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UNITED STATES GENERAL ACCOUNTING OFFICE

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This Country's Most Expensive Light Water Reactor Safety Test Facility

Consultants' Comments On The Loss-Of-Fluid-Test Facility Technical Program Objectives And Design

ENCLOSURE A

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U.S. GENERAL ACCOUNTING OFFICE

ON THE

REVIEW OF THE NRC/ERDA

LOSS-OF-FLUID TEST FACILITY

BY

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NOVEMBER 15, 1975

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REPORT TO THE GENERAL ACCOUNTING OFFICE ON LOFT QUESTIONS

November 15, 1975

1. Is the current plan not to use LOFT for a meltdown experiment in the best interests of nuclear safety?

In view of the present configuration of the LOFT facility, the program planned for the facility, and the other elements of the Water Reactor Safety Program now underway or planned, the answer is yes, the current plan not to use LOFT for a meltdown experiment is in the best interests of nuclear safety.

LOFT is a flexible, highly instrumented facility, at the largest scale for a full system that is available for thermal-hydraulic testing of ECCS-type phenomena. It will be useful for a variety of water reactor safety research experiments, and is the only facility available for many such experiments. An example is the potential use of the LOFT facility for system transient tests with successively degraded protective and emergency system conditions. These transient tests would not involve simulated pipe breaks but would deal with an assortment of system conditions that are more likely to occur than pipe breaks. In the safety analysis for commercial water-cooled reactors, a variety of system transients must be considered. The plant responses to turbine-generator trips, loss of one or more primary recirculation pumps, and accidental openings of primary or secondary safety and relief valves are among the transients calculated by means of computer codes that model the reactor system. The same reactor system codes, or extended versions of them, are used for the class of accidents referred to as "ATWS" events; anticipated transients without scram. Some checking of these computer codes is possible during startup experiments on commercial plants. The startup experiments, however, are necessarily conducted with the plant protective systems fully operative. It would be very useful to extend these checks of the computer models to transients in which the protective system actions are delayed, or are otherwise degraded below design performance. This testing could be done with LOFT, and the results cross-checked against the startup experiment results as well as used to check the computer models.

Other experiments for which LOFT should be used include loss-of-coolant runs with pressurized fuel (these are already part of the experimental program) and loss-of-coolant experiments with different ECCS configurations. Consideration should be given to LOFT experiments with a longer core, and separately to the possibility of a core of larger diameter. LOFT can also be used for certain pressure-suppression experiments, using the present suppression tank system; these would be useful in checking some of the features of the various computer codes used in analyses of pressure-suppression-type containments.

LOFT has taken many years to reach the present stage of near-startup. Several major redirections in the type of testing to be done have, of course, greatly lengthened the design and construction time of LOFT. Even so, it is clear that facilities of the LOFT type require many years of design and analysis, experiment definition and pre-calculation, instrument development and calibration, and construction and shakedown time. Having finally brought LOFT to a stage where the experiments for which it is intended can soon begin, it would be a tragic mistake to start yet another program redefinition and reorientation cycle.

The LOFT facility should be used for the experiments for which it has been so painfully constructed and reconstructed. The loss-of-coolant experiments, both non-nuclear and nuclear, should go forward without further delay. These experiments, as planned, will take several years. There are bound to be other experiments of the loss-of-coolant type that will be found to be important as a result of the planned experiments, so the ECCS-LOCA programs will certainly be extended beyond the present plans. Beyond the loss-of-coolant experiment, the other types of experiments for which the LOFT facility is well-suited, particularly the system transient experiments in abnormal conditions, should be carried out. These will be of great value in checking predicted reactor behavior in such conditions, and, as noted, have the very useful characteristic that they will apply to more likely reactor conditions than the large pipe break loss-of-coolant type tests. The system transient-abnormal condition experiments will also take several years to perform, and will also generate further experiments of the type. All together, the LOFT facility has 6 to 10 years of useful experimentation ahead of it, in testing work for which the facility is both well-suited and unique.

Meltdown experiments in LOFT are unlikely to produce data that are either useful or that give any new insights. There is first of all the open question as to whether any large fraction of the LOFT core can be made to melt under loss-of-coolant conditions, with stored energy and afterheat as the driving forces, and that is the circumstance of interest in meltdown. The limited melting of a small portion of the central core region would release some fission products, but it is hard to see what quantitative or qualitative uses could be made of such data. Second, if any substantial core melting occurs there is really only one meltdown run available. The resulting contamination in the crowded and complex array of vessels, pipes, wires, and instruments of the apparatus would be impossible to clean up to the degree necessary for the extensive personnel access needed for experiments, and in the relatively short time necessary if repetitive experiments are to be run. LOFT simply is not suitable for meltdown experiments in its present configuration, either from the standpoint of assurance of substantial core melting or with regard to cleanup provisions and possibilities to allow more than one meltdown. Since data from only one run, at best, would be available, there would always be questions about whether those data were truly representative.

The question of using the LOFT facility for meltdown experiments thus involves balancing many years of loss-of-coolant and system transient experiments, for which the facility is well-suited and unique and from which valuable data are virtually guaranteed, against a single meltdown from which the data will be suspect, if substantial core melting occurs, or against a limited series of small local meltdowns from which the data will be even less meaningful. The choice is clear: any contemplation of meltdown experiments should be held for the very end of the useful life of the LOFT facility for other experiments, and even then is unlikely to be a productive venture compared to the additional cost and difficulty of decommissioning and mothballing the facility.

A comment on what should be done about meltdown phenomena, from the standpoint of any needed safety research, is in order. In view of the results of the Rasmussen Report (WASH-1400) that core meltdown may occur, from one cause or another, with a higher probability that had previously been estimated, experimental and analytical studies of meltdown phenomena are certainly needed. This area of reactor safety research has been largely neglected until recently, on the basis that all available resources should be devoted to methods of avoiding meltdown. As we start now to devise research programs in the meltdown area, careful attention must be given to the kind of information that is needed and to the practicability of experiments in these difficult phenomena. The need is for better information upon which to base consequence modeling of the kind done in the Rasmussen studies. It is not necessary to be able to compute in detail all of the aspects of a meltdown, but rather to be better able to define and bound the phenomena that might occur. The limiting conditions for steam explosions, overall heat transfer characteristics of molten core materials and core debris beds, the general nature of molten core-concrete interactions, and limiting release fractions for fissions product species are the sorts of information needed. These aspects of core meltdown are better studied in separate effects experiments than in integral tests. Any decision on integral testing at any substantial scale could well await some results from the individual effects experiments, and may not be needed at all.

The present Water Reactor Safety Program includes a modest amount of core meltdown separate effects work. These program elements provide a start on the needed research, but the effort should be increased from present (FY 1976) half-million dollar per year level to several millions per year.

2. Should LOFT be used on a timely basis to study the means of retaining molten cores and measuring the consequences of steam explosions and radioactive releases resulting from a meltdown?

The answer is no, again in view of the configuration of the LOFT facility and of other elements in the Water Reactor Safety Program. The considerations leading to this conclusion follow, in part, from those of Question 1.

The LOFT facility is simply not suited for experiments with "core-catcher" systems or for steam explosions. As noted previously, it is not at all clear that a substantial core meltdown in loss-of-coolant conditions can be achieved with LOFT. The first essential of the proposed experiments, a molten core mass, is thus probably not available in LOFT in a prototypical loss-of-coolant configuration.

The LOFT core is small and is located in a long, narrow vessel, so that the lower plenum region is too small for installation of in-vessel core-catcher arrangements and the associated instrumentation. The vessel itself is embedded in tightly-packed and complex piping and cabling on the MTA dolly. There is little space available for ex-vessel core-catcher arrangements beneath the dolly, and what space exists would be hard to work in, considering the array of instruments and piping conditions that would be needed. Cleanup and decontamination after an experiment would be difficult at best, and probably impractical, as noted in the answer to Question 1. Much the same remarks apply to steam explosion experiment possibilities.

Rebuilding the LOFT facility for core-catcher and steam explosion experiments is certainly possible, but would be expensive and time-consuming. A new dolly, with a new reactor system unit, would likely be required. Such a course could be considered as a follow-on line of work, after all of the planned LOFT programs are carried out and after those additional experiments for which LOFT is well-suited have been done. However, in view of the long time until the facility would be available for core-catcher and steam explosion work, it would be better to deal with these matters in separate facilities, as is now being done. The data from the several separate effects programs on molten core interactions, steam explosions, fission product behavior, and heat transfer are all applicable to commercial reactors to about the same extent as data from a modified LOFT test would be. The greater experimental control and measurement capability in the separate effects testing compensate in large measure for the larger scale, integral test aspects of a modified LOFT. Further, for these effects, unlike the ECCS tests, a full-system type of test is really not needed. That is, in studying core-catcher arrangements it matters very little whether there is a complete primary system and ECCS in the facility: what counts is the molten core, its immediate environment, and the core-catcher system.

3. Will the small scale LOFT result in experimental data, including the phenomena associated with a core meltdown, that is applicable to the large commerical reactors? Is a larger LOFT type test facility needed?

First, LOFT will certainly yield experimental data that is applicable to large commercial reactors, largely through the ECCS performance computer codes, as detailed below. Next, LOFT will not yield data on meltdown phenomena, since meltdown is not contemplated in the experimental program. There are some possibilities for incipient meltdown experiments, however, as discussed below. Lastly, a larger LOFT type facility is not needed, for reasons summarized in the following discussion.

The LOFT experiments planned in the current program plan are primarily "checking" experiments for the complex computer codes used to detail ECCS performance in large reactors. The computer codes contain various models of fuel rod heat transfer, fluid behavior in the several parts of the reactor system, pump dynamic impedance to two-phase fluid flow, and the action of the several ECCS components. A series of separate effects tests are used to calibrate the computer models, and indeed to guide the construction of the models. For most of the phenomena involved, the separate effects tests provide more carefully controlled and better measured conditions from which to calibrate the computer models. LOFT experiments provide the important feature of combining the various phenomena of loss-of-coolant accident with ECCS action on a system basis that is generally representative of the large reactor systems. The LOFT experiments should provide a check on the way in which the computer codes combine the various models. Further, if there are interactions of the physical phenomena that have not been properly accounted for in the computer modeling, this should become apparent from comparison of LOFT data with the computer predictions. It is a fair guess that the LOFT experiments will lead to adjustments in some portions of the computer codes. In these ways the LOFT data is applicable to large commercial reactors.

There is an implication in some of the discussion of the LOFT program that LOFT is a proof-test for large reactor ECCS performance, that it is, or should be directly applicable to large reactors, and that LOFT will show directly whether large reactor ECCS will "succeed" or "fail" in a loss-of-coolant accident. The LOFT experiments are simply not of that nature and the LOFT facility is incapable of producing results of those sorts. LOFT is a test rig, the best in existence for loss-of-coolant-type experiments, and there will be a great number of experiments run with it over a long period of time. None of these experiments will "succeed" or "fail". All of them will provide data of some sort that will be useful for one purpose or another: some of it undoubtedly will show deficiencies in the LOFT apparatus itself, and will be the basis for improvements in LOFT components and instrumentation.

LOFT has an important place in the array of water safety experiments, and if there were no such full-system tests in the program it would be a serious deficiency. But LOFT, in and of itself, is not the definitive water reactor safety facility, and the results from the LOFT experiments, in and of themselves, are not going to be the definitive water reactor safety results. Nor were they ever intended to be. There is, in fact, considerable doubt that there can ever be a definitive set of reactor safety experiments, in the sense that such a set of experiments would settle all arguments about reactor safety. The LOFT experiments will certainly improve our knowledge of the phenomena to be tested, and thereby improve our assurance of water reactor safety, but they are unlikely to end the debate over water reactor safety.

The matter of meltdown experiments in LOFT has been discussed in connection with the previous two questions. In brief, LOFT is not suited to meltdown experiments in the present configuration and should not be used for that purpose in view of the value of the facility for loss-of-coolant and system transient testing. It is possible to develop some information about the early stages of accident conditions that would lead to meltdown by a series of runs with successively greater degradation of ECCS performance. For example, accumulator or pumped injection of emergency cooling water could be delayed by larger and larger time intervals in a series of runs until some fuel failures (small leaks in fuel rod cladding) developed. Such experiments would not lead to excessive or unrecoverable contamination conditions, and, in fact, have been considered by the LOFT group. The information developed would be useful in checking some of the margins to ECCS failure in current ECCS designs, and in extending the range of data useful for checking computer codes. This sort of experiment, however, should not be confused with meltdown experiments.

On the question of larger integral test facility, it is fair to say that in the best of all possible worlds it would be nice to have a "LOFT-II" facility intermediate between LOFT and the large commerical reactors. Such a facility would provide an additional set of data points to improve the computer codes and to follow the scaling up of all the phenomena involved in a loss-of-coolant accident toward full commercial size. But considering the time, manpower, resources, and cost that would be involved, it is a luxury we can well do without. In language dear to budget managers, it would not be cost-effective. A much more useful enterprise is to continue the thrust of the current Water Reactor Safety Program toward large-scale separate effects facilities and tests. The Plenum Fill Experiments is one example of this type of effort and the on-going FLECHT-SET program is another.

4. Should licensing of commerical reactors be modified in any way pending the results of the LOFT experiments or of experiments on a larger facility?

The answer is an emphatic no. Here again there is the implication that LOFT, or some similar but larger facility, is going to produce a single definitive test that will "succeed" or "fail" and that the safety of water reactors hangs on that result. As noted previously, the LOFT experiments are not of that character, nor would the experiments done with a larger facility be of that character.

There is a background of knowledge about water reactor safety that must be kept in mind in dealing with the question posed here. The present licensing basis for nuclear plants puts heavy emphasis on careful and high quality design and construction for reactor elements important to safety, and requires full redundancy in all safety equipment. The occurrence of piping flaws of the sort that lead to pipe breaks without prior warning by leakage is relatively rare in industrial piping built to good engineering practice, but of generally lower quality standards than nuclear piping. Estimates of the probability of nuclear piping breaks without prior warnings by leakage are based on industrial plant data and run from about one chance in one thousand per plant-year for small piping down to one chance in ten thousand to one hundred thousand per plant-year for the largest pipes in a reactor system. (The probabilities of vessel failure of a significant nature are much smaller, in the range of one chance in ten million or so per plant-year.) The events of concern for ECCS performance are thus in themselves of quite low probability.

In turn, the chances of successful ECCS performance are good, taking into account the possibilities for both equipment failure and design deficiencies. On the one hand, the redundancy in both power sources and components in ECCS greatly reduces the vulnerability to equipment failures. On the other, the considerable effort and argument that has gone into ECCS designs and performance calculations over the years means that the chance we have overlooked some basic aspect of the phenomena involved is small. This is not the say that there are no questions left in the ECCS area, or that the system performance can be guaranteed in all circumstances, but rather to point up the fact that, as the American Physical Society Study Group on Light Water Reactor Safety put it, "We have no reason to doubt that the ECCS will function as designed in most circumstances requiring its use." Overall, I find the Rasmussen study result of a one or two percent failure rate of ECCS on demand to be a reasonable one. Thus, the basic piping failure probabilties noted above are reduced by a factor of fifty to one hundred to obtain the probability of an unprotected loss-of-coolant accident that would lead to meltdown of the core.

Finally, the consequences of a core meltdown are shown by the Rasmussen study to be rather limited in the great majority of cases, when judged against the consequences of other major industrial or transportation accidents of comparable probability. Even the ultimate reactor accident was found in the Rasmussen study to have consequences within the range of very large natural disasters and industrial transportation accidents, and to have a substantially smaller change of occurring than these events.

The conclusion to be drawn here is that power reactors designed, constructed, and operated on the present licensing basis have a better public safety aspect, at least as far as loss-of-coolant accident matters are concerned, than much of the rest of our technological paraphernalia. There is, consequently, no reasonable basis for modifying the present licensing practice for commerical reactors to awaid LOFT experiment results.

At the same time, there is every reason to get on with the LOFT experiments, and with other reactor safety research, to improve both our understanding of accident phenomena and our methods of ameliorating and containing accidents. While the present safety level and licensing basis for commercial reactors are certainly adequate for the current generation of plants, and as noted are much better than for most other major technologies, there are many more plants to be built in future years. It is appropriate to improve the safety level to compensate for the greater number of plants and to reduce as far as practical the residual chance of an accident with substantial off-site effects. It is clear that there is a greatly increased public interest in making all our technologies as risk-free as practical, and wide public support for the cost and effort to do so.

5. Present LOFT plans call for the use of a pressure suppression system in lieu of blowdown to the containment. Because of this, there are no planned tests of the containment's ability to control fission product activity. Do you believe that such a test of the containment would be appropriate for LOFT?

No, because such a test would end the use of the LOFT facility for other, more important work, because it would not be a very meaningful test, and because it would increase the cost and difficulty of decommissioning the facility. These points are elaborated in the following discussion.

The pressure suppression system that has been added recently to the LOFT facility is needed to control the back-pressure on the LOFT reactor system after a blowdown, to provide post-blowdown system pressure conditions representative of those in a commercial reactor. It is an essential feature of the ECCS tests in LOFT. It has the further desirable feature of controlling the minor radioactive releases that may occur during the nuclear tests, and thus avoids the need to decontaminate the facility to allow personnel access after a run.

The possible containment-related experiments that might be done relate first to the ability of the containment to stand the resulting internal pressure and the effectiveness of the containment sprays to condense steam and reduce the pressure, and second to the effectiveness of the sprays to reduce the burden of gaseous and air-borne fission products released into the containment.

With regard to pressure retention and reduction tests, the LOFT facility is poorly suited to this work. The containment is too large relative to the contained fluid volume in the reactor system to provide any meaningful test of either pressure retention or pressure reduction. The containment was designed for a much larger reactor system volume, dating back to long-since cancelled plans for the system. There is a useful ratio in containment work by which to judge these matters. This is the ratio of the containment free volume, V, to the mass of fluid in the reactor system, M. The ratio V/M in large commercial plants of the PWR-dry containment type is typically 4 ft3/lb. The ratio for LOFT in its present configuration is about 20 ft3/lb. The result is that containment pressure resulting from a LOFT blowdown would be about 8 psig, compared to the much larger containment design pressure of 35 psig. A blowdown to the containment would hardly exercise the pressure-retaining capability at all. With regard to pressure reduction, the starting pressure of 8 psig is simply too small for the subsequent reduction by the containment sprays to have any meaning for large commercial plants.

A better case can be made for the fission product control by spray test, in that it would be the largest scale test of this type to be done. But the price of this test would be a substantial core meltdown, to produce a meaningful quantity of fission products. Running such tests with the small radioactive releases that might occur from non-meltdown LOFT tests of the type planned would not yield much of interest. As noted in discussion of previous questions, a meltdown test would have to be the last use of the LOFT facility and would have to be put off for many years while the loss-of-coolant and system transient testing was done. Even as a last hurrah from LOFT, the meltdown run would be of limited value for fission product control data. Only one run could be made, and in this kind of testing one needs many runs, with varied parameters, to develop the functional dependence of the system conditions. There is the further aspect of the increased cost and difficulty of decommissioning and mothballing a thoroughly contaminated plant.

6. LOFT is a small PWR which has been scaled so the test results will simulate the anticipated effects of LOCA's on large PWR's. Will the LOFT results be applicable to BWR's? Do you believe a LOFT experiment using a BWR mobile test assembly is needed?

LOFT results will be applicable to BWR's in much the same sense that they are applicable to PWR's, that is, through the checking and improvement of the various computer codes used for accident and ECCS performance calculations. LOFT results will not, of course, yield any information relative to the spray cooling used as part of the ECCS in BWR's. On balance, a BWR LOFT is probably not needed, and the spray cooling aspects of BWR's can be covered by the planned full-size fuel bundle tests.

The loss-of-coolant accident conditions in BWR's are somewhat different from those in PWR's, and are free of some of the complications that make PWR ECCS performance difficult to calculate. The most important example is the steam binding possibility in PWR's, in which rather delicate pressure balances of the hydraulic head of injected cooling water against the frictional flow resistances of steam and entrained water in the broken and unbroken steam generator-pump loops determine the rate of rise of cooling water in the core. These aspects of PWR ECCS performance make full-system testing an essential part of understanding and calculating correctly the PWR case. These effects are happily absent in BWR's and full-system testing of the BWR configuration is correspondingly less interesting or important.

Further, the BWR ECCS are more diversified, using both top-of-core sprays and bottom flooding to cool the core. The flooding part of the ECCS in BWR's is the dominant element in meeting current licensing requirements with regard to maximum allowed fuel cladding temperature, but in a real accident, where avoiding substantial core melting is the goal, the spray action is an important back-up.

LOFT results will be useful for BWR computer code development since many of the thermal-hydraulic phenomena are similar. Blowdown test results at the larger scale of LOFT (compared to other blowdown tests) will be applicable to BWR's, for instance, even though the BWR blowdown starts at saturated conditions. Similarly, fuel heating and cooling results from LOFT nuclear tests can be translated to some aspects of the corresponding effects of BWR's.

The spray action in cooling a BWR core, of course, has no counterpart in LOFT, and this aspect of BWR ECCS performance is calculated on the basis of spray cooling tests on electrically-heated, simulated BWR fuel bundles. An extension of these tests, funded by NRC and EPRI, is now planned. The test rig will involve an electrically-heated full size BWR fuel assembly. The tests will cover flooding action as well as spray cooling. This test series should cover the spray effects adequately. In view of this planned testing, and of the simpler ECCS problem in BWR's, the cost and time required for a new mobile test assembly with a BWR system to be run at the LOFT facility does not seem worthwhile. There is the further consideration that results from such a LOFT-BWR rig either would not be available for many years, or the PWR testing that should be done in LOFT would have to be greatly restricted. Neither propect is very encouraging.

BIOGRAPHICAL INFORMATION

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BY

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RESPONSES TO QUESTIONS ON LOFT

by N. J. Palladino

November 24, 1975

This report responds to a series of specific questions about the Loss-of-Fluid Test (LOFT) facility submitted by the U.S. Senate Committee on Government Operations to the General Accounting Office for consideration by five individual consultants, myself included among them.

I have prefaced my responses with several general comments which may provide helpful background.

A. GENERAL COMMENTS

1. Risks involved in nuclear power-plant accidents

In evaluating the safety of nuclear power plants, as in evaluating the safety of any large devices or structures, many different operating situations and postulated accident conditions must be examined and analyzed. In general, each situation will have a different probability of occurrence and will lead to different consequences. For a nuclear plant to be acceptable, normal operating conditions must be accommodated with little risk to the plant, the operating personnel, and the public; under accident conditions attention must be given primarily to protecting the public. The general philosophy that has been followed is that postulated accidents leading to severe consequences to the public must have a very low probability of occurring and accidents with a high probability of occurring must have very small consequences. 1/ The product of consequences per accident times the probability (or frequency) of occurrence represents the public exposure risk per unit time, and this risk must be acceptably low. 2/

The Rasmussen Study, WASH-1400, shows that the risk to the public from possible reactor accidents from 100 operating water-cooled reactors is much smaller than from any other man-made devices and many natural events, regardless of whether measured in terms of human fatalities per year or dollars of property loss per year. 3/ This conclusion is true even for the highly improbably Loss-of-Coolant-Flow Accident (LOCA) and for the more highly improbably core meltdown following a LOCA. This study confirms that in dealing with LOCAs and their consequences, we are concerned with highly improbable accidents with risks to the public well below those from other sources.

2. Why do more safety research?

Having estimated that the risk to the public from nuclear accidents is considerably smaller than risks from other sources, why should more safety research on water-cooled reactors be done? Aside from learning for the sake of learning, six purposes might be offered.

1) To determine if the accident conditions postulated and the ensuing consequences have been reasonably bounded in the Rasmussen study.

2) If not reasonably bounded, to determine the bounds.

3) If reasonably bounded, to establish the degree of margin, which in turn might be used for developing simpler and more economic designs.

4) To determine conditions needed to assure that the probabilities of component failure used in the Rasmussen report are achieved in the design and operation of nuclear power plants.

5) To determine if any new and unforeseen phenomena might have to be considered.

6) To learn how to reduce the risk to the public even further.

3. What areas of research are most important?

Of these purposes, numbers 1, 2, and 4 appear to be the most important at present. If, for example, purpose number 1 is achieved, there is little value in working on purpose number 6 for the nuclear field; if greater protection of public safety is sought, research in other non-nuclear areas would appear more fruitful. Achievement of purpose number 1 would also probably achieve purpose number 5; I believe that there is little likelihood of uncovering new phenomena over and above those already identified, but uncertainties about the magnitudes and importance of the currently identified phenomena during various accidents can be clarified. Also if purpose number 1 is achieved, purpose number 3 can be pursued on a more liesurely basis.

With regard to purposes 1 and 2, it is expected that there will be continuing assessment and reassessment by a variety of technical groups on the degree to which reasonable assumptions have been made in the Rasmussen study, such as presented in references (4) and (5). Detailed assessments may change as more data are obtained, but the Rasmussen report shows such a large difference between risks from nuclear plant accidents and risks from other types of accidents that the assumptions used in the Rasmussen report can change significantly without invalidating the conclusion that the relative risks involved in building and operating water-cooled nuclear reactors are low. I do not believe that this conclusion will change.

But work on bounding accident consequences, based on assumed probabilities, must not be done at the expense of work for achieving purpose number 4 which must be achieved to make the assumed probabilities valid. One of my major concerns is that the great attention being given to evaluating highly improbable accidents may very well divert attention from means for avoiding such accidents. Continuing attention must be given to developing and using knowledge about methods for keeping component failure probabilities at the assumed low level.

4. Development and testing of system codes

To achieve purposes 1, 2, and 4, primary reliance must be placed on separate-effects experiments made by appropriate specialists with facilities in which reliable measurements can be made. The phenomena involved must be then properly characterized by computational models for use in concert with other computational models to predict integrated system results via system codes. It is also necessary to confirm the integrated system codes by system tests; but this can be done only if the scenario during the test can be controlled to replicate the scenario which the computational model is to predict.

If attempts are made to test system codes on more complicated systems in which events cannot be controlled, any agreement between the results of the calculational model and the results of the given integrated test would be largely fortuitous. The reasons are as follows. At every step in the description of an accident, one is faced with selecting one of several ensuing alternative possibilities. The assumption made, with regard to which alternative applies, can significantly affect the prediction of the model even if the assumed ensuring phenomena can be well described. In actuality, at each step, the events themselves can be probabilistic in nature. (For example slug flow as opposed to bubble flow in a fuel channel, or variations in pipe flows due to burrs, etc.) If the uncertainties are large, as they would tend to be in complicated systems, the calculational model cannot give the right answer, especially if the spectrum of probabilities for various next-step scenarios is broad, that is if several possible events are about equally probable at any one time under the given conditions. Hence complicated systems where the scenario cannot be controlled are not suitable for testing system calculational codes.

System codes can be properly tested only on systems in which the scenario during the test can be controlled to replicate the scenario which is to be predicted.

5. <u>How can system codes be useful?</u>

If the foregoing conclusion is accepted, one must ask, "How can the codes be useful in estimating the consequences of a real accident?" To deal with this question it is helpful to consider two separate types of accidents, those involving core meltdown and those not involving core meltdown.

If the accident considered involves core meltdown, even "confirmed" computational models cannot predict, with any degree of precision, the course of a reactor plant accident, even one with a specified initiating event. When the core melts, the accident can take on any one of a number of significantly different paths, and thus the problem becomes computationally indeterminate.

This is the situation if none of the Emergency Core Cooling Systems (ECCS) is assumed to work. Trying to predict the course of nuclear plant accidents where none of the safety features are working is not too unlike trying to predict the course of an airplane crash when all power is lost. No one single model can cover all the possibilities because of uncertainties about what the course of the crash will be.

The situation is quite different if core melt is avoided; the range of uncertainties in the course of the accident in this case is much easier to bound. Thus, in this case, predictive models can be useful in sizing ECCS equipment and predicting its effectiveness if due allowance is made both for uncertainties in the calculational model and uncertainties in the course of the LOCA accident. Even for this situation, however, predictions cannot be very precise. Efforts to improve the calculational models and the tests used to confirm them can help reduce the uncertainty in the calculational model, but little can be done to reduce the uncertainties about the course of the accident itself.

In summary, separate-effects experiments are needed to study phenomena involved in nuclear plant accidents, and controlled systems tests are needed to confirm systems calculational codes. Such codes are useful for sizing ECCS equipment and predicting its effectiveness within reasonable error bands; but one cannot predict with any precision the course of events in a nuclear accident if the ECCS equipment is assumed not to work. The best that can be hoped for in this case is to place upper bounds on the consequences of the accidents based on various assumptions regarding the course of the accident.

B. RESPONSES TO QUESTIONS

1. Is the current plan not to use LOFT for a meltdown experiment in the best interests of nuclear safety?

This question is really a two-part question, as aptly recognized by Dr. Herbert J. C. Kouts, Director, Office of Nuclear Regulatory Research, Nuclear Regulatory Commission.6/

(a) Do we really need meltdown experiments?

(b) If such experiments are needed, should they include a meltdown of a LOFT core?

With regard to question (a), the need for meltdown experiments, one must distinguish between a test (or perhaps more than one test) involving massive meltdown of a core in a prototypical reactor and phenomenological experiments to study and evalaute the characteristics of core meltdown, including initiation mechanisms, propagation characteristics, energetics involved, and the interaction of molten UO2 with water as well as with steel, concrete, and other materials.

