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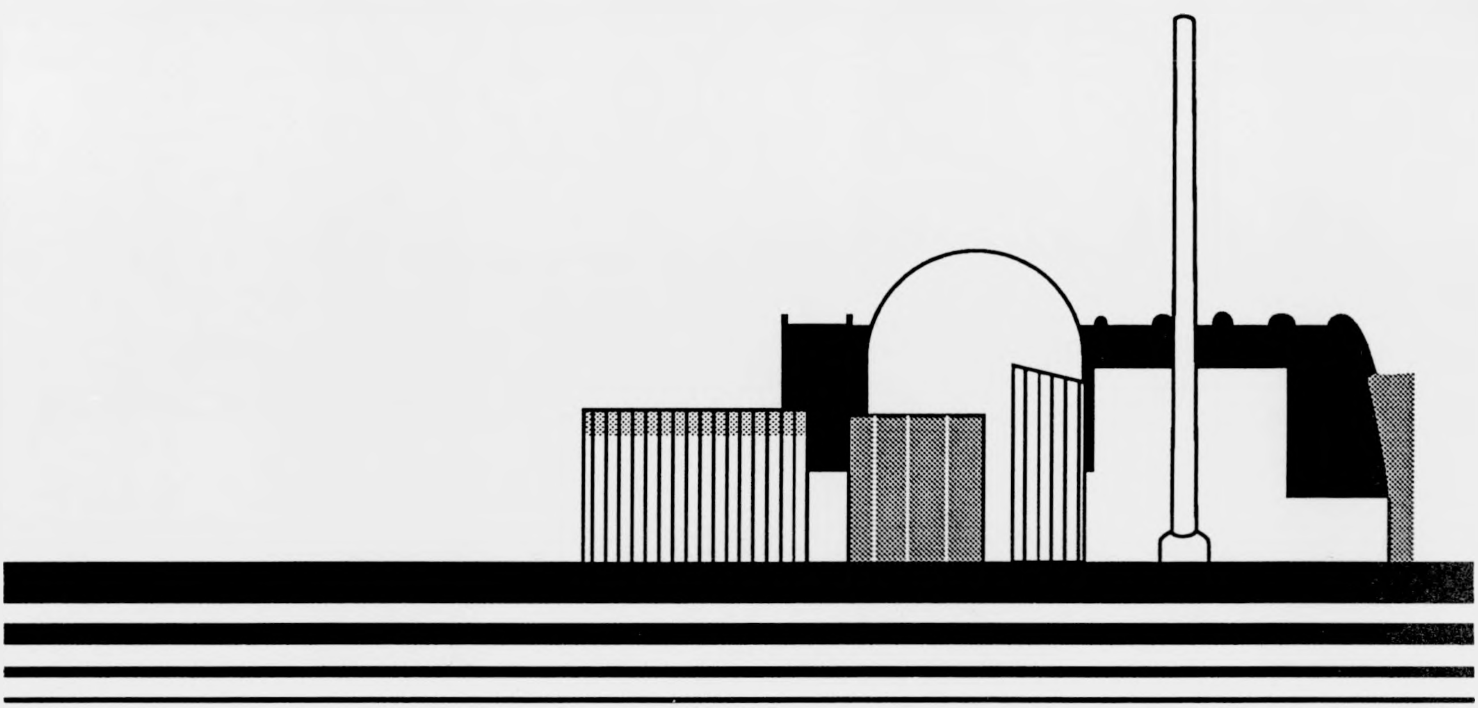
An Account of the OECD LOFT Project



Executive Summary

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AN ACCOUNT OF THE OECD LOFT PROJECT

EXECUTIVE SUMMARY

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EXECUTIVE SUMMARY[a]

Introduction

This report sets on record a Nuclear Energy Agency–Organization for Economic Cooperation and Development (NEA–OECD) initiative, following a proposal by the US authorities to set up and manage a major international project which would extend both the lifetime and scope of the US Department of Energy (USDOE) Loss of Fluid Test (LOFT) integral test facility at the Idaho National Engineering Laboratory (INEL) in Idaho, USA. The test program of 8 experiments, which extended from 1983 to 1989 followed a US Nuclear Regulatory Commission (USNRC) program of 36 experiments over the period 1976 to 1982.

The information from the OECD program has been made fully available to the international community both in the form of detailed technical reports and as a computer database containing all the quantitative experimental information, held both at the NEA Paris and at the INEL in Idaho. This account of the OECD LOFT Project can do no more than briefly summarize this mass of technical information.

The effective use of nuclear energy for power generation requires an in–depth understanding of a wide range of physical phenomena, advanced engineering, and competent and dedicated management. It raises difficult safety questions which are of concern to a wide general public and it is important to address these and to provide a credible account of the issues involved. At the same time, it is sensible to recognize that the pool of experts who can subject the technical arguments to detailed scrutiny is limited even on the widest international basis. There does exist however, a general body of scientists and engineers, not specialists in the field of nuclear energy, but who have experience in appreciating and understanding a wide range of scientific issues. It is primarily to this group that this report is addressed. This audience is international because it is clear that the problems of nuclear safety do not respect frontiers and the general scientific methodology is itself international.

The report does not confine itself solely to technical issues. The impetus for the whole project was the recognition that a major nuclear test facility was about to be decommissioned, that valuable work could still be done with it and that no comparable facility was likely to replace it within the foreseeable future. Nevertheless, while there was an international technical consensus that it would be valuable to secure the continuing use of the facility, the cost of such a program and the administrative problems of defining and managing such a project presented a major challenge. It was considered that a clear record of how the experience and resources of the OECD were used to initiate and guide such a project could be of wide interest and a guide for future similar initiatives.

