Nuclear reactor safety—the APS submits its report

Although the study group notes that power reactors have an excellent safety record, the group recommends additional studies beyond the scheduled research programs to achieve the needed level of confidence.

The Study Group on Light-Water Reactor Safety

On 28 April at the Washington meeting of The American Physical Society, the results of a year-long study of light-water-reactor safety were released. The study group, consisting of a dozen physicists, chemists and engineers, was chaired by Harold W. Lewis of the University of California at Santa Barbara; it reported to the Physical Society Council through a Steering–Review Committee consisting of Wolfgang K. H. Panofsky (chairman), Hans A. Bethe, and Victor F. Weisskopf. The study was jointly financed by the former Atomic Energy Commission and the National Science Foundation.

According to Lewis, the results of the study can be summarized succinctly as follows: "We have not found a basis for substantial short-range concern about the safety of light-water reactors, nor, on the other hand, have we found a completely satisfactory quantitative treatment of the important safety issues. For this reason, and especially because we have found that the consequences to human health of a major reactor accident are likely to be more severe than has heretofore been supposed, we have recommended a greatly expanded and somewhat redirected program in reactor safety research, designed to reduce the remaining uncertainties."

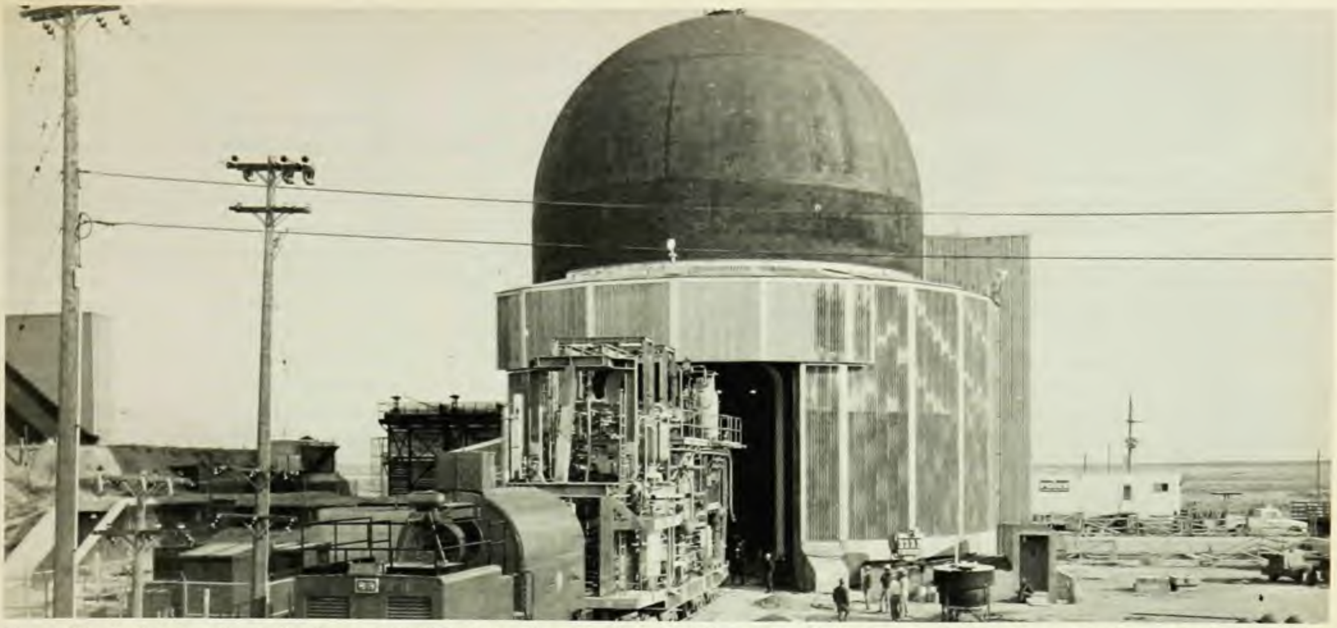
The group confined itself to the technical issues of the safety of light-water reactors typical of commercial practice in the United States now and in the immediate future and did not discuss such equally interesting questions as nuclear safeguards, waste disposal or other aspects of the nuclear fuel cycle. In particular, the group did not review the recently released AEC study WASH-1400 (the Rasmussen study), which was a major effort to estimate the probability of a core-melt accident in a reactor.