I do not believe that a test involving the complete meltdown of a core in a reactor plant is a useful undertaking. Such a test, even with the best practical instrumentation would provide little insight about what might happen in a full-sized PWR unless it was conducted in a large plant; even then the information gained would be applicable only to the particular situation involved, i.e. the type of event that led to the meltdown; whether or not water is left or introduced into the vessel during the course of the meltdown, and how much; to what extent various containment safety features were operating, etc. In essence even in a full-scale meltdown test, the test becomes a demonstration of what could happen under a given set of circumstances, without the ability to learn what would happen under a different set of circumstances. Furthermore the probabilistic nature of some of the events during the accident would not assure that the course of the accident would be the same in another test on another identical system.

If one makes a meltdown test using a less-than-full size core, such as the LOFT core, additional questions arise involving questions of scale, not only the scale of the core but of related components and systems as well.

However, I do believe that it is essential to conduct meltdown experiments of phenomenological nature to understand the characteristics of the events involved and to provide a sounder technological basis for judgement in predicting the consequences of a large core meltdown under a wide variety of postulated circumstances.

As indicated in Part A of this report, such experiments should be separate-effects experiments done by appropriate specialists in properly designed facilities. Experiments carried on in this way can be controlled to yield results under a wide range of conditions for use in bounding the consequences of a variety of possible nuclear plant accidents.

Such a range of results could not be obtained from any single core meltdown test; furthermore the events in a core meltdown in an integrated system test could not be sufficiently well measured and characterized to be useful for evaluation of other nuclear plant accidents.

With regard to question (b), I do not believe that a meltdown of a LOFT core should be made. In addition to the reasons given above, recognition must be given to the many atypical characteristics of LOFT, such as physical separation of the reactor vessel from the containment floor because of the plant's "mobile" nature and the non-typical type of PWR containment involved. 7/ 8/ 9/

2. <u>Should LOFT be used on a timely basis to study the means of retaining molten cores and measuring the consequences of steam explosions and radioactive releases resulting from a meltdown?</u>

Experiments to study the means for retaining molten cores and measuring the consequences of steam explosions and radioactive releases from a meltdown are needed, but these should not be done on LOFT, in part because LOFT has not been designed and constructed for this purpose, and in part because, as pointed out earlier, such phenomena cannot be well studied in an integrated test. These phenomena should be studied as separate effects in appropriately designed experiments which can yield qualitative and quantitative understanding under a variety of postulated conditions. A few such tests have been done or are underway 10/ 11/; others are planned. But based on the information provided to the Advisory Committee on Reactor Safeguards (ACRS) by Dr. L. S. Tong, Assistant Director for Light Water Reactor Safety Research, NRC, the amount of effort on such separate-effects experiments is far too little at present. Dr. Tong reported a yearly expenditure of $500,000 per year on such work. 12/ Inasmuch as questions about core meltdown, resulting radioactivity releases, and possible steam explosions are the bases for some of the largest areas of uncertainty in the Rasmussen report, a ten-fold increase in this effort is needed so that the needed information can be obtained in the next 5 to 8 years.

While systems such as LOFT are not well suited for phenomenological experiments, one might argue that some such system could be used to confirm the efficacy of a device to retain a molten core, often referred to as core-catcher, under accident conditions. But the LOFT facility as presently designed is not suited for this purpose. It is so compact that there is not enough room to install a core-catcher. The modifications needed to incorporate such a device would be extensive, costly, and time consuming. The LOFT system would have to be almost completely redesigned, especially if an in-vessel core-catcher is contemplated. Furthermore, appropriate redesign of LOFT could not be started until far more data from separate-effects experiments have been obtained; the results of such experiments would be needed to design both the core-catcher and the test itself. It is not believed that this would be worthwhile. Only if the separate-effects experiments disclose the need for a core-catcher, to make the Rasmussen report low-risk conclusions valid, should a core-catcher test of this magnitude be contemplated. Based on the assumptions made about core-meltdown in the Rasmussen report and the large margin for error in the low-risk results, I do not see the need for such a test.

3. <u>Will the small scale LOFT result in experimental data, including the phenomena associated with a core meltdown, that is applicable to the large commercial reactors?</u>

LOFT will result in data that are applicable to large commercial reactors, but such data will not include data on core meltdown, because meltdown tests are not planned for LOFT. Furthermore, LOFT is not suited to providing experimental data about core meltdown for the reasons given earlier.

LOFT is an integrated test facility for evaluating system-type computational codes to predict the course of a LOCA. These codes are developed by coupling calculational models derived from separate-effects experiments. This development involves making important assumptions about the way phenomena interact. The adequacy of the codes depend on both the adequacy of these assumptions and the accuracy of the separate-effects models.

To help calibrate the models, in 1973 the AEC developed standard problems whereby code predictions and results of tests performed on progressively more complicated and progressively larger systems could be compared. This program will include predictions of results from a variety of tests on the Semiscale Mod-1 system, which is an electrically-heated small-scale model of a PWR, as well as from a variety of tests on the LOFT facility, with and without nuclear heat. The comparison of predictions and tests results at various steps in the program will be valuable in identifying deficiencies in these codes and indicating where adjustments are needed. The need for such adjustments became evident early in the standard-problems program. 13/

The LOFT tests will provide the opportunity to check these codes in a system involving a nuclear core, where the heat production patterns and flow problems are more complicated than in the Semiscale loop, and where facilities for emergency core cooling injection exist. A wide variety of LOFT tests will be needed to explore the applicability of the codes under various conditions and to test adjustments found necessary in the codes. The directions which such adjustments must take will be obtained not only from the LOFT tests themselves but also from continued work on separate-effects experiments. The LOFT tests can also be useful in checking these codes under various degree of degradation of emergency core cooling systems.

But as indicated in Part A of this report, the comparison of code predictions and LOFT test results cannot be expected to be precise both because of irreducible uncertainties in the computational models and inevitable variations in the course of each test. However within these bounds, the LOFT data can be used to check and refine the system codes; greater reliance can then be placed on them for sizing ECCS equipment and predicting its effectiveness within reasonable error bands for large commercial nuclear plants.

The chief question that will remain regarding the applicability of these codes to commercial plants will be that of scale-up to systems of larger size. If LOFT were a demonstration of the response of a large PWR during a LOCA, the question of scaling would be indeed quite significant. But scaling, though difficult 14/, is not as crucial in confirming system codes as long as the processes involved in the different size plants are the same and the phenomena involved in the processes are well characterized; the approach being taken in the safety research program, of which LOFT is but one part, though an important part, will satisfy these conditions.

If building and testing a larger LOFT facility were simple and not costly, one might consider undertaking such a task to reduce the questions about scale. But in view of the costs and efforts involved and the low return in safety that would be obtained, based on the risks reported from the Rasmussen study, such an undertaking is not recommended.

4. Should licensing of commercial reactors be modified in any way pending the results of LOFT experiments or of experiments on a larger facility?

I do not believe that licensing of commercial reactors should be modified pending the completion of the LOFT experiments. The reason for this is first that the LOFT program is concerned with an exceedingly improbable type of postulated accident, namely the LOCA, and second it is concerned with confirming the efficacy of various consequence limiting devices which on large plants exist both in redundant and diverse form. The probability of fatally injuring large numbers of people because of both a LOCA and failure of all consequences-limiting devices in a single plant is considerably smaller than the probability of fatally injuring a comparable number of people from any other single natural or man-made event. By the time the number of reactors becomes large enough to significantly increase the probability of affecting many people, the LOFT data should be available to confirm the effectiveness of various ECCS provisions and related consequence limiting devices.

It must also be pointed out that licensing of nuclear power plants involves more than the evaluation of accident probabilities and accident consequences. It also involves review of measures to help assure the prevention of accidents. As a matter of fact it is important that the emphasis placed on evaluating the consequences of accidents does not divert attention from the means that must be taken to avoid them.

Prevention of accidents is basic to nuclear safety. All structures, systems, and components important to safety must be designed, built, and operated so that the probability of failure is very small. In turn, to assure a low probability of failure requires: 15/

1) Conservative bases for design (for example the most severe earthquakes, tornados, hurricanes, and floods that can be reasonably postulated),

2) An effective quality assurance program for all components, and

3) The use of redundancy and, where practical, diversity in the protective systems so that no single fault can produce failure of the system.

In the protective systems, attention must be given to preventing common-mode, or systematic failures. To reduce common-mode failures, the designer must resort to diversity (the ability to perform a function in a different way). Diversity in protective systems can be applied to instruments for measuring process variables (signal diversity), to equipment for performing a given function (equipment diversity), and to devices for taking corrective action (activator diversity). 16/

Engineering safety features also involve the use of redundancy and diversity. To be worthy of consideration, engineered safety features must be carefully designed, constructed, and installed; they must also be equipped with adequate auxiliary power and continuously maintained in working order.

Achieving safety begins with the design process and continues through manufacturing of components as well as construction, check out, start-up, and operation of the plant. Attention to these items is an important part of the licensing process. It is this effort to prevent accidents that contributes most to nuclear safety. It is the means by which accidents, such as the LOCA, are made a low probability event.

5. <u>Present LOFT plans call for the use of a pressure suppression system in lieu of blowdown to the containment. Because of this, there are no planned tests of the containment's ability to control fission product activity. Do you believe that such a test of the containment would be appropriate for LOFT?</u>

Testing of the containment's ability to control fission products is not an activity that should be undertaken in LOFT. The control of fission products in a containment is greatly dependent upon the form, temperature, and arrangement of reactor system components in the containment, as well as upon the type and size of the containment itself and the type of containment spray and air-cleaning systems within the containment. The LOFT plant is not prototypical in any of these features.

The ratio of containment-vessel volume to the coolant-system volume is much larger in LOFT than in the usual commercial nuclear plant. Thus a LOCA in LOFT would produce significantly less pressure in the containment than would be experienced by a LOCA in a commercial nuclear plant. Furthermore, a LOCA in LOFT would produce a containment pressure only about 23% of the containment design pressure whereas in a commercial plant the containment pressure would be more like 80% to 85% of the design pressure; this difference prevents confirmation of relative leak-tightness in the two plants. Hence a containment test on LOFT would not confirm the pressure-retention capability or structural adequacy of the containment in a commercial nuclear plant.

In addition, the dispersal and deposition of fission products in LOFT would be different from that which would be experienced in a commerical plant for at least four reasons: (1) the large differences in containment-to-system volume ratios referred to earlier, (2) the significant differences in the masses and arrangement of system components within the containment, (3) important differences in containment-spary and post-LOCA fission product clean up capabilities, and (4) the fact that the LOFT containment vessel is made of steel whereas commercial PWR's use steel-lined concrete containments. These differences affect the fluid flow characteristics and fission-product movement in the containment, the nature of the internal heat sources and sinks within the containment, and the heat transfer characteristics to the outside of containment, all of which influence the dispersal and deposition of fission products.

Even if a fission-product retention test were to be made on LOFT it could be only a single test and would involve core meltdown with all the attendent problems and shortcomings referred to in responses to the previous questions. The pressure suppression system being used in the currently planned LOFT tests, while not suited to testing fission product retention, does permit performance of controlled code confirmation tests without contaminating the containment and interfering with accessibility to the equipment or introducing delays for clean-up.

It is believed that, with regard to fission-product retention and removal within the containment, emphasis should continue to be placed on separate-effects experiments and tests such as being carried out in the Containment System Experiment (CSE) in this country and on related tests being done in Europe. 13/ These experiments and tests, coupled with the analytical and test programs used to confirm the structural adequacy and leak-tightness of commercial plant containments, adequately satisfy the need for data on fission-product retention and removal within containment.

6. <u>LOFT is a small pressurized water reactor which has been scaled so the test results will simulate the anticipated effects of loss-of-coolant accidents on large pressurized water reactors. Will the LOFT results be applicable to boiling water reactors? Do you believe a LOFT experiment using a boiling water reactor mobile test assembly is needed?</u>

There are enough similarities between PWR's and BWR's so that much of the information obtained from PWR LOFT tests will be applicable to BWR's, but it is not clear that this information will be sufficient to confirm BWR system codes. In both types of plants similar thermal and hydraulic phenomena are encountered during a LOCA. In both, there is a need for evaluating blowdown rates during a LOCA and for assuring rapid reflooding of the core to avoid severe clad damage. But there are several important dissimilarities between commercial PWR's and BWR's that introduce differences in system codes which will not be checked by LOFT as presently constituted.

The following differences between PWR's and BWR's bear on this question. PWR's utilize completely open bundles of fuel elements, whereas BWR's use fuel bundles enclosed in boxes open only at the inlet and outlet ends. BWR's use core spray systems to help with core cooling during a LOCA, whereas PWR's do not. PWR's have primary-loop pumps and separate steam generators through which some of the fluid must flow to escape through a cold leg break, whereas BWR's do not; this added resistance to flow can lead to steam binding in PWR's, whereas steam binding is not a problem in BWR's because the path for fluid escape is more direct; furhermore BWR's are not confronted with the possibility that tube failures in the steam generators could introduce secondary steam to the containment and futher raise containment pressure.

While a number of these differences appear to make the task of predicting the course of a LOCA in a BWR plant easier than in a PWR plant, features unique to BWR's, such as fuel assembly boxes and core sprays, introduce questions not addressed in PWR system codes. Although a number of separate-effects tests have been done and others are planned to study the effects of these features during a LOCA, no plans exist to check if they are appropriately coupled in BWR system codes.

It is believed that some attention to this matter is merited. Studies should be made to determine the extent to which features peculiar to BWR's, such as fuel-element boxes and core sprays, could be tested in a later phase of the LOFT program. It is recognized that incorporating such features will require extensive modifications in LOFT, but I believe they would be worthwhile, even though they must await completion of the PWR tests, several years away. In the interim I have no concern about proceeding with the licensing of BWR plants for the reasons given in response to question number 4.

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Nov. 1, 1966 to Present: Dean, College of Engineering, the Pa. State Univ.
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May 1, 1950, to July 15, 1959: Westinghouse Electric Corp., Pittsburgh, Pa., Manager PWR Reactor Design Subdivision. Was in responsible charge of work on the design of Submarine Prototype Reactor, Mark I, on NAUTILUS Reactor and on Shippingport Reactor. Initiated burnout studies at Westinghouse.
Oct. 1948 to April 30, 1950: Argonne National Laboratory, Lemont, Ill., Staff Assistant to the Division Manager. (On loan from Westinghouse)
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11/24/75

REPORT TO THE

U.S. GENERAL ACCOUNTING OFFICE

ON THE

REVIEW OF THE NRC/ERDA

LOSS-OF-FLUID-TEST FACILITY

BY

CARL J. HOCEVAR

UNION OF CONCERNED SCIENTISTS

NOVEMBER 6, 1975

1. Is the current plan to use LOFT for a meltdown experiment in the best interests of nuclear safety?

2. Should LOFT be used on a timely basis to study the means of retaining molten cores and measuring the consequences of steam explosions and radioactive releases resulting from a meltdown?

DISCUSSIONS OF QUESTIONS #1 and #2

Since questions #1 and #2 are closely related, they will be discussed together.

I do not believe that the LOFT facility should be used to perform core meltdown experiments. The present LOFT facility was specifically designed and built to perform simulated loss-of-coolant accidents (LOCA). The LOFT project is certainly not a panacea for the reactor safety question but it will provide some useful information. It would not make sense at this time to attempt to modify the LOFT facility to accomodate meltdown situations. A considerable amount of both time and money would be involved in such a modification program. The result would be a reduction in the rate of production of experimental data relating to the LOCA phenomena and the emergency core cooling system (ECCS) performance. I do believe, however, that core meltdown experiments should be performed in another test facility.

When the LOFT project was initiated in 1962, the intent was to investigate the core meltdown phenomena and fission product dispersal and removal mechanisms. The main objective was to demonstrate the effectiveness of the fission product removal systems and the containment and thus show that the reactors being built in the early 1960s could not undergo an accident that would affect the public.

The reactors being built in the early 1960s did not have sophisticated ECCS and, consequently, if a LOCA had occurred in the early power reactors, the core would have overheated and melted. There was no reliable analytical technique available for predicting the core meltdown process. It was not known how extensive the core melting process would be, i.e., would the core partially or completely melt and would the molten core melt through the reactor pressure vessel. Consequently, the LOFT-U (Unperturbed) experiment was initiated to provide experimental data relating to the meltdown process. This information was important because the amount of fission product release from the core was dependent on the actual meltdown process.

In addition to providing information concerning the quantity of fission product release from the molten core, the LOFT project was intended to evaluate the effectiveness of the containment structure and the fission product removal systems. Some basic data would have been obtained which would have allowed the analytical fission product models to be evaluated and improved. In addition, the overall system effectiveness would have been experimentally determined on a relatively large scale.

In the mid-1960s the reactors being proposed were much larger than the earlier plants. The larger reactor cores magnified the potential consequences of a reactor LOCA. It became apparent that not only would an uncooled reactor core melt but that it would also contain enough energy to melt through the reactor pressure vessel and through the bottom of the concrete containment building. Emergency core cooling systems were considered essential to prevent core meltdown in the event of a LOCA. The ECCS designs which were incorporated in the newer and larger reactors were not, however, based on extensive experimental data or adequate analytical techniques.

At that point a decision was made by the AEC to change the LOFT project from a core meltdown experiment to an ECCS verification program. The main objective of the revised LOFT project was, however, not stationary in time. Originally the revised LOFT project was to be a demonstration project, i.e., the effectiveness of the ECCS would be demonstrated experimentally. Then the objective was changed to a computer code verification project, i.e., the experimental results would be used to verify the adequacy of the computer codes which were being used to evaluate the performance of the ECCS. The major objective of LOFT oscillated back and forth for several years. It is now envisioned as a computer code verification project.

The reason for considering LOFT as a code verification program is as follows. Due to fluid dynamic and thermodynamic scaling problems, the small scale LOFT facility will not respond identically to a large commerical PWR during a postulated LOCA situation. Consequently, the performance of the ECCS in the LOFT system cannot be related directly to that in a large reactor. If the ECCS does not work in LOFT, it does not mean that the ECCS would not work in a large plant. Conversely, if the ECCS does work in LOFT, there is no assurance that the ECCS will function properly in a commercial reactor. The data to be obtained from LOFT can be used, however, to aid in the evaluation of the computer prediction methods currently being used.

It was probably a mistake in judgment to cancel the original LOFT-U type experiments. The original LOFT-U experimental facility may have required modification and enlargement, but the core meltdown experiment should have remained a part of the overall reactor safety program. The decision was based on the assumption that the ECCS which were being incorporated in the newer reactor designs would prevent a core meltdown from occurring. This was an over-optimistic assumption. The ECCS have never been tested under actual accident conditions and, at present, there is no guarantee that they will function as intended. The computer codes which are used to predict the LOCA and ECCS behavior have still not been verified and numerous questions remain concerning their adequacy.

Recently, however, it is becoming more apparent that we can expect LOCAs and core meltdowns in the future. For example, the Reactor Safety Study 1/ which was released last autumn in draft form has indicated that the possibility of a core meltdown is 1 in 17,000 reactor-years. Considering 1000 reactors in operation by the year 2000, as anticipated by the nuclear industry, there would be one meltdown expected every 17 years. The validity of the statistical methods used by the AEC in obtaining this value has been questioned 2/ and it is possible that the probability of a core meltdown is actually higher. The fire at Browns Ferry reactor in Alabama last March came very close to causing meltdown without a LOCA. The normal cooling systems and the ECCS were incapacitated, and only a hastily improvised pump arrangement prevented a possible core meltdown. Even though core meltdowns may not be a common occurrence, it is becoming evident that they will in fact occur.

There has been one partial core meltdown already in this country. In 1966, an accident occurred in the Fermi breeder reactor. Two fuel bundles (clusters of fuel rods) melted while the reactor was operating at only about 15% of full power. All the safety analyses which had been performed indicated that under the worst conceivable circumstances only one fuel bundle could be damaged. The safety analyses also concluded that it was practically impossible for an accident of such magnitude to occur. Yet, not only did a serious accident occur, but the melting of two fuel bundles exceeded the safety estimates of a maximum of one fuel element melting.

Since it appears that meltdowns will be a reality, it is imperative that the phenomena which might be expected to occur be understood. A recent report prepared by Sandia Laboratories for the NRC, and entitled Core Meltdown Experimental Review 3/, is probably the most complete review of the experimental core meltdown data available. The report covered numerous aspects of the core meltdown process such as the physical and chemical behavior of the melt, the structural behavior and physical motion of the core, steam explosions, release of radioactivity from the core, fission product transport and removal, non-condensable gas evolution, and hydrogen explosions. Many areas of uncertainty were identified in the technical evaluation presented. The report concluded that the present understanding of such critical events as steam explosions, melt/concrete interactions, and non-condensable gas evolution was very minimal. Since these phenomena could significantly influence the pressure levels in the containment during a LOCA, it is imperative that they be understood more completely. It is quite possible that existing containments could be ruptured during a major LOCA because these phenomena have not been properly accounted for.

As indicated in Reference 3, the NRC is conducting some separate effects tests on steam explosions, fission product release and removal, and molten core phenomena. These experiments are all being performed on a relatively small scale compared to a reactor system. Scaling effects can be important and the present small scale experiments may not be adequate.

For example, the relative proportions of the hot and cold phases, the phase composition, and the relative temperatures have all been shown to be important during steam explosions. However, there has been very little experimentation involving large quantities of both phases such as would occur during a core meltdown. In addition, much of the experimentation has been performed using materials other than uranium dioxide and water. Experiments involving large quantities of both molten core materials (uranium dioxide, zircaloy, steel) and water should be performed. Only through realistic large scale experiments can the steam explosion phenomena be adequately studied.

Other examples where scaling effects must be considered are fission product transport and heat transfer behavior. Small scale separate effects tests are not sufficient unless they are integrated with very sophisticated analytical prediction methods. Since such techniques are not currently available, larger scale tests must be employed.

Generally, the results from small scale experiments alone cannot be confidently extrapolated to large facilities such as commercial reactors. If these results are integrated with analytical prediction techniques that are capable of accurately predicting the physical phenomena occurring during a LOCA, then the techniques can be used with some confidence to predict the behavior of large scale equipment. Unfortunately, such analytical methods are currently not available. Consequently, the experimental data must be obtained over a wide range of parameters, including sizes characteristic of reactor systems. This is why large scale experiments are required in the core meltdown areas.

The present LOFT facility does not appear to be applicable to a core meltdown experiment. Any meltdown in the LOFT facility would not be representative of a large reactor because of the relatively large ratio of reactor vessel mass to core mass in LOFT. The LOFT vessel would represent a larger heat sink than a commerical vessel would. This is due to the massive steel fillers that have been placed in the vessel due to hydrodynamic considerations. This massive heat sink could possibly absorb enough thermal energy from the molten core to cool the core and, consequently, prevent a vessel melt-through. It is also possible that the large heat sink provided by the vessel could prevent extensive melting of the core itself.

As indicated in previous discussions, the response of the LOFT system will not be the same as that for a large commercial reactor. One very important difference will be attributed to the short five and one-half foot core in LOFT. The thermal and hydraulic response will be different in the short LOFT core and a larger (12 feet) reactor core. Consequently, the ECCS behavior in LOFT cannot be applied directly to a large reactor.

Another major problem in large PWRs that LOFT cannot resolve because of scaling problems is that of steam binding. Steam binding can greatly reduce the effectiveness of the ECCS.

The funding level for the research pertaining to core melt phenomena is disproportionately low compared to that for LOCA and ECCS. According to the minutes of an ACRS Subcommittee meeting on LOFT and reactor safety research 7/ the LOCA and ECCS research is receiving about $50 million, while the core meltdown phenomena research is only receiving $1/2 million. Considering the importance of understanding core meltdown phenomena, it would be appropriate to increase the funding level considerably for these studies. As mentioned earlier, the critical areas appear to be steam explosions, melt/concrete interactions, and evolution of non-condensable gases.

3. Will the small scale LOFT result in experimental data, including the phenomena associated with a core meltdown, that is applicable to the large commerical reactors? Is a larger LOFT type test facility needed?

DISCUSSION OF QUESTION #3

My answer to this question will address only the aspects of a LOCA and ECCS since that is the present purpose of LOFT. The core meltdown phenomena is not being considered in the current LOFT program. I have addressed the core meltdown question in response to questions #1 and #2.

The main purpose of the present LOFT program is to provide additional data for LOCA computer code verification. As indicated in the response to questions #1 and #2, the actual performance of the ECCS on LOFT will not be the most important aspect of the tests. The most important information obtained will be the comparison of the predicted behavior of the physical parameters with the actual experimental behavior. If the experimental data can be accurately predicted then, depending upon the particular analytical prediction methods used, a certain degree of confidence can be placed in the computer prediction methods. However, if the results cannot be predicted, a sufficient amount of basic data will not be obtained during the tests to allow the computer codes to be modified. Briefly summarized, LOFT is a computer code verification program, not an ECCS demonstration program or a computer code development project.

Since the complete ECCS have not been tested under actual accident conditions (individual safety system components have been separately tested under simulated LOCA conditions), the NRC and nuclear industry have relied upon analytical prediction methods coupled with the results from small scale experiments to determine the adequacy of the ECCS. This represents a valid engineering approach provided it is done appropriately. Unfortunately, the present licensing computer codes do not represent a soundly engineered technique. A rational program to provide a reliable LOCA and ECCS prediction technique is summarized below.

First, the appropriate equations of motion which will uniquely describe the behavior of the water and steam phases during a LOCA must be determined. Any assumptions or simplifications made in the solution of these equations must be justified by comparison with more exact analyses or with appropriate experimental data. In those areas where analytical solutions are not possible, empirical correlations must be used. When these relationships are employed, they must be valid over the complete range of parameters for which they will be used. Finally, the mathematical model must be tested against appropriate larger scale integral effects experimental data to determine its capability to predict physical events in complicated geometries.

Once a valid best estimate computer model has been developed and tested, an error analysis must be performed to provide an indication of the degree of uncertainty associated with the analytical technique. An analytical technique based primarily on empirical correlations will have a large degree of uncertainty, or error. A method which is based largely on the fundamental principles of physics such as that described in the previous paragraph will have a smaller degree of uncertainty associated with it. The best estimate and error analysis model could be used directly in the licensing process. If a special licensing model is to be used which incorporates "conservative assumptions," this model would have to be compared with the best estimate model and the error analysis to determine whether the assumptions are in fact conservative and if so by how much.

Very briefly, some of the shortcomings of the present computer models will be summarized. In general, the equations of motion for both the liquid and vapor phases are not solved, but the two-phase fluid is assumed to be uniformly mixed (homogeneous) and a set of equations is solved for these homogeneous mixtures. These assumptions are not valid during parts of the LOCA process. An attempt has been made to account for some of the non-homogeneous effects, but these correlations are based on small scale data, much of which was obtained in air-water system, not steam-water systems as exist in a reactor. In addition, many of the empirical correlations which are used are not based on applicable experimental geometries or on data obtained over appropriate parameter ranges.

At the present time, an experimentally verified analytical fuel rod deformation model does not exist. Such models are necessary if accurate predictions of important parameters and phenomena such as gas gap heat transfer coefficients and rod swelling and ballooning are to be made.

The NRC maintains that much of the conservatism in the licensing model is attributable to the heat transfer model. Claims are made that the heat transfer correlations are conservative correlations and that the use of correlations which are based on steady state data are conservative under transient conditions. These claims are simply not true. The heat transfer correlations used in licensing models are best estimate correlations, not conservative correlations.

In addition, the data which was used by the AEC to justify the claim that steady state correlations underestimate the heat transfer rates during transient situations does not in fact support the claim. There have been analyses performed for single-phase flow systems, but not two-phase systems such as occur during a LOCA, which show that transient heat transfer rates may in fact be larger or smaller than steady state rates depending on the flow conditions.

Another area where a conservatism is claimed is the break flow model. The model which is used is not very accurate and so correction factors are applied. Since the model is not accurate, the NRC requires that several computer runs be made with different correction factors, and then the run giving the worst consequence be used in the licensing of the reactor. Unfortunately, there is no guarantee that this procedure is conservative. In fact, it may actually provide a realistic or best estimate calculation and not a conservative one.

An evaluation of the adequacy or inadequacy of a number of the submodels used in the licensing model is summarized in the accompanying table. This table was taken from a recent report published by this author, entitled "Nuclear Reactor Licensing: A Critique of the Computer Safety Prediction Methods." 4/ This report, a critique of the nuclear reactor licensing computer prediction method, discusses in more detail many of the limitations of the present computer prediction methods.

The comparison of analytical predictions with experimental data from small scale integral effects tests have generally been quite poor. A number of the more important results are summarized in Reference 4. A recent set of experiments were performed this summer in the Semiscale facility by the Aerojet Nuclear Company (ANC). The Semiscale MOD-1 facility is a scaled version of LOFT. These experiments utilized a 5.5 ft. long electrically heated rod bundle to simulate a nuclear core. The total power was 1.6 MW, about 3% of the total LOFT power. The computer predictions underestimated the maximum cladding temperature by between 200°F and 250°F in several of the tests. These results were very significant because they not only showed that the computer prediction techniques were not accurate, but they also strongly indicated that the special mathematical model which is used in the reactor licensing model was not conservative under this set of possible accident conditions.