There is no doubt that the concept of international collaboration in the operation of scarce and expensive experimental facilities has important financial and political attractions, but there are obvious problems in managing such projects. The OECD LOFT Project had the benefit of the experience and support of the operating agent, the USDOE, and of the team at EG&G who had already demonstrated their technical competence. There was however, from the beginning, a clear wish from Project members to have a direct input into the detailed choice of test conditions, the choice of instrumentation, and the level and nature of the supporting computer analyses. Also a number of important managerial and technical problems had to be resolved during the course of the Project. It was felt that an account of how the Project actually worked to resolve these problems would be of general interest.

Of particular importance was the relationship between the experimental data, the pretest and post–test calculations using major thermal–hydraulic codes, and the development of these codes. Members have felt that the work on these problems under the auspices of the OECD LOFT Project had an important influence on the development of a philosophy, now widely accepted, which recognized the close relationship between these ideas. The record on these developments is an important part of the report.

Finally, there is now a situation in which the essential role of major integral test facilities is fully accepted by the international technical community but the financial climate is such that many of these facilities are lost and not replaced. This account of the OECD LOFT Project is written in the belief that it shows that

a. This Executive Summary was written as part of the document, OECD LOFT–T–3907, *An Account of the OECD LOFT Project*, May 1990.

international collaboration is a practical way to seek solutions to these problems.

The History of the US LOFT Project

From the early days of commercial PWR development, it was recognized that an accident sequence in which there was a substantial impairment of normal core cooling could lead to fuel melting with the risk of a release of fission products beyond the plant boundaries. Early in 1962, the US Atomic Energy Commission (USAEC) planned the construction of a test facility, the Loss Of Flow Test otherwise known as LOFT. This would be a single test in which a small Pressurized Water Reactor (PWR), operating at full power, would be subjected to a complete loss–of–coolant flow. The objective was to show that, in a plant provided with a containment, there would not be an unacceptable release of fission products outside the containment boundary and that the meltdown configuration was controllable.

By the time the facility had been commissioned in 1976, the increase in size of commercial PWR's had led to a profound change in safety philosophy. Plants were now provided with a comprehensive system of emergency core cooling (ECCS), which would be invoked in any loss–of–coolant accident situation (LOCA) and which would be able to maintain core cooling at a level adequate to safeguard the integrity of the nuclear core and that of the containment. It was then clear that there was a need to demonstrate the satisfactory and reliable operation of these engineered safety systems and this view was reinforced by the conclusions of the hearing held by the USNRC in 1972–73 to support the adequacy of the licensing regulations. The objectives of the US LOFT Project were therefore changed to provide quantitative data to support the calculations used in plant licensing, to provide a scientific understanding of the phenomena which occurred in these LOCA transients, and to provide assurance that unexpected effects had not been overlooked.

The facility, as completed, simulated a typical 4–loop Westinghouse type PWR (Trojan) and attempted to retain the same volume/power ratio. Figure 1 shows an axonometric projection of the LOFT system.

The nuclear core consists of 9 square and 4 triangular corner assemblies. The square 15 x 15 assemblies were of standard PWR dimensions and 1.7 m high (half standard height). The core full power was 50 MW(t) and the maximum linear heat generation rate of 52.5 kW/m was well beyond normal operating values for a commercial plant. It had two coolant loops; an active loop with a steam generator and pressurizer which rejected heat to an air–cooled condenser and a broken loop containing the simulated pipe breaks and a steam generator simulator. The simulated breaks, which could be in either the cold or the hot leg of the primary circuit, discharged to a blowdown suppression tank.

The facility is highly instrumented with a concentration on thermal–hydraulic data. These include measurements of fuel pin center and cladding temperatures, flow velocity and momentum, and coolant density.

The initial test program concentrated on major breaches of the primary circuit, typified by the 200% cold leg break, which was identified at that time as the design basis accident. Three such large break tests (LB/LOCA's), with nuclear heating, were carried out in 1978–81. These were considered to demonstrate that the ECCS designs were well able to control the peak cladding temperatures reached in the transient and that the peak cladding temperatures reached were much lower than those obtained in licensing calculations. There was however clear evidence of early rewet phenomena which had a pronounced effect in reducing peak cladding temperatures but which seemed likely to be much less important for full–size cores in commercial plants.

The emphasis in the LOFT test program was then changed as a direct consequence of the loss–of–coolant event in the plant at Three Mile Island in 1979, to recognize the importance of small–break LOCA's (SB/LOCA's) which appeared to be much less well understood and whose consequences could be strongly influenced by operator action. As a result, a further program of 25 tests was carried out covering a range of small–break LOCA's anticipated transients and a range of multiple failure scenarios. These were all relatively long–term transients of relatively high probability in which there was a risk that the emergency cooling systems might not operate as planned either because of operator action or subsequent equipment failure during the relatively long accident sequence.

By 1980, the extended program of tests in LOFT had been largely completed and the future use of the plant was under discussion. Undoubtedly an important factor was the cost of operation at $50M per year, a cost which was largely independent of test frequency.

