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AN ACCOUNT OF THE OECD LOFT PROJECT

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FIGURES

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ACRONYMS

BLHL	Broken Loop Hot Leg
BST	Blowdown Suppression Tank
BWR	Boiling Water Reactor
CCFL	Counter Current Flow Limitation
CEC	Commission of the European Communities
CFM	Central Fuel Module
CHF	Critical Heat Flux
CREST	Committee on Reactor Safety and Technology—NEA, OECD
CSNI	Committee on the Safety of Nuclear Installations—NEA, OECD
DNB	Departure from Nucleate Boiling
DRUFAN	Thermal–Hydraulic system code for PWER analysis developed by GRS
EASR	Experiment Analysis and Summary Report
ECC	Emergency Core Coolant
ECCS	Emergency Core Cooling System
FPMS	Fission Product Measurement System
FRAP	Computer code for analysis of LWR fuel under transient condition
FRG	Federal Republic of Germany
GRS	Gesellschaft fuer Reaktorsicherheit mbH.
HPIS	High–Pressure Injection System
IAEA	International Atomic Energy Agency
ICAP	International Code Assessment Program
ILCL	Intact Loop Cold Leg
ILHL	Intact Loop Hot Leg
INEL	Idaho National Engineering Laboratory
ISP	International Standard Problem
JRC	IspraJoint Research Center Ispra Establishment of the European Atomic Energy Community of CEC
KFK	Kernforschungszentrum Karlsruhe
KWU	Kraftwerk Union
LANL	Los Alamos National Laboratory
LB	Large–Break
LOCA	Loss–of–Coolant Accident

LOCE	Loss–of–Coolant Experiment
LOFT	Loss–of–Fluid Test
LPIS	Low–Pressure Injection System
LSRG	LOFT Special Review Group
LTSF	LOFT Test Support Facility
LWR	Light Water Reactor
MLHGR	Maximum Linear Heat Generation Rate
NEA	Nuclear Energy Agency (OECD)
NRC	Nuclear Regulatory Commission
OECD	Organization for Economic Cooperation and Development
PBF	Power Burst Facility
PCP	Primary Coolant Pump
PORV	Power–operated Relieve Valve
PIE	Post Irradiation Examination
PPS	Plant Protection System
PSI	Paul Scherrer Institute [previously EIR (Switzerland)]
PSS	Pressure Suppression System
PWR	Pressurized Water Reactor
QOBV	Quick–Opening Blowdown Valve
RCP	Reactor Coolant Pump
REBEKA	Cartridge type electric heater rod
RELAP	Thermal–hydraulic computer system code for LWR transient and LOCA Analysis—INEL
RCS	Reactor Coolant System
RNB	Return to Nucleate Boiling
SB	Small–Break
SG	Steam Generator
SMABRE	Thermal–Hydraulic computer system code developed in Finland
SPND	Self Powered Neutron Detectors
TC	Thermocouple
TIP	Traversing Incore Probe
TRAC	Thermal–hydraulic computer system code for PWR transient and LOCA analysis, three–dimensional capability—LANL
UP	Upper Plenum
USAEC	United States Atomic Energy Commission

AN ACCOUNT OF THE OECD LOFT PROJECT

1. INTRODUCTION

This report sets out the history and records the significant technical findings of a particularly important international collaboration on nuclear reactor safety, the OECD LOFT Project, in which a number of OECD countries, organized through the NEA, collaborated on a program to use the LOFT (Loss–of–Fluid Test) experimental nuclear test facility at the Idaho National Engineering Laboratory (INEL) in a program of safety experiments. The initial proposal was developed from an initiative of the United States Department of Energy, who also provided continuing management support. The project successfully combined the abilities and objectives of an international team with those of the reactor operation and analysis staff at INEL to provide a significant addition both to the international database of large–scale experimental data on reactor safety and to the analysis and understanding of the test results.

Nuclear power is now recognized as a major energy resource. In 1987, this amounted to over 400 plants which produced 300 GWe of electrical power. Another 200 plants are now under construction or planning and will add a further 200 GWe. As of 1987, there are 18 countries in which more than 10% of their electrical energy comes from nuclear power and 11 in which this figure is more than 30%.[1] This is a large concentration of resources and it is important to show that the industry is economically competitive and that its operation achieves high safety standards. For nuclear power, the major safety objective is to limit the radiation dose, both to operating staff and to the general population, to levels that are acceptable both to public opinion and to the licensing authorities. Nuclear power plants are designed, constructed, and operated on the basis that a potential risk to the public can arise, not only from normal operation, but from a whole pattern of accident scenarios ranging from those of relatively high probability and minimum consequences, to those whose probability is very low but which might lead to a substantial hazard extending beyond the boundaries of the power plant itself. As a consequence, all nuclear power stations include substantial built–in engineering safeguards to limit radioactive releases over a wide scenario of accident patterns, from the more likely to the highly improbable.

The design of such accident mitigation systems, or engineered safeguards, and the demonstration that they will operate effectively and reliably is a major challenge to the engineering profession. Several factors are at the heart of the problem. Full–scale testing of these systems over the whole range of accident scenarios is not a practical proposition because of the timescale and cost of such a testing program. Also, full– scale testing cannot deal with the wide range of plant designs or with changes brought about by design improvements. For major accidents, the engineered safeguards are able to ensure adequate protection to the public and operating staff, but may not avoid substantial plant damage and consequent financial loss. This in itself makes full–scale tests impractical.

Faced with this problem of demonstrating the performance and reliability of these safety systems, engineers have adopted the following strategy:

1. The behavior of individual components and features of the design is the subject of a rigorous experimental and theoretical investigation with the objective of understanding the basic physics of the phenomena involved and expressing these in terms of a detailed mathematical model. The adequacy of this model is checked against data from a "separate effects experiment" where the particular component or design feature is tested at near full–scale and under realistic operating conditions. An important feature of this investigation is that because we are dealing only with an individual component, near full–scale testing becomes feasible and "scaling" problems are not a major issue. Component damage during the test is also not a problem.

2. These individual component models are then synthesized into an overall computer model of the plant. In practice, it is useful to construct a number of different overall models, each designed to cover a limited range of accident scenarios and the individual component models they contain may have different levels of simplification. The development of these models is a continuing process since it has to be carried out within current limitations of computer speed, capacity, and numerical techniques, which change with time.

3. The overall model is then tested against a sequence of "integral experiments." An important feature of such an integral facility experiment is that it must invoke all the major physical phenomena that can influence the course of the accident

scenario and that these must then influence the result by approximately the same amount as in a full–scale test. For reasons already noted, the integral facility will not be full–scale and the specific tests may fall short of the full severity of the actual accident. As a result, a great deal of care is needed to ensure that agreement between the test results and calculations using the overall plant model is understood well enough to support the claim that the model is satisfactory for full–scale calculations. The following principles have been adopted to achieve this:

1. Calculations for both the nuclear plant and the "integral experiment" must demonstrate the same physical phenomena and show that they have substantially the same influence on the course of both transients. If not, there must be a clear and convincing argument to demonstrate that these effects are well modeled.

2. There should be a clear separation between the use of "separate effects experiments" in developing computer codes and the use of "integral experiments" to demonstrate the validity of the overall plant model. Predictions from calculations on "new" facilities are particularly valuable in ensuring this.

3. The whole program of testing and computer modeling should involve as wide a community of experts as possible. In particular, there should be independence between the groups providing experimental data and models and between those involved in their validation and use.

4. Finally, the overall plant models are used to demonstrate that over the range of accident scenarios for which they are designed, engineered safeguards hold accident consequences within design limits.

International cooperation has been an essential part of this endeavor. This was recognized by the setting up of such organizations as the International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD). The accident at Chernobyl reminded governments that the consequences of a serious nuclear accident would not respect national frontiers. Engineers and scientists also soon recognized that the investment, both financial and intellectual, to support a program of investigations could only be achieved by drawing on the full resources of the international community. It was also realized that the development of a technical climate capable of assessing and passing judgment on these safety issues would need a nucleus of informed opinion in each country. International cooperation in such safety studies was therefore seen by many countries as the most effective way of obtaining this expertise. As a consequence, those facilities that provide the essential experimental data and support theoretical work are spread widely across the international scene. Most importantly, effective measures of cooperation have been devised to make most of this information and expertise widely accessible.

2. THE USNRC LOFT PROGRAMS

2.1 Background and Initial Proposals

The approach described in the previous section comes most sharply into focus when considering the range of loss–of–coolant accidents (LOCAs) in which the rupture of a pipe in the primary coolant system of a PWR is the initiating event. This can lead to the loss of core cooling and a potential core meltdown. While it is feasible to carry out tests on small bundles of fuel rods in loop facilities attached to test reactors, the financial, technical, and safety problems of using a nuclear core in such tests has meant that almost all LOCA testing—both for special effect and component tests and in large integral facilities—has relied on electrical heaters to simulate the heating from nuclear fuel rods. The LOFT facility at INEL has been an important exception.

The earlier LOFT program in the US, initiated by the USAEC and implemented by the USNRC, has been a major contributor to LOCA research and has strongly influenced both the content and objectives of the OECD program. The intention of this section is to provide a brief discussion of the US experimental program both as important background material and as an aid in understanding the motivation of the OECD program.

Commissioned in 1976, the facility was first planned in 1962 as a "Loss–of–Fluid Test" (LOFT) with very different research objectives than those against which it was initially operated. These were to identify the physical phenomena and the course of events of a loss–of–coolant accident (LOCA) in a nuclear plant not provided with designed safety systems, and with no operational mitigation of the consequences. In order to demonstrate the behavior of the reactor and plant systems under these extreme conditions, and to provide quantitative information on the hazards associated with such an accident, a small number of nonnuclear tests would be conducted in the facility followed by a single nuclear test in which the failure of a primary coolant pipe was the initiating event leading to a core meltdown.

Parallel with the planning and construction of this facility, however, there was an increase in the thermal power level of commercial power plants [up to 3000 MW(t)] and an associated development of emergency core cooling systems (ECCS) to avoid major core damage in a LOCA and to ensure the integrity of the containment. The hearings, carried out in 1972–1973 as part of the regulatory process, then emphasized the need to demonstrate the satisfactory and reliable performance of these new ECCS systems. As a consequence, the facility design was substantially modified to meet the following redefined objectives:

1. To provide data to validate analytical methods to substantiate the claimed performance of ECCS systems and to assess quantitative safety margins.

2. To identify unexpected effects or thresholds in plant response or engineered safety features.

3. To provide experience and data to support US plant licensing practice and standards.

The test program using the modified facility began in 1976, initially using nonnuclear heating, with the first nuclear test at the end of 1978. The last transient test of the USNRC program took place in September 1982. Reference 2 discusses the history of the LOFT Project from March 1962 to March 1976 in more detail. Reference 3 presents the basis for the design of the facility and the state of knowledge of reactor safety.

2.2 Description of the LOFT Facility

The LOFT facility, as completed, simulated a typical, current generation, commercial 4–loop PWR reactor core, primary coolant system, and ECCS. It included a secondary coolant heat removal circuit and a blowdown suppression system. Reference 4 provides a detailed design description, and an account of the design philosophy and scaling approach can be found in References 5 and 6. Further plant details are provided in Appendix A.

In order to simulate the major thermal–hydraulic phenomena as accurately as possible, the design was based on volume scaling with emphasis on preserving real time scales in the saturated blowdown and refill/reflood flow regimes occurring in a large–break LOCA. The Trojan reactor, a 4–loop Westinghouse design, served as the model.

The reactor core consisted of nine fuel assemblies, each containing 15 x 15 fuel pins of standard PWR dimensions except that the pins were half–length (1.7 m) and the four corner assemblies were truncated to triangular shape. The fuel rods were 10.7 mm in diameter and the fuel was UO_2 of 4.05 wt% enrichment in zircaloy cladding. Core total power and axial profile were adjustable to achieve maximum heat generation rates of up to 52.5 kW/m (16 kW/ft), well beyond normal operating values in commercial plants.

The reactor head was designed to allow removal and insertion of the center fuel assembly. This feature allowed specially designed and instrumented fuel assemblies to be easily installed for specific experiments (for example, the study of fuel rod pressurization effects in large–break accidents).

There were two coolant loops, referred to as the intact loop and the broken loop. The intact loop contained a pressurizer and a steam generator which provided the main heat removal capability and rejected heat via a secondary coolant system to an air–cooled condenser which had an operating limit of 50 MW(t). There was no steam turbine or electrical power generating system. The broken loop was a passive system that contained the simulated pipe break. The broken loop had pump and steam generator simulators to provide the appropriate hydraulic resistance. The hot and cold legs discharged to the blowdown suppression tank. The ECCS, which was arranged as in a power plant, consisted of a high pressure injection system (HPIS), accumulator system, and a low pressure injection system (LPIS) whose characteristics could be varied for experimental purposes. The ECCS normally injects into the intact cold loop just before it enters the reactor pressure vessel, but in LOFT there was flexibility to inject into either hot or cold legs, the reactor downcomer, or the reactor vessel lower plenum. Two completely independent ECCS systems were available to permit the simulation of the FRG PWR ECCS design that injects both into the cold leg and the vessel upper plenum. A further backup ECCS was available for experiment termination.

A major feature of the facility was the instrumentation, much of which was specially developed. In addition to measuring neutron flux, the self–powered neutron detectors provided a valuable indication of core voidage, while conductivity probes measured core coolant level. Thermocouples on the fuel pins provided a very complete coverage of both fuel center line and fuel cladding temperatures. Coolant temperatures and pressures were extensively monitored. A particular feature of the LOFT facility was the use of drag–disc–turbine transducers (DDTs) and gamma densitometers to measure coolant momentum flux, velocity, and density at points in the primary circuit. Temperature, pressure, and liquid level were also measured in the blowdown suppresson tank.

2.3 The Development of the US Test Program

At the beginning of the test program, the major LOCA hazard was identified as a complete break of a major primary coolant pipe and the program concentrated on this transient. After a number of nonnuclear tests (the L–1 series) from 1976 to 1978 to determine the hydraulic response of the facility, the full program began with two large–break tests—L2–2 (December 1978) and L2–3 (May 1979) at low 25 MW(t), and at nominal full power 37 MW(t) respectively. These tests showed much earlier rewet and lower peak clad temperatures than had been expected. To some extent this was due to the shorter nuclear core and a number of other features which make LOFT not completely representative of a full–scale power plant. Overall, the results were considered to demonstrate significant conservatism in the licensing rules. Two further large–break tests, L2–5 (June 1982) and LP–02–6 (October 1983), confirmed this view, the latter test being carried out as part of the OECD LOFT program. These two tests were designed to give a better understanding of some early rewet phenomena that appear to be particularly prominent in the LOFT facility but are probably less significant in the full–scale plant, and to provide data specifically relevant to US licensing. There were also two intermediate break tests, L5–1 and L8–2, (September and October 1981).

As a direct consequence of the small–break LOCA which occurred in the Three Mile Island (TMI–2) nuclear power plant in March 1979, the large–break testing was substantially interrupted after May 1979. It caused a major USNRC review of LOCA analysis priorities, expressed in the following conclusions:

1. Large–break LOCA's appeared to be well understood and were covered conservatively by existing licensing rules. Improved computer codes were being developed which gave improved modeling of the relevant phenomena. Small–break LOCA's, however, could involve phenomena which had not been adequately researched, and their safety margins were not well quantified.