PHYSICS TODAY here reproduces the summary section of the report. Discussions, taken from the text of the report, of some of the more important issues are included below. The report is being published in its entirety as a supplement to volume 47 of Reviews of Modern Physics *(for information on how you may obtain a copy* *see page 59**).*

A central issue in the operation of light-water reactors is the prevention of a major release and widespread dispersal of radioactivity, which could have serious consequences to the public. The safety record of light-water reactors to date has been excellent in that there has been no major release of radioactivity. These reactors have been designed with numerous safety features, engineered to prevent foreseeable accidents. These safety features are backed up by other safety features intended to prevent major release of radioactivity in the event of an accident. Moreover, very conscientious efforts have been made in developing the procedures and practices involved in licensing, quality assurance, operation and inspection of these reactors to insure sound construction and safe operation.

In the course of this study, we have not uncovered reasons for substantial short-range concern regarding risk of accidents in light-water reactors. While a complete quantitative assessment of all important aspects of reactor safety—and behavior under unusual circumstances—cannot be made now, we are confident that a much better quantitative evaluation and consequent improvements of the safety situation can be achieved over the next decade if certain aspects of the safety-research program are substantially improved and the results of the research are implemented. Because of the serious potential consequences of a major release of radioactivity and in view of the safety-related technological opportunities that already exist, we believe that there should be a continuing major effort both to improve light-water reactor safety, and to understand and mitigate the consequences of possible accidents. Our recommendations are directed towards these objectives.

The safety philosophy of the nuclear



The mobile test assembly of the Loss-of-Fluid Test Facility enters its containment chamber at the Idaho National Engineering Laboratory. LOFT is a 55-thermal-megawatt reactor built to simulate possible accidents in a 3300-MW(t) representative pressurized-water reactor.

industry has emphasized design that can provide tolerance against malfunctions. This approach has laid a good foundation for reactor safety and it has resulted in reactors designed, constructed and operated for safety, not only under normal operating conditions but also in a wide range of abnormal circumstances. A great deal of research, development and quality control has gone into guaranteeing the integrity of the fuel elements and cladding, the integrity of the enclosing primary system, the general structural soundness of the entire reactor and the ability to control the reactor under both normal and abnormal conditions.

Careful design, construction, operation

Although we have not been able to analyze all of the many possible failure sequences for light-water reactors, one we have studied in detail is the possible failure of the integrity of the primary reactor pressure vessel. We find that reactor vessels are constructed of materials chosen with care and are designed with substantial safety factors. The reactor vessel is subject to careful scrutiny and testing. Based on our study, we believe that catastrophic rupture of the primary pressure vessel is not likely to be an important contributor to accident initiation; this, however, depends on maintaining a strong quality assurance program.

Primary-system piping is also subject to careful scrutiny and testing. The well known cases of cracks in pipes and failures of valves in reactor operation on the one hand reflect deficiencies in fabrication or design; but they demonstrate, on the other hand, the success of the overall safety system and the procedures that identified their existence early enough to prevent more serious consequences. Continued open discussion and analysis of such failures can lead to improvements in safety and provide the data base for a more accurate estimate of the probability of more serious incidents.

These defects underline the ongoing need for the nuclear industry and the regulatory bodies to continue improvement of inspection and test techniques. It is important that licensing and regulation be conducted in such a way as to continue to ensure openness in the quality-assurance program and to provide better-quantified evaluation of the success of the program. We also note that human error on the part of reactor operators appears to initiate or aggravate at least a few incidents of potential safety significance each year. In fact, unless diligence is maintained, quality assurance and human error may well represent a limiting factor in maintaining safe operation.