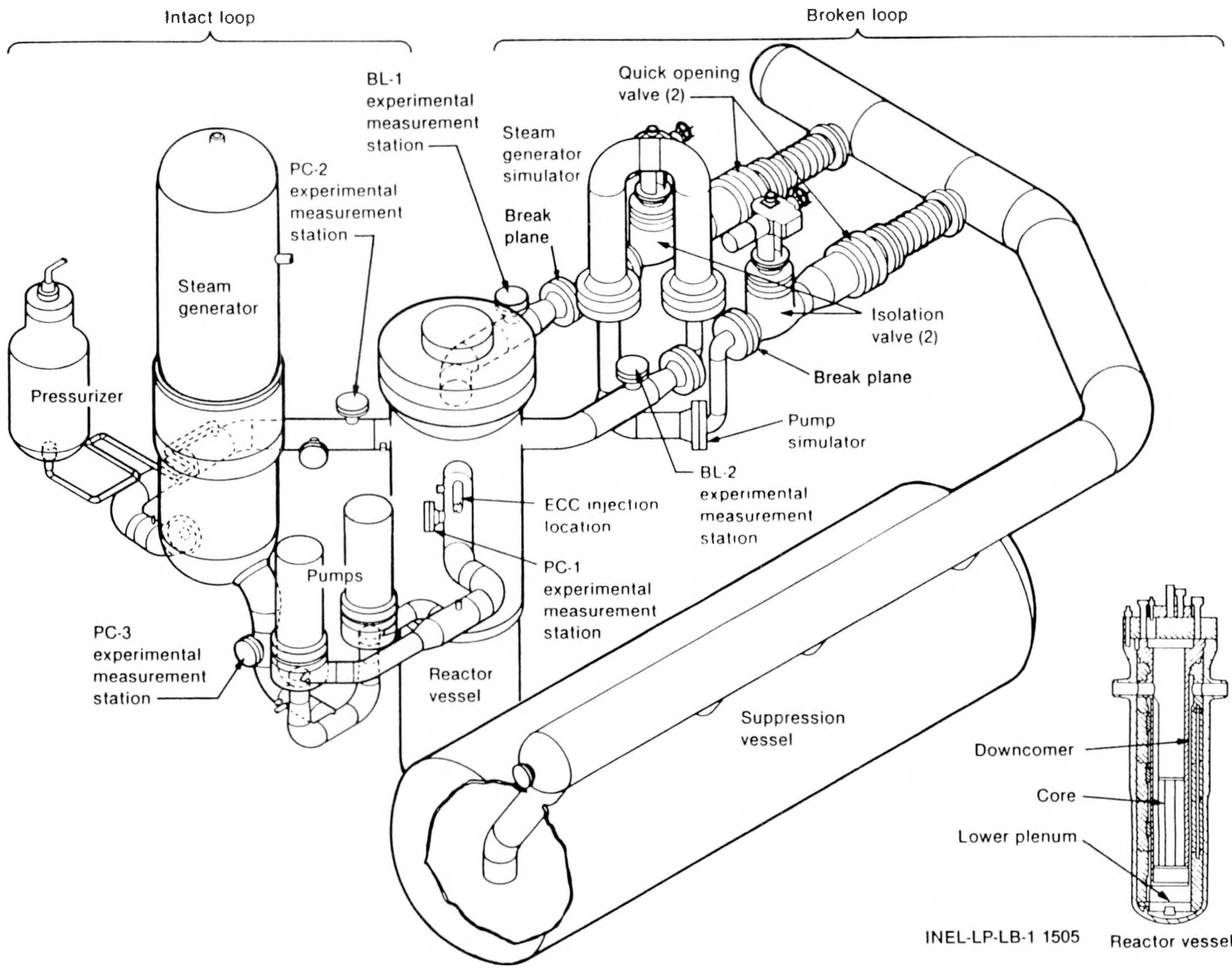

Figure 1. Axonometric projection of the LOFT system.

The US Advisory Committee on Reactor Safeguards (ACRS) then carried out a review of the USNRC LOFT research program and recommended that after a limited program of further testing lasting at most an additional two years, the reactor should be decommissioned unless a major new source of financial support could be found. The USNRC after receiving the report of its own LOFT Special Review Group (LSRG), essentially accepted this recommendation, with the provision that limited testing could extend into 1983.

Setting up the OECD LOFT Project

Although the LOFT facility and program was entirely a US initiative, there was already a strong international interest. The USNRC had continued to make the results of the experiments widely available and bilateral arrangements with a number of countries had allowed them to send staff to the LOFT team at the INEL. The international community was therefore well informed as to the value of the program. As a nuclear facility at a power and volume ratio to full scale of 1:50, it was seen as unique, with no possibility of replacement. It was a major source of validation data for thermal–hydraulic codes and it was clear that valuable work could still be done with it. There was also a case for retaining it as a flexible test facility that could be used to study unexpected results.

Against this background, the LOFT Project team held bilateral discussions with a number of countries to ascertain whether it was possible to obtain financial and technical support for a continuing program. These discussions confirmed that there was a climate of general support and a broad agreement on the possible technical scope but it became clear that bilateral negotiations would not provide an effective framework for setting up a project. As a consequence, in early 1982, the US Department of Energy made a formal approach to the OECD Nuclear Energy Agency in Paris to ask it to ascertain whether an international project could be set up to continue the LOFT experimental program from 1983.

This approach recognized that the NEA was an organization with considerable experience in the initiation and management of international projects and that it had shown a particular interest in thermal–hydraulic phenomena in water reactors. It had sponsored the first international specialist meeting in Munich in September 1972 on emergency core cooling, which developed into the CSNI Principal Working Group on Transients and Breaks. It had recognized the essential role of computer modelling and in 1973 set up a continuing program of International Standard Problem Exercises to provide an international forum to compare the performance of LOCA/ECCS codes by using them to predict the results of major LOCA experiments.

The USDOE proposal was discussed at a meeting of the NEA steering Committee for Nuclear Energy in April 1982 and a meeting of experts representing all the countries participating in the NEA was held in Paris in June. At this meeting, the US representatives put forward a set of 9 to 14 experiments as the basis from which a suitable program could be selected. Most of the items in this test matrix were a relatively straightforward extension of the previous US program (loss–of–feedwater tests, large and small–break LOCA's, and steam generator tube ruptures) but an important innovation was to carry out two to three tests which would result in fuel damage at temperatures in the range 1300 to 1800 K and consequent fission product release. These new tests would be significantly more expensive than thermal–hydraulic tests but their inclusion was regarded by most participants as an important incentive for a continuing program.

The meeting agreed that these proposals could form the basis of a three–year program costing about $80M and that a further meeting would be held in Munich in July with the objective of agreeing a detailed program.