2. There was a need to investigate accidents where the accident timescale was large enough for operator intervention to be an important factor.

3. The emphasis in the study of these longer timescale accidents should include methods, engineering systems, and operator plans to aid recovery and to minimize the consequences.

These new objectives were incorporated into the LOFT experimental program with immediate effect. As a result, from May 1979 to the end of 1982, the program covered 26 tests, classified as follows:

1. L3 series, small–break LOCAs, seven tests, March 1979 to June 1980. This was a program in direct response to TMI–2 and consisted largely of small–break tests.

2. L6 series, anticipated transients, 13 tests, October 1980 to August 1982. This was a program of relatively minor transients but whose probability is sufficiently high that some of them could be expected to occur within the lifetime of an average power plant. They are relatively long–term transients, and were therefore included to meet the changed test objectives noted above.

3. L9 series, anticipated transients with multiple failures, five tests, April 1981 to September 1982. These were designed to study multiple failure scenarios potentially more serious than design basis accidents.

Table 2.1 lists the full sequence of tests sponsored by the USNRC. The information from this very extensive test series has been reviewed elsewhere (References 7 to 10). A number of issues were identified which had an important influence on the initiation and content of the OECD LOFT program:

1. There was confidence that tests in LOFT were capable of reproducing the major thermal–hydraulic phenomena which can occur in a full–scale plant LOCA.

2. Because LOFT is not full–size and some aspects of its geometry are unrepresentative, the interaction and relative importance of these phenomena can differ from those in the full–size plant. These effects can be studied by an appropriate modification of test conditions (e.g., early pump run–down in L2–5). Despite these problems, there was confidence in using LOFT as an effective source of integral test data for code validation.

3. A number of specific issues were recognized that raise difficulties in the direct application of LOFT data and would need special consideration in any further test program:

    a. LOFT heat losses are relatively larger than in a power plant and this is particularly important in the analysis of long–term transients.

    b. Leakage paths in the primary circuit (e.g. between the vessel upper head and the downcomer annulus) are unrepresentative and have an important effect in small–break LOCAs.

    c. Difficulties in understanding the response of the fuel rod external thermocouples in LOFT have created problems in the analysis and modeling of rewet behavior in large–scale LOCAS. Although the implications for licensing are probably conservative, this has led to uncertainties in validating computer codes for these transients.

4. It was recognized that there were important gaps in the test matrix and it would be valuable to fill them.

2.4 The Termination of the USNRC LOFT Program

Like all other USNRC Safety Programs, the work on LOFT has always been subject to periodic review by the US Advisory Committee on Reactor Safeguards (ACRS) and has indeed benefited from its support over the years. Their review in July 1980 of the NRC LOFT Research program recommended that after NRC completed the LOFT program for FY–1982, the facility should then be decommissioned unless it were taken over by the nuclear industry. Several factors influenced this ACRS recommendation. The annual cost of conducting the program was approximately $50 M per year and was largely independent of experiment frequency. Also, completion of the planned experiments through FY–1982 ensured an extensive database covering a wide range of LOCA and operational transients. Finally, additional short–term transients were judged to be of only limited value and longer term tests, though possibly useful, suffered from LOFT facility limitations such as unrepresentative heat losses and core bypass flows.

The USNRC responded by setting up a LOFT Special Review Group (LSRG) to perform a technical review of the program to assist the Commission in its response to the ACRS recommendation. The conclusions of the LSRG may be summarized as follows:[11]

Table 2.1 USNRC LOFT experiment program

Experiment Identification	Date Conducted	Description
NRC L1 Series: Nonnuclear Large–Break LOCAs		
L1–1	03.04.1976	50% hot–leg–break LOCA
L1–2	05.10.1976	100% cold–leg–break LOCA; delayed cold–leg ECC
L1–3	06.28.1976	100% cold–leg–break LOCA; no ECC
L1–3A	07.15.1976	100% cold–leg–break LOCA; lower plenum ECC
L1–4	05.03.1976	100% cold–leg–break LOCA; cold–leg ECC
L1–5	04.25.1978	100% cold–leg–break LOCA; core installed
NRC L2 Series: Nuclear Large–Break LOCAs		
L2–2	12.09.1978	100% cold–leg–break LOCA; maximum heat generation, 13 kW/m
L2–3	05.12.1979	100% cold–leg–break LOCA; maximum heat generation, 39 kW/m
L2–5	06.16.1981	100% cold–leg–break LOCA; maximum heat generation, 40 kW/m, rapid pump coastdown
NRC L3 Series: Small–Break LOCAs		
L3–0	05.31.1979	Stuck open PORV from hot standby
L3–1	11.20.1979	10–cm (4–in.) cold–leg, noncommunicative–break LOCA
L3–2	02.07.1980	2.5–cm (1–in.) cold–leg, noncommunicative–break LOCA
L3–3	04.15.1981	PORV LOCA and recovery (initiated at the end of L9–1)
L3–5/L3–5A	09.29.1980	10–cm (4–in.) cold–leg, noncommunicative–break LOCA; pumps off
L3–6	12.10.1980	10–cm (4–in.) cold–leg, noncommunicative–break LOCA; pumps on
L3–7	06.20.1980	2.5–cm (1–in.) cold–leg, noncommunicative–break LOCA
NRC L5 Series: Intermediate–Break LOCA		
L5–1	09.24.1981	35.6–cm (14–in.) cold–leg, noncommunicative–break LOCA with degraded ECC
NRC L6 Series: Anticipated Transients		
L6–1	10.08.1980	Loss of steam load
L6–2	10.07.1980	Loss of forced convection
L6–3	10.09.1980	Excessive steam load
L6–5	05.29.1980	Loss–of–feedwater
L6–6A	04.19.1982	Inadvertent boron dilution: nominal recirculation flow
L6–6B	04.21.1982	Inadvertent boron dilution: doubled recirculation flow
L6–7	07.31.1981	Rapid secondary side induced cooldown
L6–8B1	08.29.1982	Slow control–rod withdrawal
L6–8B2	08.26.1982	Rapid control–rod withdrawal
L6–8C1	08.26.1982	Primary–pump–based, small–break LOCA recovery, low voidage
L6–8C2	08.29.1982	Primary feed and bleed small–break LOCA recovery
L6–8C3	08.29.1982	Primary–pump–based, small–break LOCA recovery, high voidage
L6–8D	08.31.1982	Slow natural–circulation cooldown
NRC L8 Series: Severe Core Transients		
L8–1	12.10.1981	Rapid core uncovery and reflood (initiated at the end of L3–6)
L8–2	10.12.1981	35.6–cm (14–in.) cold–leg, noncommunicative–break LOCA with delayed ECC
NRC L9 Series: Anticipated Transients with Multiple Failures		
L9–1	04.15.1981	Loss–of–feedwater with delayed scram
L9–2	07.31.1981	Rapid natural circulation cooldown (initiated at the end of L6–7)
L9–3	04.07.1982	Loss–of–feedwater ATWS
L9–4	09.24.1982	Loss–of–offsite–power ATWS

a. There was support for a limited program of tests to complete the current program. These tests (with the exception of LP–02–6) are included in the complete schedule in Table 2.1.

b. The importance of test analysis and the associated code assessment and analysis was emphasized. This work was seen as lagging the experimental test schedule and there was support for its continuance beyond FY–1982.

c. There was only limited support for using LOFT for further work on anticipated transients, and this alone was not a basis for a continuing program. It was concluded that appropriately instrumented commercial reactors, supported by a data retrieval system, were the ideal source for such data. It was, however, recognized that because they were highly instrumented, tests of this nature could be carried out quickly in LOFT, with minimum impact on other tests.

d. There was only limited support for tests leading to core damage and fission product release, principally because the facility did not possess appropriate instrumentation. It was also felt that a sequence of such tests would contaminate the facility with an adverse effect on decommissioning costs. Severe fuel damage experiments were ruled out as peripheral to the LOFT mission. It was noted that the facility did not possess equipment for handling such cores, and that its cost would be an addition to the LOFT Budget.

Therefore the LSRG essentially recommended support of the LOFT program through FY 1983, but did not then recommend any further extension. They saw merit in accepting a stretched–out budget, at a lower annual cost but with the same total overall expenditure, since this would retain the availability of the facility for a longer period to deal with unforeseen problems.

3. THE FORMATION OF THE OECD LOFT PROJECT

3.1 The Case for Continuation of the Project

The report of the LSRG was presented to the USNRC in February 1981, and their recommendations were accepted by the Commission. As a result, it was agreed that the USNRC–sponsored program would continue until mid–1983 with a sequence of experiments essentially in the area of anticipated transients. This experiment sequence would conclude with L2–6. At this time, this last experiment was identified as a severe large–break LOCA. The fuel rods would be pressurized and, following cladding ballooning, there would be fuel cladding failure leading to the release of fission products from the fuel–pellet/fuel–cladding gap.

This decision by the USNRC, taken in the context of the nuclear research program in the United States, was of international concern. In this wider forum, three issues were seen as significant.

1. As an experimental nuclear facility operating at a power/volume ratio to full–scale of about 1:50, LOFT was seen as unique, with no possibility of replacement. No other facility matched it for size, the direct use of nuclear heating, or the ability to simulate a very wide range of loss–of–coolant accident transients.

2. Substantial international use was being made of LOFT data for code validation studies. There was a consensus view that further experiments of importance for code validation could be carried out using LOFT and that adequate data to cover these cases would not be readily available from other facilities. Specific experiments were identified which could form part of a continuing program.

3. There was a good case for retaining access to the facility and the supporting team as an effective resource for studying unexpected events or accident sequences. Once the operating and analysis teams were dispersed, it would in practice be difficult to use the facility again and certainly not on short notice.

There was already an international dimension to the LOFT program. Many countries have umbrella bilateral agreements with the USNRC for cooperation on nuclear safety research and several of them expressed a strong interest in being involved in the USNRC test program from its commencement in 1976. It was agreed that the greatest benefits would be obtained by participating countries if they were able to send technical staff to the LOFT team at the Idaho National Engineering Laboratory. The LOFT Project itself would gain by acquiring additional qualified engineers and scientists with possibly different approaches to the work on hand. This option of assigning personnel to the LOFT Project Office was therefore an important part of a number of bilateral agreements between the US and other countries.

Against this background, it was decided in the US to investigate the possibility of wider international participation by extending the existing bilateral agreements through which individual countries had already taken part in the USNRC program. In support of this, presentations of a possible forward program were made by the LOFT Project team in a number of countries during the early months of 1982. At this early stage, the program was based on the recommendations of the LSRG but with the awareness that these bilateral discussions could result in additions and emendations to the program.

These discussions confirmed that there was international support for some means of ensuring the continued availability of the facility and a fairly general consensus on what might be done with it. It became clear, however, that an approach through bilateral negotiations was unable to provide a route for identifying a specific program or for setting up an organization appropriate for the handling of such a major project. Under these circumstances and encouraged by the progress that had been made in the spring of 1982, the United States Department of Energy made a formal approach to the OECD Nuclear Energy Agency in Paris to ascertain whether an international consortium could be formed to continue the LOFT experimental program after the end of the USNRC–sponsored research.

3.2 The OECD Nuclear Energy Agency Background

This approach to the OECD Nuclear Energy Agency (NEA) recognized that it was an organization with considerable experience in the initiation and operation of international projects and that it had from its earliest beginnings shown a particular interest in thermal–hydraulic phenomena in water reactors. Its first concerted attempt to deal with nuclear energy safety began in 1965 with the setting up of the Committee on

Reactor Safety and Technology (CREST). The scope of this committee and its membership were increased in 1972 when it was replaced by the Committee on the Safety of Nuclear Installations (CSNI).

Within two years of the creation of CREST, the concerns on water reactor safety first voiced in the US led it to organize a Working Group with the task of systematically assessing the status of nuclear reactor studies on water reactors and to the issue by the Working Group of a report which appeared in 1969. A topic identified in this report as of major importance was a better understanding of the phenomena and processes which occur during loss–of–coolant accidents and the response and performance of the ECCS. This led to the first international specialist meeting on emergency core cooling for LWR's, held in Munich in September 1972. This meeting marked the beginning of a concerted program within the NEA on LWR safety and provided a forum at which experts from the NEA and the United States were able to agree which issues in the LOCA/ECCS field required more detailed investigation. These concerns led the CSNI in 1973 to set up an ad hoc group on Emergency Core Cooling, its first specialist body, which developed into the CSNI Principal Working Group on Transients and Breaks.

A major contribution of this working group of the CSNI was the recognition of the importance of being able to demonstrate the ability of computer codes to simulate the important phenomena in LOCA scenarios and to be able to provide a quantitative assessment capability. To support this, it set up the International Standard Problem Exercises (ISPs) in 1973. These provided a method of comparing the performance of a number of LOCA/ECCS codes by using them to predict the results of major LOCA experiments. The objectives of such an ISP exercise were as follows:

1. To evaluate the capability of system computer codes to predict controlled experiments.

2. To suggest necessary improvements in these codes.

3. To improve the ability of code users to provide information for the quantification of safety margins.

By 1982, the CSNI had performed a substantial number of thermal–hydraulic ISP's based on both separate effect and integral tests, including LOFT analyses. It had also begun the development of a validation matrix for the assessment of the thermal–hydraulic codes used in PWR LOCA analysis. The intention in producing such a matrix was to define an agreed set of experiments with measured parameters that could be compared with code results in order to establish the accuracy of predictions. The CSNI Working Group on Transients and Breaks also provided the CSNI with a unified position on ECCS issues from time to time.

It will be seen, therefore, that after nearly a decade of intimate cooperation, this specialist body of the CSNI had become a focal point and an accepted source of ideas, views, and scientific and technical results in reactor thermal–hydraulics.

In addition to this development of expertise in thermal–hydraulics, it is important to recognize that member countries initially saw the creation of joint undertakings on research and development as one of the principal functions of the Agency. Dragon, Halden, and Eurochemie were early examples of such projects. The rapid concentration throughout the world on one reactor line, the LWR, and the early commercial exploitation of this system greatly reduced research and development on alternate concepts and appeared for a time to limit the possibilities for cooperative ventures and prevented further NEA joint undertakings after these initial successes.

For a time, it seemed that safety issues, for which individual governments were prepared to provide important financial support, would also be handled on a national basis though, as noted above, the success of CSNI itself demonstrated that a need continued to be seen for international discussion and joint analysis. However, shrinking national budgets for nuclear energy research and the recognition that informed opinion within OECD countries was concerned to see that all available information was being brought to bear in ensuring safe operation and a defensible regulatory approach meant that, when the USDOE made its approach, the time was ripe for a new major undertaking by the OECD.