TABLE 5.1 SUMMARY OF SUBMODELS USED IN EVALUATION MODEL

MODEL	STATUS		
	a	b	c
Homogeneous equations of motion			x
Relative velocity relationships			x
Pump		x	
Critical flow			x
Friction factor		x	
Radioactive decay heat	x		
Gas gap heat transfer coefficient			x
Fuel rod deformation			x
Metal-water reaction rate		x	
Zircaloy embrittlement			x
Nucleate boiling heat transfer	x		
Forced convection vaporization heat transfer			x
Critical heat flux		x	
Transition boiling heat transfer			x
Flow film boiling heat transfer		x	
Pool film boiling heat transfer			x
Forced convection heat transfer to liquid	x		
Forced convection heat transfer to vapor		x	
Transient heat transfer			x
Reflood heat transfer			x

a Adequate

b Appears adequate - however, requires further verification and development

c Inadequate

Based on the fairly poor predictions of Semiscale integral test data to date, it seems probable that the computer models will not be able to accurately predict the results from the larger scaled LOFT facility. However, even if the LOFT test results are reasonably predicted by the present computer program, the adequacy of the models will still be in question. The reason is that the present computer models are based to a very strong degree on empirical correlations which are, in turn, dependent on limited data bases. The prediction of data from one experimental apparatus will not guarantee that results from a different sized facility will be predicted. In order to develop a reasonable degree of confidence in the present mathematical models which strongly depend on empirical correlations, the results from experiments ranging from sizes smaller than the Semiscale facility to those much larger than LOFT and possibly to sizes comparable to large commercial reactors are needed. This is necessary to validate the use of the empirical correlations over a very wide range of conditions. Only then could the present type of computer model be used with confidence to predict the behavior of a large reactor.

If better computer models are developed which more realistically describe the actual physical phenomena which would occur during a LOCA, then relatively small scale test facilities such as Semiscale and LOFT could be used to develop confidence in the methods. Small integral facilities could be used because the predictions would be based more on the actual laws of physics and less on empirical correlations which are based solely on experimental data over limited parameter ranges.

A project to develop more sophisticated computer models was started at ANC about 3-1/2 years ago. Approximately 1-1/2 years ago work was started on an alternate and less fundamental approach at both the Los Alamos Scientific Laboratory (LASL) and at the Brookhaven National Laboratory (BNL). Recently, the ANC project has been terminated because the NRC officials in charge of the computer code development work did not fully understand the complexities of the problem and consequently supported a less fundamental approach which had been suggested by one of the NRC officials a number of years ago. The loss of the ANC project may result in a several year delay in the development of a badly needed analytical model.

Due to the extreme complexity of the nuclear reactor in LOFT, several critical experimental measurements cannot be made. The absence of these measurements will limit the amount of computer code verification that can be done. For example, there will be no measurement of the mass flow rate at the reactor core inlet or outlet, nor of the fluid density in the core vicinity. The mass flow rate and density represent vital pieces of information in the computer code verification program. The absence of this data will limit verification of the core heat transfer model, a very important part of the reactor licensing computer model.

The instrumentation that will be used in LOFT probably represents one of the significant weaknesses in the LOFT program. This is not due to incompetence, but to the difficulty in obtaining two-phase flow measurements in general and in-core measurements in a nuclear core in particular.

The instrument which will be used to obtain the mass flow rate and density data in the downcomer of the core, a combined drag disc-turbine meter, has both accuracy and range limitations. The drag disc which measures the momentum flux has been calibrated under steady state conditions to $\pm$ 19% accuracy. The turbine meter measures the velocity and has been calibrated to $\pm$ 8%.

If these two measurements are combined to obtain the mass flux, the combined error for this quantity would be approximately 25%. These calibrations were performed under steady state conditions. There has been no stated transient error calibrations. The error under transient conditions would probably be larger than those quoted above.

A fundamental question exists regarding the interpretation of the measurements made with the drag disc-turbine meter. In a two-phase mixture, the streamline patterns of the lighter and heavier phases will be affected differently by the drag disc which is placed perpendicular to the flow. It is not clear exactly what quantity is being measured, a mean of the liquid and vapor phases or a larger contribution from the liquid. A similar basic question arises in the velocity measurement; what does the velocity measurement actually mean? Drag disc momentum flux meters have been used in other two-phase flow situations such as a gas-solid suspension flow system. In these applications, the results were not very reliable.

An additional problem exists in the particular drag disc-turbine meter design used in LOFT. The drag disc is placed in front of the turbine wheel and shadows the turbine. It is possible that the drag disc will interfere with the velocity measurements.

Another critical measurement that will be needed in the LOFT tests is the fuel rod cladding surface temperature. The fuel rods are instrumented with external thermocouples.

These external TCs can act as fins on the fuel rods and affect the fluid flow patterns and, consequently, the heat transfer rates. The spaded TC junction can act as a nucleation site and cause premature boiling and critical heat flux. The external thermocouples could also act as wetting sites during the core reflood and alter the heat transfer processes.

The use of external TCs is dictated by the use of nuclear fuel rods. Since there appears to be no alternative available, more effort should be made to determine the magnitude of the error that will be inherent in the use of these instruments.

The amount of fundamental data to be obtained during the LOFT tests will be limited due to the complexity of the experiment. Due to this lack of data, analytical submodels will not be modifiable if they are found to be deficient during the tests. For example, ECCS by-pass, downcomer, sub-channel analysis, and heat transfer models are all critical models, but none could be modified on the basis of the data which will be obtained during the LOFT tests.

It is recommended that additional efforts be devoted to the instrumentation problems on LOFT. More emphasis should be placed on the advanced computer code development, in particular, the method which was being developed at ANC. A best estimate and error analysis project should be given high priority instead of the low priority it currently receives. Only through such a program can the degree of conservatism in the licensing models be evaluated in a quantitative manner.

4. Should licensing of commercial reactors be modified in any way pending the results of the LOFT experiments or of experiments on a larger facility?

DISCUSSION OF QUESTION #4

I believe that the validity of the computer models used in the reactor licensing process is still in serious question. The computer codes which are currently being used are basically best estimate models, not highly conservative models as claimed by the NRC. Many of the models used in the computer programs are inadequate and need further development. A summary of the weaknesses of the present NRC evaluation model is presented in a UCS critique of the computer prediction models 4/ and was discussed in response to Question #3.

The real test of the computer codes is their capability to predict the results of experiments. In this regard, the computer codes have been shown to be significantly deficient. Reference 4 summarizes the major comparisons between the analytical predictions and the experimental data which are available today. In those cases where accurate comparisons have been attempted, the computer codes have failed badly. As the computer codes have undergone improvements over the past few years, the comparisons have improved but, as the most recent comparisons with data from the Semiscale MOD-1 experiments have shown, the computer programs are still not capable of accurately predicting experimental results.

There have been only a few comprehensive comparisons of the evaluation model which is used in the licensing process with experimental data. The comparisons that have been made have generally been of fluid dynamic response but not of heat transfer behavior. The fuel rod cladding temperature is one of the most critical parameters to be considered in the licensing process, yet very few of these comparisons exist. Although comparisons of the evaluation model with the results from the recent Semiscale MOD-1 heat transfer tests have not been made, it is highly probable that such comparisons would show that the evaluation model would underpredict the fuel rod cladding temperature, i.e., the evaluation model would produce a non-conservative calculation. This statement is based on the fact that the best estimate model which was used in the analytical comparison study and which underestimated the cladding temperature is very similar to the evaluation model which is used in the licensing process.

Even though the Semiscale tests were performed last May, an evaluation model prediction has not yet been made by the NRC and released for public inspection. The NRC has an obligation to provide such information for the public.

I believe that, because of the repeated inability of the computer codes to predict experimental results, the complete LOCA and ECCS licensing policy should be reevaluated. At present, the NRC is relying on inadequate word arguments and paper studies to justify the present licensing computer models. References 4-6 discuss and summarize many of the weaknesses in the mathematical models. The conservatism of the NRC evaluation model has not been demonstrated experimentally.

It is quite doubtful that the LOFT project will result in a reduction in the conservatism of the evaluation model assuming that the model is conservative to begin with. Fluid dynamic data in selected parts of the LOFT system and cladding surface temperatures (subject to the error involved in using the finned external thermocouples) will allow the computer programs to be partially evaluated. Even if the evaluation model should be shown to be conservative for the test conditions under which the experiments will be performed, there will be, however, no way to determine which specific part of the evaluation model is conservative. The detailed data needed to check each submodel in the overall model simply will not be available from the LOFT experiments.

I believe that the present licensing process should be slowed drastically and possibly halted until the current questions regarding reactor safety are satisfactorily answered. The LOFT data will be an essential part of the computer code verification program, but other ongoing experimental programs such as Semiscale, CSE, core melt and interaction experiments, steam explosion tests, etc., and analytical computer code development programs will all provide valuable data regarding reactor safety. Much of the data necessary to determine the effectiveness of the ECCS will not be available for at least several years. The commercial reactor program has simply developed too fast; large numbers of reactors are being built and planned, but the required safety research has still not been completed. Considering the potential consequences of a major reactor accident, this is not a prudent course of action.

5. Present LOFT plans call for the use of a pressure suppression system in lieu of blowdown to the containment. Because of this, there are no planned tests of the containment's ability to control fission product activity. Do you believe that such a test of the containment would be appropriate for LOFT?

DISCUSSION OF QUESTION #5

I do not believe that the LOFT experiments would provide a good test of the containment system because neither the containment structure nor the fluid dynamic and thermodynamic conditions would be representative of a large PWR system. When the LOFT program was reoriented from an unperturbed meltdown test to an ECCS evaluation test, the fission product behavior and containment response aspects of the LOCA accident sequence were dropped from consideration. For economic reasons, the original containment structure was retained. Design and construction were well under way.

The LOFT containment structure is all steel, while large PWR containments are steel-lined reinforced concrete structures. The heat transfer characteristics of the two systems would be different, thus the results obtained from LOFT could not be applied directly to large reactor containments. The data obtained, however, would be useful in the evaluation of some portions of the containment analysis computer codes.

The LOFT containment is designed to withstand a pressure of 35 psi. Most large PWR containments are designed to withstand pressures in the range of 50 to 60 psi. The present LOFT system is only capable of generating a containment pressure of 8 psi. This would occur if all the water in the reactor system were allowed to flash to steam and fill the containment building. A larger reactor system would be required to obtain higher pressures in the containment during a LOCA.

It might be possible to inject additional amounts of steam from some other external source during a LOCA to obtain higher pressures. However, only a maximum of 35 psi could be obtained due to current design limitations. In addition, the fission product concentration in the containment building would be diluted by the addition of the auxiliary steam and, thus, a realistic test of the fission product removal systems would not be obtained. The current LOFT containment is not instrumented with appropriate equipment to determine fission product levels and removal rates. Such equipment would have to be installed if LOFT were to be used as a containment test.

Some useful information regarding the fission product removal efficiency of a vapor suppression system might be obtained. Such systems are used in current BWR designs. There are basic differences, however, between the LOFT suppression tank and a BWR suppression system. In LOFT, all of the steam injected into the suppression tank will remain in the tank, while in a BWR the excess steam would be vented to the drywell. The amount of applicable information obtained from LOFT will quite likely be very limited.

There are other containment and fission product removal tests being performed by the NRC. The Core Meltdown Experimental Review 3/ briefly discusses containment tests being carried out in the NSPP (Nuclear Safety Pilot Plant) and the CSE (Containment Systems Experiment) facilities. The largest facility, the CSE, is approximately one-eightieth as large as a typical PWR containment. In the CSE tests, non-radioactive isotopes are being used to simulate radioactive isotopes. The use of non-radioactive materials eliminates the time consuming cleanup process.

Unless more sophisticated computer programs are developed to describe the heat and mass transfer processes in the containments, larger containment experiments will probably be necessary. The current computer codes rely heavily on empirical correlations which have been developed on the basis of data from small test facilities. As long as empirical correlations provide the backbone of the computer models, data from larger scale facilities will be necessary.

6. LOFT is a small pressurized water reactor which has been scaled so the test results will simulate the anticipated effects of LOCAs on large pressurized water reactors. Will the LOFT results be applicable to boiling water reactors? Do you believe a LOFT experiment using a boiling water reactor mobile test assembly is needed?

DISCUSSION OF QUESTION #6

The majority of the government-funded safety research programs regarding LOCA and ECCS have been directed toward the PWR system. The BWR system should receive a comparable amount of attention. To date, there has actually been less verification of the computer codes for BWR systems than there has been for PWRs. The majority of the standard test problems designed to check the computer codes have been oriented to the PWRs. The Semiscale facility which has been used to provide much useful information on reactor safety simulates a PWR. There has been no extensive independent government assessment of the LOCA and ECCS phenomena which would occur during a LOCA in a BWR.

Much of the general discussion pertaining to computer model development and verification which was provided in answer to question #3 is also applicable to BWR analyses. Computer models based on realistic descriptions of the anticipated physical phenomena are highly desirable. If reliance on empirical correlations is to continue, then larger test facilities will be required.

I believe that a test of a BWR nuclear reactor is highly desirable. In designing such a facility, we would hopefully avoid many of the mistakes that have been made in the LOFT program. The size of the BWR facility required would depend on the sophistication of the computer programs which would be used to predict the test results. If the computer models relied heavily on empirical correlations, then a relatively large experimental facility would be necessary. If a more sophisticated computer model is used, then a smaller facility could be used. In any event, the size of the reactor should be large enough to employ full length (12 feet) BWR rod bundles. Only a full length core will provide a realistic test of the ECCS under simulated conditions.

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REPORT TO THE

U.S. GENERAL ACCOUNTING OFFICE

ON THE

REVIEW OF THE NRC/ERDA

LOSS-OF-FLUID-TEST FACILITY

BY

DR. FRED C. FINLAYSON

AEROSPACE CORPORATION

NOVEMBER 1975

Review of the NRC/ERDA Loss-of-Fluid-Test Facility

-- A response to questions posed by the Senate Committee

on Government Operations to the U.S.

General Accounting Office --

Fred C. Finlayson

Question:

1. Is the current plan not to use LOFT for a meltdown experiment in the best interests of nuclear safety?

The best interests of nuclear safety would be well served by an improved understanding of the physical phenomena associated with core meltdown. Uncertainty in the release fractions, transport, and removal mechanisms of certain critical fission products could have important implications with respect to the risks associated with light water reactor operation in the U.S. However, the practicality of using LOFT as the vehicle for resolving these uncertainties is not immediately apparent. The detailed justification for additional research in meltdown phenomena has been appended to these questions.

Let us consider the relative positive and negative aspects of using the LOFT facilities for a meltdown experiment. In its favor, LOFT is a large scale event. The LOFT core weighs 4,140 pounds which is much larger than any meltdown experiment to date. LOFT was also designed to resemble a pressurized water reactor (PWR) -- at least with regard to its major operational components. This might also have been a positive attribute, but use of the LOFT facility has many negative aspects. For example, the LOFT pressure vessel is relatively much heavier (containing proportionally much more steel) than a similar large PWR. The ratio of the mass of the core to the mass of the steel in the pressure vessel is nearly ten times greater in LOFT than a similar large PWR. As a consequence, the time phasing of melt processes may be substantially altered. A more serious complication is that the relatively massive amount of steel compared to fission product decay heat available furnishes such a large heat sink, that when convective and radiative heat transfer from the vessel are considered, it is not certain that the meltdown of the vessel can be assured.

The mobile test assembly upon which the reactor is constructed also complicates the containment structure configuration and meltdown processes causing them to depart further from a typical large PWR.

Some of the more critical aspects of meltdown fission product release mechanisms would be poorly simulated in LOFT. The interaction of the molten core with the concrete foundation of the containment structure would be poorly simulated, since no effort was made to adequately model, in the LOFT facility, the concrete pad beneath the reactor for a large PWR. Moreover, though the mechanism of soil scavenging of the fission products (assumed to produce a decontamination factor of 1000 in WASH-1400) is one of the more important areas requiring investigation, meltthrough of the LOFT vessel and subsequent downwind dispersal of fission products at the Idaho National Engineering Laboratory does not seem at all desirable. Similarly the uncertainties with respect to the probability and magnitude of steam explosions makes the use of the LOFT facilities an undesirable test vehicle. If an explosion should rupture the containment vessel accidently, the results again would be most undersirable. Better test facilities are needed to test these important aspects of fission product release where the risks of uncontrolled release are minimized.

From the above discussion, it is apparent that the LOFT facility is not particularly desirable for a meltdown experiment. The LOFT test bed is now designed as a vehicle for testing the effectiveness of the Emergency Core Cooling System (ECCS) against a large break in the primary system piping. The facility is evidently much better suited to the problem for which it is now designed than it would be relative to its earlier planned function as a meltdown experiment.

2. Should LOFT be used on a timely basis to study the means of retaining molten cores and measuring the consequences of steam explosions and radioactive releases resulting from a meltdown?

Four important release mechanisms have been recognized which contribute to the fission product source term in the reactor meltdown. These are respectively:

1) Gap release; A reasonably well understood mechanism pertaining to the noble gases and more volatile fission product components. This mechanism is only important with respect to understanding of the timing of release, since ultimately all of these gases and volatile fission product components would be essentially completely released at some time during the meltdown process.

2) Meltdown; Results during this phase are quite uncertain, perhaps primarily due to the small size of the fuel elements upon which experiments have been conducted. Most experiments have been conducted with particles approximately the size of a large pea - a single pellet of fuel - weighing about 30 grams. A few tests have been conducted with samples approaching 100 grams in size and the Germans are currently planning on conducting tests with samples as large as 2 Kg (using simulated fission products.) 3/ As a result of the small particle sizes and the limited thermodynamic analyses which have been conducted relative to fuel/cladding interactions, only the simplest of models have been used to date, equating the fission products released with the fraction of the core melted.

3) Vaporization; This mechanism occurs when the molten fuel comes in contact with the concrete of the containment building floor. At that time rapid decomposition of the concrete produces large quantities of gases such as carbon dioxide (CO_2) which are assumed to bubble rapidly through the molten core - "sparging" the fission products from the melt. Contact with the oxygen in the containment building atmosphere, as well as the steam contained therein, also contributes to the vaporization release component. Only highly simplified analyses have been performed for the processes involved in the vaporization release component. There are many unknown details to this mechanism concerning most of the chemical/physical, thermal, mechanical and metallurgical properties of the complex system. Results of analytical models are strongly dependent upon basic assumptions which differ widely from model to model. No large scale experimental work on relevant systems has been performed to guide the modeling. As a result, there are substantial uncertainties with respect to the magnitude of this component. Vaporization is an important fission product release mechanism since it is assumed to carry to completion the release of all the volatile components including the noble gases, iodines, telluriums, and ceсuriums. Moreover vaporization is a dominant contributor to release of the volatile and non-volatile oxides. Thus it is highly important to understand and properly model this release component because of its important relationship to some of the most hazardous fission product components.

4) Oxidation/steam explosions; Steam explosions may be produced when appreciable amounts of molten core (probably of the order of a kilogram - or more) are brought into sudden contact with water -- either by falling into the water -- or vice versa. The explosion is expected to

disperse finely divided fission product particles throughout the containment building -- and outside if the containment fails in the blast. The mechanisms of molten fuel-liquid interaction have been widely studied -- but are still poorly understood. Consequently the oxidation release mechanism is modeled only in a very gross sense. More experiments with larger samples of material need to be conducted to assure scaling mechanisms are better understood.

This rather lengthy explanation serves to highlight the depth of uncertainty in the release mechanisms as well as the disparate nature of the physical phenomena involved in each of them. The wide variations in the physical mechanisms involved in the release mechanism make it difficult to conduct an experiment which will permit all three of the objectives of the question to be satisfied. That is, the three concepts of (1) retaining molten cores (core-catchers); (2) investigating steam explosions; and (3) measuring the radioactive release components are probably mutually exclusive goals in a single experiment.

Moreover, as described above, the LOFT physical configuration is not well suited for investigations of core meltdown phenomena. The relatively massive pressure vessel complicates meltthrough mechanisms. The mobile test assembly is also a complicating factor with respect to thermal mechanisms during meltdown as well as fission product dispersal thereafter. Thus the relevance of use of the LOFT facility to investigate any of the phenomena in a meaningful fashion relative to the results in a large PWR is questionable.

If a well defined analysis method for fission product release and dispersion existed which was sufficiently general to model the complex geometry of the LOFT facility, then the test might be useful for model verification -- similar to the basic objectives of the LOFT-LOCA program. However, the meltdown models are not sufficiently well developed to justify performing this test at this time. Much of the information needed to develop such a model should be obtained initially in a well organized program of separate effects tests and theoretical analyses. Such separate effects tests would be essential prior to conducting a system level test -- perhaps at a scale similar to LOFT -- which will ultimately also be needed.

3. Will the small scale LOFT result in experimental data, including the phenomena associated with core meltdown, that is applicable to large commercial reactors? Is a larger LOFT type test facility needed?

LOFT is a system level test of the effectiveness of a reactor ECCS against a large break LOCA. System level tests fulfill an essential role in assessing reactor safety. They provide an experimental mechanism for the evaluation of a well-developed model of system performance. Their principle function in this evaluation is to assure that the model has no overlooked physical elements of significance to system performance, no synergistic effects have been missed in model development, non-linear aspects of the model are properly accounted for, and that auto-catalytic effects have not been overlooked.

They key element of the usefulness of a system level test is associated with the existenece of a well-developed physical model of system performance. If the model is based essentially completely upon fundamental theoretical physical laws, with a minimal dependence upon empirical (or semi-empirical) elements, then there would be a good possibility that an experiment of LOFT scale would be very useful in model verification. Unfortunately, however, system level models of ECCS performance for reactors are heavily dependent upon empirical elements which have complex scaling relationships. Great caution must be used in extrapolating the application of these ECCS models over ranges substantially beyond those for which measured results have been obtained. In scaling a complex system like the ECCS in LOFT to large scale PWR applications-from 55 MWt to 3300 MWt, a scaling factor of 60-the coupled thermodynamic, hydraulic, elastic-plastic mechanisms have many such scaling relationships which must be satisfied simultaneously. These range from the familiar Reynold's number (relating viscous flow regimes in the system), to the Prandtl number (heat transfer), the Froude number (relating inertia and gravitational forces), and Mach number (relating wave propagation in the multi-phase hydraulic system) to name but a few of the pertinent parameters.

It has been acknowledged that it is physically impossible to design the sub-scale model LOFT to assure simultaneous satisfaction of all these parameters 4/ in the scale model identically to their values in a full scale system during a LOCA. Consequently, it will be impossible to extrapolate LOFT results directly for application to large PWRs. Thus, the results are primarily useful for verification of model elements by comparison of experiment predictions with measured results.

It the analytical system model was essentially perfect, then model "verification" could be accomplished by the test. The probability of this occuring with the present generation of ECCS models (or any of the immediate future generations) is essentially zero. Consequently, although code verification may be unlikely, LOFT will serve the useful alternative purpose of "maturing" the codes against a new and larger system. Derivation of a new set of empirical parameters for the model is the probable result of such a maturation process. Though this is a useful and necessary function for the LOFT program, it should not be expected that LOFT will result in a "verified" code. On the contrary, it will result only in another semi-empirically defined analysis method which will next require verification against a still larger scale model system test before its verification can be adequately assured.

The inevitable conclusion is that a larger (near full scale) system test will have to be conducted before confidence in the applicability of the ECCS models is assured to the satisfaction of most reasonable members of the engineering and scientific community.

The same line of logic will probably apply to sub-scale system tests of core meltdown phenomena. In the long run, verification of results of analysis methods against a relatively large scale test program will be required.

4. Should licensing of commercial reactors be modified in any way pending the results of the LOFT experiments -- or of experiments on a larger facility?

No dramatic changes are recommended in reactor licensing procedures for commercial reactors such as restrictions on licensing of additional new reactors prior to completion of LOFT tests (or larger tests). However, changes might reasonably be made to the NRC-ECCS Acceptance Criteria, Title 10, Chapter I, Code of Federal Regulations, Part 50 (10 CFR 50) Appendix K. Specifically, limits should be prescribed on minimum allowable calculated reflood rates in PWRs and BWRs, requiring rates greater than two inches per second. A requirement for a reflooding rate this high will undoubtedly pose problems for the current PWR ECCS designs--and is probably tantamount to requiring redesign. Nevertheless, an explicit specification of minimum reflooding rate in the Acceptance Criteria is as significant a parameter as specification of the peak cladding temperature -- for which a maximum calculated temperature of 2200 F is currently prescribed. In absence of the empirical evidence for assured ECCS performance, such a minimum would reflood rate criterion, act as a redundant statement of the engineering objectives of a conservatively designed emergency core cooling system.

The reactor risk analysis of WASH-1400 has shown that core meltdown may come about as a result of several other mechanisms besides large pipe breaks in the primary system. WASH-1400 should be reviewed in detail to analyze whether requirements for additional redundancy in power supplies, critical valves, switch gear, pumps, etc. should not be levied in the reactor design criteria (e.g., as part of 10 CFR 50, Appendix A, or other appropriate Regulatory Guides).

Whether any of the conservatively prescribed regulatory criteria may be relaxed as a result of the LOFT program is uncertain. The most significant data expected to be obtained from LOFT will be associated with blowdown parameters such as critical flow models for fluid flow from the ruptured pipe and the use of transient critical heat flux (CHF) models. In the case of break flow models, criteria requirements are more "realistically" specified than conservatively, and allowable changes on the basis of LOFT results are expected to be minimal. In fact, it may be shown that more sophisticated transient break flow models accounting for metastable periods of flow -- such as the "Fauske" model -- should be explicitly incorporated into the specifications.

It is possible that the current conservative restrictions may be relaxed on the use of steady-state critical heat flux models and on the absolute restrictions against the use of nucleate boiling heat transfer coefficients after CHF occurs. Data from LOFT may be sufficient to infer the adequacy of these specifications (or conversely -- to show the continued need for conservative models). Data of sufficient adequacy to permit relaxation of other elements of the criteria is unlikely to be obtained in LOFT.

Though CHF and critical break flow models are important, relaxation of conservatisms in the models in these areas would not be expected to demonstrate an overall margin of conservatism for the ECCS criteria, or substantially increase the confidence in ECCS performance. The critical areas of uncertainty with respect to ECCS performance, probably dominating predictions of peak clad temperature histories, are: steam binding which restricts reflooding rates; and fluid flow restrictions and blockage in the core and consequent three-dimensional third diversion resulting from fuel rod swelling and rupture during the severe LOCA transient. No significant information on these vitally important problems is likely to be obtained from the LOFT program.

5. Present LOFT plans call for the use of a pressure suppression system in lieu of blowdown to the containment. Because of this, there are no planned tests of the containment's ability to control fission product activity. Do you believe that such a test of containment would be appropriate for LOFT?

The usefulness of LOFT for investigating questions associated with meltdown fission product release has already been briefly addressed. The limited usefulness of LOFT in this aspect appears to also be the case with respect to the tests of the containments ability to control fission product activity. In the first place, estimates of the containment pressure as a result of a LOCE show that if the ECCS is successful in preventing core meltdown -- but allows the release of fuel rod gap components of the fission products, by some mechanism -- the amount of steam released from the LOFT primary system would result in relatively low pressures being developed in the containment vessel. This event would result in containment pressure build-up less thatn 10 psi; compared to a containment design pressure of 35 psi. The probability of defining meaningful leakage tests from the containment or evaluating the adequacy of pressure reduction mechanisms under these conditions seems remote.

In addition, it appears that implementation of state-of-the-art fission product spray removal and heat removal systems within the containment structure has not been a high priority element of LOFT design requirements. Consequently available devices appear to be primitive and their usefulness in extrapolation of results to commercial PWR designs is probably limited.

Moreover, radioisotopic contamination of the facility, especially the mobile test assembly would be extensive. Clean-up of the facility following such an experiment would be extremely difficult, if possible. Re-use of the facility could only be made after an extensive waiting period, far in excess of customary turn-around periods between LOFT experiments. Consequently, if such an experiment were conducted, it should probably be done only after all other significant LOCA experiments have been conducted.

Since the suppression tank, with its fission product limiting characteristics, seems to be useful for expediting ECCS investigations in LOFT, and the pay-off for LOFT investigations of containment fission product control mechanisms seems low, retention of the suppression tank in the program is recommended and an investigation of the containment's ability to control fission product activity does not appear to be warranted.

6. LOFT is a small pressurized water reactor which has been scaled so the test results will simulate the anticipated effects of LOCAs on large pressurized water reactors. Will the LOFT results be applicable to boiling water reactors? Do you believe a LOFT experiment using a boiling water reactor mobile test assembly is needed?

A very limited portion of the LOFT data will be general enough to be useful for verification of elements of BWR analysis models. In particular, data for critical break flows and results related to transient CHF models may be useful for verification or maturation of models used in BWR-ECCS analysis. Data obtained during the LOFT blowdown period relative to these parameters will undoubtedly permit cross-checking and evaluation of BWR analysis routines. LOFT results in other periods (refill and reflood) will be entirely dissimilar to the thermo-hydrodynamic phenomena of BWRs during these periods. Consequently, it is not reasonable to expect to obtain any significant amount of relevant data applicable to BWRs in these periods from LOFT.

With respect to the need for large scale system tests of ECCS performance in a BWR, although performance analysis in a BWR is somewhat simpler than a PWR, there is still a need for ECCS model verification through large scale testing for BWRs also. Some of the difficult analysis problems for ECCS design in a PWR (such as steam binding) are minimized in a BWR. On the other hand, BWRs have their own set of analysis problems.