The Munich meeting was able to agree on a program of about 8 experiments for which there was consensus technical support and which were sufficiently well defined to provide a basis for proceeding with the formation of a project, but accepted that further discussion would be needed to finalize the program. The following issues were resolved:

1. Continuity with the US program and the effective deployment of staff would be helped if the first tests, a loss–of–feedwater test and two small–break LOCA's, could proceed as defined by EG&G.

2. The loss–of–feedwater test was a useful addition to the program though long–term cooling effects were difficult to simulate in LOFT.

3. Small–break tests were not well predicted and further data would help.

4. Steam generator tube rupture tests probably could not be effectively carried out in LOFT.

5. One test, L2–6 (later designated LP–02–6), remained from the USNRC program. As a

USNRC sponsored test, the US would define test conditions and support the test analysis but the test would be carried out as part of the OECD LOFT program. At this stage it was defined as an LB/LOCA but with impaired ECCS leading to cladding ballooning and fission product release.

6. There was a case for carrying out an LB/LOCA in which the test conditions were defined to minimize early rewet effects.

7. There was general support for a fission product release test but it was agreed that further work was needed to define test feasibility and test conditions.

The next meeting to review progress was held in Paris in September 1982. It had before it the report of the Munich meeting and a separate UK assessment which broadly supported the feasibility of the proposed fission product behavior experiments. A draft Agreement on Legal and Administrative Arrangements had already been issued to participants. After discussions at this meeting and during October, it was agreed that the level of technical and financial support was now sufficient to proceed with the formation of a Project. As a result of this favorable response, it was agreed at a meeting of the OECD Steering Group for Nuclear Energy in October 1982 to proceed with the formation of an OECD LOFT Project and that the transfer from the USNRC program would take place in November 1982.

Project Organization and Initial Decisions

The first formal meeting of the Project took place in February 1983. At this stage the participants were Austria, Finland, FRG, Italy, Japan, Sweden, Switzerland, United Kingdom, and the United States (DOE, NRC, and EPRI). Spain joined the Project in September 1984. The agreed budget was $91.21M. Direction of the Project was invested in a Management Board (Chairman Dr. D. Hicks, UK) with the responsibility for agreeing the technical program and approving the annual budget. A Program Review Group (PRG) (Chairman Prof. E. F. Hicken, FRG) was set up to define and review the technical program and to act as technical advisor to the Management Board. The USDOE would act as Operating Agent for the Project and would use EG&G as the performing contractor. They nominated Dr. G. D. McPherson as LOFT Project Manager and Dr. P. North, EG&G, as Onsite Contract Manager. Each signatory was encouraged to send up to three technical experts to join the Idaho/INEL team.

The first meeting of the Program Review Group in March 1983 agreed that the program would consist of 8 tests. By the time of the Management meeting in July 1983, the 6 thermal–hydraulic tests had been defined in some detail but it took more time to define the details of the two fission product release tests. The final test program and the dates on which the tests took place are summarized below.

Experiment 1 – (LP–FW–1, February 20, 1983)

This test was initiated by a total loss–of–feedwater. The objective was to demonstrate the effectiveness of feed–and–bleed cooling by coolant feed from the high pressure ECCS and coolant bleed through the pressurizer relief valve.

Experiments 2 and 3 – (LP–SB–1 & 2, June 23 & July 14, 1983)

Both these LB/LOCA experiments simulated a 3–in. hot leg break but LP–SB–1 had an early and LP–SB–2 a late primary pump trip. The choice of pump trip time was expected to have a marked effect on the probability of core uncovery.

Experiment 4 – (LP–02–6, October 3, 1983)

The USNRC finally decided that this LB/LOCA test would be a 200% cold leg break with global boundary conditions as close as possible to those specified by Appendix K rules. There would be some impairment of ECCS availability and the fuel pins would be pressurized. The objective was to show that such a transient would not lead to a significant risk of fuel pin failure from cladding ballooning.

Experiment 5 – (LP–LB–1, February 3, 1984)

The PRG accepted the UK proposal that this test should model minimum ECCS conditions for the Sizewell UKPWR. There would be an early pump trip after which the pump flywheel would be disconnected. These test conditions were chosen to minimize the early rewet effects seen in some LOFT tests which the UK believed to be less important in full–scale plant conditions.

Experiment 6 – (LP–SB–3, March 5, 1984)

The detailed proposals for this SB/LOCA test were made by Italy. It would be a 1.8–in. cold leg break with coincident High Pressure Injection System (HPIS) failure. This would lead to a core uncovery. Plant recovery would be initiated when the peak clad temperature reached 977 K by a dump of secondary steam leading to primary depressurization and the injection of ECCS from the accumulators. This would be the only LOFT SB/LOCA in which there was a core uncovery.

Experiment 7 – (LP–FP–1, December 19,1984)

Because LP–02–6 was not expected to lead to fuel cladding failure, it was agreed that LP–FP–1 would be the first fission product release experiment and that the FRG would take the initiative in defining the test conditions. It would be a large cold leg break with delayed ECCS leading to fuel clad failure and the release of fission products from the fuel–pellet/fuel–cladding gap of the 22 pins in the center fuel module that were pressurized and enriched to 6%. The experiment would be terminated by ECCS injection when the measured peak clad temperature reached 1275 K which meant that there would be no significant overheating of the fuel matrix during the course of the transient. The measurement of fission product release, transport, and retention was a major objective of this test and a special Fission Product Measurement System (FPMS) was designed and installed for this experiment. Detailed calculations were carried out to define test conditions that would ensure an upward flow of single phase steam after fuel pin failure and while fission products were being transported out of the core. Calculations were also carried out to ensure that a high fraction of the 22 enriched and pressurized fuel pins would burst during the transient.