It is difficult to quantify accurately the probability that any accident-initiating event might occur. Many aspects need to be better understood through experience and research before such calculations are tractable. Although the probabilities of major accidents appear small, their quantification deserves more attention within the reactor safety community than it has received up to now. We did not have the resources to carry out an independent evaluation of this aspect of the recent AEC Reactor Safety Study (draft WASH-1400), but we recognize that the event-tree and fault-tree approaches can have merit in highlighting *relative* strengths and weaknesses of reactor systems, particularly through comparison of different sequences of reactor behavior. However, based on our experience with problems of this nature involving very low probabilities, we do not now have confidence in the calculated absolute values of the probabilities of the various branches of these trees.

We have reservations about the present almost exclusive emphasis in the licensing process on the "design-basis-accident" concept, in which certain highly stylized accidents are used as yardsticks against which the performance of various systems is evaluated. While we agree that analysis of such accidents is an important check upon the general safety of reactor designs, we are concerned that other types of possible accidents may consequently receive insufficient attention in design, construction, licensing and operation.

Primary engineered safety features

In our study we centered much attention on the "engineered safety features." Because these features are not

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used in normal operation but are specifically intended to prevent an abnormal incident from becoming an accident, there is only limited operating experience with them. In addition, because of the complexity of the phenomena involved, these features are very difficult to simulate on a computer or to test in simulated accident conditions. There is therefore a lack of well quantified understanding of the performance of some of these special systems under some severe accident conditions.

One of the most important of the engineered safety features is the fast-acting SCRAM system for shutting down the chain reaction in the event of an emergency. Certain transients that are anticipated to occur from time to time (in pressure, temperature, reactivity) might play an important role in accident initiation. It is very important to shut down the chain reaction during a large transient. While the SCRAM designs as now prescribed appear to us to be highly reliable, not enough is known about the effects of transients in the extremely unlikely event that the reactor does not SCRAM. We believe that insufficient attention has been given to the analysis of transients, although it is encouraging that these areas are now being given intensive study. We are also concerned about transient behavior that might occur simultaneously with a massive electrical failure. While there are redundant off-site power sources, the emergency on-site (diesel) power sources are a recognized weak point.

The emergency core-cooling system is the engineered safety feature that has received the most publicity, attention and research. This system is intended to provide emergency cooling to prevent the reactor fuel from melting or losing structural integrity in the event of a loss of primary-system fluid.

We have no reason to doubt that the emergency core-cooling system will function as designed under most circumstances requiring its use. However, no comprehensive, thoroughly *quantitative* basis now exists for evaluating its performance because of inadequacies in the present data base and calculational codes. In addition, it is not clear that the present approximate calculations—even though based on generally conservative, detailed assumptions—will in all cases yield conservative assessments of the system's performance.

We have examined the AEC reactor-safety-research program intended to re-

Will LOFT scale?

The long-term goal of the AEC research program is to be able to understand and predict those aspects of reactor behavior that are relevant to safety in a convincing quantitative manner. For this purpose, its Division of Reactor Safety Research is developing advanced computer codes that are intended to give a more "realistic" calculation through more accurate treatment of results of the separate-effects tests and more realistic modeling of physical phenomena. When such advanced codes are developed, their predictive ability needs to be tested on a series of hardware-system tests that, it is hoped, will demonstrate their validity in a complete quantitative manner. The currently planned test series is centered on the LOFT (Loss of Fluid Test) facility located at the Idaho National Engineering Laboratory at Idaho Falls, intended to simulate on a scaled basis the conditions of a reactor representative of current large designs.

The LOFT Integral Test Program is the major AEC program to examine the key phenomena of loss-of-cooling accidents and the behavior of the emergency core-cooling system at the system-test level. The LOFT system is a pressurized-water reactor designed to carry out loss-of-cooling accident experiments on a size and scale, 55 MW (thermal), intermediate between a laboratory experiment and that typical of a large reactor, 3300 MW (t). The LOFT system is designed like a small pressurized-water reactor and is constructed on a mobile test assembly, which consists of the pressurized-water-reactor system mounted on a railroad dolly that can be moved between engineering buildings and the special containment (test) chamber, as shown in the photograph on page 39. A cutaway of the LOFT reactor vessel is shown on the right.