For example, considerable uncertainty exists with respect to the adequacy of ECCS core spray cooling models. Without dealing with the question of the adequacy of the tests by which the criteria core spray heat transfer coefficients were derived, it is sufficient to observe that these coefficients are acknowledged by the NCR to have large statistical error bounds associated with their definitions. Though the selected values are low and about what might be expected for the mechanisms of natural convection and radiation to steam, the uncertainty in their definition permits a variance of $\pm$ 200°F to be calculated in the peak cladding termperatures, under some circumstances. Thus the uncertainty in core spray heat transfer coefficients is evidently associated with a non-trivial factor in the BWR-ECCS performance analysis, and deserves better definition.

Similarly, claims have been made that the horizontal flow isolation associated with the use of vertically oriented channel boxes around each bundle of BWR fuel rods (7x7 or 8x8 rectangular arrays of fuel rods) eliminates problems of radial flow resulting from core blockage associated with fuel rod swelling and rupture during the LOCA. This is a somewhat deceptive argument! It is true that fluid, once entrained within the channel, cannot be lost (or gained) through radial flow to (or from) another neighboring channel. But it is not obvious that blockage in certain channels will not tend to cause preferential flow distribution of fluid from the lower plenum into unblocked channels with lower flow resistance during reflood. Under these circumstances, it is easy to visualize that the prevention of radial flow returning to the blocked channel above the swollen area of the fuel rods (by the channel box) may, in fact, exacerbate the meltdown processes, instead of aiding cooling mechanisms (as the arguments infer to be the case). Thus core blockage and resulting three-dimensional flow variations between channels in the core may prove to be at least as serious a problem in a BWR as it appears to be in a PWR.

To date, no tests have been conducted, or are known to be in the planning stages, which might investigate core blockage and resulting radial flow phenomenon in a BWR. Some single channel tests (approximately the equivalent of the Semi-scale tests at INEL -- a 1/30 scale version of LOFT) are being conducted under the joint sponsorship of NRC, GE, and the Electric Power Research Institute (EPRI). Though these tests represent a useful first step in analysis verification for BWRs, it appears that larger scale BWR tests -- at least as large as LOFT (and probably larger) -- will be required before confidence will be achieved in the adequacy of BWR-ECCS analysis methods and predicted results.

REVIEW OF THE NCR/ERDA LOSS-OF-FLUID-TEST PROGRAM

A Response to Questions Posed by the Senate Committee on Government Operations to the U.S. General Accounting Office

I. Historical Review and Statement of Problem

The Loss-of-Fluid Test (LOFT) facility is a major element of the U.S. Nuclear Regulatory Commission's (NRC) nuclear reactor safety research program. LOFT, a 55 MW (thermal power*) nuclear pressurized-water reactor (PWR), is presently designed to investigate the pehenomena associated with the principal "design basis accident" for nuclear reactors, the loss-of-coolant-accident (LOCA). Data from the tests conducted in LOFT will, in principle, provide a basis for evaluation of the design methods for the "emergency core cooling system" (ECCS), the primary element of the safety equipment which is supposed to prevent serious damage and overheating of the reactor core in the event of a LOCA.

When LOFT was initially conceived, in 1962 -- nearly 14 years ago, it was intended to provide data on the effectiveness of the reactor containment building to retain (or mitigate the loss and dispersion of) nuclear reactor fission products from an accident which resulted in meltdown of the intensely radioactive nuclear fuel in the core. At the conception of LOFT, commercial reactors were being designed with relatively low power outputs (generally less than 200 MW electrical power). For these relatively low powered commercial reactors, reactor containment buildings were expected to be able to withstand the results of reactor meltdown without danger of catastrophic failure or suffering any consequent substantial losses of fission products released by the meltdown. However, the design power output of commercial reactors increased rapidly in the next few years as utilities and vendors tried to take advantage of

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*This paper presumes a certain familiarity with the basic features of nuclear power reactors. For these readers unfamiliar with the basic features of boiling-water (BWR) and pressurized-water (PWR) reactors, an elementary description of them, their related equipment, and the physical mechanisms by which they operate is contained in the American Physical Society's review of reactor safety 1/. A brief glossary of some of the more significant technical terms (and definitions of acronyms) used in the paper has been appended to the document.

scale economies. By 1965 reactor electrical power outputs were approaching 1000 MW (equivalent to approximately 3300 MW of thermal power for typical plant efficiencies of about 30%) for several reactors for which licensing procedures had been initiated. Safety experts began to be seriously concerned about the ability of containment structures to retain a meltdown of a nuclear reactor of this size. Design emphasis shifted quickly from meltdown containment to meltdown prevention. The concept of permitting a reactor core to melt as a result of an accident became 'inconceivable' as the consequences of such an event for large reactors began to be perceived. In 1966/1967, the Ergen Committee (an AEC select committee of reactor engineers and scientists) investigated the core meltdown problem area and recommended that safety research be redirected toward development of emergency core cooling systems (ECCS) in order to prevent core meltdown.

Shortly thereafter, the LOFT program plan was revised to reflect this new emphasis on the ECCS. By 1969, the LOFT design had been revised, taking advantage of as much of the original planning for the program as possible. Thus the current LOFT facility has essentially the same containment building configuration planned for the original concept. However, the LOFT power reactor itself resembles the original design only in approximate external dimensions (the pressure vessel outside diameter is about 6 1/2 ft. height about 24 ft) and thermal power (originally planned for "about 50 MW", it is now designed for 55 MW). A complete major redesign was required of the reactor vessel and internals as well as supporting equipment for the primary coolant system to accomodate the conceptual change from investigating core meltdown, fission product release, dispersion, and control to its present objectives of supporting the verification of analysis methods for ECCS design.

From 1968 to 1973, the AEC retained their prescription of core meltdown accident unconceivability. Consequently, reactor design basis accident limits were revised to require fuel rod temperatures to be limited to peak values of less that 2200°F by action of the ECCS during the LOCA (substantially beneath fuel melting temperatures of about 4000° to 5000°F).

In 1973, a review of the probabilistic aspects of risks and consequences of reactor accidents was commissioned by the AEC, under the direction of Prof. Norman C. Rasmussen of the Massachusetts Institute of Technology. Results of the "Reactor Safety Study," WASH-1400, 2/ published in draft form in August 1974, and finalized in October 1975, reflected anew the importance of the reactor meltdown accident and raised its probability to levels where consideration of meltdown is definitely no longer inconceivable.

As a result of regrettable delays and an inefficient approach to design and construction, the LOFT experimental program has not yet begun, although most of its hardware and construction are finally complete. Since the experiments have not yet begun, a choice is once again available should the objectives of LOFT be reoriented to again include an investigation of core meltdown; or should the objectives continue to be restricted to obtaining data related to analysis of ECCS performance. In accordance with the request of the U. S. Senate Committee on Government Operations, this question is the principal object of this review. The subject has been broadened somewhat to include questions related to the probability of the current program for LOFT being able to meet and satisfy its own objectives.

II. Analysis of Technical Issues

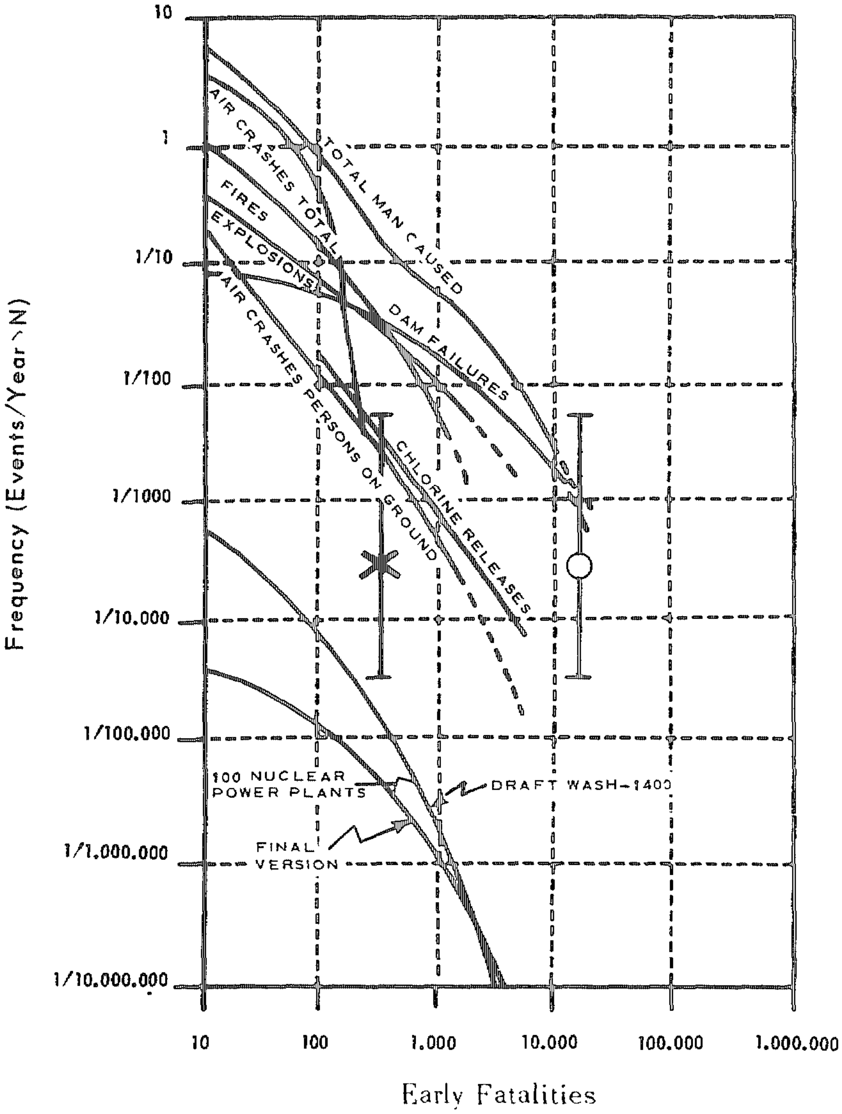
Questions related to whether the objectives for the LOFT program should be increased in scope to include core meltdown investigations, and the credibility of the program to meet its own current objectives center around several pivotal issues. A fundamental question is related to the relative significance of the core-meltdown problem to reactor safety. In addressing this question, some of the pertinent results of the WASH-1400 (Rasmussen "Reactor Safety Study") 2/ will be reviewed. A brief summary and evaluation will also be presented of the status of our understanding of the mechanisms of fission product release associated with core meltdown. The implications of the possible accident scenarios outlined by WASH-1400 leading to core meltdown, with respect to the design of the LOFT facility will also be reviewed. Finally, the basic LOFT program will be analyzed, relative to the probability of meeting current objectives. This section will attempt to make a brief, but unified, presentation of these issues and to estimate (at least qualitatively) the magnitude of the problems associated with the issues.

Probability and Consequences of Reactor Accidents and Their Implications

The most current and comprehensive analysis of nuclear reactor accidents, their probabilities, and consequences is found in WASH-1400. Figure 1 presents a composite curve summarizing the results of the draft and final versions of the study in terms of the fatality risks associated with reactor accidents. Results are presented in terms of the number of early fatalities (those occurring shortly after the accident, and clearly associated with radiation damage) from a single event as function of the calculated frequency (directly related to event probability) of the events per year -- assuming 100 operational nuclear power plants. Compared to WASH-1400 draft results, the final curves show a reduction in estimated probability of the smaller accidents to less than 1 in 10,000 that an accident will occur which causes more than 10 fatalities. Although the probability consequence curves of the final report decrease more gradually with increasing fatality levels than they did in the draft version, they still fall off rather abruptly as the number of fatalities for an event exceed 400 or 500. The largest number of fatalities predicted by WASH-1400 for a single event was 3300 deaths, with a probability/year of 1 in 10 million for the postulated one-hundred operational reactors. If for the moment we assume that the values given by the curves are correct, the rapidly decreasing event probability for higher consequence accidents implies an apparent asymptotic approach to a maximum number of early fatalities from nuclear reactor accidents of less than 10,000 with exceedingly low probabilities for such events. Under these curcumstances, the probabilistically weighted risk of death from the operation of the 100 postulated reactors of the study is much less than one person per year (i.e., about 3/1000 person/year).

Expressed on an annualized basis in this way, it is unlikely that a risk so small would be of grave concern to the public. It is, however, the potential for taking a large number of lives with a single accident, perhaps on the order of 10,000 lives, and contaminating large areas of land for years which changes the relative concern which the public feels for the problem -- no mattter how infrequent the accident may be. Few other man-made things have this potential for such large-scale disastrous consequences. Only natural events such as earthquakes, hurricanes, and famines are relatively common sources for disasters where thousands of lives are at risk from a single event. In my opinion, it is this potential for large-scale catastrophe, even though extremely infrequent, which motivates the concern of the public. There seems to be a psychological limit to the maximum number of deaths from a single man-made event which can be tolerated -- and reactors are suspected of being capable of approaching that limit.

Figure 1 shows a comparison of the relative risks deduced in WASH-1400 for other man-made accidents with large consequences. The WASH-1400 results clearly suggest



Nuclear and Non-Nuclear Accident Probabilities and Consequences-AEC Estimates and APS Corrections

The curves represent the frequency of different types of accidents, nuclear and non-nuclear, as a function of the number of fatalities as given in the Rasmussen report. The curve labeled "100 Nuclear Power Plants" (50 boiling water and 50 pressurized water reactors) includes only early fatalities, not delayed deaths due to cancer.

Two error bars have been added to this basic figure which appeared as Fig. 6.1 in the Summary volume of the Rasmussen report. The frequency range is that calculated in the Rasmussen report for the occurrence of a "reference accident" assuming the existence of 100 pressurized water reactors. This accident was assigned a probability between 1 in 20,000 and 1 in 2 million per reactor year in the Rasmussen report.

The point 'X' on the left-hand error bar indicates the total number of fatalities, 372 (62 early and 310 delayed from cancer), from the 'reference accident' as calculated in the Rasmussen report. Using the Rasmussen report's probability estimate, but including the corrections to the estimated number of cancer deaths calculated in the APS study, gives the point 'O' or 10,000 to 20,000 cancer deaths.

*A pressurized water reactor core meltdown with a release of radioactivity to the atmosphere almost as great as if there were no containment building at all.

FIG. 1

that several other man-made activities could lead to accidents with much larger probabilities of occurence and with very large numbers of associated fatalities. The implication of the report is that since these non-nuclear mechanisms for high fatality accidents are apparently tolerated by society, it should also find the much lower estimated probabilities for nuclear accidents acceptable.

Not everyone, however, has accepted the WASH-1400 results without challenge. Superimposed upon the curves of Figure 1 are variations upon the results of WASH-1400 suggested by Dr. Frank von Hippel, based upon an independent study of reactor safety conducted by the American Physical Society 3/. The result shown by point "x" on the curve indicates the possible increase in the consequences associated with one particular accident scenario calculated by WASH-1400, if all deaths (delayed as well as early) are included in the estimate. Fatalities calculated for the draft report would be increased to a total of 372 (62 early and 310 delayed -- from cancer) under these circumstances. Based upon corrections to the WASH-1400 estimates of the biological consequences of the referenced accident recommended by the APS review 4/, von Hippel suggests that total fatalities for the accident could potentially be increased to values as high as 10,000 to 20,000 cancer deaths -- as indicated by the point marked "o" on the curves. It should, however, be observed that von Hippel's consequence calculations are dominated by delayed cancer deaths, which would probably be spread rather uniformly over about 30 years. Thus the accident produces an equivalent increment in the annual cancer death rate of about 300 persons/year, compared to a natural cancer death rate in excess of 300,000/year in the U. S. This will represent an increase of only about 0.1% in the cancer death rate. It will clearly be difficult to even identify the increase in the cancer rate, against this background, in spite of the potentially large total number of added deaths as a result of the accident.

Nevertheless, it appears that the public awareness of the possibility for accidents with such large numbers of fatalities, irrespective of the rate at which deaths occur, coupled with the mystique of radioactivity as a cause of death, is the essential source of a major stumbling block to public acceptance of nuclear reactors as an energy source.

Large vertical error bars are shown in von Hippel's estimates of the probability of the accident -- in accordance with WASH-1400 estimates of the uncertainty in the probability of the referenced accident. Subsequently in this review, the need will be discussed for adding horizontal errors bars showing possible additional perturbations to estimated consequences resulting from uncertainties in the quantities of fission products released in the meltdown.

Factors Contributing to Accident Risks

Before considering in detail any of the individual factors influencing the potential risks of a reactor core meltdown, let us consider the several elements contributing to the overall picture. There are basically three dominant factors which control nuclear reactor accident risks. Risks may be considered the product of the probability of the event times the consequences resulting from it. Thus the contributing factors to nuclear reactor risk may be represented as: 1) the initiating event and resulting accident scenario, along with its estimated probability; 2) the magnitude of fission product release estimated to be associated with the accident scenario; and 3) the predicted biological consequences of a fractional release of the fission products to the environment -- outside of the control of the reactor containment building.

Prior to WASH-1400. little quantitative work had been done to carefully define the probability of the potential initiating events and accident scenarios leading to reactor accidents. WASH-1400 applied logical methods of fault and event tree analysis to the problem. In this manner, sequential steps leading to an accident, along with estimates of the probability of each element in the sequence, were generated for a very large number of possible accident scenarios. Though the absolute values of the probabilistic results of the study have been challenged, it is generally acknowledged that the results of the analysis represent a significant contribution to providing more insight and credibility to estimates of reactor risks. The probabilistic elements of the accident scenarios are only peripherally significant to this review of the LOFT study. Consequently, no serious attempt was made to evaluate the probabilistic aspects of the WASH-1400 results. They have been generally utilized in this study where they were applicable.

On the other hand, the magnitudes of fission product release factors are critically relevant to this review. Results of an evaluation of the WASH-1400 results and the general state-of-the-art in prediction of fission product release in a nuclear reactor core meltdown are briefly presented in the next section. A detailed review of the biological consequences of the reactor meltdown was felt to be beyond the scope of the objectives of this study.

Fission Product Release Estimates

The first element in estimating fission products released to the environment (and ultimately their consequences) is to define the source terms -- the fission product release mechanisms and the respective quantities released from the fuel during the several physical processes associated with the meltdown. Table I presents an integrated description of the important isoptopic subgroups of the fission products; relative fuel release fractions in terms of the several recognized release mechanisms; estimates of the influence of several natural and man-made fission product reduction mechansims utilized within the containment building to reduce the quantities released to the environment; and estimates of biological consequences expressed in terms of whole body doses received as a result of exposure to the penetrating radiation of the fission products.

An operating reactor develops an inventory of highly radioactive fission product isotopes in excess of a billion curies, with half-lives of an hour or longer, after a relatively short period (a few weeks) of operation. Although a substantial fraction of the radioactivity decays rapidly away during the first few hours after the reactor is shut down, the intensely penetrating radiation of the remainder must be controlled (retained) or it can induce the serious consequences discussed earlier. The basic elemental subgroups of radioisotopes are given in Table I, in terms of isotope groups exhibiting similar chemical behavior. Details of the isotopic breakdown have not been presented. Those who wish more detailed discussions of specific elements of the radioisotopes should consult the APS review 1/ or WASH-1400 2/.

Table I Radioactive Isotope Inventory, Release Relationships, and Selected Biological Conversion Relationships

	Isotopic Group	Shutdown Inventory[a] (10^6 curies)	Meltdown Source Release Fractions[b] Gap	Melt-down	Vapor.	Expl**	Cumul	Fractional Release to Envir.[c] PWR-1	PWR-2	PWR-3	Whole Body Dose-[d] APS Est(PWR-2) (10^6 man-rem)	"Max Dose" Risk Factor (10^6 man-rem)
1)	Noble Gases Xe, Kr	351	.03	.87	0.1	(.9)	1.	0.9	0.9	0.8	.044	.049
2)	Iodines	725	.017	.883	.1	(.9)	1.	0.7	0.7	0.2	1.631	2.33
3)	Telluruims and Cesiums	173 14	10^{-4} .05	.15 .76	.85 .19	(.6) --	1. 1.	0.4 0.4	0.3 0.5	0.3 0.2	.399 .810	1.329 1.620
4)	Volatile Oxides Mo, Tc, Ru, Rh	535	--	.03 (.01-.1)*	.05 (1/5 to 5x)	(.9)	.08 (.02 to .35)	0.4	.02	.03	.033	0.578
5)	Alkaline Earths Ba, Sr.	405	10^{-6}	.01 (.02-.2)	.01 (1/5 to 5x)	--	.11 (.03 to .25)	.05	.06	.02	.996	4.15
6)	Nonvolatile Oxides Y, Zr, Nb, La, Ce, Pr, Nd, Pm, Pu	1472	--	.003 (.001-.01)	.01 (1/5 to 5x)	--	.013 (.006 to .06)	.003	.004	.003	.534	8.01

a) WASH-1400(D), App VI, p. 6
b) WASH-1400(D), App. VII, p. 20
c) WASH-1400, Main Report, p. 97
d) APS Review(1), App. II, p. S105

* Numbers enclosed in parentheses represent the ranges of uncertainty for the specified fission product release fractions as they appear in Draft WASH-1400.

** Numbers (in parentheses) listed under explosive release column represent multiplicative factors to be applied to remaining unreleased (at time of explosion) fractions of individual isotopic groups to obtain explosive release fraction.

Unique among the isotopic subgroups are the noble gases, xenon and krypton, which are gases at all temperatures of interest to reactor analysis and not strongly bound chemically to the melt. These gases would be expected to escape from the melt under almost any set of circumstances in which the fuel rods are postulated to at least rupture. The elements of various meltdown scenarios do not generally restrict the quantity released, but only dictate the timing of the release of the noble gases.

The relatively volatile elements among the fission products are next in their ease of release from the binding matrix of the fuel. In decreasing order of volatility, these are represented by the iodines, tellurium and cesium isotopes, and the volatile oxides (formed from the isotopes of the elements molybdenum, technetium, rhenium, and Ruthenium). In this latter category (volatile oxides), the boiling points of the pure forms of the elements are well above the melting temperature of the uranium oxide of the fuel elements. However, if there is sufficient free oxygen in the core during the processes leading to meltdown, the elements can form stable oxides which are volatile at much lower temperatures and would consequently be expected to escape the melt reasonably readily. As indicated in Table I, the range of uncertainty in release fractions cited in Draft WASH-1400 for these radioisotopes is from 2 to 35 percent, depending upon the degree of oxidation.

The alkaline earths, barium and strontium, have chemical reactions which are almost the opposite of the volatile oxides. Barium and strontium are relatively volatile in their pure elemental forms, but in the presence of free oxygen, they form nonvolatile oxides. Barium and strontium are important contributors to radiation dose to the body. They represent a large fraction of the shutdown core inventory; and if they were released to the atmosphere at the upper limits of the uncertainties suggested in WASH-1400, could contribute an increment to the whole body dose equivalent to the total estimated value of the dose received in the WASH-1400 reference accident case (PWR-2).

The nonvolatile oxides: including yttrium, zirconium, niobium, lanthanum, cerium, praseodymium, neodymium, promethium, and plutonium (and several other trace isotopes) are all elements which react with water and carbon dioxide to form stable oxides. Carbon dioxide will be formed in abundance by thermal decomposition of concrete in the containment vessel. Thus the stable radioisoptopic oxides are expected to be mixed intimately with the molten uranium oxide fuel and be released in roughly the same proportion that the fuel itself is vaporized. Considerable uncertainty exists concerning the amount of vaporization to be expected with the molten fuel mixture. Simple energy balances indicate that in the absence of constraints relative to the volume (and hence carrying capacity) of the containment, or unless limited by reduced decay energy due to loss of vaporized fission product themselves from the melt, a vaporization rate of from 10 to 40 tons per hour might be expected. For a molten core mass of about 100 tons, this would represent a maximum vaporization loss rate of from 10 to 40% per hour. Other simple estimates of the maximum carrying capacity of the containment for the vaporized fuel aerosols due to natural gravitational settling processes, indicates a maximum steady-state capacity of approximately two tons of vaporized fuel aerosol would be expected to fill the containment. 5/

Estimates of this sort appear to have been used to establish the limits on the range of nonvolatile oxide release used in Draft WASH-1400, as shown in Table I. However, it should be noted that if the containment building leaks, there appears to be ample energy to volatize the fuel at any given leak rate up to the energy balance limits of from 10-40% of the fuel per hour (a containment leak rate equivalent to 5 to 20 complete changes of the containment atmosphere per hour). Thus, it is not all all obvious that the 1 to 6% vaporization limits suggested by WASH-1400 represent upper limits to nonvolatile oxide release. Moreover, if the vaporization rate of the molten fuel were increased, the ranges of expcected limits on release of volatile oxides and alkaline earths would also appear to require at least similar increases.

The four most important core meltdown fission product release mechanisms, providing the source terms for subsequent release to the environment are: gap release; meltdown; vaporization; and fuel-water interactive explosions. Of these four meltdown source release mechanisms, the WASH-1400 analyses indicate substantial uncertainties exist in essentially three of them 6/ -- especially in regard to the relatively low volatility elements of the fission product groups.

Gap release is a relatively well understood fission product release mechanism. As soon as the fuel rods swell and rupture (very early, in any accident scenario) the gaseous and volatile fission products derived during normal reactor operation -- principally, Xe, Kr, and the iodines gradually accumulated under pressure within the intact fuel rods -- would escape through the gap between the fuel pellets and the zirconium cladding of the fuel rod. The relatively small fractions shown in the gap release column of Table 1, represent only that portion of the fission product available at the time of rupture. Even if emergency core cooling measures were effective, there is a high probability that essentially all of the noble gases and iodines would be expected to be released to the containment vessel, in addition to the relatively small fractions designated as gap release.

As the meltdown processes continue, the less volatile components will be driven off. However, the release mechanisms associated with the meltdown process itself are quite uncertain. This is probably largely due to the small sizes of experimental samples which have been examined to evaluate this element of the fission product release mechanisms. Most experiments conducted to date have measured releases from samples about the size of a large pea -a single pellet of fuel -- weighing about 30 grams 7/. A few tests have been conducted with samples up to 100 grams in size and the Germans are planning on conducting tests with samples as large as two kilograms (using simulated fission products) 8/. Scaling of these results to equivalent masses of a melting core (on the order of 100 tons) is clearly uncertain and data on fuel meltdown in real reactor configurations is unavailable. As a result of uncertainty in empirical results and the absence of definitive thermodynamic analyses for meltdown release mechanisms (evidently correlatable weaknesses), only the simplest of models of meltdown fission product release have been used to date. These models equate fission product release, from products of suitable volatility, with the fraction of the core melted. 9/

Vaporization is a very poorly defined release mechanism. The customary boil-off mechanisms themselves have not been thoroughly investigated. Estimates of vaporization rates depend upon gross extrapolation of experimental results for thermodynamic properties of the elements and oxides beyond their measured temperature ranges by approximately 1000°C (from about 2000°C or 2500°C to over 3000°C). These large uncertainties in the basic vaporization processes are further compounded (in fact probably overwhelmed) when the supplemental vaporization mechanisms associated with interaction of the molten core with the concrete of the containment building floor are considered (after melt through of the reactor pressure vessel). Gases released during concrete decomposition are expected to pass rapidly through the melt, "sparging" the fission products from the molten mass.

Only highly simplified analyses have been performed for the processes involved in the vaporization release component. There are many unknown details concerning most of the chemical, physical, thermal, mechanical, and metallurgical processes of this complex system. Results of analytical models are strongly dependent upon basic assumptions which differ widely from model-to-model. No large scale experimental work on relevant systems has been performed to guide the modeling. As a result, as previously noted, there are substantial uncertainties associated with estimates of the magnitude of this component. Based upon the simple bounding estimates previously discussed, there is little evidence that the vaporization release component will be constrained to be as small as the WASH-1400 estimates suggest in the presence of a ruptured and leaking containment building, especially in connection with the failure scenarios included under the referenced WASH-1400 accident groups (PWR-1, PWR-2, and PWR-3).

An additional, and in some ways supplemental, release mechanism is associated with the rapid oxidation of the molten fuel occurring during an explosive fuel-water interaction. This explosive release mechanism is also poorly understood. Steam explosions resulting from such interactions may be produced when appreciable amounts of the molten core (probably of the order of a kilogram or more) are brought into sudden contact with water. The resulting explosion is expected to disperse finely divided fission product particles throughout the containment building -- and outside also if the building fails during the blast. The mechanisms of molten fuel-liquid interactions have been widely studied, but are still poorly understood. Consequently, the oxidation/explosion release mechanism (like the meltdown and vaporization processes) is also modeled only in a very gross fashion. More experiments, with larger samples of material, apparently need to be conducted to assure that the scaling mechanisms for this process are adequately understood.

The estimated results shown in Table 1 for releases by this mechanism are intended to indicate that if an explosion occurs, it will disperse and release the indicated fraction of whatever portion of the fission product in that category had not been released at the time of the explosion. For example, if only 10% of the volatile iodines had been released at the time of the molten fuel-water explosion, 0.9 of the remaining 90% -- or 81% -- would be released in the explosion -- for a total cumulative release of 91% of the nonvolatile fission products. This obviously is an important fission product release mechanism which deserves further experimental investigation to support development of meaningful methods of analyzing the molten fuel-water interactions.

The fundamental message of this brief examination of the source terms for meltdown release fractions is that physical models for essentially all the dominant mechanisms (with the exception of the relatively insignificant gap release terms) are only defined in the crudest of fashions.