Experiment 8 – (LP–FP–2, July 3, 1985)

From the beginning, it was accepted that a major objective of this experiment would be to measure fission product release and transport during a transient in which there was an appreciable release of fission products from the fuel matrix as a consequence of fuel overheating during the transient. In the original proposal it was suggested that fuel temperatures might reach 1800 K. There was in fact considerable discussion before the conditions for this experiment were finally agreed and indeed important changes were made as late as May 1985. There were a number of reasons for this. There was a wish to take maximum advantage of the fact that this would be the last test in the facility and to increase the severity of the test conditions while remaining within the limits set by safety and cleanup costs. There was an increased interest in the severe core damage aspects of an experiment on a large fuel bundle. Finally, there was a desire to have a test in which a significant part of the heat input during the transient came from the exothermic zirconium/steam reaction.

In the final test configuration, the center fuel bundle contained a 10 x 10 array of enriched (9.75%) fuel pins and a representative number (12) of control rods. It was surrounded by a thick zircaloy/zirconium dioxide shroud. The model for the transient would be a WASH 1400 V sequence involving an interfacing valve failure. The fuel temperature would be allowed to rise during the transient to a level above 2100 K for three minutes. Fission product transport would be strongly influenced by an aerosol from the failed control rods and a special FPMS was designed and installed.

Issues Resolved during the Course of the Program

The issues that arose in defining test conditions for the first six thermal–hydraulic experiments have been briefly described in the previous section. A number of other problems arose during the course of the program and were successfully resolved. They are summarized in the following paragraphs.

The Option 5 Choice and the Cooperative Analysis Program

By the end of 1984, the thermal–hydraulic program had been finalized and considerable progress had been made in defining the two fission product release tests, but by then it had become clear that the proposed technical program could not be completed within the committed financial resources. The PRG reviewed the position and identified the following issues:

1. By reducing the extent of postirradiation examination, abandoning some advanced instrumentation proposals and accepting some reduction in post–test computer analyses, the total program cost, which had threatened to exceed $100M could be held to $93.5M. This was still a significant increase on the agreed budget of $91.25M and it was unlikely that it could be funded.

2. Postexperiment analysis was particularly important and should be done using the best available computer codes.

3. A number of countries were indicating problems in funding the agreed budget of $91.25M. This problem was not resolved until October 1984 and it was clear that there was insufficient support for an increase to $93.5M.

4. There was support for the so–called Option 5 proposal which retained the new baseline program but reduced the cost to $91.25M by agreeing that the responsibility for the definitive post–test analyses would be accepted by Project members, with the support of EG&G.

The Option 5 proposal was agreed to by the Management Board at their meeting in February 1984. The post–test analysis work would appear in two documents:

1. Experiment Analysis Summary Report (EASR). This would contain a definitive review of the experimental data and a best estimate postexperiment code calculation.

2. Comparative Analysis Report. This would contain a summary of the independent analyses carried out by members and a comparison of the results, with a critical commentary. It would attempt to reach conclusions on the ability of the available computer codes to provide a quantitative prediction of the data.

   It was agreed to split up these tasks as follows:

Experiment	EASR	Comparative Analysis Report
LP–FW–1	EG&G	EG&G
LP–SB–1 & LP–SB–2	EG&G	EG&G
LP–02–6	No EASR	Switzerland
LP–LB–1	UK	UK
LP–SB–3	Italy	Italy
LP–FP–1	FRG	FRG
LP–FP–2	EG&G	EPRI

This represented a significant extra input from member countries into the work of the Project. Although the proposal was accepted on financial grounds, there is no doubt that all members benefitted from the consequent requirement to work more closely with one another and with the team at the INEL.

The Availability of Computer Codes

It was clear from the beginning that members attached particular value to using the experimental data for the validation of computer codes and saw the comparison of this data with predictions using state–of–the–art computer codes as an essential part of the Project. It was understood that the pre–experiment and postexperiment calculations would be carried out by EG&G and the attached staff using the most recent versions of codes developed by the USNRC and that copies of the code versions used by EG&G would be made available to Signatories. This issue became of particular importance when PRG members accepted their obligations to carry out the analyses set out in the Option 5 proposal.

This problem of computer code availability ran into two separate difficulties. The original intention of EG&G was to use RELAP5/MOD1 to analyze all the thermal–hydraulic experiments. It soon became clear that this did not provide satisfactory predictions for SB/LOCA tests. Although some work was carried out with an interim version RELAP5/MOD1.5, the position did not stabilize until the middle of 1984 when the USNRC made RELAP5/MOD2 available for general release. There was general agreement that this code was satisfactory for SB/LOCA analysis. For the LB/LOCA analysis there was early agreement that RELAP/MOD1, which did not include a reflood model, should not be used and the pretest calculations for both LP–02–6 and LP–LB–1 were carried out with TRAC–PD2. However, the USNRC released a much improved version TRAC–PF1/MOD1 in mid–1984.

The difficulty was that the USNRC initially released earlier versions of the codes to any users who were able to use them effectively and to provide direct feedback on performance to the code originators. By 1984 it was clear that thermal–hydraulic code development had become much more costly and time–consuming than originally expected. The USNRC therefore wished to release these new code versions only through bilateral arrangements in which there would be a contractual obligation to provide the USNRC with specific analyses of experimental data.

The Management Board accepted the arguments presented by the USNRC but made the following two points:

1. They wished EG&G to use the best codes to which they had access, independent of any issue of wider availability.

2. The Board expressed the wish that all Signatories who desired would be able to conclude a mutually satisfactory bilateral agreement with the USNRC.

In fact, satisfactory arrangements with all PRG members were completed by July 1984. The result has been that members have had access to analyses based on either RELAP5/MOD2 or TRAC–PF1/MOD1 for all the experiments in the program, with the exception of the early test, LP–FW–1.