3.3 Initial Discussions and the Formulation of the Technical Program

The NEA Steering Committee for Nuclear Energy discussed the USDOE proposal on April 27 and 28, 1982. It already had before it the conclusions of its specialist Working Group which, at a meeting held in February 1982, had endorsed the conclusions set out in Section 3.1, which identified the unique capability of the facility, the value of further data for code validation studies, and the case for retaining access to the facility. At this meeting, a number of countries expressed interest in exploring mechanisms to implement such a proposal. As a result, the secretariat was instructed to convene a meeting of experts to attempt to define a technical program and to examine the financial and administrative arrangements that would be needed to attract wide participation by member countries.

A meeting of experts representative of all the countries participating in the NEA was held in Paris on June 2–3, 1982. At this meeting, the US representatives put forward for discussion the broad outlines of a possible experiment matrix as follows:

1. Steam generator tube rupture, including the effect of reactor coolant pump operation and pressurizer spray (approximately 2 experiments).

2. Loss–of–feedwater, including feed and bleed operations with different high pressure injection pumps (approximately 2 experiments).

3. Small–break LOCA/pressurized thermal shock (approximately 3 or 4 experiments).

4. Large–break LOCA, including the effect of steam generator rupture and fission product behavior (approximately 2 or 3 experiments).

5. Fuel damage at temperatures in the range 1000 to 1500° C, including the study of fission product behavior (approximately 2 or 3 experiments).

It was agreed that the objective should be to run a three–year program at a total cost estimated at approximately $80M. This was understood to permit a program of about 15 thermal–hydraulic experiments assuming little additional instrumentation and no major changes to the facility. Experiments resulting in severe fuel damage or fission product release would be significantly more expensive and would reduce the total number of experiments carried out. Nevertheless, the inclusion of such fission product experiments was generally seen as a major attraction of this new LOFT program.

This meeting was therefore successful in defining the broad scope of an international program and setting targets for the probable duration and cost. It was, however, recognized that further discussion was needed to identify a specific program against these technical objectives, which would also be consistent with the proposed budget. This was agreed as the task of the next meeting, held in Munich on July 15–16, and chaired by Prof. E. F. Hicken, GRS Munich. Representatives were present from the USA, Austria, Finland, France, Federal Republic of Germany (FRG), Japan, Netherlands, Sweden, Switzerland, and the United Kingdom (UK).

For this meeting, the US representatives had taken note of the comments made at the June meeting in Paris and they presented a somewhat modified proposal at the July meeting. In particular, they noted the strong interest expressed by the FRG in fission product experiments. In order to support the discussion, information was provided on the following wide spectrum of experiments:

1. 8 small–break LOCA experiments.

2. 4 steam generator tube rupture experiments.

3. 2 loss–of–feedwater experiments.

4. 2 large–break LOCA experiments.

5. 3 fission product release experiments. For these, a 5 x 5 array of more highly enriched fuel pins would be included in the center fuel module.

6. The L2–6 experiment was also included in the program. The US still saw this as a large–break LOCA leading to cladding ballooning and the release of fission products from the fuel–pellet/fuel–cladding gap.

The US representatives also presented two possible experiment scenarios taken from this spectrum of experiments as an example of a program that would be consistent with the overall budget.

Table 3.1 Proposed Experiment Plan July 1982

	Number of Experiments	
Experiment	Plan 1	Plan 2
Loss–of–feedwater	2	2
Small–break	3	3
Large–break	2	1
Steam generator tube rupture	2	1
Fission product	1	2
LP–02–6	1	1

Against a total cost projection of $80M, they proposed the following cost breakdown:

US (DOE + NRC), for experiment program	$30M
International support	35M
US, Decommissioning	15M
Total	$80M

These proposals were accepted as a basis for a detailed technical discussion. The conclusions on the priorities of the proposed experiment were as follows:

1. There was overwhelming support for the L2–6 experiment, identified as a large–break LOCA leading to cladding ballooning and fission product release. There were, however, important features of this experiment that still required definition.

2. It was agreed that small–break experiments were not well predicted and that further data would be valuable, particularly if they included core uncovery.

3. It was recognized that only one large–break LOCA experiment had been carried out in LOFT that was not dominated by early rewet to the extent not likely to be experienced in a large power plant.

4. Loss–of–feedwater was seen as a high probability fault for which there was a need for accurate prediction. It was recognized that, where long–term cooling effects were important, as in these transients, there were difficulties in carrying out representative experiments in LOFT.

5. There was general support for a fission product release experiment, but it was unclear how this could be used for code validation, given the current status of relevant codes. Further work was needed to define experiment conditions, but there was general support for further studies of feasibility. Members were asked to carry out an independent study with specific reference to an assessment of the value of the experiment data.

6. It was felt that the proposed steam generator tube rupture experiments were not well defined and suffered both from the limitations of the LOFT configuration and the sparsity of the steam generator secondary instrumentation. It was agreed to omit these experiments from further discussion.

Therefore, the Munich meeting was effective in defining a detailed program with consensus technical support. It was, however, accepted that some additional pruning would be needed to come within the expected budget, and that further technical review of the fission product experiments and a firmer definition of L2–6 were needed.

3.4 The Final Agreement

The next meeting to review progress was held in Paris on September 20–21, 1982, with representatives from Austria, Finland, France, FRG, Italy, Japan, Netherlands, Sweden, Switzerland, United Kingdom, United States (NRC and DOE), and the Commission of the Europeon Communties (CEC). The report of the technical experts from the Munich meeting was discussed. The meeting had also been provided with a separate UK assessment of the fission product behavior experiments, which broadly supported the feasibility of such experiments but drew attention to the requirement that the central assembly fuel rods should contain a sufficient quantity and the correct ratios of fission product isotopes. It was agreed that the experiment program as presented to the meeting now provided a sufficient technical basis for the formation of the Project and that the Project Management Board would be able to amend the program in the light of future developments.

A draft agreement on Legal and Administrative Arrangements had already been issued to participants, and this was accepted as a basis for proceeding. Following this resolution of technical and administrative issues, members were asked for a statement of support against an agreed level of financial contributions. After discussions at this meeting and during October, the position reached was as noted below:

Full support (conditional on the participation of sufficient members to meet the financial target):

Austria	$ 0.3	M
FRG	5.0	M
Italy	2.5	M
Japan	5.0	M
Sweden/Finland	1.0	M
Switzerland	0.7	M
UK	5.0	M
US (DOE, NRC, EPRI)	55.2	M[a]

a. The US contribution included $22.5M already committed for L2–6.

Firm interest was expressed by Spain and the CEC but it was not possible to resolve their participation on this timescale. France offered support for the continued use of the LOFT facility but stated that it was not able to participate in the proposed OECD LOFT project. As a result of this favorable response from eight countries, at the meeting of the Steering Group for Nuclear Energy on October 18–19 in Paris, a decision was reached to proceed with the formation of an OECD LOFT Project. It was agreed that the transfer from the USNRC Program would take place in November 1982.

3.5 Project Organization and Initial Decisions

The first formal meeting of the Project took place in Paris on February 9–10, 1983. The countries participating in the Project were Austria, Finland, FRG, Italy, Japan, Sweden, Switzerland, United Kingdom, and United States (DOE, NRC, and EPRI). EPRI had agreed to join as an associate member. After discussion, it was decided that separate representation from CEC Ispra would not be feasible unless all CEC countries agreed to participate in the Project. The meeting also noted, with regret, that Spain was unable to join at this stage of the Project but it did in fact do so in September 1984. On this basis the agreed funding was \$91.21M.[a]

The participating countries agreed on the following Project organization:

1. Direction of the Project would be vested in a Management Board with one representative from each signatory. The Board would be responsible for agreeing on the program of research and development and for approving the budget on an annual basis. It was also required to ensure that the management of the Project was sound and consistent with the objectives of the agreement. It would also review the reports of the Program Review Group.

2. A Program Review Group was also set up to act as technical adviser to the Management Board, again with one representative from each signatory. Its main task was to review the technical program and to advise the Management Board and the Operating Agent (USDOE). It was also responsible for reviewing the technical reports and ensuring that the experimental results were adequately documented.

3. The Project was to be operated by the USDOE as Operating Agent. They would use EG&G Idaho, Inc. as the Operating Contractor, responsible for planning, carrying out, and documenting the experiments.

4. Based on the previous successful experience under bilateral agreements, each signatory was encouraged to assign up to three technical experts to Idaho to participate in the Project.

Dr. D. Hicks (UK) and Mr. J. D. Griffith (USDOE) were elected as Chairman and Vice–Chairman of the Management Board. Dr. G. D. McPherson was nominated by the USDOE as the LOFT Project Manager. Dr. P. North at EG&G, Idaho was the Onsite Contract Manager responsible for the day–to–day operaton of the facility. At the subsequent meeting of the Project Program Review Group in Idaho on March 14–16, 1983, Prof. E. F. Hicken (FRG) and Dr. K. Tasaka (Japan) were elected Chairman and Vice–Chairman respectively.

Following the first meeting of the Program Review Group, further progress was made in defining the experimental program.

Experiment 1

This experiment, LP–FW–1, was initiated by a total loss–of–feedwater and hence of secondary cooling. The objective was to demonstrate the effectiveness of feed and bleed cooling, the coolant feed into the primary being via the low–head, high–pressure injection pumps, with coolant bleed through the power operated relief valve on the pressurizer. Because of scheduling requirements, the experiment conditions were agreed at the Management Board meeting in Paris and the experiment was carried out on February 20, 1983.

Experiments 2 and 3

It was agreed that both these experiments would simulate a 3–in. hot leg break. LP–SB–1 would be an early

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a. This included USDOE decommissioning costs of \$21.51M.

pump trip and LP–SB–2 a later pump trip. Significant differences in minimum coolant inventory were expected between these two cases with LP–SB–2 leading to a core uncovery and fuel cladding temperature excursion. These experiments were scheduled for June and July respectively and in the time available there was essentially no opportunity to influence or modify the experiment conditions from those originally proposed.

Experiment 4

The USNRC, who was solely responsible for defining the conditions for this experiment, LP–02–6, decided to change the experiment objective. The objective now became that of conducting an experiment with 200% cold leg break with global boundary conditions as near as possible to those specified by Appendix K rules. Coincident loss of offsite power and the availability of only one ECCS feed train would be simulated. The core power would be 50 MW(t) and the center assembly fuel rods would be pressurized to 350 psig. No special measures (e.g., early pump rundown) would be taken to minimize the early rewet effects seen in other LOFT LB/LOCA experiments. The basic objective now was to show that an Appendix K LB/LOCA at full power would not lead to significant cladding ballooning or fission product release. This experiment was scheduled for December 1983.

Experiments 5 and 6

These would be thermal–hydraulic experiments, but there were four contenders:

1. A large cold leg break with minimum ECCS and operating conditions in LOFT chosen to minimize early rewet effects.

2. A large cold leg break with coincident steam-generator tube rupture. Calculations using RELAP5/MOD1 showed a sharp peak in the peak cladding temperature at the equivalent of a 15 tube rupture and consequently optimum test conditions might be difficult to achieve.

3. A further small–break experiment at present undefined.

4. A large hot leg break. There was, however, evidence that peak cladding temperatures in such an experiment would not rise significantly above normal operating values.

Experiment 7

Since the probable result of LP–02–6 would be no fission product release, it was agreed that Experiment 7 would be a large cold leg break with delayed ECCS leading to fuel failure at a fuel cladding temperature of about 1200 K. There would be cladding ballooning and a release of fission products from the fuel–pellet/fuel–cladding gap. The experiment would be terminated in time to avoid any further heating of the fuel in the transient and the associated release of fission products. Agreement was reached on the definition of the experiment conditions largely as a result of a meeting of experts at the INEL in Idaho Falls on February 22–23, 1983.

Experiment 8

It was confirmed that the objective of this experiment was to study the release and transport of fission products where this release had been augmented by a rise in temperature of the fuel to approximately 1700–1800 K during the accident transient. There were, however, two important issues that remained unresolved:

1. Although the original intention had been to carry out a large–break experiment, it now appeared that a small–break experiment could be more representative of a probable core degradation event. However, this would require the fission product measurement system to operate at high pressures and would involve considerable extra expense. It would also be more difficult to achieve high fuel temperatures within the limits of the facility safety clearance.

2. Any experiment would have to use fuel irradiated to relatively low levels and hence with unrepresentative ratios of the various fission product isotopes (e.g., the ratio of cesium to iodine). Using fission product simulants could avoid the problem, and it was argued that they could be present at levels which could simulate some of the surface saturation effects expected to occur with highly irradiated fuel. It was accepted that it would probably be difficult to obtain a concensus on the appropriate chemical form for these simulants.

4. THE DEVELOPMENT OF THE OECD LOFT PROJECT

4.1 Planning Issues from July 1983

By the time of the second Management Board meeting in July 1983, the main features of the six thermal–hydraulic experiments appeared settled. Indeed, three of them, LP–FW–1, LP–SB–1, and LP–SB–2 had already been carried out and the conditions for LP–02–6, now a thermal–hydraulic experiment, had been firmly defined by the USNRC. Of the four contenders for the two remaining thermal–hydraulic experiments, two of them, the steam generator tube rupture in a large cold leg break experiment, and the large hot leg break, had been shown by further analysis to be unattractive, while the UK and Italy had put forward firm proposals for the experiments known as LP–LB–1 and LP–SB–3, respectively. All six thermal–hydraulic experiments were scheduled for completion by March 1984. Agreement was reached on the first of the fission product release experiments and suggested no change was likely other than in detail. On the other hand, for LP–FP–2, there were still difficulties in defining an appropriate experiment scenario and in selecting the instrumentation. Although the resolution of these issues was pursued in parallel, it simplifies the presentation to deal with them separately. This section describes the issues that arose during the completion of the thermal–hydraulic program and the implications of the so-called Option 5 decision.

4.2 The Large–Break Experiments LP–02–6 and LP–LB–1

Experiment LP–02–6 was conducted on October 3, 1983, from an initial power of 46 MW(t). Before the experiment, a further assessment had been carried out to confirm a negligible risk of fuel failure from cladding ballooning, particularly in rods fitted with thermocouples. In the event, there was no evidence of fuel cladding failure and the peak clad temperature achieved was too low to cause cladding ballooning. There was evidence of a bottom–up early rewet at 4 s into the transient and a partial top–down quench at 15 s. This experiment was seen as strengthening the case for test conditions proposed for LP–LB–1, because the LOFT experiments L2–2, L2–3, and LP–02–6 in one group and L2–5 and LP–LB–1 in another group were seen as presenting consistent patterns.

A number of detailed planning calculations for LP–LB–1 were carried out by the UK using TRAC–PD2/MOD1 to meet the following objectives:

1. The experiment should model minimum ECCS conditions for the Sizewell B UK PWR. This implies effective operation of only two out of the four accumulators, no high–pressure pump injection, and only two of the four low–pressure injection pumps operating. The initial power would be 50 MW(t).

2. There would be an early pump trip after which the pump flywheel would be disconnected. Researchers agreed that both early top–down and bottom–up quenches would occur, but calculations suggested that conditions could be arranged to minimize both these effects.

The essential philosophy in this experiment was to use LOFT to model the expected dominant phenomena in the full–scale minimum ECCS transient, rather than carry out the experiment in LOFT by directly scaling the global parameters. The opposite design approach was used for LP–02–6, and the two experiments were expected to complement each other. The prediction calculations for the LP–LB–1 specified boundary conditions were completed with TRAC–PD2/MOD1, and the experiment was carried out on February 3, 1984.

