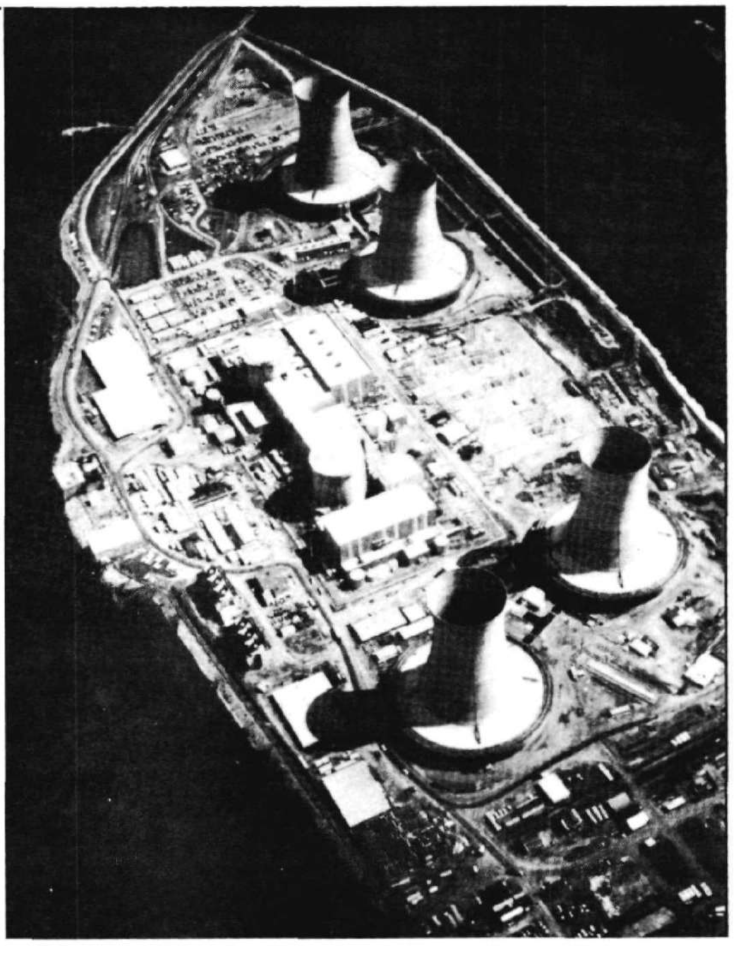

Site of the world's most serious commercial nuclear power accident. Three Mile Island, Pennsylvania, now serves as a valuable laboratory for studying the potential effects of reactor mishaps. One of the major findings was that passive physical features and processes greatly limit the accidental release of radioactive materials from the plant.
(Credit: AIF, Inc.)

Reassessing radiation releases: A closer look at source term

by Morris Rosen and Michael Jankowski

The Three Mile Island (TMI-2) accident in 1979 accelerated the process of obtaining and applying the best technical information to estimate the release of radioactive material during postulated severe accidents in commercial light-water reactors. Particular emphasis has been placed on the accident behaviour of radioactive iodine since it is predicted to be a major contributor to public exposure. Current regulatory accident analysis procedures focus on iodine.

But in the past few years, existing regulatory requirements have been challenged. The latest experimental and analytical research results now predict that, in general, the amount of radiation released during severe accidents may be lower than earlier estimates, as compared to assumptions underlying current regulations.

It is expected that findings at TMI-2 will contribute even more to the present understanding of severe accidents and the ongoing discussion on how these types of accidents should be treated in the regulatory and licensing process.

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Experience at TMI-2

Recent information from TMI-2 reveals that core damage from the accident was considerably more severe than previously estimated. Despite significant damage to the core and subsequently to the plant, the consequences of the TMI-2 accident remain unchanged — the reactor pressure vessel was not breached, the containment building integrity was not challenged, radiation releases to the environment were very low, and at the same time, in-plant airborne releases during the accident were very small.

The most recent evaluation of the core damage concludes that approximately 30% of the upper core region is now a void and as much as 20% of the core may have been relocated to the bottom of the reactor vessel.

Further, the TMI-2 core rubble contains evidence of the complete spectrum of fuel rod damage, from localized cladding failures to complete clad dissolution, fuel fragmentation, and melting. During the accident, temperatures in portions of the upper half of the core reached the fuel component melting points (up to 2800°C) and some of the cladding became embrittled due to oxidation as temperatures increased. Some molten material, including structural and fuel material, flowed down and resolidified in the lower portions of the core.