Fission Product Simulants

It was never going to be possible to achieve high fuel pin irradiations in LP–FP–2 and indeed the program review, carried out at the time of the Option 5 decision, limited the target value to 250 MWD/MTU. A correct simulation of commercial power reactor conditions would need figures nearer 30,000 MWD/MTU. Thus, for example, iodine/cesium ratios would be unrepresentative. It was also argued that deposition effects would not be modelled correctly unless there was at least a monolayer coverage of surfaces and that this would be difficult to achieve.

These arguments prompted a proposal to incorporate a mixture of fission product simulants (i.e., a mixture of non–radioactive isotopes) into the center of each fuel pin in the center fuel module during manufacture and to demonstrate the feasibility of this technique by irradiations in the Halden reactor. It would be a simple matter to model fission product concentrations typical of irradiations up to 30,000 MWD/MTU. The case was strongly argued but the final decision was to not use simulants. The decisive arguments were as follows:

1. Initial fission product deposition would be on the aerosol from the control rods and subsequent transport and deposition would be largely a function of the properties and concentration of this aerosol and would not be strongly dependent on fission product concentration.

2. Their use would give no data on the source term.

3. The uncertainties, particularly on chemical form and release behavior, were considered to be too large for a final and unrepeatable experiment in the LOFT facility.

The Extended Analysis Program for LP–FP–2

The LP–FP–2 experiment was carried out in July 1985. At a very early stage, although the analysis was incomplete, there was confidence that the test had met its stated objectives—the study of the transport and retention of fission products released from fuel overheated in the transient in the presence of a large aerosol source. It also became clear that the test was probably even more valuable in providing severe fuel damage data on a large fuel bundle in which there were important material relocation effects after several minutes at temperatures in excess of 2100 K under conditions in which heating from both decay heat and the zirconium/steam reaction were important. Unfortunately, the Option 5 program had involved severe restrictions on the extent of the postirradiation analysis program and there was only limited time before the program terminated at the end of 1986. This finding indicated that only severely limited severe core damage data could be obtained.

It was agreed that to take advantage of these features of the experiment, a wider postexperiment analysis program was needed. This would include neutron radiography and the sectioning of the center fuel module after filling with epoxy resin to produce about 20, 2.5–cm thick sections. These would then be subjected to a detailed metallographic and chemical analysis. There was also a proposal to include neutron tomography and this appeared a promising technique which would help to interpolate data between sections. The tomography proposal was only abandoned when EG&G produced information from other tests which show that widespread and severe neutron attenuation from indium dispersed from the control rods made it unlikely that useful data could be obtained. The program extension would cost an additional $3.5M and would last about three years.

Attempts were made to finance this extended analysis program by seeking additional members, in particular from countries which had participated in the early discussions on the OECD LOFT Project. A proposal for French participation was received in which the major contribution would be data from two PHEBUS PHASE III tests and a proposal to carry out the destructive examination of the LP–FP–2 bundle at Saclay, France. The Management Board finally decided not to pursue this proposal because the PHEBUS data fell outside the primary objectives of the LOFT Consortium Agreement and the additional cost and delays which would be incurred in moving the work to Saclay left only a small cost incentive.

After further negotiations it was found possible to obtain the additional funding from the existing membership. A proposal by the USDOE to accept the financial responsibility for the additional cleanup costs associated with the decision to increase the severity of the LP–FP–2 experiment was of considerable assistance in setting up these new arrangements. As a result, it was possible to complete the work included in the Extended LOFT program by the end of 1989.

A Summary of the Experimental Results

The results of the OECD LOFT program can be summarized briefly as follows:

1. LP–FW–1 was a loss–of–feedwater transient. The test showed that feed and bleed from the HPIS and the pressurizer relief valve (PORV) was an effective method for the safe termination of such a transient. Calculations showed that detailed modelling improvements were needed and that long–term transients of this nature were difficult to model accurately.

2. The two small–break LOCA's LP–SB–1 and LP–SB–2 were intended to show the value of an early trip of the primary pump in avoiding core uncovery in a hot leg break. In fact, the minimum coolant inventory in the core was very similar in both experiments. It is now clear that this was due to the effects of a gravitational separation of steam and water in the hot leg near the break. These effects were not well predicted and led to modelling changes in the codes.

3. LP–SB–3 was the only small–break LOCA test which led to core uncovery. It was designed to study core cooling under these conditions and to demonstrate the effectiveness of secondary feed and bleed in terminating the transient. Calculations with RELAP5/MOD2 and TRAC–PF1/MOD1 show that this complex transient was in general well predicted though there were some three–dimensional effects during core uncovery that were not modelled in the analysis.

4. The two large–break LOCA's LP–02–6 and LP–LB–1 were both intended to be directly relevant to licensing calculations and reflected the different approaches in the US and the UK respectively to the use of LOFT to provide data that would be relevant to transients in full–size plants. Both transients were interpreted to show significant margins against fuel damage and also provided a better understanding of the early rewet effects that have always been a feature of such transients

in LOFT. They also provided data that was useful in interpreting the performance of the external thermocouples on the fuel pins which were used to measure peak cladding temperatures. Calculations of the peak cladding temperature were satisfactory but there were problems in modelling the core thermal–hydraulics and the early quench phenomena.

The first attempt to carry out LP–FP–1 was aborted when there was an indication (later found to be false) that one of the quick opening blowdown valves that simulated the break had failed to open. When the test was repeated, by inadvertence, one of the accumulator lines was not completely purged and therefore contained a gas bubble. The resulting expansion of this gas bubble as the primary system depressurized during the test then caused water to be injected into the upper plenum, with the effect of an early unplanned ECC injection. Although only a small volume of water was injected, it had a profound effect on the course of the transient. Only 8 of the 22 prepressurized fuel pins burst and the thermal–hydraulic conditions in the core were complex and certainly not as planned. It does however seem that the objective of achieving an upward flow of steam for fission product transport was achieved. As a result of detailed studies using TRAC–PF1/MOD1, it is believed that the magnitude and effect of the unplanned ECC injection and the general pattern of core flow during the transient can be reasonably well simulated.