4.3 Decisions on LP–SB–3

The detailed proposals for this experiment were made by Italy and based on extensive calculations using RELAP5/MOD1. The proposal was a 2–in. cold leg break, with coincident failure of the HPIS. For this break size, the energy flow through the break is insufficient to remove decay heat and leads to a slow core uncovery. A dump of secondary steam leading to depressurization of the primary and the initiation of ECCS from the accumulators would lead to prompt recovery. Experiment conditions in LOFT were sensitive to the choice of break size, to heat losses in the system relatively much higher than in the full–scale plant, and to the time into the transient at which the primary pumps were tripped. Supporting calculations were also presented by Italy to demonstrate the direct relevance of the LOFT experiment to full–scale plant conditions. The LP–SB–3 experiment was of particular interest because it would be the only example of a core uncovery in a LOFT small–break transient and the method of recovery via a secondary depressurization was of interest to operators. Calculations to finalize the initial and boundary conditions were carried out at the INEL by the EG&G staff. The experiment was conducted on March 5, 1984.

4.4 Financial Problems and the Option 5 Choice

During the period from July 1984 to the end of the year, the thermal–hydraulic program was finalized and substantial progress (which will be discussed later) was made in defining the two fission product experiments. It had become clear by the end of the year that the agreed technical program could not be fully supported with the available financial resources. As a result, a special meeting of the Program Review Group (PRG) was held in December 1983, in Paris, to review the situation and make recommendations to the Management Board. The financial problem was attributed to several major causes:

1. Some of the technical proposals being considered by the PRG (particularly on instrumentation and postirradiation examination) were likely to result in a total cost near $100M. Subsequent evaluation brought the estimated cost down to $93.5M. The cost estimate prompted the following decisions:

   a. There would now be no postirradiation analysis of either the LP–FP–1 or the LP–FP–2 center fuel assemblies.

   b. The irradiation of the LP–FP–2 center fuel assembly would be limited to 250 MWD/MTU.

   c. The advanced instrumentation for aerosol deposition analysis using tape recorders would be replaced by filters, thus losing any possibility of time resolution of this data.

   d. There would be some postexperiment analysis, but there would be no topical reports on general issues (e.g., a review of all large–break experiments in LOFT had been proposed).

2. The Program Review Group felt that insufficient funding existed for postexperiment analysis and was not satisfied that the best available computer codes would be used.

3. Reasonable expectations existed that, after the initial decision to form the Project in October 1983, a number of other countries would join and provide additional funding. This had not happened.

4. Initially, the FRG, Japan, and the UK indicated their funding would include a significant contribution from their nuclear industry and that negotiations to provide such funding could not be finalized by the October 1983 date. Resolution of this position took longer than expected in the FRG and Japan, while the UK indicated it would only provide funding at the same level as these two countries. On the December timescale, a total of $3M was uncertain because of this unresolved issue.

In summary, the position in December 1983 was that the proposed funding of $91.2M was at risk by $3M and the modified baseline program would cost $92.5M. This modified baseline program was still seen by the PRG as an inadequate funding source for postexperiment analyses.

At the PRG meeting in September, EG&G presented five options as follows:

Baseline Program—$93.5M
This is the baseline program as summarized above. It included LP–LB–1, LP–SB–3, LP–FP–1, and LP–FP–2:

Option 1 – $87.25M
This replaced LP–FP–2 by a cladding ballooning experiment similar to the earlier specification of LP–L2–6.

Option 2 – $84.95M
This replaced the LP–FP–2 experiment by three thermal–hydraulic experiments.

Option 3 – $88.23M
LP–FP–1 was removed with some modifications to LP–FP–2 which claimed to provide some of the information that would have been obtained from LP–FP–1.

Option 4 – $85.55M
This is very similar to Option 2 in that LP–FP–2 was replaced by three thermal–hydraulic experiments. However, LP–FP–1 was also replaced by a cladding ballooning experiment similar to that described in Option 1.

Option 5 – $91.2M
This was essentially the modified baseline program, but with no postexperiment analysis or comparative reports. EG&G would therefore be responsible only for issuing the Quick–Look Reports and the digitized Output Data Tapes.

Faced with these proposals, the PRG made the following recommendations:

1. It agreed that the list of options provided was an adequate basis for a decision and did not propose to add further items.

2. It agreed with the decisions taken to produce the modified baseline program.

3. It was clear that there were strong objections to Options 1 to 4. Both LP–FP–1 and LP–FP–2 had significant support from committee members, and no consensus could be reached for any program omitting either.

4. Option 5 was acceptable only if a cooperative arrangement could be agreed upon between Project members for them to carry out postexperiment analysis and the comparative analysis reports, but with full support from EG&G on experiment information and features. It was noted that this proposal transferred some of the costs directly to member countries and further reduced their overall cost saving. The PRG also drew the attention of the Management Board to the importance of using the most up–to–date versions of computer codes and that special measures would be needed to make these available to participating countries.

The PRG noted that considerable effort had already been expended in support of the baseline program and that their decisions had undoubtedly been influenced by the view that major program changes were not feasible within the timescale and without extra cost.

The Management Board, at its meeting in February 1984, accepted the recommendations of the Program Review Group and made the following decisions:

1. The experimental proposals as set out in the modified baseline program were accepted.

2. It was agreed that the post–test analysis would be carried out by member countries but with full support from EG&G.

3. These decisions were consistent with a Project budget of $91.2M.

4. The Project Review Group was given the task of organizing the postexperiment analysis.

Although the Project operated on this basis through 1984, it was not until October 1984 that arrangements were made to secure payment of the final $3M. The position was also eased by the accession of Spain as a full member with a subscription of $0.75M.

4.5 OECD LOFT Technical Documentation and Cooperative Analyses

Although the formal initiative was taken in response to financial pressure, there is no doubt that the agreement among members to work more closely to carry out a comparative postexperiment analysis was one of the major achievements of the Project. It provided a continuing international forum for comparing code performance and for building up modeling experience. Because much of the analysis was carried out using common codes, in particular various versions of TRAC and RELAP5, it had an important influence on the development of these codes. This experience, in parallel with the work on NEA standard problems and the code validation matrix, (in which members were also involved) provided important background support to other cooperative ventures such as ICAP, the International Code Assessment and Applications Program.

It had already been agreed that each experiment would be supported by the following documentation:

1. Experiment Specification Summary—This summary sets out the experiment objectives and major parameters for Management Board approval.

2. Experiment Specification Document—This provides a detailed account of the experiment to support a full analysis and review by the PRG and for Management Board briefing.

3. Experiment Prediction Document—This is a best–estimate prediction carried out before the experiment to provide operational support and to demonstrate that the experiment objectives are achievable. Agreement was reached that this analysis would use the best available codes though it was recognized that not all members would have access to them.

4. Quick–Look Report—This would be issued within one month of experiment completion and would give a preliminary listing and assessment of experimental results to compare with the pretest prediction.

5. Data Tapes—These would contain completely qualified and processed experimental data in a digital data bank format.

6 Experiment Analysis Summary Report—This should contain definitive review of the experimental data and a best-estimate postexperiment calculation.

7. Comparative Analysis Report—This would include the analyses carried out by OECD LOFT members, a comparison of the results and a critical commentary. The report would attempt to assess the extent to which the phenomena displayed in the experiment could be quantitatively predicted by state-of-the-art codes.

It was agreed that under the new arrangements, this documentation would be handled as follows: All the documentation relating to experiment preparation, preprediction, preliminary results and the final database documents 1, 2, 3, 4, and 5, that is to say all the documentation relating to experiment preparation, preprediction, preliminary results, and the final data bank, would remain the responsibility of EG&G. The postexperiment analyses, represented by the Postexperiment Analysis and the Comparative Analysis Reports, would be the responsibility of members.

As shown in Table 4.1, these proposals were successfully implemented.

4.6 The Availability of Computer Codes

The impetus for the OECD LOFT program was the recognition that a unique experimental facility, still capable of producing data not likely to be available from any other source, was under the threat of closure. It rapidly became clear that project signatories attached major importance to the following three aspects of the program:

1. The selection of experiments that would add significantly to the world database for code validation and that would take full advantage of the special features of the LOFT facility.

2. The provision of qualified and processed data.

3. The comparison of this experimental data with predictions using state-of-the-art codes.

There was a brief reference to the availability of computer codes in the formal Agreement in the following form: "Article 6 Information and Intellectual Property Copies of computer programs used under this Agreement shall be given to each signatory for each and all uses."

There is no doubt that the general view among the signatories was that this clause was to be interpreted as implying three significant commitments by the Project:

1. Pre-experiment and postexperiment calculations would be carried out by EG&G and the attached staff.

2. These calculations would be carried out with the most recent versions of codes developed by the USNRC. There was also an understanding that the codes available to do this were capable of modeling the relevant thermal-hydraulic phenomena with quantitative accuracy.

3. Copies of the code versions used at EG&G would be made available to Signatories at no additional cost.

It could be argued that nothing in the wording of the agreement prevented the application of this clause to codes developed by organizations other than the USNRC, but in fact this question was never raised. Also, though most of the discussion centered on thermal-hydraulic codes, as time passed, it was recognized that this was also an issue for codes that modeled fission product release and transport and severe core damage phenomena.

These issues came sharply into focus for the thermal-hydraulic experiments because they were the first to be completed and there was perhaps a reasonable expectation that, after over 10-years work in this field, the codes would have reached maturity. In fact, the difficulties arose from two causes:

1. The original intention of EG&G was to use RELAP5/MOD1 to analyze all the thermal-hydraulic experiments, but it soon became clear that this was not providing good predictions of small-break LOCAs and was significantly inadequate for large-break LOCA scenarios. A new version RELAP5/MOD1.5 was expected to provide better modeling for small-break LOCA transients and in July 1983, EG&G expressed their intention to use this code. However, shortly afterwards, the USNRC concluded that this code was not sufficiently tested for general release outside the US, and that their policy would be to replace RELAP5/MOD1 with RELAP5/MOD2 (unavail-

able for general release until the middle of 1984). Since the thermal–hydraulic experiment program was completed by LP–SB–3 in March 1984, there was general agreement that the use of RELAP5/MOD2, at least for the small–break experiments was a satisfactory solution.

Table 4.1 Postexperiment analyses reports prepared by the project members

Experiment	EASR	Comparison Report
LP–FW–1	Project (EG&G)/December 1983	Italy/December 1984
LP–SB–1 & LP–SB–2	Project (EG&G)/May 1984	Japan/December 1987
LP–02–6	No EASR[a]	Switzerland/February 1987
LP–LB–1	UK/January 1986	UK/February 1989
LP–SB–3	Italy/December 1985	Italy/February 1987
LP–FP–1	FRG[b]	No comparison report[c]
LP–FP–2	USA (EG&G)/March 1989	USA (EPRI)

a. Since this experiment was not the responsibility of the Project, no EASR was issued.
b. Four volumes issued between November 1986 and April 1989.
c. Comparative analyses were included in the EASR.

2. For the two large–break experiments LP–02–6 and LP–LB–1, there was substantial pressure to use TRAC–PD2 because, unlike RELAP5/MOD1, it contained a reflood model and had been tested against large–break LOCA data. Also, the planning calculations for LP–LB–1 had been carried out in the UK using this code. As a result, the pre–experiment calculations for both these experiments were carried out with TRAC–PD2. However, an improved version–TRAC–PF1/MOD1–became available in mid–1984 and was used for postexperiment analyses for both experiments.

Earlier versions of both RELAP5 and TRAC had been widely distributed by the USNRC against the philosophy that a corpus of experience in the use of these codes outside the US and the provision of a direct feedback to code originators would be an adequate recompense. However, the development of a satisfactory base of thermal–hydraulic codes had proved significantly more time consuming and costly than originally expected. Therefore, the USNRC decided to release TRAC–PF1/MOD1 and RELAP5/MOD2 only through bilateral arrangements in which some part of their expenditure on code development would be balanced largely by work in kind on experimental analysis and assessment by code recipients.

The Management Board readily accepted that the further development of the thermal–hydraulic codes had proved more time consuming and costly than perhaps the USNRC had anticipated when the Agreement was drawn up, and that the interpretation of the clause in the Agreement cited above should be that it applied only to codes that were freely available. They also adopted two further resolutions:

1. The pre–experiment planning and analysis should be carried out, as far as possible, using the best codes available to EG&G independent of any question of their wider distribution.

2. The Board expressed the wish that all members who desired would be able to conclude a mutually acceptable bilateral agreement with the USNRC on codes relevant to the analysis of the OECD LOFT experiments.

In fact, the USNRC agreed to make TRAC–PD2 freely available to signatories in July 1983 and February 1984 accepted that it would act specifically to expedite bilateral agreements with Signatories on other codes. This issue was effectively resolved when Switzerland and Italy were presented with their copies of RELAP5/MOD2 and TRAC–PF1/MOD1 at the PRG in July 1984. Other members came to acceptable arrangements at an earlier date.

This whole issue became particularly important as a result of the Option 5 decision because most of the postexperiment analyses carried out by members and presented in the EASR and Comparison Reports were largely carried out with a consistent set of up–to–date codes. This can be seen in Tables 5.2 through 5.7, which set out the calculations carried out by signatories for each thermal–hydraulic experiment. In addition to the USNRC codes RELAP5 and TRAC, calculations were also presented using DRUFAN (FRG), RETRAN (UK, EPRI) and SMABRE (Finland).

In practice, further development of advanced thermal–hydraulic codes has continued with the release of RELAP5/MOD3 and TRAC–PF1/MOD2 and with parallel work in the FRG on ATHLET. It is however, realistic to recognize that this happened largely outside the framework and timescale of the OECD LOFT Project.

In principle, similar issues could well have arisen in the analyses of LP–FP–1 and LP–FP–2 but the later timescale of these experiments meant that suitable arrangements for access to relevant codes were concluded before there was a substantial need for analysis. Also, codes in this area were very much in the developmental stage and there was no pressure for the issue of definitive versions. Furthermore, it was not possible in these experiments to make a sharp distinction between the use of the computer codes to help understand the phenomena encountered in the experiment and the direct use of the experimental data to provide an independent validation of the codes.

4.7 The Fission Product Release Experiments

A major attraction of the OECD LOFT program was that, as well as completing the program of thermal–hydraulic experiments (described in the previous section), it also included two experiments leading to fuel failure and the consequent release of fission products into the primary coolant circuit. As the program continued, the objectives of these two experiments were seen to be quite distinct:

1. LP–FP–1 would be typical of an accident in which fuel failure led to the release of fission products into the coolant. The rise of fuel temperature in the accident would be safely terminated by ECCS injection before any fission products held in the fuel pellet matrix could be released. The appropriate transient for this experiment was identified as a large–break LOCA with delayed ECCS injection.

2. LP–FP–2 would be typical of an accident in which the fuel continued to overheat after cladding failure so that fission products were released both from the fuel–pellet/fuel–cladding gap and from the fuel matrix itself. Some form of small–break LOCA appeared to be an appropriate initiating event.