The crucial issue is the comparison between LOFT system parameters and those typical of a pressurized-water reactor. The initial pressures and temperatures are typical of the operation of such a reactor. In the Table, the volume distributions and volume/power distributions are compared.

To ensure that the same relative amounts of fluid are available for energy exchange in LOFT as in a large pressurized-water reactor, the scaling criterion chosen is volumetric and volume/power (that is, power density) scaling. [More fundamental scaling studies are also under way.] In addition, scaling of fluid channels and fuel diameter in the core is 1:1. The break area for LOFT is chosen so that time scale for a loss-of-cooling accident should be approximately the same in each case tested. Core-volume/vessel-volume ratios are also made comparable, so that gross core–vessel energy-transfer effects should also be representative. On these bases, the designers of LOFT believe that its behavior will be comparable with a typical pressurized-water reactor. However, surface-to-volume ratios cannot be matched at the same time; one also cannot match hydraulic resistances or the core-area/break-area ratios, and the vessel-volume/system-volume ratio differs by roughly a factor of two. LOFT will be qualitatively representative of pressurized-water reactors in characteristic behavior, but many points are compromises whose effects are hard to estimate at the present state of knowledge.

In view of these scaling uncertainties, the study group believes that it is fair to regard LOFT not as a proof test of the emergency core-cooling behavior of pressurized-water reactors but rather as an important test of the understanding of the various separate effects in interaction with each other, with basic thermodynamic variables and a time scale for loss-of-cooling accidents that will probably be roughly typical of pressurized-water reactors. For example, LOFT affords an excellent chance to examine the extent to which we understand the crucial critical-heat-flux behavior; LOFT will certainly throw light on emergency-core-cooling-bypass phenomena and reflood-heat-transfer behavior so important in determining the temperature history of the core—even though results

Comparison of coolant-system volume distributions and volume/power ratios

Component	Fraction of total volume PWR[1]	LOFT	Volume/power [ft^3/MW (t)] PWR[1]	LOFT
Reactor vessel	0.380	0.366[2]	1.36	1.81[2]
Combined volume, steam generators	0.352	0.252	1.26	1.25
Combined volume, primary coolant pumps	0.026	0.037	0.09	0.18
Pressurizer	0.147	0.125	0.53	0.62
Volume of intact loop(s)	0.355[3]	0.340	1.27	1.68
Volume of ruptured loop, reactor to break	0.118	0.170	0.43	0.82

(1) A four-loop pressurized-water reactor of selected, typical design.
(2) Based on downcomer gap of 2.0 in.
(3) Three of four loops, including the three steam generators and three pumps.

solve these uncertainties. Expanded experimental tests and advanced calculational code development are now under way with the goal of accomplishing a sufficiently quantitative comparison between calculation and experiment so that the technical community can reach consensus on the effectiveness of the emergency core-cooling system. That consensus can only be reached through several years of effort with improved research techniques and more open publication and review of the results. We doubt that a complete quantitative evaluation of this system's effectiveness can be achieved through the present program. We recommend several possible approaches for improvement in the box on page 42.

The last line of defense in preventing or mitigating the release of radioactivity is a further set of engineered safety features designed as a backstop in case of significant failure of the preceding safety features. The greater part of this last safety umbrella is the containment machinery and the building that encloses the entire primary system of the reactor. These containments, which have worked well in controlling routine and minor radioactive emissions, have not yet been subjected to the test of a large-scale controlled or accidental release. More research toward increasing the effectiveness of containment devices, along with more vigorous pursuit of the possibilities for major improvements in their design, would be prudent.

Accident containment, consequences

Although a major release of radioactivity is unlikely, it is important to calculate the types and extent of consequences of releases under various circumstances. We have found that these calculations are very difficult. There are significant uncertainties in nearly every category of potential consequences: immediate deaths, latent cancers and property damage/denial [loss of use through actual damage or potential hazard].

We have made no independent studies of acute effects, estimates of which are particularly dependent upon details of local siting, weather and population, and upon important uncertainties in acute biological effects of radiation. However, for the same releases and the same basic references for the biological effects as taken in Draft WASH-1400, we estimate substantially larger long-term consequences, particularly con-

may not be completely typical of operational pressurized-water reactors.