The recent discovery of large pieces of reconfigured core material resting on the bottom inside surface of the reactor vessel provides evidence that the core reheated, became molten, and subsequently entered the lower plenum region. Currently very few, if any, fuel assemblies are expected to be intact.

Many estimates of the amount of hydrogen generated from metal-water reactions also have been prepared. The current estimates are that about 50% of the core zircaloy inventory was oxidized, producing about 460 kilograms of hydrogen. Temperature and pressure data as a function of time and location have been measured. These data have also been reconciled with the physical evidence of the burn, including evidence of burning of materials. This information is now being used to benchmark computer codes to analyse hydrogen burn phenomena.

Information on the transport and deposition of fission products in the TMI-2 containment building is limited for other than noble gases because only small amounts of fission products were distributed in the containment atmosphere – even though more than 20% of the iodine and greater than 50% of the caesium left the primary system. The majority of these radionuclides were found in the basement. It is unlikely that large amounts of core material will be found in the reactor coolant system loops if the estimates of core debris in the lower plenum are correct. Nevertheless, the current effort is concentrating on gamma scanning of the reactor coolant system loops to identify if there are large repositories of fuel material. To date none have been found. This illustrates the validity, as well as the limitations, of the current approach to design basis accidents (DBAs).

Source term for DBAs

From the beginning of nuclear power programme development, "source term" has been recognized as a very important factor influencing both the design of certain safety equipment and safety evaluations, including risk assessment. Simply stated, source term refers to the radioactive material released during a nuclear reactor accident. As presently used, it is synonymous with "fission-product release", "accident release", or other similar terminology.

Yet because of technology limitations – especially in the understanding of the physical and chemical processes involved in the complex sequence of events during an accident – the source term factor was not available for regulatory consideration. Instead, conservative, non-mechanistic assumptions were developed and incorporated into regulatory requirements. These assumptions still are being used today.

One of the most important assumptions focused on radioiodine releases as the principal substance of concern because of its volatility and biological concentration in the thyroid.

Current safety evaluation approach

Currently, the safety evaluation of a nuclear power plant includes analyses of the plant's response to postulated disturbances in process variables and to postulated malfunctions or failures of equipment. Such analyses contribute to the selection of design specifications for components and systems.

In design basis accidents, the source term is based on assumed "worst case" conditions. Intentional conservatism in the iodine source term has been used to create what is perceived as a substantial safety margin. This, in turn, was expected to compensate for uncertainties in the analysis and for non-conservative omissions made to simplify the day-to-day analyses used in the licensing process. The most significant of these simplifications is the omission of all non-gaseous fission products other than iodine from the source term.

DBAs are postulated as an aid in developing and evaluating a variety of safety-related systems and equipment, and may involve a wide range of postulated fission-product releases. DBAs which consider the release of substantial amount of fission products include:

- Accidents involving the release of radioactivity normally circulating in the primary system
- Release of radioisotopes contained in the void space (gap) between fuel and cladding
- Releases postulated for site analyses (siting DBA-LOCA*) from the fuel in addition to coolant and gap activity.

The DBAs are, therefore, a set of accidents that have been chosen to envelop the most credible conditions anticipated, or postulated, in what has been perceived to be a very conservative manner. Thus, although they are not representative of expected or realistic conditions, they have been judged to bound any credible accidents.

Alternative source term approach

The existing non-mechanistic DBA structure does not lend itself to easy adjustment to reflect experience and specific research findings concerning one aspect of fission-product release. An alternative approach is to perform the evaluations of postulated accidents on a mechanistic basis.

A mechanistic treatment of an accident would necessitate the specification of important parameters and environmental conditions affecting actual fission-product release and behaviour. This includes, for example, temperatures, pressures, timing of release, oxidation potential, chemical reactions, and particle size distributions.

Such details concerning the history and physical/chemical environment of the postulated release require the specification of event sequences. Since the mix of fission products released would vary with core,

*Loss-of-coolant accident (LOCA).

primary system, and containment conditions, a wide spectrum of accidents should be considered to arrive at a realistic estimate for source term.