This section discusses the issues that arose in the final planning of these two experiments and the action taken by the Management Board to maximize the information obtained in LP–FP–2.

4.7.1 The Planning of Experiment LP–FP–1. It was agreed that the FRG would take the lead in cooperation with EG&G in defining the conditions for this experiment. The experiment would be a large–break LOCA with the ECCS delayed until after fuel–cladding rupture. The ECCS would operate in a mode representative of the German PWR ECCS in nominal (best estimate) conditions for combined upper plenum and cold leg injection. There would be minor modifications to the plant to incorporate the upper plenum injection facility. ECCS injection would be triggered on a thermocouple indication of peak fuel–cladding temperature and would be designed to avoid further fuel overheating after fuel–cladding burst.

The major objectives of the fission product measuring system were measurement of the retention of the noble gases and iodine within the PCS and their release to the blowdown header and the suppression tank, and the washout of fission products for up to 12 hours after system recovery. This was spelled out in more detail as follows:

1. Determine the fraction of volatile fission products released from the core region during the heatup period.

2. Determine the retention of volatile fission products in the upper plenum for a representative surface.

3. Determine the quantity of volatile fission products within the liquid and vapor in the blowdown suppression tank.

4. Determine quantitatively the fission products washed off of the reactor vessel and internals and leached out of the fuel after the transient.

5. Determine the general mass balance of volatile fission products in the fuel, the RCS, and the blowdown tank and header.

6. Qualitatively determine the chemical species of iodine transported to the plenum region during heatup, and out of the RCS during reflood.

The fission product measuring system was designed assuming that in the period between cladding burst and the final reflood, the fission products would be transported upward through the core by steam and this period would exhibit no flow through the cold leg to the break. The original intention was to ensure this by early injection of ECCS water to seal the bottom of the downcomer. This would also provide evidence of the effectiveness of upper plenum injection in reducing fission product transport out of the reactor vessel. However, TRAC–PD2 calculations showed that the initial effect of ECCS injection was condensation leading to downward steam flow. This problem was solved by delaying the initiation of ECCS injection until after cladding burst and by closing the Quick Opening Blowdown Valve at 65 s, thus leaving only the path through the hot leg for fission product transport to the break. There was also considerable discussion on what signal should be used to initiate the ECCS. It was finally agreed to do this when the peak clad temperature, as measured by the installed thermocouples, exceeded 1275 K. A careful statistical study was carried out to establish that this criterion predicted a high probability that all 22 enriched and pressurized fuel pins would burst, while still ensuring a low probability of further fuel heating after cladding burst. The 11 x 11 segment of the center fuel bundle containing the enriched pins was enclosed within a zircaloy shroud to assist in the sampling of released fission products.

The first attempt to carry out this experiment, which was later designated as LP–FP–1A, was made on Dcember 12, 1984. However, this experiment was aborted because the position indicator for the Quick Opening Blowdown valves indicated that this valve had failed to open. As a consequence, operator action was taken to terminate this experiment after 50 s. Later analysis showed that the position indicator had given a false reading and that, for the first 19 s, conditions fully met the requirements for a successful experiment. The data for this experiment is included in the OECD LOFT database. The experiment was then carried out on the second attempt on the 19th of December. On this occasion, experiment conditions did in fact meet the requirements of the experiment operations schedule but there was a premature release of a small amount of ECC water into the upper plenum and this severely compromised the scientific objectives of the experiment.

When the instrumentation and analysis requirements for this experiment were agreed upon prior to carrying it out it was decided not to carry out any Post Irradiation Examination (PIE) of the center fuel element or of the enriched pins because researchers were confident that all of the 22 enriched and pressurized pins would fail after cladding ballooning. Investigations after the conclusion of the experiment showed that the number of failed pins and the source term for the total fission product release were in doubt. It was therefore decided to puncture all the high enriched pins (leading to the conclusion that 8±2 pins had failed) and to analyze the two enriched but unpressurized pins, which had been designed to be removable, to determine their cesium and iodine content as a function of axial position.

4.7.2 The Planning of Experiment LP–FP–2. The major objectives of the LP–FP–2 experiment, as noted above, remained essentially unchanged since the start of the project. It was agreed and further confirmed in the Option 5 discussions that the US, and in particular, EPRI, would take the lead in consultation with EG&G in defining the experiment conditions. As part of these discussions, a serious attempt was made to see if the information to be obtained from the two fission product experiments could be supplied by an appropriately defined single experiment and this confirmed the separate role of these two experiments. Nevertheless, it was not until March 1984 that final agreement was reached on the major experiment features and subsequent important changes were made in experiment conditions as late as May 1985. The reason for this was the desire to take maximum advantage of the last experiment in the LOFT facility, while remaining within the restrictions set by safety considerations and the overriding requirement to avoid serious damage to the facility or intractable cleanup problems.

Two proposals were initially under consideration:

1. The first proposal was essentially a repeat of LP–02–6, i.e., a LB/LOCA but with a considerably delayed ECCS and with the fuel rods pressurized to 600 psi. About 150 fuel rods would fail after cladding ballooning and these pins would then overheat to about 1900 K. Only a small extra release of fission products from this overheating would occur (a few percent of the total held in the fuel matrix), but the total fission product release would be large because of the number of failed fuel pins. This was considered to be important in achieving representative conditions for fission product transport and retention.

2. The second proposal was to fail an array of more highly enriched pins in the center fuel element by allowing them to reach peak fuel temperatures of up to 2100 K for several minutes. This was considered possible while still holding the peripheral fuel within the agreed safety limit of 1500 K so

that fuel pin failure would only occur for the enriched pins.

Researchers concluded that the first proposal did not take adequate advantage of what could be done in a final experiment in LOFT. There is also no doubt that the strengthening of the LOFT team with EG&G personnel from the PBF Program influenced this decision, coupled with the knowledge that the PBF Phase II program would not be undertaken. In particular, it was stated that there would be no PBF experiment using control rods as an aerosol source. There was also a wish to carry out an experiment that would have direct relevance to some of the phenomena seen in the TMI–2 accident. As a consequence, the second proposal was further developed to include the following features:

1. The experiment would simulate the early features of a severe accident in which there was a loss of fuel geometry but successful termination of the accident by ECCS injections.

2. The model for the accident would be a WASH 1400 V sequence involving an interfacing check valve failure. The fuel temperature would be allowed to rise (from decay heat) to a level above 2100 K at which the heatup would be strongly driven by the exothermic zirconium/steam reaction..

3. An 11 x 11 array of enriched pins in the center fuel element would be surrounded by a thick zircaloy/zirconium dioxide shroud (essentially replacing two rows of fuel pins). This would allow the achievement of fuel temperatures above 2100 K for about 3 minutes while still maintaining a temperature limit of 1462 K for the peripheral fuel pins. A further limit of 1573 K was set for the temperature of the outer surface of the shroud to ensure its integrity and to permit easy removal of the center fuel element after the experiment. The shroud was also beneficial in increasing the fission product concentration in the upper plenum above the failed fuel and this made the experiment conditions more representative with a better signal for the fission product instrumentation system.

4. The fuel element would contain a representative number (12) of control rods. Upon failure, they would provide a strong aerosol source that could be expected to have a dominant influence on fission product transport and deposition on surfaces.

The detailed planning for achieving these experiment conditions proved unexpectedly difficult and there were problems in ensuring that temperatures in the center fuel bundle would increase sufficiently faster than those in the peripheral bundle. The main reason for this was because it was difficult to provide a steam supply to the central fuel bundle which was large enough to ensure that the progress of the zirconium/steam reaction was not limited by steam starvation. A number of techniques to improve the situation were investigated, and the final one was adopted to reduce the initial reactor power before the experiment from the original value of 33 to 25 MW. The lower decay heat reduced the rate of temperature rise in the core periphery while having much less effect on the central bundle that was being driven by the exothermic chemical reaction. The choice of a 5 sequence transient meant that fuel failure and fission product release all occurred at high pressure. This choice had two consequences:

1. The design of the fission product measurement system had to be modified from the low–pressure system used for LP–FP–1, with cost implications.

2. There were problems in providing zirconium dioxide tiles for the shroud capable of accepting these high–pressure conditions without mechanical failure. Those problems were solved by developing a higher density material. The detailed design of this shroud involved a number of very difficult technical problems and their successful solution by EG&G was crucial to the success of this experiment.

A major issue in the design of the experiment was whether to use fission product simulants to bring their concentration nearer to full–scale values. Arguments arose that claimed deposition effects on surfaces would not be representative unless there was at least monolayer coverage. This prompted a proposal to incorporate a mixture of fission product simulants into the center of each CFM fuel pin during manufacture. During operation, these materials would be uniformly dispersed within the fuel matrix. It was claimed that this would ensure that these materials would retain the same chemical form as the true fission products and would behave chemically and physically in the same way. It would, for example, be entirely feasible to achieve fission product concentrations and iodine/cesium ratios equivalent to an irradiation of 30,000 MWD/MTU, whereas the actual irradiation in the experiment could not exceed 1000 MWD/MTU on cost grounds. This issue proved difficult to resolve and plans were made to prove the adequacy of the simulation approach using irradiation experiments in the Halden reactor. The final decision to exclude simulants was made essentially on the following grounds:

1. Their use meant the experiment would give no data on the source term.

2. A careful study by EG&G concluded that their use would give a maximum factor of 10 on fission product levels.

3. The main deposition mechanism was likely to be condensation, therefore a realistic simulation only required that the fission product concentrations reached vapor saturation levels. This requirement was relatively easy to meet without using simulants.

4. In the presence of a strong aerosol source from the control rods, the initial deposition would be on this aerosol, its large area more than compensating for its higher temperature. Subsequent deposition on surfaces would then largely be a function of aerosol concentration and characteristics which could be expected to be representative of full–scale.

5. There was some evidence that simulants could affect fission product release behavior. Testing to resolve this was likely to be costly.

6. Finally, it was felt that this was a relatively untried and unproven technique and that the risks involved in using it were inappropriate to a final and unrepeatable experiment in the facility.

The target irradiation for the experiment was also subject to change. The original proposal was that this should be at least 500 MWD/MTU, but financial pressure reduced this at one stage to 180 MWD/MTU. Careful financial control made it possible to increase this and the value achieved in the experiment was 430 MWD/MTU.

The fission product measurement system closely followed that for LP–FP–1. A new proposal considered was to use a system being developed for the PBF experiments in which a tape recorder fitted with a sticky tape fed by three cyclone separators was considered to give a time dependent record of aerosol concentration and particle size. In the end, it was concluded that schedule pressures would not allow for the extensive testing of the total measurement system needed before such a novel system could be used in LOFT without an unacceptable risk of data loss. It was replaced by a system in which the three cyclone separators fed into a filter, thus retaining particle size discrimination but with no time resolution. In view of the importance of the zirconium/steam reaction, a measurement of final hydrogen concentration was also included.

The LP–FP–2 experiment was successfully carried out on July 3, 1985. The test requirement to achieve fuel temperatures in the center fuel bundle of more than 2100 K for more than 3 minutes was met with a substantial margin. The experiment was terminated when the maximum temperature in the peripheral fuel and outside shroud reached their simultaneous trip settings of 1462 K and 1517.5 K respectively.

4.7.3 The Extended Analysis Program for Experiment LP–FP–2. The OECD LOFT Agreement envisioned completion of the program by the end of 1986. The remaining time after the conclusion of the experimental program would be taken up by further comparative analyses of the thermal–hydraulic experiments, the conclusion of the work on LP–FP–1, and the reporting and analysis of LP–FP–2. The latter would be confined to calculations of the thermal–hydraulics up to the time of significant loss of fuel element geometry from fuel relocation and to the presentation of the data from the fission product measurement system.

At an early stage after the experiment, although the performance of the instrumentation still had to be fully assessed, there was substantial confidence that its stated objectives, the study of the propagation and the retention of fission products derived from both the fuel–pellet/fuel–clad gap and fuel overheating in the transient in the presence of a large aerosol source from the control rods, would be met and good data obtained. It was also clear that, in addition to its original objectives, the experiment would provide substantial information of value for severe core damage studies. The major features of the experiment considered important in this context included the following:

1. The bundle was large enough for the course of the temperature excursion, the temperature attained, and the final quenching behavior to be relevant to conditions in a PWR core.

2. At the temperatures reached, there were important material relocation effects arising from the Zr/steam reaction and from fuel melting.

3. Both the zirconium oxide content and hydrogen levels would be measured after the experiment.

4. Direct chemical analysis could provide evidence on fission product retention.

However, to take advantage of these features of the experiment, a wider analysis program would be needed. The original proposal was that it would have the following four components:

1. Gamma scanning and neutron radiography of the center fuel bundle.

2. An extended program of neutron radiography to support analysis by tomography.

3. Sectioning the bundle after filling with epoxy resin to produce twenty 2.5-cm thick sections (after polishing) for a visual record of the final state of the center fuel module.

4. Subsequent metallographic, chemical, and other analyses of these sections to identify material compositions.

This would provide information on the final distribution of fuel, fuel clad, and control rod materials and their chemical and metallurgical form, some evidence on the maximum temperatures reached as a function of position in the fuel bundle, and data on fission product distribution in nonfuel material. It was envisioned that such a program would last for about three years, i.e., until the end of 1989 at a cost of an additional $3.85M.

The original proposal was to assist the funding of the extended program by offering participation to countries outside the existing OECD LOFT Consortium. Initial approaches were made to France, Belgium, the Netherlands, Norway, Canada, and Denmark. As a result, there was a presentation to representatives of Belgium, Denmark, and France in September 1985. Following this, a proposal for French participation was received, based as follows:

1. France would receive the results of the LP-FP-1 and LP-FP-2 experiments and would participate fully in the Extended Analysis program for LP-FP-2.

2. France offered the following support:

    a. Access to the data from two PHEBUS Phase III experiments.

    b. To carry out the destructive examination of the LP-FP-2 center fuel bundle at Saclay, France.

    c. A balancing contribution in cash.

After further negotiations, the Board decided at its November 1985 meeting that it was not able to pursue this offer because of the following reasons:

1. The proposal to provide access to PHEBUS data did not fit in with the primary objectives of the LOFT Consortium agreement.

2. The additional cost of shipping material to France and of achieving satisfactory liaison with the general program of work on LP-FP-2 left only a marginal cost incentive and was likely to lead to program delays.

3. The Consortium had operated to date on the basis that work carried out by signatories, e.g., Option 5 and the provision of attached staff, would not be offset against financial contributions.

This left no alternative but to seek additional funding from the existing membership. The problem of limited funds was assisted by an offer from the USDOE to take over full responsibility for cleanup and Project termination (including the increased commitment consequent on the early decision to increase the severity of the LP-FP-2 transient) against the nominal sum of $10.78M set down in the Project accounts. Members also accepted that the US contribution to the extended program would be set at 33% of the total. Although the Extension Agreement was not signed by all parties until July 1987, early indications of support were sufficient to ensure a smooth transition and no holdup to the work program.