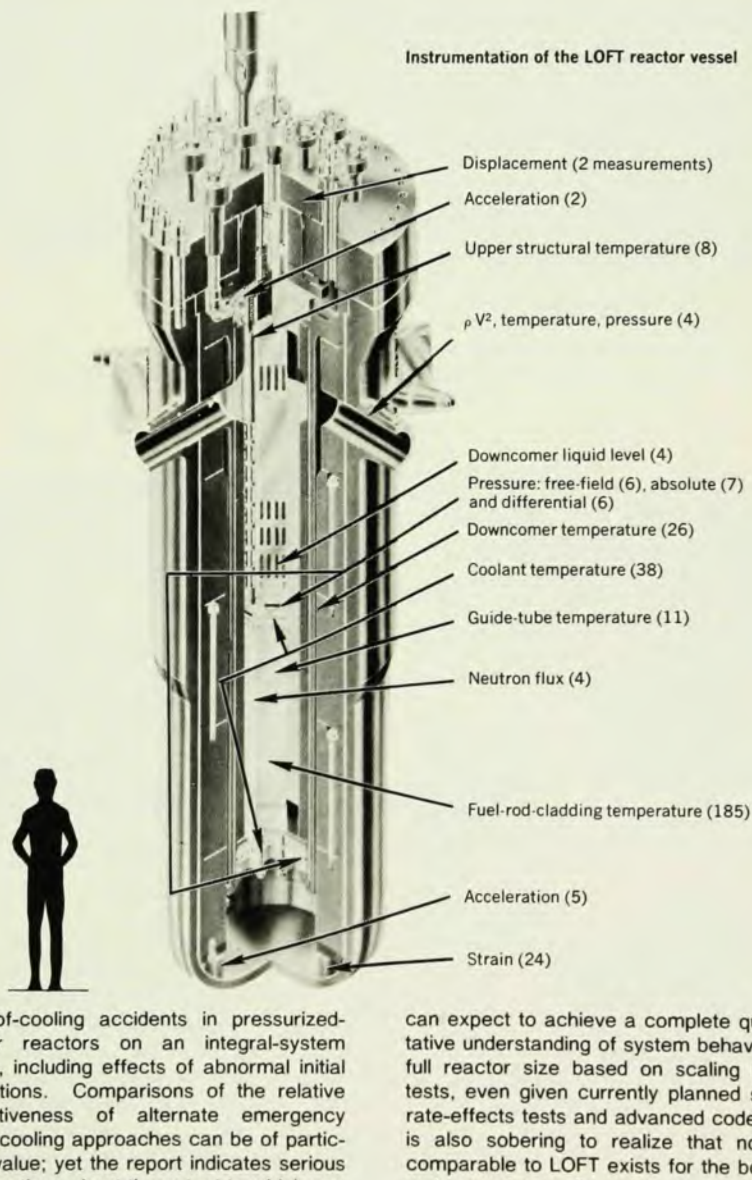
Recommendations. LOFT provides the type of facility in which a variety of accident conditions can be created with tests that can take into account previous test results. The APS group strongly urges that a flexible and responsive test approach be maintained. LOFT also provides a unique test bed for study of the efficacy of alternate approaches to emergency core-cooling (for example, direct injection of fluid into the lower and/or upper plenum) in comparison with the present approach, and they strongly recommend that emphasis be given to examination of such alternate approaches in LOFT.

Because of a tendency to choose typical *normal* reactor conditions as initial conditions for the simulated accident, the authors of the APS report recommend that a wide range of initial conditions, including those corresponding to abnormal light-water reactor conditions, be included in the LOFT series. Such a range is also important to put computer-code calculations to a better test.