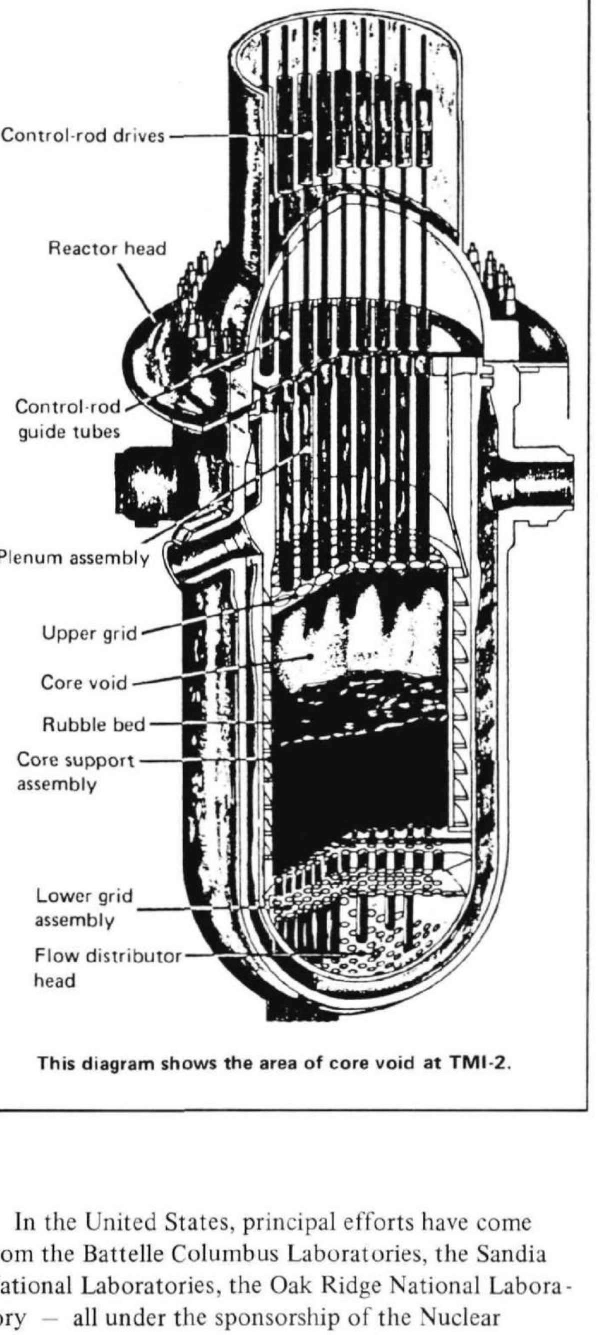
The Reactor Safety Study

The first attempt to use quantitative assessment techniques to evolve the sequences of events necessary to cause core damage, and to assess the probabilities associated with such sequences, was the Reactor Safety Study (RSS), also commonly referred to as WASH-1400. The study was undertaken by the US Nuclear Regulatory Commission (NRC) and published in 1975.

In short, the study concluded that the risk from reactor accidents was small, and that accidents more severe than "maximum credible accidents" dominate the risk. Such accidents involve not only core melt, but deterioration of the capacity of the containment, the ultimate barrier to limit the release of radioactive materials to the environment.

The study utilized generic, bounding estimates for radionuclide release. A major difficulty with this approach is that it does not lend itself to systematic improvement, for the reason that the attenuation of the source term is highly dependent on the actual physical processes in a given accident and a given plant. An independent review of the accomplishments and limitations of the RSS reached the general conclusions that the uncertainties associated with the absolute values used are large, and that caution must be applied whenever using its techniques.

Despite large uncertainties in the RSS, probabilistic risk assessment has and should have a place in the safety assessment and licensing process. The derived numerical source term values in WASH-1400 are used and being frequently quoted in the absence of any other.



This diagram shows the area of core void at TMI-2.

Reassessments of the source term

The March 28, 1979 accident at the Three Mile Island plant illustrates the validity, as well as the limitations, of the current approach to design basis accidents.

On the one hand, the existing design concepts (low leakage containment, containment spray additives for pH control) accommodated accident conditions (in particular severe core damage conditions) beyond the design basis. On the other hand, the current limitations of the safety system design to anticipate accident conditions beyond the design basis is illustrated by the retention of most fission products (and especially iodine) by the auxiliary building and through auxiliary building filters; transfer of highly radioactive liquid from the containment sump to the auxiliary building; and lack of adequate capability to deal with high levels of activity in the auxiliary and fuel-handling buildings.

Since TMI-2, a number of worldwide studies have been undertaken to re-evaluate the source term in light of experimental and analytical research and actual experience.