In fact, work on the center fuel bundle was not able to start until the autumn of 1987 because of the need for a cooldown period. The PRG used this time to review the program and to take a position on the relative merits and cost effectiveness of neutron radiography, neutron tomography and sectioning plus the associated metallurgical and chemical analysis. In particular, the proposal to use tomography was novel and outside the experience of most members. EG&G was able to present work that they had carried out as part of the US Severe Fuel Damage program and results from the tomographic examination of a mockup of a TMI-2 fuel bundle. This was valuable evidence that the technique would work for a large fuel pin array. It was clearly beneficial that these techniques would be developed in parallel by EG&G for other projects. Although this involved an exercise of judgment, the PRG was able to reach the following conclusions:

1. Tomography appeared to be capable of providing detailed qualitative evidence of the TMI-2 fuel bundle internal geometry.

2. There was a reasonable expectation that this could provide quantitative geometrical data, e.g., fuel blockage areas could be used to interpolate the material geometry between the cut sections.

3. There was some evidence that quantitative density measurements could be used to a limited degree to identify materials. These techniques did not in fact develop as fast as had been expected.

As a result, the PRG concluded in February 1987 that all three techniques–radiography, tomography, and sectioning–should be used and would be cost effective. The investigation did not go precisely as planned, but the objectives of the program were fully achieved. The final position may be summarized as follows:

1. In the summer of 1987, EG&G was able, for the first time, to take a neutron radiograph of a severe fuel damage experimental assembly containing Ag–In–Cd control rods. They found that indium was widely dispersed through the fuel element, and because this strongly attenuated the neutron beam, the radiograph lacked definition in important areas of the fuel bundle, therefore, making a satisfactory tomographic reconstruction impossible. They concluded that a reasonable extrapolation from this evidence found this technique could not be used effectively for LP–FP–2. It was therefore agreed to take two neutron radiographs at 90° and to abandon tomography.

2. The PRG was able to work very closely with EG&G in defining the sectioning program.

3. The techniques used for sectioning were very successful. Photographs of the sections produced before polishing could be used to obtain a good general impression.

5. OECD LOFT EXPERIMENTS AND THEIR ANALYSES

As discussed in Chapter 4, the experimental program of the OECD LOFT Project comprised eight experiments, six thermal–hydraulic experiments, and two fission product release experiments as shown in the following table:

Table 5.1 OECD LOFT experiment program

Experiment Identification	Date Conducted	Description
LP–FW–1	02/20/1983	Loss–of–feedwater, primary feed and bleed recovery procedure
LP–SB–1	06/23/1983	Hot leg SB LOCA, early pump trip
LP–SB–2	07/14/1983	Hot leg SB LOCA, delayed pump trip
LP–SB–3	03/05/1984	Cold leg SB LOCA, core uncovery, secondary feed and bleed recovery procedure, accumulator injection at low–pressure differential
LP–02–6	10/03/1983	200% large–break LOCA, US licensing case
LP–LB–1	02/03/1984	200% large–break LOCA UK, licensing case
LP–FP–1	12/19/1984	Gap fission product release, large–break LOCA, German licensing case
LP–FP–2	07/03/1985	Fission product release at high fuel temperatures (above 2100 K), V–sequence

This chapter provides a description of the experiments, including objectives, experimental findings, and analyses. The discussion of the thermal–hydraulic experiments is provided in three sections: for the loss–of–feedwater experiment, for the small–break LOCA experiments, and for the large–break LOCA experiments. Because of the special character of the fission product experiments, the two experiments are discussed separately.

Because one of the most important objectives of the OECD LOFT experimental program was to provide data for assessment of computer codes used to predict accident behavior of nuclear power plants, this chapter also includes summaries on performance of those codes used in analyses of the experiments. In assessment of code performance, pre–experiment prediction calculations are especially important because they provide evidence of the current ability of computer codes to predict accident sequences in nuclear power plants. Each of the sections on code performance, therefore, contains a short discussion of the predictions followed by a summary of the postexperiment analyses. The postexperiment calculations included some adjustment of initial and boundary conditions on the basis of experiment results, and were performed to evaluate code performance and to support the understanding of the physical phenomena that occurred during the experiments. Since many factors can influence the quality of code simulation, the discussions of code performance in postexperiment analyses has purposely been kept fairly general and emphasizes only major trends. Detailed information on models and associated code performance can be found in the referenced literature.

5.1 Loss–of–Feedwater Experiment LP–FW–1

5.1.1 Experiment Objectives and Description. The first OECD LOFT experiment was conducted on February 20, 1983. This experiment, designated LP–FW–1, was designed to evaluate the generic PWR system response during a complete loss–of–feedwater transient.[12] The unrestored loss–of–feedwater places a PWR system in a situation where the plant safety is dependent on use of the pressurizer Power Operated Relief Valve (PORV) for pressure reduction and energy removal, and on emergency core coolant (ECC) injection to maintain coolant inventory adequate for core cooling. The objective of the experiment was to investigate the performance of primary "feed and bleed," using a "bleed" from the PORV and "feed" from the HPIS to provide decay heat removal and system pressure reduction while maintaining the primary coolant inventory. Further, the experiment had to provide information on transient characteristics and data for the evaluation of the capabilities of computer system codes to predict system response during a loss–of–feedwater transient. These issues and the significant phenomena of this experiment are discussed in the next section. Ability of the computer system codes to address the issues of Experiment LP–FW–1 is summarized in Section 5.1.3.

The experiment was initiated with typical conditions for operation of a commercial PWR by tripping the feedwater pump.[12,13] The auxiliary feedwater system was not activated in this experiment. The degradation of the primary–to–secondary heat transfer consequent to the pump trip caused an increase in primary system pressure and temperature. Negative reactivity feedback, as result of this increase in coolant temperature, then produced a decrease in reactor power. The primary system pressure rise activated pressurizer sprays, but when the pressure continued to increase, the reactor scrammed automatically on high pressure signal. The same signal caused the PORV to open and the main steam control valve to close. At that time, the steam generator secondary coolant inventory was depleted to the bottom of the indicating range. The open PORV then slowly reduced the primary pressure. At 8.72 MPa, the primary coolant pumps were tripped and the HPIS injection initiated because of the coolant loss void formed in the primary system outside the pressurizer. The open PORV, however, resulted in liquid level rise in the pressurizer above the top of the indicating range. The decay heat was dissipated through the PORV and by steam generator primary to secondary heat transfer. Natural circulation then developed and was aided by the HPIS PORV through–flow. The experiment was terminated about two hours after initiation.

5.1.2 Experimental Findings.

This section summarizes the findings on the safety issues addressed with Experiment LP–FW–1 i.e., the effectiveness of primary feed and bleed, the energy transfer from primary to secondary, the possibility of pressurized thermal shock, and methods for transient identification.

Primary Feed and Bleed. The experiment was initiated with conditions typical for the operaton of a commercial PWR by tripping the secondary system main feedwater pump which resulted in degradation of primary–to–secondary heat transfer, and therefore increased primary pressure and temperature. The high primary system pressure caused a reactor scram which was automatically followed by closure of the main steam control valve.[13,14] Immediately after the scram, the operator opened the PORV. After the subcooled water was expelled from the PORV line, the flow through the PORV changed to low density flow. The void formation in the primary system resulted in liquid insurge into the pressurizer so that the liquid level reached the top of the pressurizer. The mass flow rate through the PORV increased and slug flow with a frequency of approximately 1Hz was measured. For the rest of the transient, the pressurizer level varied and so did the slug frequency, decreasing as the pressurizer level reduced.

The coolant loss through the PORV was compensated by a primary feed from the HPIS. The primary system remained full of subcooled liquid except for steam domes in the upper plenum above the intact loop nozzles and in the pressurizer and for steam bubbles due to partial subcooled nucleate boiling in the core. Through the entire transient, about 150 kg more coolant was expelled than injected. The PORV and HPIS flow rates were power scaled to a typical commercial PWR (Calvert Cliffs).[12] It was assumed that only one of the reference HPIS trains was available.

Energy Transfer. The primary energy sources during Experiment LP–FW–1 were core decay power, pump power (at the beginning of the transient), and the HPIS flow enthalpy. The energy removal mechanism was mainly via the PORV flow.

The decay energy was removed from the core by circulation established between the inflow of the HPIS through the core and the outflow through the PORV. This circulation was supported by buoyancy forces provided by the core decay heat. Detailed investigations[14] showed that up to about 25% of decay heat was removed through primary–to–secondary steam generator heat transfer. This heat transfer was primarily maintained by boiling liquid on the tube sheet of the steam generator. After steam was condensed on the inside walls of the steam generator, the condensate was evaporated on the tube sheet and then superheated by primary–to–secondary heat transfer across the dry steam generator tubes. Leakage in the main steam control valve was a further source of energy removal from the secondary side of the steam generator. The heat transfer through the steam generator provided some natural circulation, though the PORV heat sink was much larger than that of the steam generator. Some of the energy in the primary system was also removed through the heat loss from primary components to ambient. In summary, all the heat transfer mechanisms were effective in removing the decay heat and reducing the primary system pressure.[14]

Possibility of Pressurized Thermal Shock. A review of fluid temperature measurements indicated no temperature stratification in the cold leg of the operating loop. Temperature gradients measured in the downcomer were also small. It was therefore concluded[14] that the circulation established during this experiment provided sufficient mixing of the ECCS flow with the primary coolant to reduce the possibility of pressurized thermal shock.

Transient Identification. One accident management concern is the correct and timely identification of the accident transient. It is often difficult to distinguish between various transient scenarios. For example, the beginning of an undercooling transient might be indicated by the standard instrumentation of primary system pressure and by the pressurizer liquid level measurements in the same way as the beginning of a small-break LOCA would. Further data from a display based on Reactor Coolant Pump (RCP) shaft power, however, may provide a way of distinguishing between these two possibilities. In fact, data from LP-FW-1 showed that pump power alone could be used to identify the transient.[14] The possibility of using pump power for transient identification was also investigated in OECD LOFT small-break experiments, and will be discussed later.

5.1.3. Performance of Computer Codes in the Analyses of the LP-FW-1 Experiment. The RELAP5/MOD1 code was used for the prediction of the LP-FW-1 Experiment.[13,15] Results of these calculations are considered invalid because of modeling error. Additional code analyses also using RELAP5/MOD1 were performed by the INEL team for the EASR.[14] Several project participants performed independent code calculations as shown in Table 5.2.

The results of these calculations and those of the EASR were compared in the Comparative Analysis Report.[16] In general, the calculations were well able to simulate the experiment, particularly the early phases. Figure 5.1 shows the calculational envelope for the primary system pressure and the experiment data. The overall error band of the calculations is very small—less than 2% for the short-term (up to about 60 s) and less than 20% for the long-term of the transient. The individual code perfomance is summarized as follows:

RELAP5/MOD1

The short-term phenomena, including primary system pressure and temperature rise, reactivity feedback, pressurizer response, and secondary system depletion were, in general, well calculated by this code.[16] Simulation of phenomena such as level swell in the pressurizer, secondary side degradation, and density reactivity feedback, however, need some improvement. For the long-term phase of the experiment, problems associated with simulation of primary-to-secondary heat transfer were noted. There was also an overprediction of coolant losses through the PORV. A number of suggestions to improve the code capabilities were made which were later considered in the development of newer versions of the code (RELAP5/MOD2 and MOD3). Interesting sensitivity studies with a very simple model of LOFT were performed in Finland. These sensitivity studies indicated that the atypical behavior of the LOFT steam generator (main steam control valve leakage and resulting primary-to-secondary heat transfer) did not have a major influence on the outcome of this experiment.

Table 5.2 LP-FW-1 Project Members Analyses

Country	Organization	Code
Finland	VTT[a]	RELAP5/ MOD1/CY19
German	GRS	DRUFAN-02
Italy	ENEA	RELAP5/ MOD1/CY19
Japan	JAERI	RELAP5/ MOD1/CY18
Great Britain	UKAEA	RETRAN-02 MOD2

a. Short-term (100 s) calculations.

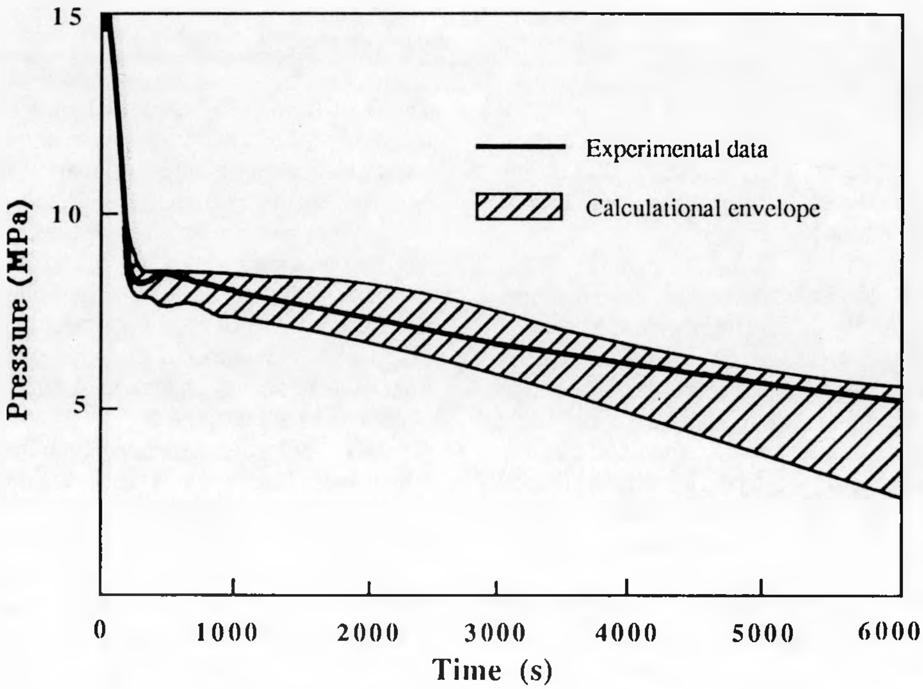

Figure 5.1. Primary system pressure, LP–FW–1.

DRUFAN–02

Code to data comparisons [14,16] indicated that this code performed well in the early phases of the experiment. The long–term calculations were less satisfactory, with areas identified for improvement such as: two–phase flow through the PORV, phase separation in the pressurizer, and primary–to–secondary heat transfer.

RETRAN–02

RETRAN reproduced the short–term part of the transient effectively.[16] The analyses showed that it is mandatory to calculate accurately the sequence of events at the end of the initial transient, otherwise the calculations move progressively further from the physical reality. Further studies with this code indicated that reasonable simulation of the experiment can be obtained by explicit modeling of the pressurizer surge line and relief line, and by adjusting parameters such as steam leakage, PORV area, and HPIS flow rate.

5.2 Small–Break Experiments

5.2.1 Experiment Objectives and Description. A small–break LOCA is usually defined as any break in the PWR pressure boundary that has an area of 0.046 m^2 or less (where a typical cross–sectional area of a primary system main coolant piping is about 0.64 m^2). This range of break areas also encompasses all small lines that penetrate the reactor primary coolant pressure boundary, including relief and safety valves, charging and letdown lines, drain lines, and various instrumentation lines. The major concern during a small–break LOCA is the loss of primary coolant inventory and the ability of the protection systems to detect this in time and restore the water inventory. The physical phenomena during a small–break LOCA and the magnitude and timing of these phenomena, however, vary as a function of break size, break location, and plant design parameters. The long time scales in such an accident also mean that operator intervention can be an important factor in aggravating or ameliorating the accident consequences. The small–break experiments of the OECD LOFT program addressed in a generic way several of the most important small–break LOCA phenomena and issues which aid in the understanding of PWR system behavior under small-break LOCA conditions, provide data for assessment and evaluation of computer system codes, and provide data on different plant recovery procedures.