A well defined program of experiments and analyses of each of the components and a subsequent integrated large-scale test program to verify the models appears to be needed. It was not possible to thoroughly evaluate the current NRC program in this area to determine whether it will meet these broad goals. However, a brief review of the past history of NRC studies shows that investigations of meltdown processes have received the lowest priorities. When funds were needed to supplement ubiquitous overruns in expensive experimental programs (such as LOFT), meltdown studies were commonly expendable. Future programs in this area should be given priorities commensurate with the importance of meltdown to reactor safety.

The columns of Table 1, labeled "fractional release to the environment" show the relative importance of estimates of fission transport and removal mechanisms as they function within the containment. These mechanisms are reportedly well enough understood to permit "conservative overall" prediction of fission product reduction processes within the containment following meltdown. However, it is also acknowledged that insufficient data exists to be able to accurately predict individual isotopic removal processes. 10/

As a result of concentration on the design basis accident goal of successful ECCS performance, most attention in decontainmination studies has been given to understanding and developing removal mechanisms for the more violatile fission product components, especially the iodines. Short of cryogenic removal, the noble radioisotopic gases Xe and Kr are not readily accessible to removal during their residence within the containment. Thus, except for the fraction retained naturally within the containment building during its decompression (as a result of an accident induced leak), essentially all of the noble gases will escape to the environment.

To provide quantitative insight into the significance of fission product transport and removal mechanisms for decontamenation within the containment building, specific results from WASH-1400 for several accident/consequence categories (designated PWR-1, PWR-2, PWR-3) have been shown in Table 1. These three referenced accidents have the following characteristics: 11/

PWR 1 This release category is characterized by an accident sequence initiated by various mechanisms, but dominated by a core meltdown followed by a steam explosion when the molten fuel contacts residual water in the reactor vessel. The steam explosion is assumed to rupture the upper portion of the reactor vessel which becomes a missile and breaches the containment barrier resulting in a substantial amount of radioactivity being expelled from the containment. The containment spray and heat removal system are also assumed to have failed.

PWR 2 This category includes failure of core cooling systems, and core melting concurrent with a loss of containment spray and heat removal systems. Failure of the containment barrier occurs through overpressure causing a substantial fraction of the containment atmosphere to be released in a "puff" from the containment.

PWR 3 This category involves an overpressure failure of the containment due to failure of containment heat removal. The core cooling systems are operating until the containment overpressure failure occurs. These systems are assumed lost when coolant, at the point of incipient boiling in the containment sump, flashes to steam as a result of the containment decompression and results in cavitation of the core cooling pumps. Core melting then proceeds to release fission products through a ruptured containment barrier. This meltdown case occurs over a substantially longer time period than the preceding cases.

As a result of the failure of containment spray and/or heat removal mechanisms, these three reference cases from WASH-1400 result in the largest estimated releases of fission products to the atmosphere. When spray mechanisms fail, only natural deposition mechanisms (discussed in greater depth subsequently) are effective for fission product removal. According to WASH-1400, only natural fission product removal mechanisms were considered for these types of accident/consequence categories. Moreover, no credit was reportedly taken in these cases, for leakage path decontamination factors (through the break in the containment) which would probably, in fact, be operative. 12/ Thus the basic assumptions relative to the cases examined appear to be conservative (i.e., would tend to increase estimates of fission products released to the environment). The application of the assumptions, and their implications to ultimate results, will be discussed in more detail, together with the discussion of the basic fission product transport and removal mechansims.

Iodine removal mechanisms are reasonably well understood and developed. If the containment spray removal mechanisms function properly, iodine concentrations can be reduced by factors of 100 to 1000 in relatively short times. Until the concentrations fall below one percent of initial values, iodine removal models are well substantiated by experimental results. If iodine removal with containment sprays is successful to at least these levels (a decontamination factor of 100 or more) the hazards associated with core meltdown could be greatly reduced. The more serious accidents (including those designated PWR-1, PWR-2, and PWR-3 in Table 1) are those in which the containment spray devices fail by any of several mechanisms, investigated in detail in WASH-1400. If the containment spray devices fail, only natural (gravitational) deposition mechanisms are operative. When iodine concentrations are high, natural deposition processes have been estimated to produce reductions by factors of about 1/4 of the initial concentration in an hour. 13/ Thus even if the sprays fail, if containment failure is delayed, or leaks are small, then substantial reductions in iodine levels could be achieved in relatively short times by natural deposition. For large leaks occurring while meltdown is still in process, natural deposition may not be this effective, as may be observed in cases PWR-1, and PWR-2.

In the case of the other fission product aerosols (all others except the noble gases and the iodines) spray removal mechanisms are not as well understood. Though the models are generally held to be conservative (i.e., they underpredict measured removal rates) they are acknowledged to be physically unreliable. Moreover, reproducibility of results in similar experiments is poor. Deviations by factors of 10 may be observed in measured decontamination factors for otherwise apparently similar experiments. 14/

For the cases of particular emphasis in this study, the evaluation of maximum consequence events, spray removal mechanisms have been assumed inoperative for the accident scenarios. Under these circumstances, concentration reduction for non-iodine aerosols was estimated to be very slow -- relative concentration factors being reduced only to about 9/10 of initial concentrations in an hour. As previously discussed, the inherent fission product decay energy within the melt evidently has the capacity to readily replenish the aerosols of the low volatility fission products so that the aerosols removed by natural deposition could apparently be maintained at the natural carrying capacity of the containment for extended periods -- even in the presence of large leaks.

Examination of Table 1, does not indicate that this fission product replenishment mechanism was recognized by the authors of WASH-1400. In reviewing the draft document, no explanation was found for the very low fractional releases (i.e., relatively high attenuation factors) for volatile and non-volatile oxides, in particular.

Considering both source release fractions and fission product transport and removal mechanism analyses for WASH-1400, it appears that results have not necessarily been conservatively (or sometimes even realistically) derived.

The implications of higher release rates were examined relative to the particular biological consequences of the PWR-2 release category. Detailed analyses of the whole body dose resulting from the PWR-2 release model were presented in the APS reactor safety study. 15/ Results were obtained on the basis of simplified, but adequate, dose-deposition models for dose evaluation once the fission products were released from the containment structure. Results of the study are summarized in Table 1.

In an attempt to assess the implications of the ranges of uncertainty relative to fission produce source release and transport and removal models, values of fission products released near the upper limits of variable uncertainty ranges were assumed to have reached the environment and biological consequences, in terms of whole-body dose, were estimated on that basis. The results have been labeled "Maximum Dose" Risk Factors in Table 1. The resultant whole body dose would apparently be increased by about a factor of four if release fractions were to approach these values. Results also demonstrate the substantially heightened roles of the alkaline earths and non-volatile oxides. If this upper range estimate were correct, the importance of the plutonium, cerium and zirconium isotopes would be significantly enhanced -- whereas they played a relatively minor role in the WASH-1400/APS results. Note that the iodines dominated the source of the dose in the WASH-1400/APS calculations of the PWR-2 results.

Assuming the validity of the standard linear dose-fatality relationships, increasing the whole-body dose by a factor of four would induce four times as many deaths from that source. Though the whole-body dose is just one element of a complex biological dose-conversion/fatality picture, it is interesting to extrapolate the implied increment in results to the curves of Figure 1. Since PWR-2 is one of the highest consequence accidents, if the consequence estimates for the tail of the curve were increased representatively, fatalities would exceed 12,000 -- and would begin to be similar to von Hippel's estimate of fatalities. If they were applied to von Hippel's estimates, the extrapolation could imply 40,000 to 80,000 deaths resulting from the accident.

How significant is an increase in estimated fatalities by about a factor of four? Applied to the annualized individual risk factor of about 3/1000 deaths/year from 100 operating reactors, it increases the result to only about 1/100 death/year -- a seemingly insignificant perturbation. When the factor is considered in terms of the difference between about 3,000 and 12,000 deaths, perhaps the significance depends upon how close the public is to reaching a tolerance limit on the acceptable number of fatalities from a single incident (or conversely on how abstract the number appears considering the extremely low probability predicted for the event). Considering the factor in isolation however, unless there is reason to believe the value should be rather substantially larger, there does appear to be reason to feel that there are probably more significant problems in nuclear safety than the uncertainties associated with fission product release from meltdown. Taken collectively along with the other uncertainties implied by the APS reactor safety study, and others, there is reason to believe that investigation of the physics of meltdown source release fractions and fission product transport and removal processes should be included as part of a systematic theoretical and experimental program for investigation of the problems associated with the most severe problem imaginable for the light water nuclear reactors, the meltdown accident.

Probable Initiating Events for Reactor Meltdown and LOFT Design Constraints

One of the more significant results of WASH-1400 was the quantification of the probabilities of many different initiating events relative to their leading to an accident with consequences ranging from serious to minor. Prior to publication of WASH-1400 it was generally conceded that the large double-ended "guillotine" break of the "cold" leg (the pipe -- approximately one meter in diameter -- containing the relatively colder fluid returning to the reactor, for recirculation, from the steam generator) LOCA led to the most severe consequences which were expected to be met by the reactor. Table 2 presents a synopsis of some of the WASH-1400 results which have led to altered concepts with respect to the most probable scenarios for these severe accidents.

Table 2 - WASH-1400 Estimates of the Probabilities of Certain Initiating Events Leading to Severe Consequences. 16/

Initiating Event	Probability (by Consequence Category) (Events/year/reactor) PWR-1	PWR-2	PWR-3
1. Large LOCA (D > 6')	2×10^{-9}	1×10^{-8}	1×10^{-7}
2. Medium LOCA (6" > D > 2")	3×10^{-9}	2×10^{-8}	2×10^{-7}
3. Small LOCA (D < 2")	1×10^{-7}	3×10^{-7}	3×10^{-6}
4. Check Valve	4×10^{-7}	4×10^{-6}	4×10^{-7}
5. Transient (electrical)	3×10^{-7}	3×10^{-6}	4×10^{-7}
Median Probability	9×10^{-7}	8×10^{-6}	5×10^{-6}

The results indicate that for the three most serious consequence categories, that other initiating events are from 10 to 100 times more probable to lead to a meltdown than the large break LOCA. In particular, failure of check valves which isolate the low-pressure ECC injection system from the high pressure of the primary reactor coolant system will lead to a 6" diameter break which not only has a direct piping path outside the containment, but also simultaneously fails one of the most important elements of the ECCS. The dominant transient failures (unanticipated events producing reactor shutdown) leading to serious consequences are those associated with electrical failure (both offsite and on-site power) to the decay heat removal systems for the reactor and containment vessel. Although a longer time is required for meltdown in this mode, unless power is restored to the heat removal systems within a period of between 1 and 3 hours, failure of the containment by overpressure is predicted. The small LOCA sequences contribute the largest overall probability to PWR core melt (when all other consequence categories are included). These sequences have relatively low leakage rates for which make-up fluid is added to the primary system by high pressure ECCS elements. Failure of the high pressure ECC system along with the break leads to the indicated consequence categories.

LOFT has been designed to evaluate ECCS repsonse under large break LOCA conditions. Would it be suitable for investigation of response under other conditions? Probably major redesign and reworking of hardware and perhaps instrumentation would be needed to make the system suitable for investigation of any of these other mechanisms. Dr. H. J. C. Kouts, Director of NRC's reactor safety research, noted to an Advisory Committee on Reactor Safeguards (ACRS) LOFT Subcommittee meeting that LOFT response to small break LOCAS would probably not be typical of the response in a commercial reactor. Differences in the LOFT high pressure injection system and the predicted dynamic system pressure responses would tend to make results atypical of small breaks in large commercial reactors. 17/ Similar problems would evidently exist in adapting LOFT to investigations of other types of initiating events.

Although LOFT may not be directly applicable to the investigation of other initiating events, because of their significance to reactor safety, it would be appropriate to now begin to perform the advanced planning for utilization of the LOFT facility to meet revised objectives of investigating the more probable accident initiation sequences. Perhaps in this fashion it would be possible to have a firm design for facility revision before it was time to start construction and fabrication activities. The practice of simultaneous program planning, facility design, and hardware fabrication during the current LOFT exercise appears to have been one of the major contributors to cost overruns and schedule slippages. It would be wise to avoid such practices, if future revisions are to be made to the facility.

Evaluation of LOFT Relative to Its Current Design Objectives

In a recent presentation of the status of the LOFT program to the ACRS, the following objectives were listed for LOFT: 18/

1) To verify realistic code predictions of the transient coupled thermo-hydrodynamic behavior of a reactor to a simulated LOCA in an integrated reactor system, and to verify the conservatism of "evaluation" models used in reactor licensing.

2) To check the correlations developed in separate effect and "semi-scale" tests with predicted scaling effects. Such correlations include: Time to Critical Heat Flux (CHF) and break discharge flow;

Post-CHF and reflood heat transfer; ECC coolant bypass of the core (flowing out the break instead) during blowdown.

3) To explore the ability of computer codes to predict the system behavior under varying modes of ECCS fluid injection, such as varying the injection location from the conventional cold leg location to direct injection into the lower or upper plenum, or to the hot leg.

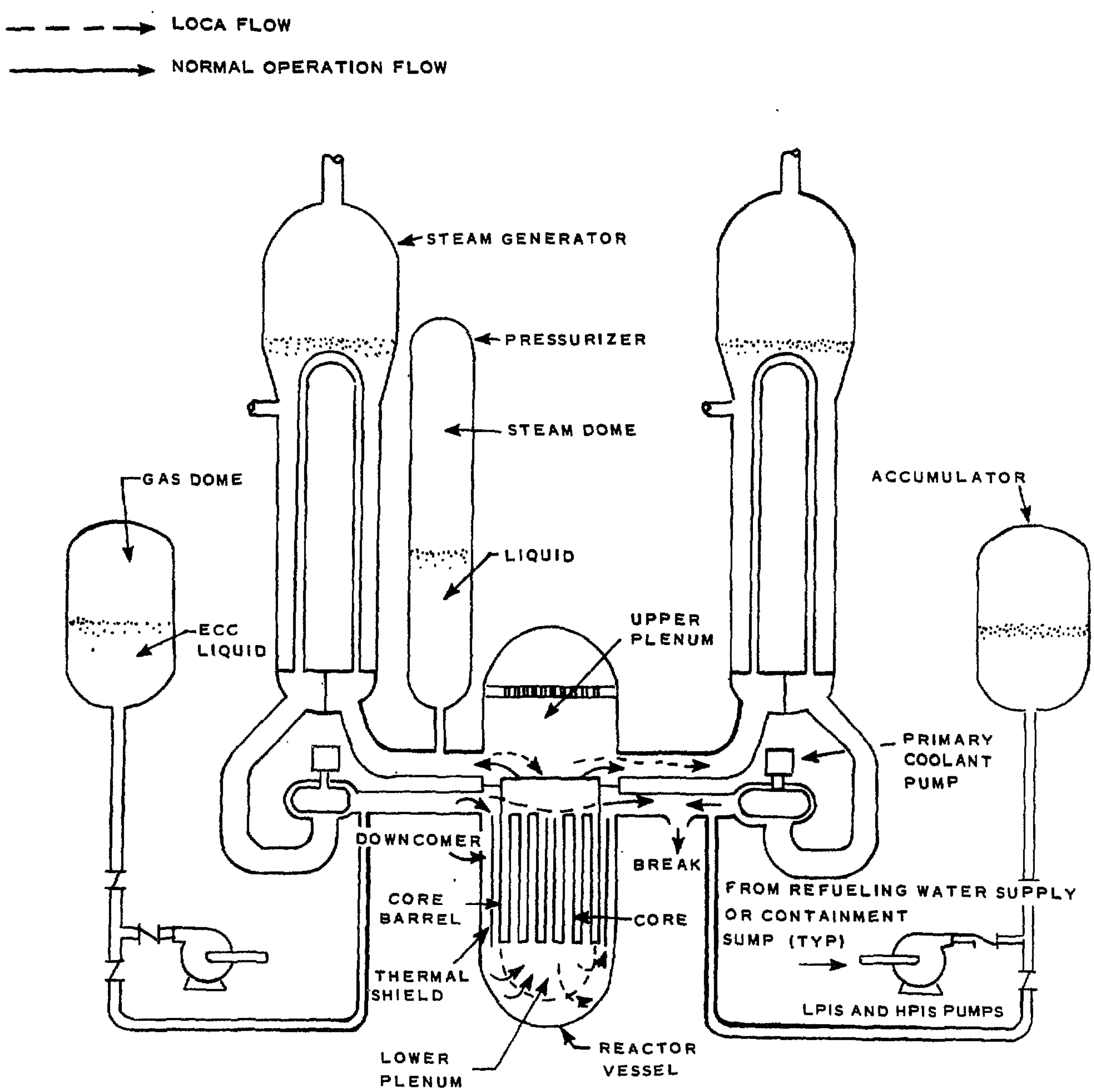
Only an integral system test-combining in one complete facility all the functional elements of the reactor nuclear steam supply subsystems could hope to satisfy the above objectives. LOFT represents NRC's culminating program in which all of the elements of the individual separate effects investigations conducted can be integrated into a complete unified system for verification. Dr. Kouts described the function which a system test like LOFT performs for reactor safety research. 19/ He observed that only through such a test can calculational methods and models for evaluation of reactor response to a LOCA be examined to: (1) detect potential oversimplifications in the analysis routines; (2) discover significant phenomena which may have been overlooked in models; and (3) reveal failure of the model to account for non-linear, synergistic or auto-catalytic effects which may occur during the transient response of the reactor.

It is true that a balanced program for reactor safety research must contain detailed examination of separate effects of isolated elements of the system, separated from other complicating elements of the system. Tests and analyses must be conducted on these individual subsystem elements until adequate models have been developed to describe the individual components. However, only an adequately simulated system test will provide a means for detection of these critical elements of the problem -- perhaps uniquely related to the integrated system performance.

Adequate system simulation, however, is not assured simply by integrating scale models of the subsystem elements into a whole without regard to critical aspects of system scaling. A nuclear reactor presents a complex physical picture to describe during the sequence of events associated with a LOCA. In the brief course of the accident, fluid flow in the system changes from relatively incompressible high pressure liquid to two-phase (steam-liquid) flow, and finally to relatively stagnant conditions of saturated and/or superheated steam. Heat transfer during the process is equally difficult to analyze on the basis of first principles or with complete theoretical rigor. Heat transfer analyses have historically been conducted on a semi-empirical basis (i.e., combined theoretical and experimental analyses are pre-requisites to modeling the processes). Application of semi-empirical analysis methods under unusual conditions, or to a new configuration for a piece of equipment, or for larger or smaller visions of geometrically similar equipment requires a thorough understanding of the scaling relationships upon which the semi-empirical models have been constructed. In the case of heat transfer in the reactor, many of the important analysis methods are being used in regimes where these applications are uncertain and considerable extrapolation from measured data is required. In these cases, the appropriate scaling relationships to use with the analysis methods may be quite uncertain.

Even the mechanical response of the system is important during the LOCA and intimitately coupled to the fluid flow and heat transfer processes. Preservation of the mechanical integrity of the core, prevention of fuel rod bending and distortion, and minimization of fuel rod swelling and rupture (as their temperatures increase) are vitally important in the design of the reactor. Loss of core integrity or possible development of blockage can lead to restricted flow (analysis of locally three-dimensional flow is beyond the scope of current LOCA/ECCS computer codes) with strong coupling between resulting fluid flow patterns and consequent altered heat transfer.

This discussion has highlighted only a few of the complicating factors which make the development of methods difficult for analyzing the transient response of a reactor during a LOCA. As a consequence, it should be recognized that computer codes for LOCA/ECCS transient response analysis are of necessity simplified engineering analytical tools. The are <u>not</u> ideal codes derived from the basic principles of physics; such as a simple application of numerical methods to Newton's laws of motion, coupled with fundamental relationships for conservation of mass, momentum, and energy, and equations of state for the materials involved. On the contrary, the simplifications required to provide an analysis tool for these complex geometries and phenomena have required analysts to model the system in terms of a large number of semi-empirically defined individual "components". Figure 2 presents a schematic diagram of the system relationships of most of these components. Models for each of these components are based upon individual "separate effects" tests and analyses and upon representative individual scaling studies. These "components" include a considerable breakdown of the reactor system. For example, the following elements are



SCHEMATIC DIAGRAM OF PWR STEAM SUPPLY SYSTEM SHOWING EMERGENCY CORE COOLING SYSTEM (ECCS) ELEMENTS

FIG. 2

considered to be "components" for modeling purposes: the reactor vessel subdivided into eight distinct portions (upper plenum, upper head, reactor core, fuel cladding, fuel pellet, lower plenum, downcomer, and upper annulus); the steam generator, the pressurizer; the primary coolant pump; the ECCS (including separate descriptions (models) of the accumulator, the low-pressure injection system (LPIS); the high-pressure injection system (HPIS), and ECCS injection method/location); the piping; the break; and the containment building. Each of the component pieces is then integrated into the LOCA/ECCS analysis code. The adequacy of this code then depends not only on the adequacy of the individual component models (and their own scaling relationships) but also on the adequacy of the integration routines (including descriptions of inter-relationships between "components" -- some of which were briefly alluded to earlier) and the codes completeness in modeling all aspects of the system.

To verify the validity of the integrated code, integral systems tests must be conducted. As Dr. Kouts noted, there is no other way that over-simplifications in the code, overlooked phenomena, or unpredicted effects which are nonlinear, synergistic, or autocatalytic in nature can be detected. But since the codes themselves are dependent heavily upon semi-empirically derived models, for which scaling may in most cases be uncertain, then scaling of the experiment becomes a critically important part of the test equipment design.

The scaling of LOFT has been reviewed in considerable detail. In over a century of engineering practice, classical scaling relationships have been developed by which models of facility designs can be evaluated. These scaling relationships show important interrelationships between physical variables which must be preserved between sub-scale and full-scale pieces of equipment. Generally speaking, in a problem involving as many physical phenomena as a reactor undergoing a LOCA, it will not be possible to scale the equipment dimensions in such a way that all of the important scaling relationships can be simultaneously satisfied. If the difference in physical size is not large between the sub-scale model and full-scale equipment, then the effects of the necessary compromises between the more important scaling parameters on the system response may be relatively insignificant. As a general rule of thumb, extrapolation of the results of complex hydrodynamic systems (or solid-elastic plastic systems) over a factor of no more than 3 or 4 in volume scaling has been reasonably successful. Extrapolation of results over much larger ranges is generally impractical.

There are basically two integral system tests for evaluation of LOCA/ECCS models, the LOFT and Semi-scale facilities. At 55 MW of nuclear thermal power, LOFT is about a 1/60 scale model of a commercial 1000 MW (electrical power) reactor -- which will have approximately 3300 MW of thermal power. However, not all elements of LOFT have been scaled to the same geometric relationships. LOFT required many scaling compromises to attempt to model the LOCA response phenomena in the way which the designers felt would be representative of actual practice. These design scaling compromises were generally based upon analyses of the reactor performance made with the analysis methods which the models are intended to verify. It is evident that many opportunities for circularity in the facility design and consequent measured performance are possible in the implementation of a program involving such scaling compromises and interrelated design and performance analysis methods. Though at 1/60 scale, compromises may have been required to improve the probability of simulation of full-scale system performance, they are certainly undersirable for assuring that verification of the adequacy of the code predictions will be acheived.

"Semiscale" is a 1.07 MW (maximum thermal power) electrically heated, "little brother" of LOFT. As a result of many scaling compromises included in the Semiscale design, it is difficult to make a direct comparison of its scaling relative to LOFT or a full-scale commercial reactor. It is frequently asserted to be approximately 1/30 scale of LOFT -- and hence 1/1800 of the scale of a commercial reactor. However, on the basis of its thermal power to volume scaling, it may be nearer 1/3000 scale of a commercial reactor.

In spite of its extremely small scale, Semiscale plays a very important role in LOCA/ECCS system analysis. It is the only integrated system test facility available for which any serious attempt has been made to incorporate all of the previously described individual "components" properly into the system facility. Thus practically all of our current evidence for system code adequacy is now dependent upon correlation of Semiscale results and code predictions.

Observations Relative to LOFT Program Adequacy

1. There is an important need for integral system tests of reactor performance under accident conditions. As such, LOFT performs a significant role in increasing confidence in the evaluation of ECC system performance. It is not, however, designed to address many other significant

elements of reactor performance with equal or greater significance to reactor accidents.

2. At a scale of about 1/60 of a commercial reactor, LOFT performance cannot be expected to be extrapolatable to commercial reactor performance -- whether LOFT results are good or bad. Nor can LOFT be considered as a "demonstration" test of the adequacy of ECCS performance, as a result of the requisite scaling compromises incorporated into the facility. These scaling compromises assure that the similarity of the LOFT response to that for a LOCA in a full-scale reactor will not be complete.

3. LOFT will provide an opportunity to test the validity of integral system performance codes. Some subsystem models have a fairly high probability of being adequately verified such as break flow and time-to-critical heat flux estimates, etc. Other important elements of the analysis will be poorly simulated such as ECC fluid bypass during the blowdown; and as a consequence, time required to refill the lower plenum; steam binding phenomena; reflood rates, fuel swelling and rupture with consequent influences on core blockage and resulting three-dimensional flow effects about the blocked portions of the core. Many of these phenomena are of great apparent significance to the thermal response of the core during a LOCA, irrespective of uncertainties in their modeling in the integrated LOCA analysis methods. Poor simulation of these phenomena in LOFT, with consequent lack of model verification for the pehnomena in the LOCA/ECC codes, makes the phenomena increase in relative significance -- almost in direct proportion to the uncertainty in their predictability.

4. Important information pertaining to the relative performance of alternate ECC delivery modes will be obtained in LOFT. The results of investigations of ECC fluid insertion into upper and lower plenums as well as the hot and cold legs of the reactor will provide significant insight into relative strengths and weaknesses of such alternate ECCS concepts.

5. LOFT results will not be complete enough to provide verification of ECCS performance analysis methods to the satisfaction of the majority of the reasonable members of the scientific community. They will provide an important basis for maturation and improvement of the codes -- but this is not the same as code verification.

6. The probability that another, larger scale, more definitive test will be needed to truly provide code verfication is very high. Planning for such a test should be initiated at once.

Supplement

F. C. Finlayson

14 February 1976

Cost Effectiveness of Large-Scale Testing of Reactor Core Meltdown Prevention Systems

If nuclear power is to remain a viable energy source in this century, a high probability exists that LOFT will ultimately need to be supplemented with a large-scale test program of reactor core meltdown prevention systems. Convincing demonstration of the effectiveness of Emergency Core Cooling Systems (ECCS) is obviously cost-effective, even if such large scale testing is required. These tests would no doubt be expensive, with costs conceivably approaching the order of a billion dollars. However, compared with the annualized busbar costs of energy production from the reactor industry (of the order of tens to hundreds of billions of dollars) the experimental program costs seem relatively insignificant. This is especially true when it is recognized that the accumulated value to electrical utilities of the energy production from the reactor industry over the period from 1975 to 2000 AD is of the order of a trillion dollars.

Demonstration of the effectiveness of the ECCS, and other related meltdown prevention systems, would eliminate a large portion of the basis for public concern over the risks of high consequence accidents. In the absence of such a demonstration, the potential appears to be high for continued growth in public concern over reactor problems -- when amplified by outspoken, highly visible nuclear critics. The growth of legal action (similar to the current California anti-nuclear initiative) is the apparent alternative to failure to recognize the need and rapidly initiate the necessary supporting planning programs for a large-scale test program. Those arguments which oppose large-scale testing based largely upon its costs, appear to be insensitive to the potential magnitude of the ultimate costs of failure to convincingly demonstrate the effectiveness of systems designed to prevent core meltdown and resulting high-consequence accidents.

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BIOGRAPHICAL INFORMATION

Fred C. Finlayson, Staff Engineer, Energy Programs Group
Energy and Resources Division

SPECIAL QUALIFICATIONS

As Staff Engineer, Energy Programs Group. Dr. Finlayson has been responsible for a variety of projects dealing with nuclear power, energy supply and demand, and advanced power conversion concepts. He has completed an independent review of the effectiveness of the emergency core cooling systems (ECCS) for nuclear power plants and has recently served as a member of the American Physical Society's select committee on Light-Water Reactor Safety. He has also directed and conducted recent research in the general aspects of the safety of nuclear power generation including a study of hazards associated with transportation of spent nuclear fuel in the western regions of the United States. Dr. Finlayson has also been actively conducting research in the broader areas of energy supply and demand, having conducted recent investigations of the conceptual design and evaluation of hybrid solar/geothermal power systems, as well as a variety of studies in energy consumption and the effectiveness of specific conservation measures.

EDUCATION

B.S., Mechanical Engineering, Brigham Young University, 1958
PH.D., Mechanical Engineering, Northwestern University, 1964

EXPERIENCE

The Aerospace Corporation (1972 - Present)

Dr. Finlayson is currently responsible for planning and conducting programs in energy systems analysis and hazards analysis of elements of the nuclear fuel cycle. In a previous assignment, he was temporarily attached to the Environmental Quality Laboratory of the California Institute of Technology where he was responsible for evaluation of problems in nuclear power plant safety.

Physics International Company (1968 - 1972)

Dr. Finlayson was Manager of the Systems Development and Assessment Department where he directed and conducted research related to strategic and tactical weapon systems survivability/vulnerability and numerical analyses of the propagation of strong shocks in geologic media and structural materials as well as structure-medium interactions.