Because not all the phenomena can be scaled in the same way, LOFT scaling is necessarily a compromise, which makes the issue of code calculation and verification, with LOFT data, extremely complex. The Study's evaluation of the present lumped-flow calculations is that they are not likely to provide a satisfactory quantitative comparison of the various separate-effects experiments *vis-a-vis* the same effects as they occur in LOFT, although with sufficient adjustment of parameters one may get a reasonable fit. The group believes that better codes, with a more comprehensive treatment of the physics and incorporating a minimum of arbitrary parameters, are a necessity for this job of careful data comparison, particularly for the more problematic job of scaling such results to full reactor conditions.

Overall, the APS study group believes that LOFT can provide a very valuable test of our understanding of the phenomena of

Instrumentation of the LOFT reactor vessel



loss-of-cooling accidents in pressurized-water reactors on an integral-system basis, including effects of abnormal initial conditions. Comparisons of the relative effectiveness of alternate emergency core-cooling approaches can be of particular value; yet the report indicates serious reservations about the extent to which one can expect to achieve a complete quantitative understanding of system behavior at full reactor size based on scaling LOFT tests, even given currently planned separate-effects tests and advanced codes. It is also sobering to realize that nothing comparable to LOFT exists for the boiling-water reactor configuration.

Recommendations for the future of the research program

In the face of past criticisms of the reactor-safety research effort, how do the members of the study group feel about the research program proposed for the future by the Nuclear Regulatory Commission?

The authors of the APS study believe that it is essential to push separate-effects tests on a broader basis at large scale to establish the upper range of our engineering knowledge. With regard to tests of nuclear-fuel behavior, they are concerned about the slow progress in the Power Burst Facility program and recommend a major push to make it successful, to get the necessary large-scale fuel data in the near future. The LOFT instrumentation needs to be upgraded in numbers, range, redundancy and quality if we are to learn what we should from this important test series.

Two very sobering aspects of the study group's assessment are emphasized. First, one must face the fact that important results of the separate-effects tests and of the LOFT series are years away; moreover, the new series of advanced computer codes are only being started and will themselves require a period of years for maturation. Second, having carried out this task, we will be in the position of making experiment-code comparisons only on integral-system effects at a scale well below that typical of operational light-water reactors.

Given the results from planned separate-effects tests, LOFT data and the advanced codes now envisaged, the group is very skeptical about the extent to which one can expect to scale *quantitatively* our understanding of loss-of-cooling accidents in pressurized-water reactors and the transient system behavior to full size and conditions; they have similar, perhaps less firm, reservations about the case of boiling-water reactors: "It is true that we are applying very high standards; however, we believe that the situation warrants such standards."

They therefore conclude that the research program is at a critical juncture. The following avenues, which are not mutually exclusive, are possible for this program:

Option A: Keep reactor-safety research within the scope of the present program, with somewhat expanded and larger-scale separate-effects tests, power-burst facility experiments and LOFT series, plus development of advanced codes.

Option B: Augment the research program by pushing the investigation and development of alternative concepts to cope with loss-of-cooling accidents and other transients of concern.

Option C: Augment the present program by larger integral-system tests combined with a move toward some standardization of light-water reactor designs.

Option D: Augment the research program by placing additional emphasis on better containment, mitigation of consequences and accident recovery.

Option E: Complement the research program through emphasis on remote (and other conservative types of) siting.

Summary of options. Full evaluation of any combination of these or other program alternatives really involves a cost/benefit/risk analysis that is beyond the scope of anything the study group has done—as far as they have seen, it is beyond anything the AEC has done. What they can, and do, say is: "We are not convinced by the available experimental data and code calculations that a complete quantitative assessment of loss-of-cooling accidents and transient behavior can be made at present," and "We are skeptical that, given the results from the presently planned experiments and the advanced codes, the difficult scaling question can be unequivocally resolved." They view it as extremely important that means be found to bring about a convincing resolution of these issues.

Within the presently planned level of resources, the AEC's current program gives evidence of a shift toward limited investigation of parts of Option B (alternative emergency core-cooling systems), D (containment) and E (siting). If the budgets are to be restricted to this level, the study group would advise more emphasis on engineering-hardware-level studies of Options B and D with funds diverted, if necessary, from Option A (present program).