Although there were some unexplained failures in the fission product measurement system, it was possible to assess the partition of the noble gases iodine and cesium between the primary circuit and the blowdown suppression tank (BST). There was also a satisfactory agreement between estimates of the total release of fission products (source term) obtained from measurements on two enriched but unpressurized and intact fuel pins and the measured release to the primary circuit and the BST.

Experiment LP–FP–2 met all its objectives and must be considered as the major achievement of the OECD–LOFT program. All the instrumentation, particularly the fission product measurement system, appears to have worked as expected. After the test, the center fuel module (CFM) was removed from the reactor, epoxy resin was successfully injected into it, and 20, 2.5–cm thick sections were sawn from it. These were then polished and subjected to detailed metallographic and chemical analysis. A cross–sectional view of the LP–FP–2 center fuel module showing typical postexperiment configuration is illustrated in Figure 2.

Although the data from the test have been available since the latter half of 1989 and a number of calculations on different aspects of the transient have been carried out, it took considerable time and a number of attempts before it was possible to put together a consistent model of the major features of this experiment. For this reason it would be reasonable to expect that further studies could still lead to important changes in the interpretation of the data. Nevertheless, largely due to the work carried out at EG&G, it is now possible to present a coherent story.

The main features are as follows:

1. The center fuel module achieved temperatures exceeding 2100 K for more than 4.5 minutes and localized regions probably exceeded 2800 K.

2. There were two blockage regions. One near the bottom of the fuel element consisted of metallic melt material mainly from the control rods. The other in the middle of the fuel element consisted of previously molten (UZr)O formed from the interaction between previously molten zircaloy and uranium dioxide. The formation of eutectics caused materials to melt at temperatures significantly below their melting temperatures as pure materials.

3. There appears always to have been a free flow path through the CFM. Reflood and cooldown were rapid after ECCS injection although the blockage regions could only have been cooled externally.

4. During the main part of the transient, heat release from the exothermic zirconium/steam reaction appears to have been limited by a lack of steam. During reflood, substantial amounts of steam became available and this led to a significant heat input and a further temperature excursion leading to the highest temperatures reached during the experiment. This conclusion is based on the following four pieces of evidence:

   a. Most of the hydrogen (about 80%) appears to have been produced during reflood.

   b. There was a substantial additional release of fission products during reflood and this is indicative of a temperature transient at this time.

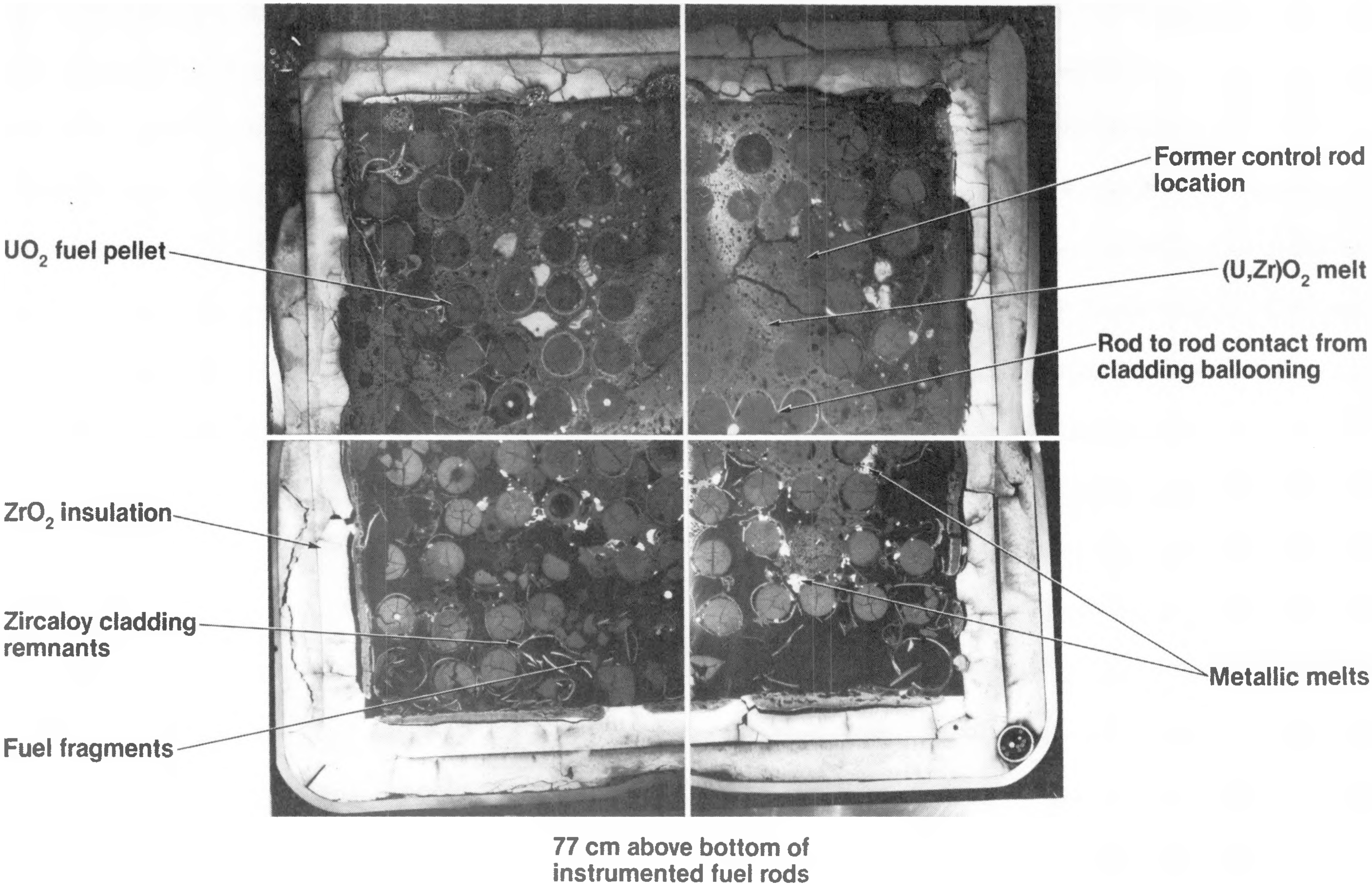

Figure 2. Cross–sectional view of the LP–FP–2 center fuel module showing typical postexperiment configuration.

c. Metallographic examination of both fuel and the upper tie plate provides evidence that these achieved much higher temperatures than were indicated by the thermocouples during the main transient.

d. Failure of a number of thermocouples due to high temperatures appeared to take place after the end of the transient and to have occurred during reflood.