The Aerospace Corporation (1964 - 1968)

Dr. Finlayson was Manager of the Ground Systems Survivability Section of the Nuclear Effects Department where he directed and conducted investigations of ground based system survivability to all relevant effects of nuclear weapons.

The General American Transportation Corporation (1960 - 1964)

As a Researach Engineer in the MRD Division, Dr. Finlayson conducted research on the interactions of strong shocks in air and earth materials with above-ground and buried structures.

PROFESSIONAL ACTIVITIES

Dr. Finlayson is the author of a number of papers and reports on the dynamics of strong shocks in solids and fluids and their interactions with structures, as well as the safety of nuclear power reactors, energy consumption and conservation. He is currently a member of the American Geophysical Union and the American Nuclear Society.

REPORT TO THE

U.S. GENERAL ACCOUNTING OFFICE

ON THE

REVIEW OF THE NRC/ERDA

LOSS-OF-FLUID-TEST FACILITY

BY

ROMANO SALVATORI

WESTINGHOUSE ELECTRIC CORPORATION

NOVEMBER 14, 1975

OUTLINE

LIST OF FIGURES

1.0 SUMMARY AND CONCLUSIONS

The U.S. General Accounting Office requested that I and four other consultants review the question of whether the LOFT test program and facility are adequate to answer today's relevant light water reactor safety questions. I truly believe that, because of the numerous "separate effect" and "system effect" tests (some key ones in support of the LOFT program) already performed and the philosophy used in the design of nuclear power plants, i.e., upper bound and range of assumptions rather than best fit assumptions, there are no unanswered relevant safety questions in the area of ECCS performance following a pipe rupture. Therefore, I do not expect any "safety breakthrough" from the LOFT program.

The LOFT program will contribute, however, toward improving the understanding of localized phenomena following a pipe rupture such as pump performance, break flow, flow regimes in various components, test instrumentation adequacy, nuclear fuel rod behavior, steam generator heat transfer, etc.

Should we then redirect the LOFT program? I do not believe so. We should go ahead with the current plan of "producing experimental Nuclear Steam Supply System (NSSS) data capable of validating or maturing analytical LOCA predictive codes over a full range of ECCS performance levels." We should also finalize plans for utilizing the LOFT facility for non-LOCA experiments (Attachment 4).

In particular, I do not believe that LOFT should be used "to study means of retaining molten cores and measuring the consequences of steam explosions and radioactive releases resulting form a meltdown" or to study "the containment's ability to control fission product activity."

This report is organized in three main parts. The first part illustrates the approach I chose in addressing the GAO questions; the second part gives my ideas on the overall philosophy of a safety R&D program and the role of LOFT in it; and the third part addresses each GAO question in detail.

2.0 INTERPRETATION OF ASSIGNMENT

The U.S. General Accounting Office requested that I and four other consultants review the LOFT program from both the standpoint of cost and schedule and whether the test program and facility is adequate to answer today's relevant light-water reactor safety questions. They also requested that we express ourselves in as non-technical terms as possible.

At the briefing on September 18 in Idaho Falls the GAO representative also informed the consultants that GAO was looking for individual reports to them and not a consensus report. GAO would undertake the task of responding to the Senate Committee on Government Operations utilizing whatever they felt appropriate in the consultants' reports.

In carrying out the assignment, I have chosen not to address every scientific or engineering detail under controversy in this or that arena. In order to do this I would have needed a significant additional amount of time which may not have been of sufficient benefit to GAO. My report would have been another scientific or engineering critique that would have added my opinion to already existing thousands of opinions on this or that microscopic detail. I strongly feel that we have already been polluted, above safe limits, by opinions on various types of details. We must leave discussions and resolutions of scientific and technological details to constructive and cooperating scientists and engineers, in the proper forums like the pertinent departments of universities, national laboratories, regulatory agencies, manufacturers, consulting agencies, etc.

I have chosen, instead, to a) consult with selected specialists, b) study selected material, c) utilize my more than 10 years experience in the nuclear safety field, and d) formulate broad, microscopic answers to the questions posed to us by GAO. Today's vast amount of printed material and large number of experts and pseudoexperts forced me to be selective in order not to make a career project out of this assignment. I am not a specialist in any single field. Instead, I consider myself a nuclear safety engineer/manager. By this I mean I consider myself an "integralist" with the capability to ask questions of specialists, listen to them and their answers, put these in perspective with regard to their costs and their benefits, draw an overall judgment and translate this judgment back into "microscopic" terms so that scientists and engineers can design and build separate pieces which will have high likelihood to fit together and yield something that works usefully and safely.

This concept of safety engineer/manager is graphically illustrated in Figure 1.

I wil stand behind my overall recommendations and the reasoning that led me to them. If the GAO or the Senate Committee on Government Operations is interested in pursuing a "microscopic" scientific or technological point, I am sure many experts can be found to address that detail. If requested, I will be happy to assist in the identification of such specialists.

ROLE OF SAFETY ENGINEER/MANAGER

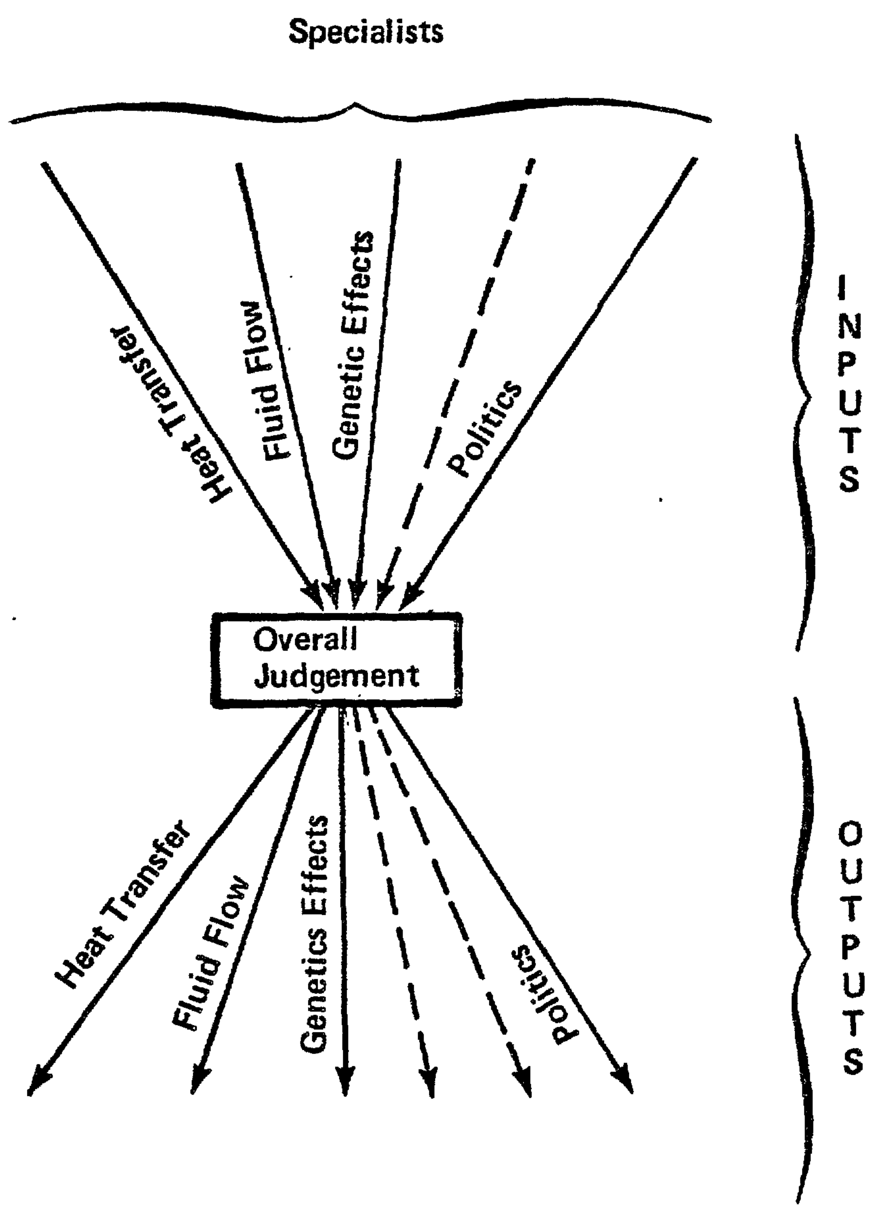

FIGURE 1

3.0 SAFETY R&D AND THE ROLE OF THE LOFT PROGRAM

I have chosen to spend some time in the following pages to present my viewpoint on the overall philosophy of an R&D program based on my own experience as well as that of many scientists and engineers I have been in contact with over the years. I believe that this discussion is pertinent to the assignment at hand and will make much easier future discussions more closely related to the LOFT program role.

Therefore, I plead with you to bear with me for a few minutes.

In planning a safety R&D program or any R&D program in general, the whole "system" is first investigated theoretically. An R&D program does not spring out of nowhere. Instead its scope is defined relative to its impact on the "final answer." For example, when the requirement for "maintaining the core in a coolable geometry following rupture of any reactor coolant pipe" was imposed, all affected parties (e.g., reactor vendors, regulatory agencies, consulting outfits, etc.) increased their efforts in analyzing the behavior of the reactor coolant system and the nuclear core contained in it. Overall theoretical system models were developed. Using these models, sensitivity analyses to variations of all pertinent parameters were conducted. These studies contributed to the identification of those parameters or areas which not only had a major impact on the "final answer" but also could cause large variations in such "final answer" as a consequence of only small changes in their value. Some areas were so complex that the status of the art could not allow a complete analytical representation of their behavior. For the sake of this discussion, let us assume that the "final answer" is the peak temperature of the uranium fuel cladding pellet (PCT). The intent is to keep such PCT below a pre-established value, say 2200°F. We will not discuss here the need for this limit and whether we could tolerate higher PCT or even localized melting.

At that point in time, the nuclear industry adopted a four-pronged approach. The approach was a logical one and it is applicable to any other industry. Firstly, the industry concentrated in making the initiating event, i.e., reactor coolant pipe rupture, even more unlikely than before by improving the quality of the pipe, installing leak detection systems to give early warning of small crack appearance well in advance of when they may propagate around or along the pipe, improving techniques and procedures for periodic inspections of the reactor coolant system, etc. Details on this point can be found in the licensing documentation of a typical nuclear power plant.

Secondly, a significant, high priority effort was started to improve the analytical tools in order to get a better understanding of the behavior of various systems and components under accident conditions. This effort not only addressed the hydraulic, thermal and nuclear behavior but also the mechanical one and, in the most critical areas, their relative interactions.

Thirdly, whenever available analytical tools were not advanced enough at the time to give a realistic representation of system/component behavior under accident conditions bounding assumptions were adopted: either an upper bound or a range of assumptions wide enought to have reasonable assurance to have bracketed the actual value. Whenever knowledge is not complete, a scientist and a safety engineer sometimes depart in their viewpoint of how much knowledge is necessary before something can be built and operated safely. The scientist tends to search for the exact behavior of a given parameter or a given component. A safety engineer starts the same way but he does not wait until he knows everything about everything. When he has reached an amount of knowledge that allows him to establish upper bounds or safe ranges, he studies the pros and cons of waiting for more knowledge or going ahead. If the benefits of going ahead outweigh the costs he will decide to go ahead in a safe way. For this reason, critiques by specialists must be viewed in context. They are very useful in making microscopic decisions in the area of specialty of that given expert. However, these critiques are <u>only</u> one of many inputs necessary to make a policy decision. Policy decisions should be made by "integralists" not by "specialists".

An example which illustrates this point is represented by the report to the American Physical Society by the Study Group on Light Water Reactor Safety (28 April 1975). This report contains a series of good "scientific" suggestions. The report states "Many (if not most) of the scientists and engineers involved with reactor design feel that the requirements of the ECCS Acceptance Criteria are excessively conservative and would be relaxed if better quantitative data were available. Nevertheless in our opinion, there is a substantial need for quantification of ECCS adequacy." But, the report also states that "We have not studied the benefits of nuclear power, much less attempted to weigh them against the risks; therefore, we cannot answer whether existing reactors are safe enough." Thus, these specialists have recognized that, before people take the American Physical Society report and run with it to either slow down nuclear power plants or to invest millions of dollars in additional safety research, the benefits and the costs of such actions must be studied and balanced.

Going back to the main train of thought (i.e., the four-pronged approach to safety), the fourth direction adopted, in parallel, by the industry was an agressive R&D program. The nuclear safety R&D program in general proceeded along the following main directions:

a) Obtain experimental results in the areas where, due to limitations of the state of the art, unrealistic conservative assumptions had to be made. The intent here was to get a better handle of reality so that, at a later date, the excessive conservatism could be reduced and used to either reduce the cost or to increase plant availability through more maneuverability.

b) Obtain better analytical and/or experimental knowledge in areas where the state of the art might have been extrapolated too much but still considered adequate because of high confidence of large conservatism in other areas. The intent here was to shift, in time, from high confidence of an overall conservatism, i.e., PCT less than a safe value, to high confidence that each separate area or assumption having an impact on the final result (e.g., peak clad temperature) is conservative by itself.

c) Obtain pure and simple verification that interpolations or extrapolations of existing knowledge with the added tough of conservatism were indeed adequate.

I am not including here various R&D programs undertaken with private goals in mind, e.g., to develop less expensive systems or to improve verification to obtain a market edge.

As a result of numerous meetings, private and public, among scientists, engineers and safety engineers, many different R&D programs were initiated.

No matter whether the experimenters were national laboratories, NSSS manufacturers, universities, etc., they all decided to run separate effect tests first. I will cover later on how the LOFT program fits in the picture. The reason is obvious: if you try to understand a phenomenon, you do not cloud it with many other phenomena in a complex integral test, otherwise you do not know what affects what and it is very difficult to develop correlations. For this reason, you waste a lot of time, money, sleep and achieve very little with integral tests. On top of it, pseudoexperts, parasites and people with their own goal in mind raise hell everytime you run an integral test without exactly matching your ante-facto prediction forgetting that the main reason for running the test was to learn. The same people also forget that to take care of the temporary lack of specific knowledge in a given area, upper bound assumptions or more margin in another area were adopted so that the final result, e.g., PCT, is conservative.

Going the route of separate effect tests really leads to getting an answer. Furthermore, separate effect tests can be directed and run by the experts in that particular field. If you run integral thermal, hydraulic, mechanical, nuclear, etc. tests at one time and in various areas like vessel, pumps, steam generators, etc., it is pretty difficult to pull together a team covering all these disciplines. Also, while earlier I said that we do not want specialists to make policy decisions, at the same time we do not want "integralists" to run specific tests. Separate effect tests can also be run in the proper test facilities since they are limited in scope and size and they can be properly instrumented. The approach of concentrating on separate effect tests and running system tests only when necessary to bound the "system inputs" to the separate effect tests is not peculiar to the nuclear industry. Industries involved in large structures which, if they fail, could put public safety in jeopardy, such as ships, dams, airplanes, buildings use the same approach. I have not heard of any large building, seismically designed and provided with anti-fire systems, subjected to the large forces of an earthquake or put on fire to check whether the structural design and the fire extinguishers are adequate.

Attachment No. 2 contains a list of all the core cooling related separate effect tests since the mid-sixties at the best of my recollection and the recollection of my files. As you can see the list is impressive. But before going on, I believe it is worthwhile to elaborate on what a separate effect test really is. Figure 2 contains a schmatic of the reactor coolant system which provides a boundary to the core coolant. PWR vendors and the NRC and their consultants using different computer codes have concluded that the behavior of the Reactor Coolant Pump during all phases of the accident plays a significant role in what final temperature the nuclear fuel cladding reaches. In reaching this conclusion, not just one analysis was performed but literally hundreds of analyses varying all significant parameters to make sure that there was not combination of parameters which gives a surprise. Data were not available on the actual behavior of a pump of this type under the extreme conditions represented by the double-ended severance of a reactor coolant pipe, e.g., two-phase flow with changes from one phase to the other, high flow rates, etc.

While planning a test program, agreement was reached on what would be a conservative behavior or set of behaviors for safety design of nuclear power plants. Again , in order to proceed with the design of nuclear power plants, the characterizations of a reactor coolant pump which gave the highest uranium fuel clad temperature were adopted independent of whether they were real or not. In parallel various test programs were initiated by private industries with and without government funding to better characterize the pump behavior and remove the excessive conservatism in the nuclear reactor design at a later date.

It may be worthwhile to mention at this point the significant contribution to safety that comes from keeping the results of private R&D programs confidential. By proceeding this way affected vendors are obliged to run their own test program since they do not get the results of their competitors' tests. The NRC then gets all of them with the benefit of comparing one against the other and making sure that nothing has been overlooked.

Going back to the sample of the separate effect test program on the reactor coolant pump, the entire system was analyzed in order to determine what the pump had to be tested against. By running a series of analyses varying all pertinent parameters, including various size breaks from a simple crack to the rupture of the largest pipe, the test conditions (e.g., coolant flow, temperature, pressure, density, etc.) and how they vary in time, were selected. Figure 3 illustrates this point. Attachment 3 describes the Westinghouse separate effect test programs on the Reactor Coolant Pump. The intent here is to give an idea of the extent and complexity of these separate effect tests. Sometimes I get the feeling that many people to not really appreciate separate effects tests but they feel they are quick and dirty tests run in somebody's garage.

As Figure 3 shows the inputs to the pump test program are represented by the overall system response to the initiating event, e.g., pipe rupture. These inputs are determined by running a series of sensitivity analyses. Sensitivity studies are analyses performed by varying the imput parameters to determine how sensitive the "final answer" is to these variations. A controversy starts at this point. The typical question asked is: "We believe your separate effect tests on the pump are okay. By this, we mean that you know how your pump behaves under the conditions you have specified as 'system inputs.' But how do you know that what you call 'system inputs' is correct? After all they are only based on your theoretical analyses performed with your imperfect codes. You need 'system tests' to make sure that the 'system inputs' to your 'separate effect tests' are accurate. You need 'full scale system tests.'"

The answer to this combination of statements and questions is as follows. First of all the "system inputs" for the separate effect tests (the pump test in this case) are not superficially determined. As I said earlier, hundreds of sensitivity analyses are performed before the test facility is built and during the period the actual testing takes place. Such analyses are reviewed by experts from the manufacturers who decided to run the tests, the regulatory agencies and any consultant they feel appropriate. Not just one set of test conditions is selected but a long series going even outside any reasonable system behavior following a catastrophic pipe rupture. Also the results of the tests are plugged back into the sensitivity studies to again confirm applicability of the separate effects tests. Let's remember again the different role of the safety engineer and the pure scientist. The safety engineer does not want necessarily to exactly understand nature but he wants confirmation that his upper bounds or ranges of assumptions are reasonably conservative. When this goal is kept in mind, analytical studies of overall system behavior with today's knowledge are quite reliable.

I do not want to give the impression that I am flatly against "system tests." I am not. What I am strongly against is the implication that separate effect tests are no good unless they are combined with full scale systems tests. People who support this theory either have never run R&D programs, especially safety R&D programs, or have different objectives in mind. The request for a full scale or near full scale test facility is, in my opinion, completely unwarranted. Could the objective of their proponents be to kill the nuclear program by slow death? Let us assume we find a couple of billions of dollars or more to invest in such a facility. In today's environment with a great majority of Doubting Thomases and very few Saint Augustines, it might take 3 to 5 years to agree on what we want to do with such a facility and to get a construction permit. It might take about 10 years to build it and probably an additional 3 years before any meaningful nuclear test can be run. Hence, with a decision to go ahead today, it will take more than 15 years before we get any useful answer. And we know very well that a decision to appropriate that amount of money will not be made overnight. Hence, the question of whether we need a larger LOFT facility approaching present reactor size is an academic one. We must decide on whether to exploit nuclear power to its fullest potential with no such facility. We cannot wait more than 15 years to make such a decision. It would then take more than an additional 10 years beyond that before commercial nuclear power plants can be put on line assuming that we can turn off and on the nuclear industry.

But, would it be desirable? Additional knowledge is always desirable. Only broad cost/benefit analyses can determine whether such desire warrants such large investment with a return more than 15 years from now when a decision on the extent of nuclear power utilization must be made today. I would like to submit that a broad cost/benefit analysis has already be performed. You are all aware of the so-called Rasmussen Safety Study. Rasmussen and his team have performed a study of the adequacy of ECCS, if called upon, put such results in perspective and concluded that the risk to the public from potential accidents in nuclear power plants are very small. He has also concluded that non-nuclear events are about 10,000 times more likely to produce large accidents than nuclear plants and that nuclear plants are about 100 to 1000 times less likely to cause comparable large dollar value accidents than other sources. The table on Figure 4 is taken from the August 1974 Draft Summary Report by the U.S. AEC on the Rasmussen Study.

At this point, I would like to submit that, if we have one or two billion dollars to invest in public safety, we do not improve it a darn bit by running more ECCS tests or by increasing reactor safety in general. Such money should be invested in making automobiles, firearms and airplanes safer, or in medical research or in many other things that control our lives to a much larger degree.

Going back to the point I made earlier, I do not want to leave you with the impression that I am flatly against "system tests." I am very much in favor of using them when appropriate and not to verify every system input to every separate effect test before the results of such tests can be used. Attachment 2 includes system tests already performed or planned. As this attachment shows, system tests have been performed in many areas. Such "system tests" (e.g., Flecht-SET, Semiscale, etc.) have confirmed the adequacy of the safety assumptions made in designing nuclear power plants based on separate effect tests.

The LOFT program fits logically in the progression of R&D aimed at improving the understanding of the phenomena associated with a sudden rupture of a reactor coolant pipe. Will LOFT contribute to the understanding of the reactor behavior following a pipe rupture? The answer probably is yes. Scientific knowledge will be improved in localized areas, such as pumps, break flow, flow regimes in various equipment, test instrumentation, fuel rods, steam generators, etc. Will the LOFT program significantly contribute to improving the safety and licenseability of commercial reactors? I do not believe so. Let me say it again, a safety engineer bases his design of power plants on upper bounds and ranges of parameters. The LOFT program will mainly provide, as formulated by the Aerojet Nuclear Company: ". . . experimental NSSS data capable of validating or maturing analytical LOCA predictive codes over a full range of ECCS performance levels." I do not believe LOFT will provide a major breakthrough in safety-related areas. The answer would have been different 5 years ago. As shown in Attachment 2 a significant number of "separate effect" and "system effect" tests have already been performed. Some of them were directly in support of the LOFT program, e.g., semiscale, etc. These tests due to their scope and their timeliness have been very useful. Further discussion of the various aspects of the LOFT program is contained in the subsequent sections which deal directly with the specific questions asked by GAO. I would like to address in this section only the general question whether the test program and the facility is adequate to answer today's relevant light water reactor safety questions. I truly believe that, because of the numerous "separate effect" and "system effect" tests (some key ones in support of the LOFT program) already performed and the philosophy used in the design of nuclear power plants, i.e., upper bound and range of assumptions rather than best fit assumptions, there are no unanswered relevant safety questions in the area of pipe rupture and ECCS performance. The LOFT program missed its chance to directly address relevant safety questions in this area when it started running more and more behind schedule. As I said earlier, the LOFT program will surely contribute to a better scientific understanding of many phenomena but this understanding will have little impact on the safety design of nuclear power plants.

The other question that can be asked is whether the LOFT program will improve public confidence in the adequacy of the Emergency Core Cooling System and therefore in the safety of nuclear power. I am sorry to be obliged to give another negative answer because of the way the nuclear controversy has shaped up. Some of the most outspoken critics of nuclear power still reject the claim of adequate safety because ECCS did not work as proved by "six tests at Idaho." These tests conducted by Aerojet Nuclear Company and labeled tests 845 through 851 have been time and time again recognized as completely atypical of commercial light water reactors and they disappeared even from the list of contentions in the ECCS rulemaking hearing after a few months of discussions. What confidence do I have that vociferous critics who oppose nuclear power for completely different reasons will believe the verification of adequacy that will come from LOFT? They will stress the atypicalities between LOFT and current commercial reactors, the small size, etc. The public will be as in doubt as ever, in this area.

Should we then mothball the LOFT program? My answer is clearly no. The goal of "producing experimental NSSS data capable of validating or maturing analytical LOCA predictive codes over a full range of ECCS performance levels" is a valid one and will be achieved. This will give confidence to a large sector of the scientific community about the adequacy of ECCS. Also, as I said earlier, it will give a closer insight into many phenomena and the facility can be used to run a series of tests not related to reactor coolant pipe rupture and ECCS performance. Balancing these benefits with the additional relatively modest cost to continue the program or the large political and psychological costs that will be incurred if the program is stopped, my recommendation is clearly to go ahead with LOFT and not delay it any further. Significant effort should be invested, however, in carefully planning each test, predicting the key results and writing comprehensive but clear reports on each test phase.

WESTINGHOUSE PWR FOUR-LOOP NSSS

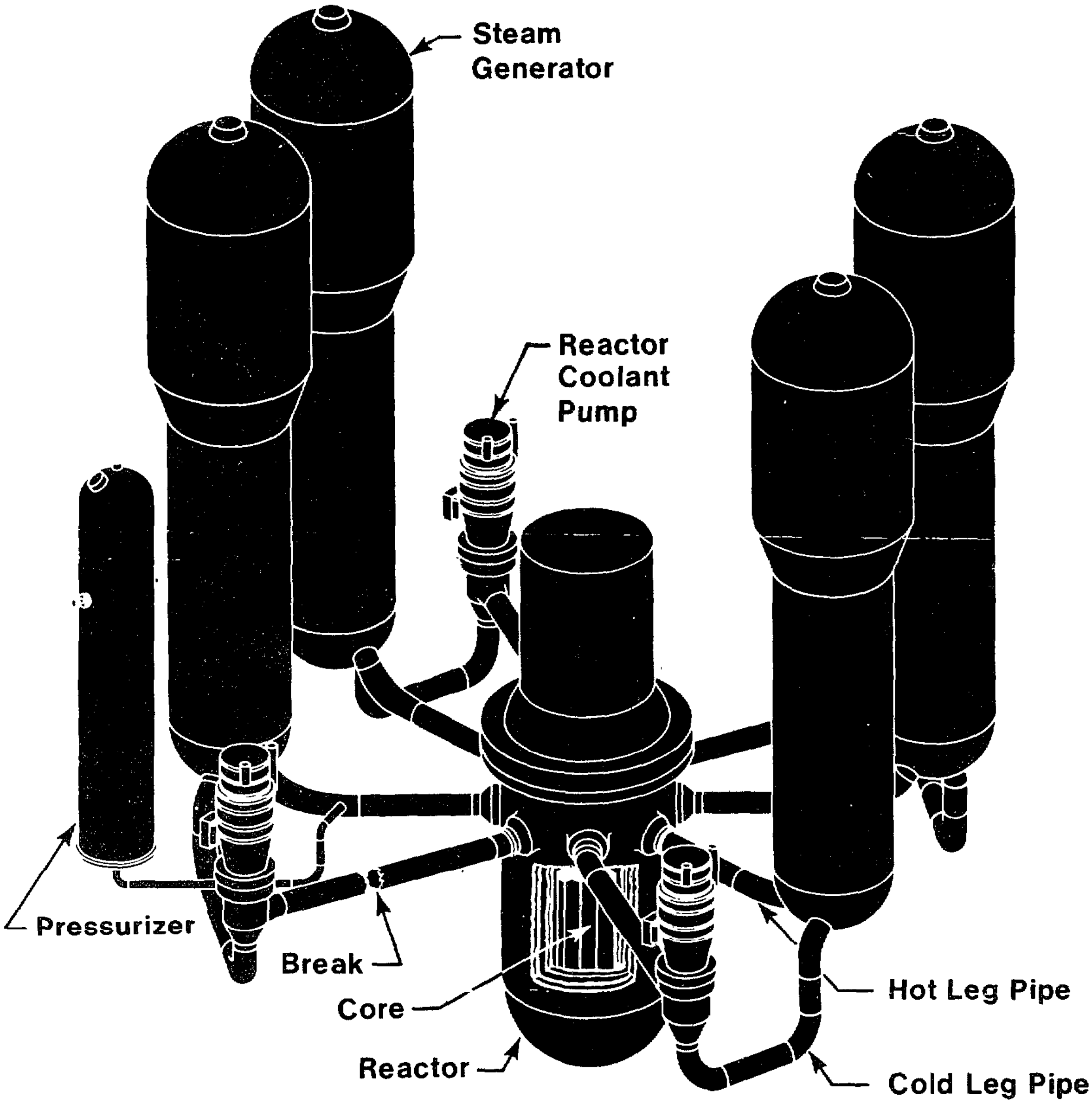

FIGURE 2

REACTOR COOLANT PUMP-SEPARATE EFFECT TESTS

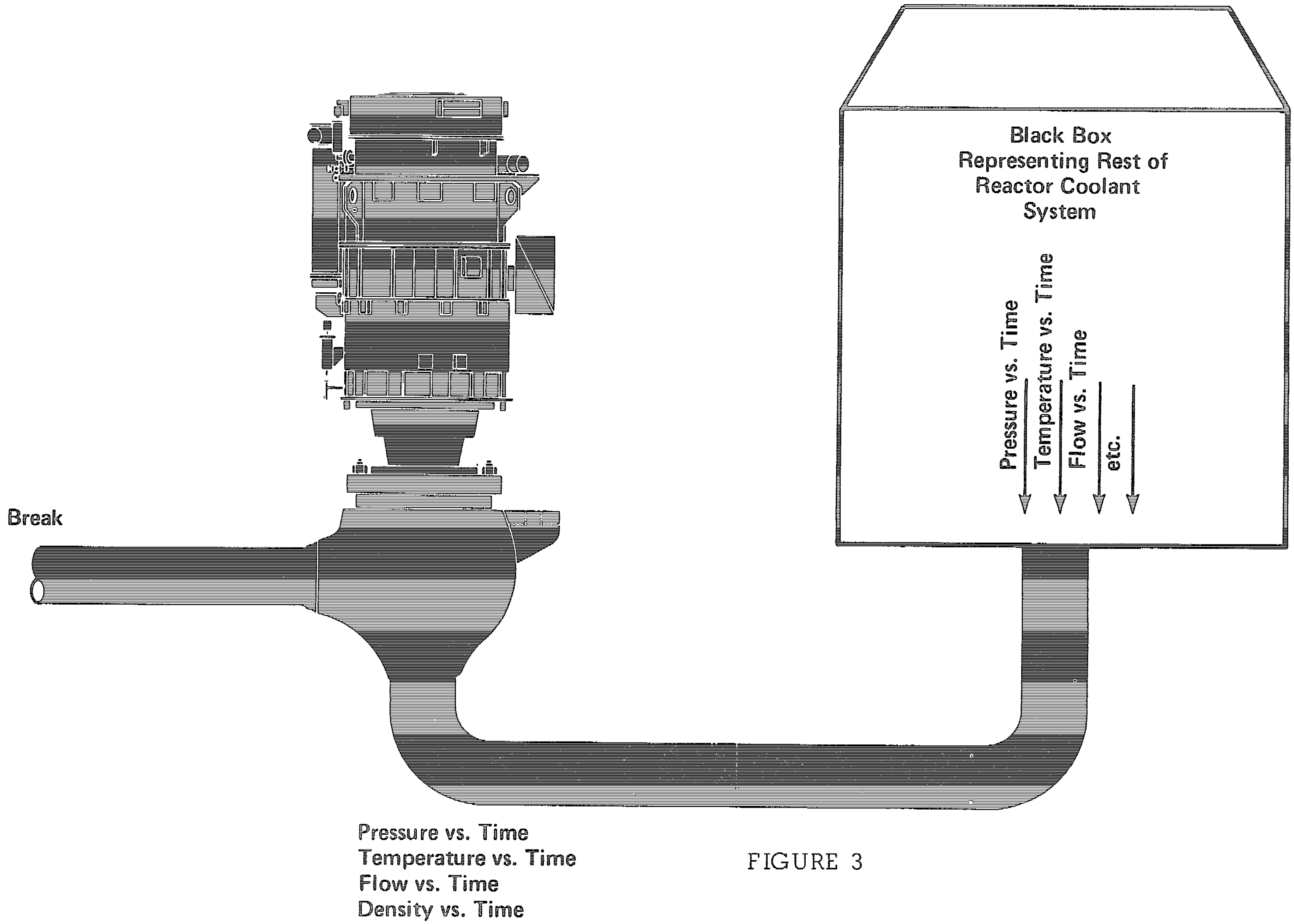

FIGURE 3

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
Lightning	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)	0	1 in 300,000,000

Risk of Fatality by Various Causes

(from U.S. AEC August 1974 Summary Report on the Reactor Safety Study)

FIGURE 4

4.0 ANSWERS TO QUESTIONS NO. 1 AND NO. 2

Q. 1 Is the current plan to not use LOFT for a meltdown experiment in the best interest of nuclear safety?

Q. 2 Should LOFT be used on a timely basis to study the means of retaining molten cores and measuring the consequences of steam explosions and radioactive releases resulting from a meltdown?