The following paragraphs describe the experiments, their objectives, and general results. Section 5.2.2 provides a summary of the issues and phenomena addressed in these experiments which include: the effects of RCP operation, the primary coolant inventory and ECCS effectiveness, plant recovery using secondary feed and bleed, and the effectiveness of natural circulation heat removal, pressurized thermal shock, accident diagnostic techniques, and core uncovery and dryout. The ability of currently available computer system codes to address the issues of concern and to simulate important phenomena is summarized separately in Section 5.2.3.

Experiment LP–SB–1 and LP–SB–2. The first two small-break experiments of the OECD LOFT Project, Experiments LP–SB–1 and LP–SB–2, simulated a 3–in. (7.62 cm) equivalent break diameter located in the hot leg of the operating loop. The major objective of these experiments was to determine system transient characteristics for small hot leg break loss–of–coolant accidents with early and delayed pump trip, and to provide integral nuclear system data for assessing the ability of computer codes to predict system response during a small–break LOCA.[17] In Experiment LP–SB–1, the RCPs were tripped at the beginning of the transient and, in Experiment LP–SB–2, they were left running until a substantial amount of coolant was depleted from the system. Both experiments were conducted from initial temperature and pressure conditions representative of typical commercial PWRs.

The timing of the trip during a small–break transient has a substantial influence on the primary coolant inventory, and therefore on the possibility of core uncovery and fuel cladding heatup. The USNRC cold leg break experiments have shown that continued pump operation during the transient results in larger coolant losses than for early pump trip and can result in core uncovery and rapid core heatup.[18] Some vendor analyses indicated that, for breaks located in the hot legs, core uncovery can still occur, despite continuous pump operation.[17,19] These general uncertainties, the known inadequacies in computer codes for prediction of these transients, and lack of integral experimental data on hot leg breaks[19] were some of the incentives for OECD LOFT Experiments LP–SB–1 and LP–SB–2.

In Experiment LP–SB–1, the pump trip shortly after transient initiation (24 s) resulted in a smooth transition from forced loop flow into natural circulation. The steam generated in the primary system due to coolant loss through the break resulted in stratified flow formation in the hot leg of the operating loop. The break inlet, at the midplane of the hot leg piping, was covered with saturated liquid until about 715 s into the transient. After the break inlet uncovered, the coolant depletion was significantly reduced and eventually the HPIS injection rate exceeded the break mass flow rate and the primary system started to refill. The minimum coolant inventory reached in this experiment resulted in a liquid level in the reactor vessel just above the bottom of the vessel nozzles and well above the top of the core.

Experiment LP–SB–2 was conducted in the same manner as the Experiment LP–SB–1, with the exception that the RCPs were left running for almost 50 minutes. Initially, continued pump operation maintained fairly homogeneous distribution of the generated vapor through the primary system. However, at about 200 s into the transient, a vertical density gradient began to form in the hot leg of the operating loop. At about 600 s, this eventually changed into stratified flow with liquid level above the break inlet. The break uncovered at about 1200 s and the coolant loss rate was reduced significantly.

Experiments LP–SB–1 and LP–SB–2 showed that the timing of the pump trip during a hot leg small–break (SB) transient has only a small effect on the minimum coolant inventory and does not significantly effect the risk of core uncovery. The formation of stratified flow in the hot leg makes both transients similar. Figure 5.2 shows the coolant inventory for both experiments. The conclusion from these experiments, combined with those of the USNRC experimental program, is that primary pump operation has much less effect on fluid distribution in the primary system for a break in the hot leg, than for a break in the cold leg .[19,20,21] Experiment predictions performed with RELAP5 indicated that the inadequacy of the flow regime map used in the code[19,21,22] was the main reason for deviation of the predictions from the experimental data. Flow stratification, which occurred in the hot leg early in Experiment LP–SB–2, was not reached in the calculations until the pump trip, and resulted in overpredicted coolant losses and core uncovery at the pump trip (see Section 5.2.3).

Experiment LP–SB–3. The last LOFT small-break experiment (LP–SB–3), representing a 1.8–in cold leg break LOCA with no HPIS available, was designed mainly for investigation of plant recovery effectiveness using secondary feed and bleed during core uncovery and addressed accumulator injection at low pressure differentials. This experiment was proposed by Italy. Conduct guidelines and details of this experiment [23,24] were then developed in cooperation between the Project and Italian research organizations.

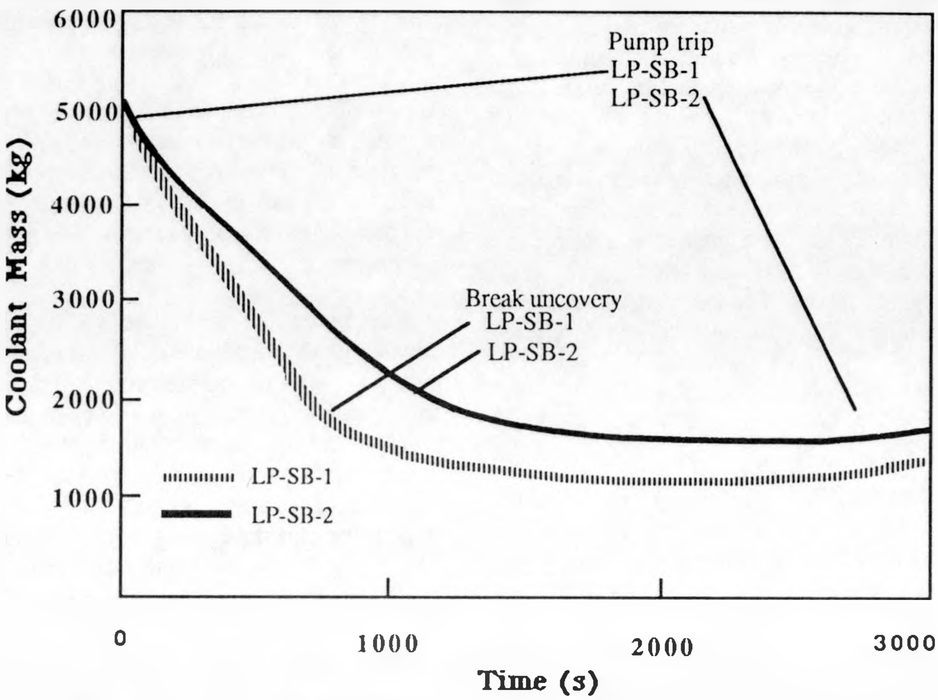

Figure 5.2. Primary coolant inventory during experiments LP–SB–1 and LP–SB–2.

The experiment was initiated from conditions representative of those in a commercial PWR by opening a valve in the intact loop cold leg break piping. Figure 5.3 shows the primary system pressure and the fuel cladding temperatures, and indicates the major events during this experiment. The primary system depressurized rapidly until fluid saturation conditions were reached in the hot leg at about 100 s, which resulted in a decrease in the primary system depressurization rate. Because a void then formed in the coolant, there was a consequent reduction in the break mass flow rate. The continued pump operation homogenized the fluid in the primary loop and the system void fraction increased steadily as fluid was discharged through the break. The pumps were tripped when the break flow instrumentation indicated that approximately 2000 kg of coolant remained in the primary system. The primary system coolant inventory continued to decrease until at 4200 s the core began to uncover. The break was isolated to stop the primary system depressurization when the cladding temperature reached 811 K. At the indicated temperature of 977 K, the steam generator feed and bleed procedure was initiated which resulted in primary system depressurization and cladding cooling. Accumulator injection commenced when the primary system depressurized to 2.73 MPa, and this then quenched the fuel cladding effectively from the core bottom upward.

The experiment showed that operator initiated secondary feed and bleed in a small–break transient in which HPIS flow is not available is effective in primary pressure reduction. Despite low pressure differential, the accumulators were able to provide enough water to the system to quench the core effectively. The RELAP5 predictions indicated some deficiencies in the modeling of liquid entrainment from the lower plenum, affecting core heatup and mechanisms of system recovery via steam generator bleeding. The code also calculated significant oscillation in the delivered accumulator flow, and this was not observed in the experiment.

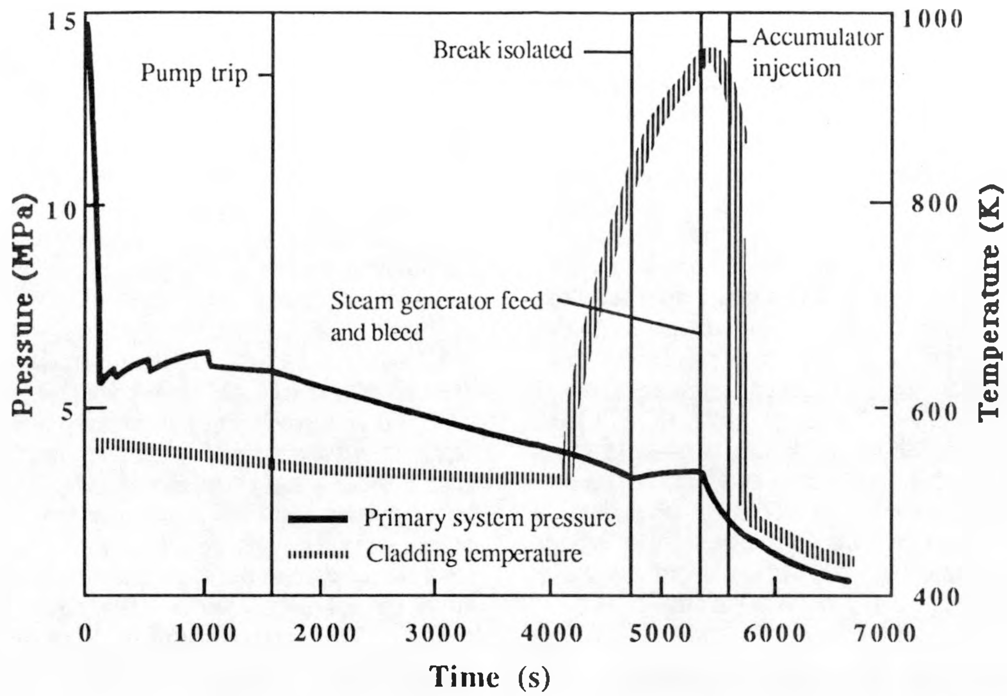

Figure 5.3. Primary system pressure and fuel cladding temperature during experiment LP–SB–3.

5.2.2 Experimental Findings

Effects of RCP Operation. The TMI–2 accident showed that operation of the reactor coolant pumps could have an important effect on system behavior and coolant inventory during the course of a small–break LOCA accident. The USNRC LOFT Experiments L3–5 and L3–6 were designed to investigate this problem for breaks located in the cold leg. They showed that continuation of the reactor coolant pump operation during the transient results in larger coolant losses than if the pumps are tripped early (for example, because of loss of offsite power or inadvertent operator action) and can result in core uncovery and rapid core heatup.[18]

The first two small–break experiments of the OECD project, LP–SB–1 and LP–SB–2, simulated a 3–in. (0.00456 m^2; 0.7%) break located in the hot leg of the operating loop.[17,20,21] These experiments can be considered as complementary to the cold leg break experiments conducted during the USNRC program. The OECD LOFT experiments showed that the timing of the pump trip during a hot leg small–break transient does not have an important effect on the minimum coolant inventory, nor does it significantly change the likelihood of core uncovery.[19] During the experiment with operating pumps, a flow stratification in the hot leg leading to a break uncovery was observed, and this had the effect of making the loss–of–coolant inventory similar to that for early pump trip. These experiments showed that in LOFT, the timing of the reactor coolant pump operation has much less effect on fluid distribution in the primary system for breaks in the hot leg than for breaks in the cold leg.[19]

Experiment predictions performed with RELAP5 indicated an important inadequacy of the flow regime map used in the code. Flow stratification which occurred in the hot leg early in LP–SB–2 was not reached in the calculations until the pump trip, resulting in overpredicted coolant losses and core uncovery at the pump trip.[19,21,25,26]

Experiments LP–SB–2 and LP–SB–3 confirmed findings of the Experiment L3–6 that the overall performance of the RCP degrades with the increasing void fraction at the pump inlet with rapid drop of performance at an inlet voidage of about 40%.[19,27] The rapid degradation of the RCP performance enhanced phase separation in the hot leg, but had less effect on the flow in the cold leg which remained relatively homogeneous until pump trip.

Primary Coolant Inventory and ECCS Effectiveness. Figure 5.1 showed the behavior of the primary coolant inventory during Experiments LP–SB–1 and LP–SB–2. There were obvious similarities in the coolant depletion in both experiments. The mass inventory decreased more quickly with early pump trip, but after break uncovery (defined as transition from single phase liquid or two–phase flow at the break to predominantly vapor flow), the primary coolant inventory decreased at approximately the same rate in both experiments. There was an initial larger coolant loss during LP–SB–1 because the break line inlet was still covered with saturated liquid after the flow stratified in the hot leg. In LP–SB–2, continued pump operation over the same time period produced a two–phase mixture at the break inlet until stratification and break uncovery, which occurred later than in LP–SB–1. As a consequence, the coolant losses were smaller for LP–SB–2 than for LP–SB–1. The minimum mass inventory was reached in both experiments at about the same time, with most of the coolant in both experiments being in the reactor vessel with the liquid level above the bottom of the loop nozzles.

For both experiments, the ECCS flow was scaled to simulate degraded ECCS injection in a commercial power plant. This scaling was based on the assumption that only one of the three charging pumps and one of the three HPIS pumps in the reference plant were available. Accumulators and LPIS were not used in either experiments. It is evident from Figure 5.2 that despite degraded ECCS injection, the high pressure injection systems were capable of arresting the coolant losses and ensuring a completely covered and cooled core.

Accumulator and LPIS effectiveness were separately investigated in Experiment LP–SB–3. If no HPIS is available, it is possible that for a range of break sizes for which the depressurization rate is small, the accumulators may be required to operate when the difference between accumulator pressure and that of the primary system is very low. Under such conditions, there can be doubts about the effectiveness of ECCS injection. LP–SB–3 results showed that despite very low pressure differential between the accumulator and the primary system, the injection was effective in providing radially uniform quenching on the core at a rate of 0.67 cm/s.[28,29]

Plant Recovery Effectiveness Using Secondary Feed and Bleed. It is conceivable that during certain small–break transients when the depressurization rate is small the core may become uncovered while the primary system pressure remains above accumulator injection pressure. In this situation, an operator–initiated secondary feed and bleed procedure can be used as an accident mitigation method to reduce primary system pressure and restore core cooling. Experiment LP–SB–3 addressed this procedure and showed that feeding and bleeding in the secondary side of the steam generator is able to reduce primary system pressure at a rapid rate of 8.8 KPa/s.[28] The condensation on the primary side of the steam generator, induced by a feed and bleed procedure, caused an increase of steam velocity in the core and liquid entrainment from the lower plenum and consequently initiated cooling of the fuel cladding.