In view of the urgency of the situation the group believes that the safety-research program is underfunded to do the job required of it. Assuming more resources are available, they recommend that Options B and D be fully implemented without slowing down investigation of the current emergency core-cooling concept. Furthermore, to provide an alternative to continuing dependence on a system design for which unequivocal conservatism may not be demonstrated, they recommend that immediate steps be taken to implement detailed analysis and planning for a large-scale integral-system test program, Option C.

cerning land damage/denial and possible latent cancers from exposures to individuals who live in areas that are contaminated below the evacuation thresholds used in Draft WASH-1400. (We were recently informed, however, that substantial revisions are being considered before publication of the final WASH-1400 report.)

The social significance of the long-term consequences depends in part upon the probability of the assumed release, of which we have made no independent assessment. However, the uncertainties in estimates of consequences need to be resolved because they have important implications in reactor design, siting policy and protection against potential sabotage. In analyzing the societal risk–benefit balance of commercial nuclear reactors, one must be able to estimate with reasonable confidence both the probability and the consequences of system failure; research must continue on both.

Considering the great social importance of reactor safety and the large present and future capital investment in light-water reactors, the current funding of safety research is relatively small. We believe that the many technological opportunities for the enhancement of reactor safety warrant the investment of additional funds in safety research.

Major recommendations

Many recommendations are made in the body of the Report. A few of the major ones are summarized here, but in each case the reader is referred to the main text for detailed discussions of the background and rationale. Our major recommendations (not ranked according to their importance) include the following:

1. Human engineering of reactor controls, which might significantly reduce the chance of operator errors, should be improved. We also encourage the automation of more control functions and increased operator training with simulators, especially in accident-simulation mode.
2. Measures should be taken to quantify the effectiveness of the present quality-assurance program, using both the analysis of experience already reported and new measurements on the quality-assurance system.
3. The techniques used in Draft WASH-1400 for the calculation of accident sequences and their probabilities should be:
   - employed to estimate quantitatively whether assumed subsystem-failure data are compatible with the observed individual small accidents;
   - used to provide parametric studies of the effects of phenomena that are ill-understood in the identified sequences;
   - refined so that they can be used for continuing risk assessment on a routine basis with a growing data base of failure data.
4. The Draft-WASH-1400 analysis of accident consequences should be redone with account taken of the modifications discussed in our Report, in order to obtain corrected consequence estimates. The results will help determine the magnitude of the benefits that might be obtained from the introductions of design changes and means of consequence mitigation.
5. The problem of sabotage and its ef-

fect on increasing the risk of radioactivity release should be studied carefully. We have no way of estimating the present likelihood of sabotage; however, we believe that reactor security can be improved and have specific recommendations for studies that go beyond those already under way.

6. The safety margin of the emergency core-cooling system should be quantified and, if necessary, improved through one or more of the following approaches:

- the substitution of more easily analyzable or more effective system concepts;
- a much stronger theoretical and calculational development effort combined with a much improved experimental program, the results of which must be published openly for evaluation by the technical community;
- a series of large-scale experiments along with some standardization of reactors. Detailed planning and analysis for this approach should begin immediately in case it should be decided in the future that it is needed.

There should be increased emphasis on realistic calculations and experiments as opposed to those that merely attempt to set upper limits on the behavior of a reactor in an accident. In view of the number of reactors now operating or being planned, we believe it is important that the reactor-safety-research program quickly take major steps to bring about a convincing resolution of the uncertainties in emergency core-cooling system performance.

7. In the area of safety research, more emphasis should be placed on seeking improvements in containment methods and technology. In particular, controlled venting of the containment building in case of overpressure should be studied. A careful assessment should also be made of the benefits and costs of alternative siting policies, such as remote, underground and nuclear-park siting.

8. There should be more effort to resolve major uncertainties in estimating consequences, including improvement of the biological-effects data base. Techniques for the mitigation of consequences should be developed, especially in connection with the problems of decontamination after a large accident.

9. While we strongly endorse the substantial improvements that have been made in the safety-research programs and in the openness to scrutiny by the technical public in the last two years, additional measures should be taken to continue to improve the research program and techniques and to assure that the results of both experimental and computer-code-development work related to safety are openly published. □