The answer to the first question is, in my opinion, affirmative. I do believe that we should continue with the LOFT program without consideration to core meltdown. Hence, no plans should be made at this time to experiment, at the LOFT facility, the consequences of core meltdown or means of retaining molten cores, for various reasons. Even though I do not expect any major safety breakthroughs from the current LOFT test program because of the use of upper bounds or ranges of parameters in the design of commercial nuclear power plants, as explained in Section 3.0, I do expect LOFT to provide closer insight in many specific areas, like nuclear fuel, flow regimes in components and at the pipe break, etc. I also would expect the use of the LOFT facility for getting more understanding on transients other than the loss of reactor coolant and related performance of ECCS. By this I mean the long list of transients categorized as ANS conditions one through four. By simulating such transients, we can gain additional verification and maturity in other potential chains of accidents which, according to the Rasmussen study may have an equivalent impact on the overall nuclear risk as reactor coolant pipe ruptures. Attachment 4, provided by Aerojet Nuclear Company, contains a list of areas other than reactor coolant pipe ruptures in which LOFT can contribute in gaining verification as I said earlier or in optimizing current design. With the proper allocation of time for meaningful data collection as a result of such tests, I see a useful utilization of the LOFT facility up to the mid-eighties. Core meltdown tests cannot be intermingled with the prior tests because of their high potential for significant radioactive contamination and damage to the delicate testing instrumentation. The question of potential use of the LOFT facility for core meltdown testing may be reexamined in the early eighties.

In regard to the original LOFT meltdown experiment, I would like to make two points. First of all, the original LOFT meltdown experiment was based on the wrong premise. Prior to 1966, it was the general belief that, if a core meltdown would occur, the containment would contain it so the only unknown was the fission product evolution from the molten uranium and their transport to the containment and the outside environment. A significant portion of the safety R&D program, at that time, was therefore related to fission product transport, e.g., dependence of fission product evolution from the uranium as a function of temperature, their physical status, the efficiency of containment sprays or filters in removing fission products, especially iodine, from the containment atmosphere so they would not be available for leakage to the outside, leak rate through concrete cracks, etc. But, just about 1966, the automatic assumption that the molten core would be contained was questioned by the nuclear community itself and serious concerns were raised about the capability of cooling a molten core. Emphasis then was put into preventing the core from melting by providing augmented and reliable Emergency Core Cooling Systems. The shift in direction for the LOFT program was a proper one. The original program with the wrong premises would not have helped much, because effects like steam explosions, molten metal interactions, molten uranium-concrete interaction, generation of considerable gases, etc. were not known, hence ignored. I am sure that these effects would have surfaced during the design of the LOFT program for core meltdown experiments with significant, periodic changes in the facility design. I believe that the LOFT facility would have been much more behind schedule and still today far from being ready for final shakedown before nuclear testing.

I would also like to address the question of whether core meltdown experiments are of primary importance. The Rasmussen Report contains in its Appendix VIII an assessment of core meltdown consequences. As it can be seen from such a study, there are quite a few areas of uncertainty but it is possible to put a reasonable upper bound on such uncertainties and, when the upper bound consequences are weighed against their likelihood, the overall risk to public and environment is quite small. Therefore, I respectfully submit that a detailed investigation of the various phenomena associated with core meltdown is not of primary importance. References 2 and 3 contain a detailed analysis of what is known and what is not and what has a significant impact on the final answer and what has not. These two references could be used to enlarge current R&D in this area. But I would like to make two comments at this point. Before significant amounts of money are allocated to studying core meltdown in detail, other areas in the nuclear energy field as well as in the non-nuclear energy field and areas outside the energy field must be considered and priority assigned. By allocating money to an area which, in the broad picture, is not controlling makes less money available for research in areas which have more severe impact on our health and environment and this is directly in conflict with the intent of any safety R&D program. The study of priorities among R&D programs is outside the scope of the current GAO assignment.

The second point I wish to make is that if funding is made available for core meltdown experiments, I strongly recommend not to invest them at LOFT or, worst, at a larger integral facility. There is nothing better than separate effects for an accurate understanding of what goes on. A detailed discussion on the philosophy of "separate effects" and "systems effects" tests is contained in Section 3.0.

4.0 REFERENCES

1. Reactor Safety Study, Appendices VI, VII, and VIII, August 1974 Draft, WASH-1400

2. "Core-Meltdown Experimental Review," SAND74-0382, August 1975.

3. "An Evaluation of the Applicability of Existing Data to the Analytical Description of a Nuclear Reactor Accident - Core Meltdown Evaluation," Battelle Columbus Laboratories, BMI-1910, July 1971.

5.0 ANSWER TO QUESTION NO. 3

Q. 3 Will the small scale LOFT result in experimental data, including the phenomena associated with a core meltdown, that is applicable to the large commercial reactors? Is a larger LOFT type test facility needed?

I will address the above broad question first, then the other points the GAO asked the consultants to consider.

The part of Question No. 3 dealing with core meltdown testing is addressed in Section 4.0 of my report. In that section I strongly recommend that meltdown tests not be contemplated in the LOFT facility . For further discussion on this point I refer the reader to Section 4.0.

Once reference to core meltdown tests is eliminated, I do believe that the experimental data which will be obtained as a result of the LOFT program will be applicable to large commercial reactors. We should repeat here, at the onset, that the LOFT facility is <u>not</u> a demonstration facility. By this, it is meant that the results obtained with the LOFT program are not directly representatives of the behavior of large commercial reactors under the same circumstances. In other words, if a pipe break of a given size in a given location with the LOFT facility in a given pre-selected set of conditions yields a peak fuel rod clad temperature of 1500 F, this does not mean that, if we postulate a break of similar size in a similar location in a large commercial reactor assumed in a similar set of pre-selected conditions, we will calculate a peak fuel rod clad temperature of 1500 F. The calculated peak fuel rod clad temperature in a large commercial reactor can easily be much higher or much lower because of the many physical differences between LOFT and any large commercial reactor. References 1, 2, 3 and 4 contain ample data and discussions on the similarities and dissimilarities between LOFT and a large commercial reactor.

The LOFT facility is a small PWR which will manifest the same overall behavior of a large commercial PWR when subjected to a sudden reactor coolant pipe break. Therefore, the LOFT program will "produce experimental NSSS data capable of validating or maturing analytical LOCA predictive codes over a full range of ECCS performance levels," to use ANC words, I would like to make clear that the aforesaid goal is not simple to achieve. In order to achieve it, the following steps must be performed:

a) Carefully study the LOFT facility and a reference large commercial reactor to identify all key physical differences;

b) Modify licensing computer codes to reflect such differences;

c) Modify the computer codes developed under b) above to remove, whenever possible, the intrinsic conservatism of the codes so that "best estimate" analyses can be performed;

d) Fix a set of "initial conditions" for a given LOFT test and perform prediction analyses using both sets of computer codes developed in b) and c) above;

e) Perform the planned test with the LOFT facility. Extreme care must be used in employing the right instrumentation at the right places. This is necessary to characterize the performance of the LOFT "peculiarities," i.e., the key differences between the LOFT facility and the reference large commercial reactor. The behavior of the "final answer" is expected to be close to the prediction obtained using the codes developed under c) above and "bound" by the predictions obtained using the computer codes developed under b);

f) Modify, wherever appropriate, the computer codes developed under b) and c) as a result of the information obtained from the test performed;

g) Replace, in the computer codes developed under f) the LOFT "pecularities" with the reference large commercial reactor "pecularities". This is a very delicate step for which careful selection and location of instrumentation is needed, as I pointed out earlier;

h) Use the licensing computer codes developed under g) to repeat ECCS performance tests for the reference large commercial reactor to check whether these results are less severe than the original ones.

The above steps have been described to stress the point that the LOFT facility can yeild results applicable to large commercial reactors if extreme care is used in running such tests and in fully characterizing the behavior of the key differences between LOFT and a reference large commercial reactor.

The latter part of the referenced GAO Question No. 3 (i.e., "Is a larger LOFT type test facility needed?") has been addressed in Section 3.0 of my report. In that section I explained why, in my opinion, a larger test facility is not needed.

I would like to address now the additional points raised by GAO. The question of effectivity of LOFT scaling and applicability to large commercial reactors has been already addressed in part in this section. I believe that the criteria adopted in scaling LOFT to achieve applicability to large commercial reactors are, in general, adequate. References 1 through 4 give a fair representation of the areas where LOFT deviates significantly from a large commercial reactor. These are the areas that require careful and extensive instrumentation for proper characterization.

With regard to--how can NRC avoid these criticisms?--I honestly do not know. Safety is not based on "yes" and "no" answers but on the delicate balance of probabilities and consequences. This balance is not easily understood and digested by the "little person" with tennis shoes. Hence, critics do not need to work very hard in confusing the average person in the pursuit of their personal goals whether related to nuclear safety or not.

The question of core meltdown tests at LOFT is addressed in Section 4.0.

5.0 REFERENCES

1. "LOFT Integral Test System Design Basis Report," January 1974, DBR-1, Aerojet Nuclear Company.

2. "LOFT/LPWR Blowdown/ECCS Typicality Evaluation," October 1973, by H. W. Massie, Jr. and R. P. Vijuk, WOAP-8196, Westinghouse Electric Corporation.

3. "LOFT Program Description," January 1975, LPD-1 Rev. 0, Aerojet Nuclear Company.

4. "Examination of LOFT Scaling," by L. J. Ybarrando, P. Griffith, S. Fabic and G. D. McPherson, presented at the ASME Winter Annual Meeting, New York, N.Y., November 17-22, 1974.

6.0 ANSWER TO QUESTION NO. 4

Q. 4 Should licensing of commercial reactors be modified in any way pending the results of the LOFT experiments or of experiments on a larger facility?

The answer to this question is clearly: No. As I mentioned in Section 3.0, I do not expect any "safety breakthrough" as a result of the LOFT program. The LOFT program, again, will increase the scientific knowledge here and there but it will not find, in my opinion, any area in the reactor design which may raise doubt, serious enough to require a change in the licensing of nuclear power plants. For this reason I do not believe that LOFT is the major light water reactor safety project. This might have been true in the late sixties but not in the last few years. Due to continuous delays other R&D programs, some of them in support of the LOFT program, have been planned and carried out to completion. These programs have provided the necessary basic knowledge to identify the upper bounds and ranges of parameters adopted in the safety design of commercial reactors. This point is treated in greater extent in Section 3.0.

With regard to the second part of the question, i.e., whether the licensing of commercial reactors should be modified in any way pending the results of experiments on a facility larger than LOFT, the answer is still no. This point is treated in detail in Section 3.0. The main reasons for my negative answer is twofold; a) over the years enough basic knowledge has been accumulated to identify upper bounds and ranges of parameters which assure an overall conservative design as the Rasmussen Report has proven; and b) a large integral facility, authorized today, will need 15 to 20 years before it can yield results that can be used in licensing.

Another question asked by GAO is whether or not LOFT is a means to reduce the conservatism in reactor design. The answer theoretically is yes but practically it will likely be no. By this I mean that the LOFT program will identify areas where the design of nuclear power plants is conservative but it would be difficult to quantify and characterize such conservatism in detail. Hence, manufacturers would not be able "to take credit" for this excessive conservatism and reduce the cost of a plant. Unless they are major, changes in well established designs may have financial savings offset by costs incurred in changing drawings, equipment specifications, licensing documentations, etc. I am sure that the Committee is aware of the many steps that can be taken in shortening the licensing time and in freezing the ever changing safety requirements if we are sincere in trying to reduce conservatism and associated costs to the consumers.

If anything has happened in the last few years which could really modify plant licensing, it is the Rasmussen's Reactor Safety Study. This study could be used to freeze escalating safety requirements and reduce the licensing time (between one-half to one-third of a nuclear power plant cost is due to interests during construction and escalation because now it takes closer to 10 years to build a plant rather than 5 as it was the case until a few years ago).

7.0 ANSWER TO QUESTION NO. 5

Q. 5 Present LOFT plans call for the use of a pressure suppression system in lieu of blowdown to the containment. Because of this, there are no planned tests of the containment's ability to control fission product activity. Do you believe that such a test of the containment would be appropriate for LOFT?

At least for the next 5 years, I do not believe it is appropriate to consider containment tests for LOFT. The reasons are the same as those mentioned in Section 4.0 against running meltdown tests in the LOFT facility. Very few containment tests could be run at LOFT before the facility is either seriously contaminated or, as a minimum, seriously affected by blowdown of steam, water and radioactivity to prevent resumption of testing without a major overhaul. As mentioned in Section 4.0, considerations of this type of tests may be appropriate in the early eighties when the planned series of LOFT tests have already been conducted. I would not consider however as a reason for not running containment tests the claim that the LOFT containment is not representative of a large commercial reactor. I believe that experimental data could be obtained to then verify presently employed computer codes. The main reasons, in my mind, are those mentioned earlier in ths Section and in Section 4.0.

The other obvious question that could be asked is: If the LOFT facility is not appropriate for such a test, do we need to run an integral test somewhere else?

With regard to fission product removal systems, such as containment spray, containment filtration systems there exists already a large amount of analytical and experimental data not only to give a good data base but also to confirm the adequacy of such systems in post-accident conditions. Tests on containment safeguards have been performed by both industry and the Nuclear Regulatory Commission (see Attachment 3 for references.)

With regard to containment structural adequacy, both analytical and experimental programs have been conducted over the years to give a good data base for selecting upper bound parameters in the design of commerical power plant containments. Examples of large scale test programs are the Carolinas-Virginia Tube Reactor (CVTR) containment tests in this country and those on the Marvekin plant containment in Sweden. Furthermore an extensive program is being conducted in Germany and the NRC will have access to the results under a bilateral exchange program.

For these reasons I do not see any sense of urgency for such types of tests.

8.0 ANSWER TO QUESTION NO. 6

Q. 6 LOFT is a small pressurized water reactor which has been scaled so the test results will simulate the anticipated effects of LOCAs on large pressurized water reactors. Will the LOFT results be applicable to boiling water reactors? Do you believe a LOFT experiment using a boiling water reactor mobile test assembly is needed?

I am not familiar with the details of the Boiling Water Reactor design to be able to address this question properly. I believe that some of the LOFT results will be applicable to a BWR. How many and to what extent I do not know.

Considering the large amount of "separate effect" and "system effect" tests performed for PWR's and the even hand policy applied by AEC/NRC, I would be surprised to hear that a LOFT experiment using a boiling water reactor mobile test assembly is needed. Furthermore, assuming that it is needed, can we afford to wait 10 years to get the "needed" results? I hope that the other consultants have more familiarity with BWR's than I do so they can address this question properly. I would like to just add that were LOFT a BWR now, my previous remarks would apply equally to the question of a PWR LOFT facility.

BIOGRAPHY

OF

ROMANO SALVATORI

Romano Salvatori is Manager of the U.S. Projects Department of the Westinghouse Electric Corporation. In this position he is responsible for the project management and coordination of all Nuclear Steam Supply System contracts with domestic customers.

Mr. Salvatori was graduated from the University of Rome (Italy) with a degree of "Dottore in Ingegneria Electrotecnica" in 1962. From 1962 until 1965 he was employed by the Italian Atomic Energy Commission (Comitato Nazionale Energia Nucleare) in Rome, Italy, as a nuclear safety engineer. During that period he was concerned with the safety evaluation of nuclear safety laws and standards, nuclear power plant systems inspection and approval. Since coming to Westinghouse in 1965, he has held engineering and management assignments in various aspects of the design, technical specification, reliability and safety for pressurized water reactor components and systems. From 1965 to 1969, he participated in the evaluation of safeguards, in licensing, and in the safety analysis review for the projects then under construction by the Westinghouse Atomic Power Department, beginning with the R. E. Ginna reactor of Rochester Gas and Electric Company. From 1969 to 1970 as Manager of Reliability, he was responsible for design reviews, for establishing and reviewing reliability criteria and for solving a variety of design problems concerned with reliability of PWR plants. From 1970 to July 1973 as Manager of Safety and Licensing, he was responsible for establishing safety criteria, for conducting safety evaluations of system and component design and for preparing Safety Analysis Reports and Environmental Reports documentation. From July 1973 to January 1975 as Manager of the Nuclear Safety Department, he was responsible for establishing safety criteria, for conducting safety evaluations of system and component design, for preparing documentation for safety analyses and environmental reports, for providing safety system performance requirements and for safety analyses and analytical methods development.

During his employment at Westinghouse, he has completed post-graduate courses in reliability and nuclear engineering at the University of Pittsburgh, Penn State University and Carnegie-Mellon University. He has lectured on nuclear engineering and safety at the University of Pittsburgh and Carnegie-Mellon University. In the summer of 1974 he has lectured at the Reactor Safety Course at the Massachusetts Institute of Technology. He has authored and coauthored several papers and articles on the safety and licensing of nuclear power plants. He is a member of the American Nuclear Society and the Atomic Industrial Forum(AIF). He is the AIF representative on the Nuclear Technical Advisory and Chairman of N-177. He is also a member of the Sierra club.

Mr. Salvatori was born in Foggia, Italy, in 1938. He moved to the United States and became a U.S. Citizen in 1971. He married Maria C. Rizzi in 1963. They and their two children, Pia and Olga, reside in Pittsburgh, Pennsylvania.

Core-Related
Separate/System Effects Tests

(Government and Multi-Industry Sponsored Programs)

I. BLOWDOWN

1. Title : Analysis and Correlation of Post-CHF Heat Transfer
   Project Number: A4055
   Organization : Lehigh University
   Sponsor : NRC
   Purpose : Develop post-CHF heat transfer correlations based upon experimental data.
   Reference : NUREG 75/046

2. Title : Pressurized Water Reactor Blowdown Heat Transfer Program
   Project Number: B0125
   Organization : ORNL
   Sponsor : NRC
   Purpose : Determine time to CHF and heat transfer rates during pre- and post-CHF phases of blowdown as influenced by variation in power, pressure, flow and break location.
   Reference : NUREG 75/046

3. Title : Semiscale
   Project Number: A6038
   Organization : ANC
   Sponsor : NRC
   Purpose : Investigate blowdown, refill and reflood heat transfer characteristics and performance of simulated ECCS systems
   Reference : NUREG 75/046

4. Title : Evaluation of Pressure Drop Across Area Changes and Fittings During Blowdown
   Project Number: AT2152
   Organization : University of Cincinnati
   Sponsor : NRC
   Purpose : Determine the appropriate one-dimentional models which may be used for the estimation of two-phase pressure drop across area changes under blowdown conditions for reactor system modeling codes.
   Reference : NUREG 75/046

5. Title : Phenomenological Modeling and Experiments in Water Reactor Safety
   Project Number: A2026
   Organization : Argonne National Laboratory
   Sponsor : NRC
   Purpose : Analyze and formulate models related to two-phase flow during blowdown and conduct and analyze transient CHF tests.
   Reference : NUREG 75/046

6. Title : Single- and Multi-Parameter Blowdown Heat Transfer Tests
   Porject Number: RP494
   Organization : Westinghouse
   Sponsor : EPRI
   Purpose : Provide experimental data and analysis on key heat transfer parameters during simulated pressurized water reactor (PWR) LOCA conditions, including transient critical heat flux (CHF) and post-CHF heat transfer.
   Reference : EPRI Research and Development Programs, September 3, 1975

7. Title : PWR Transient Critical Flux
   Project Number: RP 292
   Organization : MIT
   Sponsor : EPRI
   Purpose : Determine a correlation for CHF under high-pressure, steady-state countercurrent flow conditions and provide data on the time and location of transient CHF for a well-defined geometry and for thermal-hydraulic conditions in the range of early PWR blowdown conditions.
   Reference : EPRI Research and Development Programs, September 3, 1975

8. Title : PWR Blowdown Heat Transfer Performance
   Project Number: RP289
   Organization : Combustion Engineering
   Sponsor : EPRI
   Purpose : Improve the understanding of transient critical heat flux (CHF) and early post-CHF heat transfer phenomena expected to occur in PWR cores during transient loss-of-coolant accident (LOCA) blowdown conditions.

	Reference :	EPRI Research and Development Programs, September 3, 1975
9.	Title :	Boiling Water Reactor-Blowdown Heat Transfer (BWR-BDHT)
	Project Number:	B3014
	Organization :	GE
	Sponsor :	NRC, EPRI, GE
	Purpose :	Obtain information on transient thermal hydraulics following a postulated recirculation or steam line rupture in a BWR.
	Reference :	NUREG 75/046
10.	Title :	BWR Blowdown/ECC
	Project Number:	RP495
	Organization :	GE
	Sponsor :	EPRI
	Purpose :	Provide system and core response data for the latter stages of blowdown with actuation of the emergency core cooling system (ECCS).
	Reference :	EPRI Research and Development Programs, September 3, 1975
11.	Title :	LOCA Thermodynamics and Fluid Mechanics
	Project Number:	RP229
	Organization :	CREARE, INC.
	Sponsor :	EPRI
	Purpose :	Analytically assess the state of the art in modeling the important thermal-hydraulic processes which can occur during a postulated LOCA.
	Reference :	EPRI Research and Development Programs, September 3, 1975
12.	Title :	Scaling of Two-Phase Fluid Dynamics and Heat Transfer
	Project Number:	RP228
	Organization :	University of California, Berkeley
	Sponsor :	EPRI
	Purpose :	Study experimentally and analytically the fundamentals of scaling vapor-liquid flow systems.
	Reference :	EPRI Reserach and Development Programs, September 3, 1975

13. Title : Transient Analysis Code for Reactor Safety Studies
Project Number: RP227
Organization : MIT
Sponsor : EPRI
Purpose : Development of a computer code for safety studies of light-water reactors in which three-dimensional kinetics will be coupled with thermal-hydraulic calculations to provide necessary feedback effects.
Reference : EPRI Research and Development Programs, September 3, 1975

14. Title : Scaling Laws for Transient and Steady-State Boiling Heat Transfer
Project Number: RP344
Organization : California Institute of Technology
Sponsor : EPRI
Purpose : Analyze incipient boiling heat transfer at a heated surface with turbulent and laminar boundary layers.
Reference : EPRI Research and Development Programs, September 3, 1975

15. Title : Separated Flow Model for Two-Phase Flow
Project Number: RP443
Organization : Dartmouth College
Sponsor : EPRI
Purpose : Provide a verified model based on experimental investigations that could be applied to predict certain types of separated two-phase flows which are relevant to LWR safety analysis.
Reference : EPRI Research and Development Programs, September 3, 1975

II. PWR REFILL

1. Title : Creare Downcomer Effects Experimental Program
   Project Number: A4070
   Organization : CREARE
   Sponsor : NRC
   Purpose : Conduce experiments in a 1/15 scale model of a multiple loop LPWR downcomer-lower plenum geometry to investigate the effects of downcomer hot walls, pressure level and internal hardware on ECC delivery and bypass.
   Reference : NUREG 75/046

2. Title : Plenum Fill Experiment
   Project Number: B2039
   Organizaion : Pacific Northwest Labs
   Sponsor : NRC
   Purpose : Provide data for evaluating the potential for ECC bypass and the type of two-phase flow phenomena which might occur during the latter stages of blowdown and during the early portion of reflood, thereby providing data to establish when "end-of-bypass" might have occurred.
   Reference : NUREG 75/046

3. Title : BCL Steam-Water Mixing Programs Coordination and Supportive Testing
   Project Number: A4042
   Organization : Battelle-Columbus Laboratories
   Sponsor : NRC
   Purpose : Conduct supportive testing in subscale geometries, representative of LPWR cold leg/downcomer/lower plenum regions, to investigate ECC-steam interactions and penetrations.
   Reference : NUREG 75/046

4. Title : Cold Water Steam Mixing Program
   Project Number: RP294
   Organization : Westinghouse
   Sponsor : EPRI, Westinghouse
   Purpose : Perform tests and analyses to understand the thermal/hydraulic behavior of the steam/water mixing process in a PWR cold leg.
   Reference : WCAP-8307

5. Title : Steam-Water Mixing
   Project Number: RP286
   Organization : CE
   Sponsor : EPRI, AEC
   Purpose : Provide the broader data base required for the further verification and continued development of analytical methods for safety analysis.
   Reference : EPRI Research and Development Programs, September 3, 1975

6. Title : Two-Phase Pump Performance Program
   Project Number: RP301
   Organization : CE
   Sponsor : EPRI
   Purpose : Investigate through scale-model testing the hydraulic and mechanical performance of PWR circulating pumps during a postulated LOCA.
   Reference : EPRI Research and Development Programs, September 3, 1975

7. Title : Water Entrainment Intercompartmental Flows Resulting from Pipeline Breaks
   Project Number: RP275
   Organization : Drexel University
   Sponsor : EPRI
   Purpose : Develop a pressure model that includes the effect of phase separation.
   Reference : EPRI Research and Development Programs, September 3, 1975

8. Title : Two-Phase Flow
   Project Number: RP295
   Organization : Battelle Pacific Northwest Laboratory
   Sponsor : EPRI, NRC
   Purpose : Determine, under simulated loss-of-coolant accident (LOCA) conditions, how much of the injected emergency core coolant (ECC) would reach the reactor lower plenum during the blowdown phase.