In the United States, principal efforts have come from the Battelle Columbus Laboratories, the Sandia National Laboratories, the Oak Ridge National Laboratory — all under the sponsorship of the Nuclear Regulatory Commission (NRC) — and the Industry Degraded Core Rulemaking Programme (IDCOR), supported by the US nuclear industry. These efforts have analysed a number of similar accident sequences for selected US-designed plants.

In Europe, the Federal Republic of Germany has conducted analyses of certain sequences for the 1300-megawatt pressurized-water reactor (PWR). In these cases, the calculated releases are very low. However, they are difficult to compare with the US results since the containment designs differ greatly.

Studies also have been done in other countries. In the United Kingdom, a considerable amount of work has been carried out on the source term for postulated severe accidents for the proposed 1300-megawatt PWR at Sizewell. French studies have analysed PWRs in

France, and a study in Denmark has analysed the Swedish type of boiling-water reactor (BWR) design. In addition, many countries have undertaken experiments to derive supporting data and have developed analytical methods that can be used as part of the overall sequence and source term analyses.

Recently, three separate reviews of the state-of-knowledge of source term have been published in the United States. These are the report of the Special Committee of the American Nuclear Society on Source Term, the report on the Industry Degraded Core Rulemaking Programme (IDCOR), and the report of the Study Group of the American Physical Society (APS) reviewing the programmes sponsored by the US NRC. Long awaited, the US NRC report on this subject (known as NUREG-0956) was scheduled to be published in August 1985.

The latest findings

Some general conclusions emerge from a summary of the latest findings in published reports. First, it must be emphasized that results have confirmed once again that the risk to the public from most severe nuclear accidents, especially those posing risks from iodine, would be significantly below that predicted by previous studies using earlier source term assumptions. This conclusion is supported by analyses of many accident sequences in which fission-product source terms are likely to be significantly lower than had been calculated in previous studies.

This overall reduction can be attributed to three principal factors:

- Recognition that reactor containments are stronger than previously assumed and therefore fail at later times, if at all
- Inclusion in analytical methods of the modelling of previously neglected physical and chemical phenomena that would lead to the retention of fission products
- Inclusion in analyses of additional sites (auxiliary buildings, pools of water) that would retain radionuclides more efficiently than previously assumed.

The second and most significant conclusion emerging from recent studies is that sequences and specific plant design details are very important in estimating the source term. It has been demonstrated that results obtained from analysis of a variety of sequences for a number of different plants are not transferable from one plant to another without careful examination of plant specific details in system design, component design and selection, and structural detail.

This means that it is not possible to characterize source term for light-water reactors by a single table of numerical values. How the basic and generic findings could and should be used in the practical aspects of the source term application is still an open question. It is clear, however, that some modifications in the areas of emergency planning and response and probabilistic risk assessment would be justified. It is also clear that the substantial improvement in understanding the fission-product source term from postulated accidents of severe damage to the core leads to many changes in the way such accidents are viewed.

In particular, improved estimates of the source term from different conceivable accidents and from different ways of reacting to them makes it possible to include accident management into broad aspects of operational safety. It is possible to begin to reduce or eliminate conservatisms and non-physical assumptions that were formally made to compensate for the omission of real effects. This would permit nuclear plant designers, operators, and regulators to concentrate their attention even more on practices that can provide real benefit.

October 1985:
International symposium on source term

More than two dozen IAEA Member States and organizations have designated representatives to attend an International Symposium on Source Term Evaluation for Accident Conditions scheduled in the United States from 28 October to 1 November 1985.

The symposium is being organized by IAEA in co-operation with the US Nuclear Regulatory Commission and Battelle Columbus Laboratories and will take place in Columbus, Ohio. As an international forum to review results and conclusions from ongoing re-evaluation and research, the meeting is expected to assist in updating the determination of radionuclide releases during postulated accidents at nuclear power plants.

Major areas of discussion include.

- In-vessel release of radionuclides and generation of aerosols
- Transport and retention in the reactor coolant system
- Ex-vessel release of radionuclides and generation of aerosols
- Retention and behaviour of fission products and aerosols in containment
- Release of fission products from the plant, including engineered safety system performance and evaluation.

The symposium also includes special panel discussions summarizing the current state of knowledge on containment response and loads following accident conditions well beyond design basis accidents, including hydrogen generation and control; and identification of unresolved technical issues and needs for regulatory application.