixing process also reduces the mixture power density and the heat fluxes on the vessel walls. The core catcher is full of bricks made of oxidic material containing Fe_2O_3 and other oxides. The purpose is the same as in EPR: to reduce the temperature of the discharged corium and to keep it liquid over a larger range of temperature. The core catcher walls are steel but they are lined with oxide bricks. The chemistry of the materials with the corium has been subject of several experiments and the chosen oxide composition is such that the Uranium and the metals in the corium combine to form a dense metal layer which sinks to the bottom of the melt pool. There, supposedly is no metal layer on the top of the oxidic pool. The melt pool is flooded with water with the argument that the probability of a stratified steam explosion is much reduced, since the metal is at the bottom under the oxidic material pool.

The melt pool in the Russian core catcher design also may remain molten for considerable time and will perform natural convection. There is not sufficient information in the literature to assess the long term operation of the core catcher and the state of the melt pool inside.

The General electric company has designed a Generation 3+ BWR, which also is equipped with an innovative core catcher below the vessel, which is cooled by a set of steel pipes embedded in the floor and walls, which are lined with a non-ablating material. The steel pipes have natural-circulation water flow in them to remove the decay heat generated in the ex-vessel melt-debris pool. This core catcher design is currently under development, testing and peer review.

14. CONCLUSIONS

The march of history for LWR safety has shown a definitly positive direction. The potential of nuclear power for public good accompanied by its potential for serious public hazard was recognized early and this is to the credit of the scientists and engineers, who pioneered the civilian nuclear power. Much credit also has to be given to the regulatory commissions (bodies) of the various countries, who have been the guardians of public safety during the development and spread of nuclear power in the World. Most credit, however, has to go to the scientists and engineers, who have diligently raised every safety issue or question, performed research and provided solutions. It has been a splendid history and it deserves praise.

The challenges posed by the TMI-2 accident were met through patient hard work, severe accident research, and design innovation. The presently-installed LWR plants made improvements in components, systems, operator training, man-machine interface, safety culture, etc., thereby significantly reducing the probability of a severe accident occurring. They also instituted severe accident management backfits, systems and procedures, which are providing assurance of the elimination of an uncontrolled and large release of radioactivity even in case a severe accident occurs. Still, the presently-installed plants can not provide assurance of coolability of a melt pool/debris bed, which could be formed during a bounding severe accident. In that situation, the LWR owner can not assure the public that the accident has been terminated and that there is no further danger of the release of radioactivity.

The new, generation 3+, LWR designs, exemplified by EPR, VVER-1000, AP-1000, APWR-1400 and ESBWR, which employ in-vessel, or ex-vessel, cooling and retention of the core melt/debris bed that would be produced in the postulated severe accident, are reaching the end state of the development for public safety for LWRs. They, not only provide systems, which have an extremely low probability for a severe (core-damage) accident but also assure that there will not be any large release of radioactivity to the environment. The public living in a low population zone near a nuclear plant does not want to move from, or abandon the return to, their homes. Although such assurances have not been explicitly provided by the designers (vendors) of these new LWRs, they may be able to do so. Convincing the public will not be easy, however, these designs have the potential for making such convincing arguments.

The challenges of LWR safety have diminished and the new designs are great accomplishments. We can not, however, forget the incidents like Mihama, or the potential near-incident due to corrosion of vessel-head in TMI-1 or several others, e.g. partial power black-out in FORSMARK-1, which attract much publicity. The human component of LWR safety needs to be improved. In particular, complacency has to be banned. The operators, staff and the management of the nuclear plants have to be more responsive and reliable than what the plant components and systems are.

Plant aging is an issue that will be making itself visible more and more in the future years. Some of the LWR plants are reaching near the end of the original estimate of their life spans. The reliability of components and systems will become less and less in future. The need for vigilance on the part of plant staff and management has to increase. The utility companies have to recognize this and be prepared to spend the money to replace/renew old equipment, instrumentation and systems.

It should be remembered always, that the public has given civilian nuclear power a very short leash with respect to safety and in order to keep the confidence of the public, the human and management component of LWR safety can not afford to fail. It should be stressed that, presently, the nuclear power industry can not even afford a successfully-terminated and contained severe accident. There, still, would be too many 'what-if' questions and perceptions.

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