9. Title : Basic Investigation of Two-Phase Pump Performance
   Project Number: RP493
   Organization : MIT
   Sponsor : EPRI

	Purpose :	Develop from experimental data a realistic analytical pump model for possible incorporation into LOCA analysis codes.
	Reference :	EPRI Research and Development Programs, September 3, 1975
10.	Title :	Model Pump Tests and Analysis for LOCA Application
	Project Number:	RP598
	Organization :	B&W
	Sponsor :	EPRI
	Purpose :	Evaluate through model devleopment two-phase flow performance of large-size model pump.
	Reference :	EPRI Research and Development Programs, September 3, 1975

III. PWR REFLOOD

1. Title : FLECHT Reflood Heat Transfer
   Project Number: S7045
   Organization : Westinghouse
   Sponsor : NRC, Westinghouse
   Purpose : Obtain experimental heat transfer data under simulated LOCA conditions for use in evaluating the heat transfer capabilities of PWR ECCS.
   Reference : WCAP-7665, 7931

2. Title : FLECHT-SET
   Project Number: A4071
   Organization : Westinghouse
   Sponsor : NRC, EPRI, Westinghouse
   Purpose : Provide experimental data on the influence of system effects on ECC behavior during the reflood phase of a LOCA for use in the verification and development of reflood models.
   Reference : WCAP-7906, 8238, 4810, 8431, 8583

3. Title : Development of Liquid Carryover and Reflood Heat Transfer Correlations
   Project Number: A4060
   Organization : MIT
   Sponsor : NRC
   Purpose : Develop liquid carryover and reflood heat transfer correlations for use in predicting FLECHT experiment.
   Reference : NUREG 75/046

4. Title : Study of Reflood Heat Transfer During LOCA
   Project Number: RP248
   Organization : University of California, Berkeley
   Sponsor : EPRI
   Purpose : Develop an analytical model capable of predicting the local flow conditions and heat transfer along a nuclear fuel bundle under reflood conditions. A laboratory-scale experimental program will be conducted to assist the model development.
   Reference : EPRI Research and Development Programs, September 3, 1975

5.	Title :	Combined Core Cooling Injection Following PWR Blowdown
	Project Number:	RP341
	Organization :	State University of New York, Buffalo
	Sponsor :	EPRI
	Purpose :	Obtain information on an alternate ECCS system (top and bottom injection) performance during a LOCA.
	Reference :	EPRI Research and Development Programs, September 3, 1975
6.	Title :	Core Thermal-Hydraulic Studies
	Project Number:	RP345
	Organization :	Columbia University
	Sponsor :	EPRI
	Purpose :	Improve the capabilities for performing simulated core thermal-hydraulic tests and generate data to verify reactor core thermal-hydraulic code predictions.
	Reference :	EPRI Research and Development Programs, September 3, 1975
7.	Title :	COBRA - Coolant Boiling and Rod Arrays
	Project Number:	B2041
	Organization :	PNL
	Sponsor :	NRC
	Purpose :	Develop and verify experimental methods of evaluating the thermal hydraulic performance of water cooled nuclear fuel rod bundles during postulated accidents.
	Reference :	NUREG 75/046

IV. FUEL/CLADDING BEHAVIOR AND CORE MELTDOWN

1. Title : LWR Core Meltdown Study
   Project Number: A1019
   Organization : SANDIA
   Sponsor : NRC
   Purpose : Prepare a state of the art survey of the experimental data applicable to analyses of hypotehtical core meltdown accidents.
   Reference : NUREG 75/046, SAND-74-0382

2. Title : Mechanical Properties of Zr Containing Oxygen
   Project Number: A2017
   Organization : Argonne National Lab
   Sponsor : NRC
   Purpose : Obtain quantative information on the effect of oxidation on the mechanical behavior of zircaloy in order to evaluate the conservatism of the acceptance criteria for the ECCS.
   Reference : NUREG 75/046

3. Title : Molten Core Interactions
   Project Number: A1019
   Organization : Sandia Laboratories
   Sponsor : NRC
   Purpose : Characterize the chemical and physical interactions between prototypical materials likely to be encountered during hypothetical core melt accidents in LWR's.
   Reference : NUREG 75/046

4. Title : Multi-Rod Burst Test
   Project Number: B1020
   Organization : ORNL
   Sponsor : NRC
   Purpose : Characterize deformation behavior of LWR fuel cladding under conditions predicted for LOCA.
   Reference : NUREG 75/046

5. Title : Natural Convection in Molten Pools
   Project Number: A4061
   Organization : Ohio State University
   Sponsor : NRC

	Purpose :	Dvelop correltions for determining heat transfer rates by natural convection from enclosed fluid volumes having interral heat generation for use in evaluating post-accident heat removal capabilities under postulated core meltdown conditions.
	Reference :	NUREG 75/046
6.	Title :	Power Burst Facility
	Project Number:	A6041
	Organization :	ANC
	Sponsor :	NRC
	Purpose :	Obtain data on the performance of fuel rod clusters under abnormal power flow and energy density conditions.
	Reference :	NUREG 75/046
7.	Title :	Steam Explosion Phenomena
	Project Number:	A1030
	Organization :	Sandia Laboratories
	Sponsor :	NRC
	Purpose :	Identify and characterize the physical conditions which must be met in order for a steam explosion to occur when molten LWR core materials contact water.
	Reference :	NUREG 75/046
8.	Title :	Transient Fuel Response and Fission-Product Release
	Project Number:	A2016
	Organization :	Argonne National Labs
	Sponsor :	NRC
	Purpose :	Develop a comprehensive fission product release model based on mechanistic understanding of fuel behavior in LWR fuel elements undergoing a wide range of accidental overheating conditions.
	Reference :	NUREG 75/046
9.	Title :	Vapor Explosion Triggering
	Project Number:	A2029
	Organization :	Argonne National Labs
	Sponsor :	NRC

	Purpose	:	Identify and understand the physical conditions which must be met in order to trigger explosive interactions between two liquid phases for use in estimating the probability of occurrence of a steam explosion during hypothetical LWR accidents in which contact between molten LWR core materials and water is postulated.
	Reference	:	NUREG 75/046
10.	Title	:	Zircaloy Fuel Cladding Collapse Studies
	Project Number	:	B0124
	Organization	:	ORNL
	Sponsor	:	NRC
	Purpose	:	Determine the factors that affect the collapse behavior of Zircaloy fuel cladding.
	Reference	:	NUREG 75/046
11.	Title	:	Zircaloy Metal-Water Oxidation Kinetics
	Project Number	:	B0128
	Organization	:	ORNL
	Sponsor	:	NRC
	Purpose	:	Provide highly reliable kinetic data pertaining to zirconium metal-water oxidation phenomena expected to occur in case of a LOCA.
	Reference	:	NUREG 75/046

V. CONTAINMENT

1. Title : Containment System Experiment
   Project Number: AT(45-1)-1830
   Organization : Batelle Pacific Northwest Laboratory
   Sponsor : AEC
   Purpose : Obtain data in support of safety analyses on the transient response of several aspects of the reactor containment system during and following a LOCA.
   Reference : BNWL 1592 and attached list of reports

2. Title : Nuclear Safety Pilot Plant
   Project Number: W7405
   Organization : ORNL
   Sponsor : AEC
   Purpose : Obtain data on the removal of radioactive particles from reactor containments by sprays
   Reference : ORNL 4623, ORNL 4671, ORNL 4602, etc.

3. Title : Marviken Full-Scale Containment Experiments
   Project Number: MXA
   Organization : Marviken Project Board
   Sponsor : USAEC, Sweden, Denmark, Norway, Finland, Germany
   Purpose : Perform test on a full-scale reactor containment to determine its response to a LOCA.
   Reference : MXA-0-401, 402, 403, 404 and attached list of reports

4. Title : Containment Analysis Development
   Project Number: A6042
   Organization : ANC
   Sponsor : NRC
   Purpose : Development of a multi-dimensional transient flow program for independent evaluation of water reactor containment systems.
   Reference : NUREG 75/046

5. Title : Evaluation of Radioiodine Emanations from Nuclear Power Plants
   Project Number: RP274
   Organization : Science Applications, Inc.
   Sponsor : EPRI
   Purpose : Obtain in-plant measurements at several LWR sites to evaluate, quantify and identify diverse radioiodine releases.
   Reference : EPRI Research and Development Programs, September 3, 1975

LIST OF MXA-REPORTS

THE MARVIKEN FULL SCALE CONTAINMENT EXPERIMENTS

APPENDIX

LISTING OF PUBLICATIONS ISSUED BY THE CSE PROGRAM

<u>Topical Reports</u>

1. G. J. Rogers. "Program For Containment Systems Experiment. A Summary of Hanford Laboratories Progress During 1964 Under General Electric and Program for Future Work", HW-83607, Hanford Atomic Products Operation, Richland, Washington, September 1964, 40 p.

2. C. E. Huck, "Instrumentation Development for the Containment Systems Experiment." BNWL-26, Battelle-Northwest, February 1965.

3. B. M. Johnson. "Containment Systems Experiment Part III. Mathematical Models of Pressure-Temperature Transients," BNWL-233, Battelle-Northwest, May 1966, 142 p.

4. N. P. Wilburn, and L. D. Coffin, "The Combination of On-Line Analysis with Collection of Multicomponent Spectra in a PDP-7," BNWL-CC-700, Battelle-Northwest, July 1966.

5. L. D. Coffin, "On-Line Computer Storage and Retrieval of Processed Gamma Spectra Data, "BNWL-506, Battelle-Northwest, July 1967.

6. W. R. Weissenberger and E. L. Wells, "Computer Retrieval of CSE Multiplexer Data," BNWL-693, Battelle-Northwest, September 1967.

7. P. C. Owzarski, "Fortran IV Computer Program for Calculation Transient Heat Transfer Coefficients from Wall Temperatures," BNWL-552, Battelle-Northwest, October 1967.

8. E. L. Wells, "UNIVAC 1108/PDP-7 Magnetic Tape Compatibility Program I, "BNWL-610, Battelle-Northwest, December 1967.

9. H. D. COllins, Sonic Anemometer for Harsh Environments," BNWL-604, Battelle-Northwest, January 1968.

10. N. P. Wilburn, Multiplexer Codes for the PDP-7," BNWL-604, Battelle-Northwest, February 1968.

11. R. K. Hilliard, L. F. Coleman, and J. D. McCormick. "Comparisons of the Containment Behavior fo a Simulant With Fission Products Released From Irradiated UO2", BNWL-581, Battelle-Northwest, Richland, Washington, March 1968, 140 p.

12. J. G. Knudsen and R. K. Hilliard. "Fission Product Transport by Natural Processes in Containment Vessels", January 1969, 90 p.

13. L. F. Coleman. "Preparation, Generation, and Analysis of Gases and Aerosols For the Containment System Experiment", BNWL-1001, Battelle-Northwest, Richard, Washington, April 1969, 95 p.

14. J. D. McCormack. "Maypack Behavior in the Containment Systems Experiment--A Penetrating Analysis" BNWL-1145, Battelle-Northwest, Richland, Washington, August 1969, 44 p.

15. M. E. Witherspoon and G. J. Rogers. "Air Leakage Rate Studies on the CSE Containment Vessel", BNWL-1028, Battelle-Northwest, Richland, Washington, September 1969, 121 p.

16. R. K. Hilliard, L.F. COleman, C. E. Linderoth, J. D. McCormack and A. K. Postma. "Removal of Iodine and Particles From Containment Atmospheres by Sprays--Containment Systems Experiment Interim Report", BNWL-1244, Battelle-Northwest, Richland, Washington, February 1970, 196 p.

17. C. E. Linderoth. "Containment Systems Experiment Part I. Description of Experimental Facilities", BNWL-456, Battelle-Northwest, Richland, Washington, March 1970, 100 p.

18. J. G. Knudsen. "Properties of Air-Stream Mixtures Containing Small Amounts of Iodine", BNWL-1326, Battelle-Northwest, Richland, Washington, April 1970, 75 p.

19. N. P. Wilburn, "Void Fraction Profile in a Nuclear Reactor Vessel During Coolant Blowdown", BNWL-1295, Battelle-Northwest, Richland, Washington, April 1970, 137 p.

20. R. K. Hilliard and L. F. COleman. "Natural Transport Effects on Fission Product Behavior In the Containment Systems Experiment", BNWL-1457, Battelle-Northwest, Richland, Washington, December 1970, 181 p.

21. A. K. Postma and L. F. COleman. "Effect of Continuous Spray Operation on the Removal of Aerosols and Gases in the Containment Systems Experiment", BNWL-1485, Battelle-Northwest, Richland, Washington, December 1970, 127 p.

22. R. T. Allemann, A. J. McElfresh, A. S. Neuls, W. C. Townsend, N. P. Wilburn, M. E. Witherspoon. "Experimental High Enthalpy Water Blowdown From a Simple Vessel Trhough a Bottom Outlet", BNWL-1411, Battelle-Northwest, Richland, Washington.

23. R. T. Allermann, A. J. McElfresh, A. S. Neuls, W. C. Townsend, N. P. Wilburn, M. E. Witherspoon, "High Enthalpy Blowdown of a Reactor Simulator Vessel Containing a Perforated Sieve Plate Separator", BNWL-1463, Battelle-Northwest, Richland, Washington, February 1971, 273 p.

24. R. T. Allemann, A. J. McElfresh, A. S. Neuls, W. C. Townsend, N. P. WIlburn, M.E. Witherspoon, "High Enthalpy Blowdowns, From a Simple Vessel Through a Side Outlet", BNWL-1470, Battelle-Northwest, Richland, Washington, February 1971, 66 p.

25. R. T. Allemann, A. J. McElfresh, A. S. Neuls, W. C. Townsend, N. P. Wilburn, M. E. Witherspoon, "Coolant Blowdown Studies of a Reactor Simulator Vessel Containing a Simulated Reactor Core", BNWL-1524, Battelle-Northwest, Richland, Washington.

26. W. C. Townsend, "Defect Study - Pressure Vessel CSE Reactor Vessel Simulator", BNWL-1554, Battelle-Northwest, Richland, Washington.

27. M. E. Witherspoon. "Leakage Rate Tests on the CSE Containment Vessel With Heated Air and Steam-Air Atmospheres", BNWL-1475, Battelle-Northwest, Richland Washington, 67 p.

28. R. K. Hilliard and A. K. Postma. "The Effect of Flow Rate on the Washout of Gases and Particulates in the Containment System Experiment", BNWL-1591, Battelle-Northwest, Richland, Washington.

29. M. E. Witherspoon. "Leakeage of Fission Products From Artificial Leaks in the CSE Containment Vessel", BNWL-1582, Battelle-Northwest, Richland, Washington.

30. J. D. MCCormack, R. K. Hilliard, A. K. Postma, "Removal of Airborne Fission Products by Recirculating Filter System in the Containment System Experiment", BNWL-1587, Battelle-Northwest, Richland, Washington, June 197.

31. B. M. Johnson and A. K. Postma. "Containment Sytems Experiment Final Program Summary", BNWL-1592, Battelle-Northwest, Richland, Washington.

necessary. The purposes of these visits would be to conduct group briefings on veterans benefits and to counsel individual veteran inmates.

After discussing the results of our review, the Chief Benefits Director suggested that he have an opportunity to poll all VA regional offices so that he could provide us with a more complete report on the service that was being given to incarcerated veterans.

VA furnished us its report with supporting schedules in November 1974 (see app. II and III). VA's report stated, among other things, that:

> "There is a wide range of involvement. Veterans Services Officers have expressed deep concern towards socially and educationally disadvantaged veterans, and particularly towards the incarcerated disabled veteran.
>
> "We find that most VA hospitals have considerable contact with Federal and State prisons through their Social Work Service. There are some 40 VA Drug Treatment Centers nationwide which have very active liaison with the prison systems."

Our analysis of VA's report and supporting schedules showed that, of the 280 Federal and State penal institutions, in which more than 44,000 male veterans were imprisoned, VA regional representatives had provided service to 142 institutions or about 51 percent. This report also showed that 14 VA regional offices were not providing any service to the 79 institutions within their jurisdictions. For the 43 VA regional offices reported to be servicing incarcerated veterans, many offices indicated on-call type service rather than scheduled periodic visits.

VA's November 1974 report did not indicate whether VA was still considering the proposed policy change to require semiannual visits to Federal and State prisons.

CONCLUSIONS

VA is required under the Veterans Education and Training Amendments Act of 1970 to provide an outreach services program for all veterans, including those incarcerated in, or recently released on parole from, penal institutions. Incarcerated veterans, in particular, do not have the mobility to seek out information regarding their benefits. Consequently, to have an effective outreach service program for these veterans, VA must take the initiative to reach them. However, many Federal and State institutions are not being served by VA in any manner. For the institutions that are being served, VA's outreach efforts have generally been limited to responding to specific inquiries from incarcerated veterans and prison officials.

RECOMMENDATIONS

The need for an effective outreach program is evident by the many veterans we interviewed who were not aware of their entitlement to benefits. Accordingly, we recommend that the Administrator of Veterans Affairs require VA regional representatives to visit Federal and State penal institutions at least semiannually to advise prison officials and inmates about the various benefits available to incarcerated veterans, as was proposed by VA officials at our October 22, 1974, briefing.

In view of the special circumstances applicable to incarcerated veterans, who as of October 1974 numbered more than 44,000, we also recommend that VA develop and distribute, to incarcerated veterans and veterans released or paroled from penal institutions, pamphlets and other literature specifically aimed to motivating them to use the benefits available to assist them in readjusting to society.

- - - -

We do not plan to distribute this report further unless you agree or publicly announce its contents. In this connection, we want to invite your attention to the fact that this report contains recommendations to the Administrator of Veterans Affairs. As you know, section 236 of the Legislative Reorganization Act of 1970 requires the head of a Federal agency to submit a written statement on actions he has taken on our recommendations to the House and Senate Committees on Government Operations not later than 60 days after the date of the report, and the House and Senate Committees on Appropriations with the agency's first request for appropriations made more than 60-days after the date of the report. When we obtain your agreement to release the report, we will make it available to the Administrator and the four committees for the purpose of setting in motion the requirements of section 236.

Sincerely yours,

R. F. Keller

Acting Comptroller General
of the United States

October 18, 1974

COUNSELING OF INCARCERATED VETERANS BY

VETERANS SERVICES PERSONNEL

1. Background:

a. At one time regularly scheduled visits were made by VA personnel to prisons, hospitals, remote locations, and homes for the aged. A change in policy over the years has placed emphasis on serving these veterans by telephone and mail. WATS lines were installed and additional counselors hired to man them, Vet Reps have been placed on IHL campuses, and the energy crisis has caused stations to cut back on long-distance driving.

b. A problem exists in serving incarcerated veterans in Federal and State prisons. In some institutions there is a security problem, and prison officials do not encourage visits by VA personnel. In other institutions correctional counselors expect a VA counselor to drive 150 miles to interview and assist one veteran. The majority of cases lie in between--the prison officials would welcome VA counselors on a regularly scheduled basis.

2. A check around the Country to see what is happening right now produced this information:

a. New York: Visits are made to prisons as needed on call. Some prison officials, because of security reasons, are not in favor of VA visits. Community Service Specialists, Veterans Benefits Counselors, or Vet Reps make prison visits. The Veterans Services Officer would welcome a requirement to visit prisons on a semiannual basis.

b. Philadelphia: There has been a pretty active program in the past year. Community Service Specialist has been coordinating prison visits. Drug program people have also been making visits. Veterans Benefits Counselors have been going to Graterford and Holmbsburg prisons for group orientations. Penn State University and Northampton Community College have a tie in with the prison system. The Vet Reps assigned to these colleges assist the incarcerated veteran.

c. Boston:

(1) Crockett Reformatory: Visit scheduled this week.

(2) Massachusetts State Prison, Walpole: Four visits have been made in the past year.

(3) Norfolk: Three visits in the past year.

A job fair was recently held in Boston for ex-offenders. VA participated in this. The Community Service Specialist is in frequent contact with prison officials in all institutions in their area. He also goes along with State social workers in their visits to prisons. A very active program.

d. Detroit: There are two major prisons in the area. The Vet Rep in the area is taking care of Jackson, Michigan, prison by frequent visits. Southern Michigan Prison allows inmates to attend community colleges on campus. The college Vet Rep is in constant contact with these inmates.

e. Washington, D. C.: Visits are made by the Community Service Specialist to Lawton Reformatory four times a year. A considerable amount of good has come from these visits.

f. St. Petersburg: The Community Service Specialist has visited all prisons in the Miami area several times a year. Personnel from the drug treatment program have made regular visits as part of their outreach program. Belgrade Prison recently requested a visit by VA personnel. This was taken care of right away by the local Vet Rep.

g. Chicago: No regularly scheduled visits are made. It has been handled on an on-call basis. Four calls for service were received in the past year. All calls resulted in a visit by a Veterans Benefits Counselor within a few weeks. In addition, a Chicago Vet Rep recently assisted one prison to get approval for its adult high school program. The Veterans Services Officer would welcome a directive requiring scheduled visits.

h. Houston: There are 10 prisons in the area. One has a college level program in the institution. Several calls have been made at the prisons by Veterans Benefits Counselors and Vet Reps. The Assistant VSO feels that Vet Reps are too new and inexperienced to give good quality counseling to incarcerated veterans.

i. Boise: The Veterans Services Officer states that there is only one prison in his area. The State of Idaho has an extremely progressive penal system, with emphasis on rehabilitation. Visits to the prison by Veterans Benefits Counselors are made on the average of one every 6 weeks. DAV and VFW also assist greatly in this program. An interesting note--of the 197 veterans in the prison last month, 102 had bad discharges. An outstanding program.

j. Seattle: Visits are made to all Federal and State prisons on an on-call basis. In eastern Washington, visits to prison are made by the Veterans Benefits Counselor at the local VA hospital. Several requests were received for group briefings in the past year. All were taken care of within 2 weeks of request.

k. Los Angeles: Monthly visits to California Rehabilitation Center have been made for the past 2 years. Two counselors go there each month. Monthly visits are made to Terminal Island Federal Prison. Visits were made in the past year to Lompoc, Chino and Tehachopi, and Atascadero on an on-call basis. All prison visits are coordinated by the Community Service Specialist at Los Angeles Regional Office. A very active program.

3. The above sampling gives a pretty fair picture for the past year. The Veterans Services Officers at all of the above stations were asked if a requirement of at least

semiannual visits to all Federal and State prisons in their areas would place an undue hardship on them. All except one answered that they would welcome such a directive and that it would not be any problem for them to comply. The proposed new revision of chapter 13, M232-1, will reflect this change. A requirement has been written into the manual that all Federal and State prisons will be visited by Veterans Services personnel at least semiannually where the prison authorities deem that this is desirable and necessary. These visits are to be made for the purpose of conducting group briefings on veterans' benefits and individual counseling for veteran inmates.

4. Care should also be taken when these visits occur to acquaint incarcerated veterans with the services that are available to them by mail and WATS telephone. Distribution of benefits pamphlets and brochures to prison officials and distribution of needed VA forms to each prison should be arranged by VA regional office. Prison officials can also be assisted by letting them know what assistance is available from local county and/or State service officers and veterans' service organizations, particularly with regard to specialized help in the matter of applications for review of discharge.

5. VA counselors who are experienced in counseling incarcerated veterans tend to agree that frequently the most helpful and interested official in each institution is the Education Officer. He is usually a good first contact.

VETERANS ADMINISTRATION
DEPARTMENT OF VETERANS BENEFITS
WASHINGTON, D.C. 20420

Date: NOV 12 1974

In Reply Refer to: 271

Mr. Joseph A. Vance
Supervisor Auditor
McPherson Building
1425 K Street
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Washington, D. C. 20524

Enclosed is a narrative summary of our findings regarding Veterans Administration service to incarcerated veterans.

ODELL W. VAUGHN
Chief Benefits Director

Enclosure

VA REPORT ON SERVICE BY THE VETERANS ADMINISTRATION

TO INCARCERATED VETERANS

On October 22, 1974, representatives of the U.S. General Accounting Office made inquiries of the Veterans Assistance Service regarding services provided to incarcerated veterans.

On that same date we queried all regional offices in an effort to find out what we are doing right now for incarcerated veterans.

The replies show that there are approximately 287 prisons, that we are servicing 145 of them, and that there are an estimated 44,473 veterans incarcerated. When we say that we have serviced a prison our people frequently mean that they have an agreement whereby we will service them on an on-call basis.

(See GAO note on p. 14.)

There is a wide range of involvement. Veterans Services Officers have expressed deep concern towards socially and educationally disadvantaged veterans, and particularly towards the incarcerated disabled veteran.

We find that most VA hospitals have considerable contact with Federal and State prisons through their Social Work Service. There are some 40 VA Drug Treatment Centers nationwide which have very active liaison with the prison systems.

Our review includes only Federal and State prisons for adult males. While the VA occasionally services county prisons, these are generally serviced by County Veterans Services Officers who are county employees. The VA provides backup and follow-through at regional offices for applications and inquiries coming from these county officers.

Several stations reported that they have an understanding with all penal institutions that VA personnel will visit these sites on an on-call basis. However, our stations report that they seldom receive calls from a high percentage of these prisons, indicating a less than enthusiastic response from prison officials.

Although most VA regional offices now have provision for toll-free telephone service, it should be noted that incarcerated veterans are generally not permitted to use the telephone to obtain counseling on veterans' benefits. This is one area where the prisons could perhaps meet us half way by making arrangements whereby prisoners can make supervised telephone calls from the office of a correctional counselor or education officer.

GAO note: VA's supporting data (see app. III) indicate that 142 prisons were being serviced out of a total of 280 identified prisons having a total of 44,475 incarcerated male veterans.

REPORT ON SERVICE TO INCARCERATED VETERANS
AREA I

STATION	NUMBER OF PRISONS	NUMBER SERVICED	NUMBER OF VETS	REMARKS
Baltimore	8	-	Unknown	
Boston	11	11	418	Community Service Specialist (CSS) visits all prisons
Buffalo	4	4	1,100	By agreement, visits are made by State and County Veterans Affairs personnel
Hartford	10	4	1,100	4 visits per month by Veterans Benefits Counselors (VBCs)
Manchester	1	1	240	4 visits made this year
Newark	3	3	1,285	21 visits made to explain veterans benefits and develop programs
New York	17	17	3,000	On call only, by CSS and VBC
Pittsburgh	2	2	327	7 visits made in past year
Providence	3	3	65	On call visits are made by VBC. Visits are also made by Drug Program personnel
San Juan	1	1	150	Monthly visits are made by VBC
Togus	4	4	183	On call visits only
Washington, D.C.	1	1	200	Active program at Lorton, Virginia prison

STATION	NUMBER OF PRISONS	NUMBER SERVICED	NUMBER OF VETS	REMARKS
White River Junction	1	1	60	Visits are made by regional office and hospital personnel
Wilmington	3	-	125	No visits made this year
Totals, Area 1	69	52	8,253	

AREA 2

STATION	NUMBER OF PRISONS	NUMBER SERVICED	NUMBER OF VETS	REMARKS
Atlanta	12	3	800	On call visits only
Columbia	5	5	2,000	On call visits only
Houston	14	-	5,500	
Huntington	5	-	419	On call visits only
Jackson	1	1	300	On call visits only
Little Rock	3	1	150	Visits made weekly by Veterans Representative (Vet Rep)
Louisville	9	-	Unknown	
Montgomery	3	3	849	Monthly visits are made
Nashville	3	3	Unknown	On call visits only
New Orleans	3	2	508	12 visits are made this year to date
Roanoke	5	-	Unknown	
St. Petersburg	12	-	2,100	
Waco	7	3	1,000	15 visits made this year
Winston-Salem	10	4	Unknown	4 visits were made by Vet Rep, 1 by Rehabilitation Specialist
Totals, Area 2	92	25	13,626	

AREA 3

STATION	NUMBER OF PRISONS	NUMBER SERVICED	NUMBER OF VETS	REMARKS
Chicago	7	-	2,100	On call visits made. Also many visits are made by Drug Program personnel.
Cleveland	7	2	Unknown	Visits made on request, by VBC or Field Attorney
Des Moines	5	-	500	
Detroit	6	4	1,685	Weekly visits made by Vet Rep, 2 visits per month by VBC
Fargo	1	1	74	4 visits made by VBC
Indianapolis	4	-	Unknown	1 visit made by Vet Rep
Lincoln	3	3	325	Visits made on request by CSS
Milwaukee	6	1	850	2 VBCs make visits every other month
Muskogee	3	-	Unknown	
Philadelphia	9	7	600	14 visits made by VBC and Veterans Assistance Center
St. Louis	7	3	1,740	1 visit made per month by Vet Rep, 2 visits per month by CSS
St. Paul	4	4	1,735	On call visits only
Sioux Falls	1	1	100	1 visit made per month
Wichita	5	5	500	5 visits made this year
Totals, Area 3	68	31	10,209	

AREA 4

STATION	NUMBER OF PRISONS	NUMBER SERVICED	NUMBER OF VETS	REMARKS
Albuquerque	1	1	280	2 visits made per month
Boise	1	1	197	Very closely coordinated (our best program)--2 visits made per month plus group briefings
Cheyenne	1	-	40	No visits this year
Denver	3	2	877	Have made quarterly visits plus 2 group presentations
Ft. Harrison	1	-	75	
Honolulu	2	1	85	On call visits only
Juneau	5	5	72	3 visits made this year
Los Angeles	5	4	2,781	26 visits made through October (1974)
Phoenix	7	3	700	1 visit made per month by Vet Rep
Portland	2	-	600	On call visits only
Reno	3	3	175	5 visits made by Vet Reps
Salt Lake City	1	1	202	Visits made twice a month
San Diego	7	1	1,603	Visits made every other month
San Francisco	8	8	3,500	Monthly visits are made as well as visits by Drug Program personnel

STATION	NUMBER OF PRISONS	NUMBER SERVICED	NUMBER OF VETS	REMARKS
Seattle	4	4	1,200	12 visits made this year
Totals, Area 4	51	34	12,387	

Totals: Number of prisons - 280
Number of prisons serviced - 142
Estimated number of incarcerated veterans - 44,475