5. The failure of the control rods occurred by bursting under pressure. This produced a large aerosol source, mainly of silver.

6. Between 2 and 5% of the CFM inventory was released during the transient and 14% during reflood. The primary fission products were probably CsOH and AgI with little evidence of CsI. Only small quantities of fission products (1 to 2%) were transported to the BST. Small quantities of Xe, I, and Cs were measured in the LOFT containment.

Only preliminary calculations have been carried out in support of the experimental analysis. In particular they have not included the reflood stage when the peak temperatures were reached. It is clear that unlike the thermal–hydraulic test, where the primary volume of the experimental data is in its use for code validation, the role of computer codes for LP–FP–2 is likely to be that of supporting and confirming our understanding of the experimental phenomena.

Conclusions

Nuclear power plants are designed with engineering features which provide an in–depth defense against the risk of major accidents. For example, in a PWR any breach of the primary coolant circuit can lead to a loss of effective cooling of the fuel with a possible rise of the temperature of the fuel pin cladding which could then lead to a cladding failure and a release of fission products to the environment. These reactors are therefore provided with an emergency cooling system (ECCS) as an auxiliary source of cooling in such a LOCA so that the accident transient can be safely terminated. The approach adopted both by reactor operators and licensing authorities to ensure the effectiveness of these engineered safely systems is by using computer code simulations supported by data from laboratory experiments, component tests, and large–scale integral test facilities.

The LOFT facility at the INEL was one of the most powerful of these facilities because of its size (1/50 full–scale), nuclear core, and general flexibility. This report describes an international program sponsored by the OECD to continue the use of LOFT after the conclusion of the USNRC program in 1982. The OECD program has now been completed and it is possible to draw some general conclusions. They cover the value of the data, the experience to members of the Consortium, the experience of operating an international project, and the lessons to be learned.

The thermal–hydraulics experiments undoubtedly provided valuable new data. LP–SB–1 and LP–SB–2 showed that phase separation effects in large pipes were not well modelled and LP–SB–3 provided heat transfer data in a core uncovery situation. The two large–break LOCA tests, LP–02–6 and LP–LB–1 are a source of new data on early rewetting and helped to obtain a better understanding of the performance of the fuel pin external thermocouples.

LP–FP–1 did not proceed as planned, but useful fission product release and transport data were obtained. Perhaps an unusual aspect was that it showed that the new thermal–hydraulic codes could be used to unravel a complex and unexpected set of thermal–hydraulic phenomena. As a result, flow conditions were well enough understood to support the fission product transport analysis.

LP–FP–2 was an important success. It was a difficult test to plan and to carry out and as the last test in the facility could not have been repeated. EG&G is to be congratulated on this achievement. It is a valuable source of data on severe core damage accidents in a large fuel bundle heated both by decay heat and by the exothermic zirconium/steam reaction. The latter was particularly important in reflood. These data are an important link with the phenomena seen in the TMI–2 accident. The other objective of providing data on fission product emission from fuel overheated in the transient and its transport in the presence of a silver aerosol from the control rods which burst during the accident was also achieved. Work on the full analysis of this test is still in its early stages.

The detailed results of these experiments are now archived at EG&G and at the NEA Paris. The thermal–hydraulic data is included in the PWR section of the CSNI Code Validation Database. It has therefore been made widely available internationally outside the membership of the LOFT Consortium.

As well as direct access to the experimental data, members obtained other important benefits from the

Project. They found it valuable to obtain access to the advanced technology being developed within the Project by attaching staff from whose expertise the Project also benefitted. They contributed to the design of the experiments and to their analysis and were provided with direct evidence on the performance of state–of–the–art computer codes. Perhaps of particular importance is that the experience of working together has helped to reinforce the view that in addition to the formal approach to licensing calculations, it is important to obtain a full understanding of the phenomena involved and to be able to model these effectively in computer codes.

As well as its organization of the technical program, the Project had to deal with a number of difficult financial and management problems. The Option 5 proposal was adopted because of financial stringency, but the close collaboration of members in experimental analysis was of mutual benefit. After the completion of LP–FP–2, it was realized that the original Project arrangements gave inadequate support to the severe core damage aspects of this experiment and this led to the LOFT Extension Agreement. Finally, it was possible to identify effective routes for continuing the work of the Project via the CSNI Specialist Committees, the International Code Assessment and Analysis Project, and the USNRC Severe Core Damage Program.

This report records the response of the international community to the opportunity offered by the USDOE to extend the life of a unique facility. It describes the organization of the Project, the detailed technical program, and the various problems encountered by the Project during its lifetime. It recognizes the important role played by the NEA–OECD in providing a project framework and offering its experience in initiating and operating such an international collaboration.

It has become increasingly difficult to retain facilities and expertise in a number of areas of reactor safety and international ventures of the kind exemplified by the OECD LOFT Project are a possible solution. The members of the Project wish it to be seen as an encouragement to further ventures of this kind and as evidence that nuclear safety can be effectively supported by international collaboration.