Effectiveness of Natural Circulation Heat Removal. Under small–break accident conditions, energy is removed from the primary system via the break, steam generator, and ambient heat losses. For certain break sizes, however, the decay heat can only be removed effectively via the steam generator. If the reactor coolant pumps are tripped early, the heat transfer from the core to the steam generator can occur by either natural circulation or reflux cooling. Experiment LP–SB–1 addressed this problem, despite difficulties in instrumenting a facility like LOFT for measurement of very small flow rates and possible counter–current flow conditions. It was shown[19] that natural circulation was established in the primary loop after RCP coastdown and lasted for about 450 s, mostly under two–phase conditions. During that time, the energy was removed via the steam generator at about the same rate as the heat added to the primary coolant from the core. The steam generator acted as heat sink for about 200 s longer, possibly in a reflux mode. However, the primary mechanism for the energy removal from the primary system was the break flow.

The secondary bleed and feed during Experiment LP–SB–3 showed that as soon as the steam generator can be restored to effectiveness as a heat sink, even in a highly voided system, sufficient flow can be induced between reactor vessel and steam generator to ensure effective removal of the decay heat with a consequent reduction of the primary system pressure and also to provide some cooling to the fuel cladding .[28,29]

Pressurized Thermal Shock. Small hot leg breaks may create a potential for pressurized thermal shock for some operating nuclear power plants. Data from Experiment LP–SB–1 were examined to assess the degree of mixing of the cold high pressure injection water with the coolant flow in the cold leg when the reactor coolant pumps were tripped.[19] The results of this assessment indicated good mixing of the cold high pressure injection liquid with the cold leg flow, thus minimizing the potential for pressurized thermal shock occurring during the periods of natural circulation. During stagnant loop flow conditions, however, the mixing effect at the injection point was relatively small.

Accident Diagnostic Techniques. The role of the reactor operator in the accident management depends on the receipt of information which will enable him to control the plant safely. This information must be quickly and correctly interpreted. Experiments LP–SB–1 and LP–SB–2 provided valuable data to show the efficiency and the limitations of using pressure–temperature displays or subcooling displays as an accident diagnosis tool.[19] The LOFT results showed that these displays must be backed up by displays of other key parameters that are sensitive to differences between accident sequences. For example, monitoring pump power during upset plant conditions could be useful in reducing the reactor operator's "blind spot" between the loss of subcooling and the onset of data from, say, a reactor vessel liquid level detector. This would require operation of the RCPs for some period during the accident. The running pumps would also permit the use of pressurizer sprays to reduce system pressure and would also limit the number of relief valve operations. Running RCPs would also reduce the potential for void formation in the reactor vessel head and for pressurized thermal shock.

Core Dryout. High pressure injection was not used in the Experiment LP–SB–3, since the objective was to let the core slowly uncover and heatup.[30] The core heatup progressed from top to bottom. The estimated boil–off rate was 1.8 mm/s in the upper part of the core, and about 1.1 mm/s in the lower part of the core (about 0.57 kg/s and 0.35 kg/s respectively). The change in the boil–off rate is associated with the core uncovery, and therefore the amount of energy transferred to the fluid. The core heatup was almost uniform radially. Lower temperatures were measured in the upper part of the fuel module, which was closest to the hot leg of the operating loop, than in the other modules. Investigations[29] showed that the condensation in the steam generator and the reflux flow back to the reactor vessel were responsible for this cooling.

5.2.3 The Performance of Computer Codes in the Analyses of Small–Break Experiments.

This section provides an overview of the reported calculations, for each of the experiments. Pre–experiment analyses for all the small–break experiments were performed by the Project using the RELAP5/MOD1 code. This code was also used for the analyses of experiments LP–SB–1 and LP–SB–2 in the framework of the EASR. For the EASR analyses of Experiment LP–SB–3, RELAP5/MOD2 was applied. Several participants performed additional code analyses with different codes which are reported in the Comparison Reports and some individual reports.

Experiment LP–SB–1. For planning and predictions of Experiment LP–SB–1, the RELAP5/MOD1 CY18 code was used. Results of the predictions compared to the experiment results, are discussed in detail in References 19 and 21. The primary system pressure and temperature behavior were calculated correctly by the code, and all major experiment phenomena were predicted to occur in the correct order. The major difference between the predictions and the experiment is related to underprediction of the saturated break flow, which resulted in calculated late break uncovery and in discrepancies in coolant inventory. The calculated minimum coolant inventory was about twice the measured value. The underprediction of the saturated break flow is directly attributable to the branch model used in RELAP5/MOD1. Despite flow stratification in the hot leg of the operating loop, the code used average fluid density for break density in the hot leg, which was in fact lower than the density of the liquid covering the break pipe inlet. The postexperiment EASR calculations, performed with the same code but using the measured boundary conditions with some nodalization and modeling changes, showed that only marginally better results could be obtained without significant improvement of the stratified flow model.

Several postexperiment code analyses of the LP–SB–1 experiment were carried out by Project members. Most of these analyses are summarized and compared in the Comparative Analyses Report,[22] prepared by JAERI of Japan. The members analyses are summarized in the following table:

Table 5.3. LP–SB–1 Project Members Analyses

Country	Organization	Code
Japan	JAERI	RELAP5/MOD1/CY18
Finland	VTT	SMABRE
F.R.G.	GRS	DRUFAN 02
U.K.	CEGB	RELAP5/MOD1 CY19
U.K.[a]	CEGB	RELAP5/MOD2
U.K.[a]	AEE Winfrith	TRAC–PF1/MOD1

a. These calculations are not included in the comparative analysis report.

Calculations indicated in Table 5.3 were also compared with the predictions. Additionally, although the calculations using RELAP5/MOD2 and TRAC–PFI/MOD1 noted in this table were carried out too late to be included in the Comparison Report, separate documents setting out these calculations were distributed among the Project members.[26,31]

The comparisons of the calculations with the experimental results showed that, like the pre–prediction calculations, codes used in the postexperimental work were able to simulate major thermal–hydraulic parameters such as system pressure and fluid temperatures. Figure 5.4 for example, shows the envelope of the primary system pressure calculations and the measured values. However, the codes were not quite able to simulate the multidimensional two–phase flowand had problems in modelling the flow stratification in the hot leg of the operating loop. As result, the primary system coolant inventory was usually overpredicted (Figure 5.5). Based on these analyses, the following summary statements on code performance in simulation of the LP–SB–1 experiment can be made:

RELAP5/MOD1

The analyses showed need for improved stratification criteria and for improved modeling of liquid/vapor entrainment upstream of a break. Lessons learned in application of this code were used to implement improvements when developing its replacement, RELAP5/MOD2.

RELAP5/MOD2

Performed analyses indicated significant improvement over the results obtained with the older code RELAP5/MOD1. The new horizontal stratification model implemented in the RELAP5/MOD2 code improved the density of the fluid fed from the hot leg into the break line, and therefore improved the calculated break mass flow rates and primary system coolant inventory. There were still some problems with the critical discharge model, which does not take into account non–equilibrium effects in the discharge nozzle.

DRUFAN 02

Because this code lacks a specific stratified flow model, calculations were performed[22,32] in which three parallel vertically–stacked volumes were used to mimic the stratification in the hot leg. These calculations showed that if stratification can be modeled correctly, significant improvements in calculation of the break flow and consequently the coolant inventory can be achieved.

SMABRE

This small, fast running code was able to calculate the major parameters as well as the more complex codes.

TRAC–PF1/MOD

Results obtained with this code were generally consistent with the other calculations presented in the Comparison Report. The major problem was the deficiency of the stratified flow model. Implementation of an EPRI correlation, for determining side branch quality as a function of main pipe stratified liquid level, was effective in improving the calculated break mass flow rate early in the transient. Further analysis has shown that inclusion of the CATHARE correlation for predicting the level at which vapor pullthrough occurs further improves the calculations.

The main lesson from these calculations was that accurate simulation of small–break transients with breaks located in the hot leg of the main coolant piping requires models which accurately predict the transition into the stratified flow regime and predict the relation of fluid conditions in the break to those in the hot leg near the break.

Experiment LP–SB–2. Because both Experiments LP–SB–1 and LP–SB–2 addressed the same issue, the predictions for both experiments were carried out with the same version of the RELAP5/MOD1 code. Comparison of the predictions with experimental data[19,21] showed that parameters such as primary system pressure and fluid temperatures were again relatively well predicted. The code calculated that the continuous pump operation would also provide both a continuous loop circulation, and a homogeneous two–phase mixture at the break location, and consequently lead to relatively large coolant losses. As a result of this, it was predicted that the late pump trip would result in collapse of the two–phase mixture in the reactor vessel and consequent core uncovery and fuel cladding heatup. The code completely failed to predict the flow stratification and break uncovery in the hot leg of the operating loop, which occurred in the experiment and which prevented the large coolant losses and core uncovery.

The following table shows the reported postexperiment analyses of the LP–SB–2 experiment performed by the Project members. The first four calculations are reported in the Comparative Analysis Report, prepared by JAERI; the sixth calculation is reported in Reference 22.

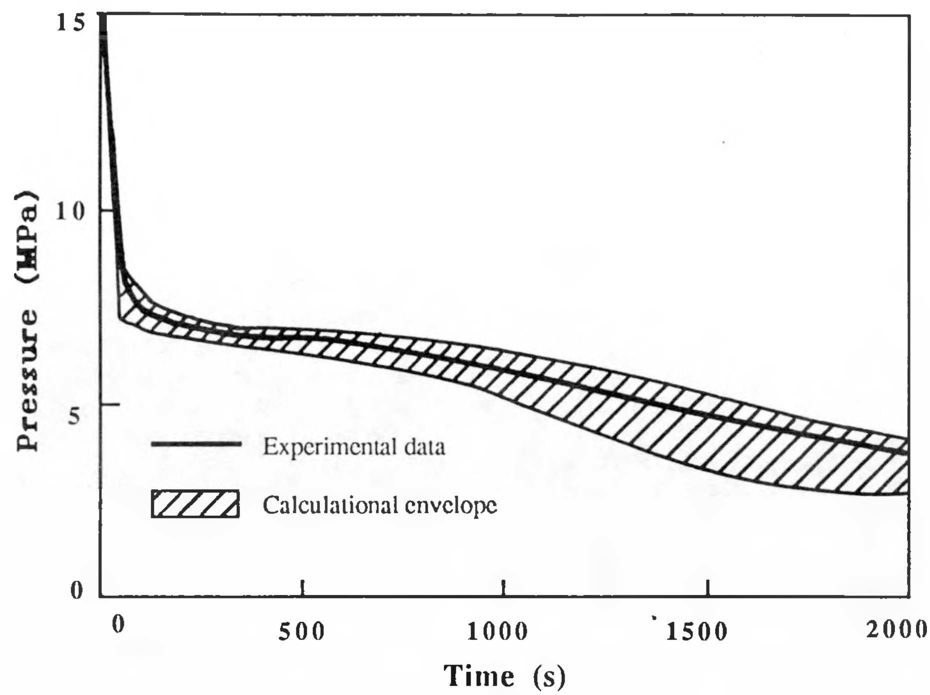

Figure 5.4. Primary system pressure during experiment LP–SB–1.

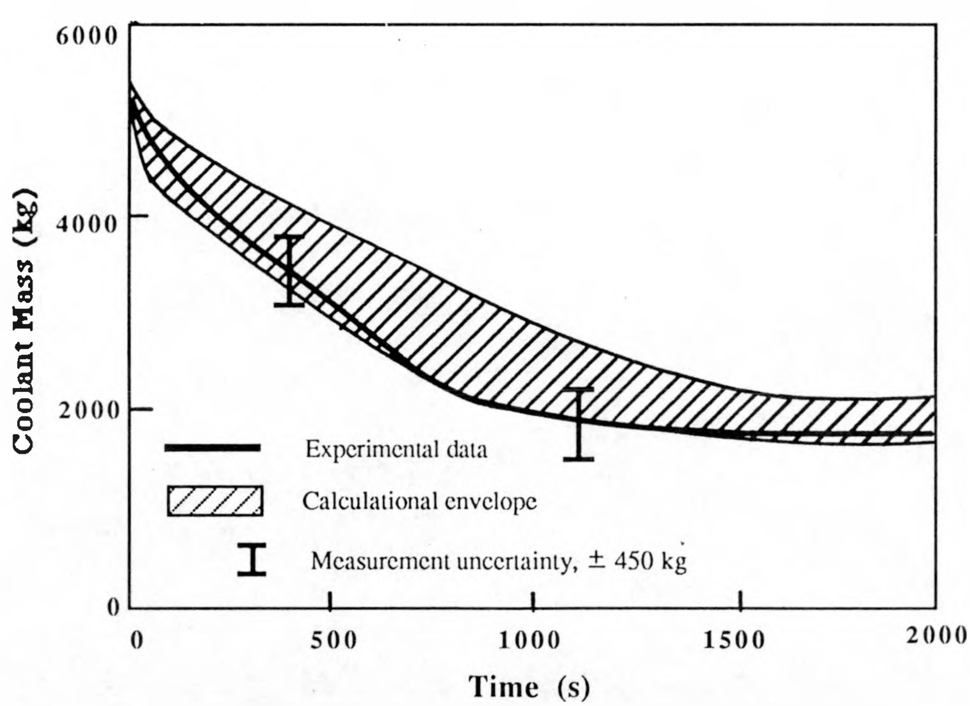

Figure 5.5. Primary coolant inventory during experiment LP–SB–1.

Table 5.4. LP–SB–2 Project Members Analyses

Country	Organization	Code
Italy	NIRA	RELAP5/MOD1/ CY19
Japan	JAERI	RELAP5/MOD1/ CY18
Finland	VTT	SMABRE
F.R.G.	GRS	DRUFAN 02
U.K.	CEGB	RELAP5/MOD1 CY19
U.K.[a]	CEGB	RELAP5/MOD2
Spain and U.K.[a]	AEE Winfrith	TRAC–PF/MOD1

a. These calculations are not included in the Comparative Analysis Report.

Postexperiment analyses carried out for the EASR[19] and analyses performed by the Project members for the Comparative Analysis Report [22] showed that the general parameters, such as primary pressure (Figure 5.6), were simulated relatively well. These analyses, however, also confirmed the weakness of the computer codes in proper prediction of stratification onset in the hot leg and calculations of fluid quality in the side branch for stratified flow. These modeling deficiencies are mainly responsible for the incorrect calculation of coolant inventory (Figure 5.7) and differences between calculated and measured loop flow behavior. Based on these analyses, the following general comments can be made on code performance in simulation of the LP–SB–2 experiment:

RELAP5/MOD1

Performed analyses showed significant deficiencies in modeling of the flow regimes, particularly with regard to onset of stratified flow at higher mass fluxes and liquid/vapor entrainment behavior at branches. These deficiencies are the main reason for the failure to calculate the experiment conditions correctly.

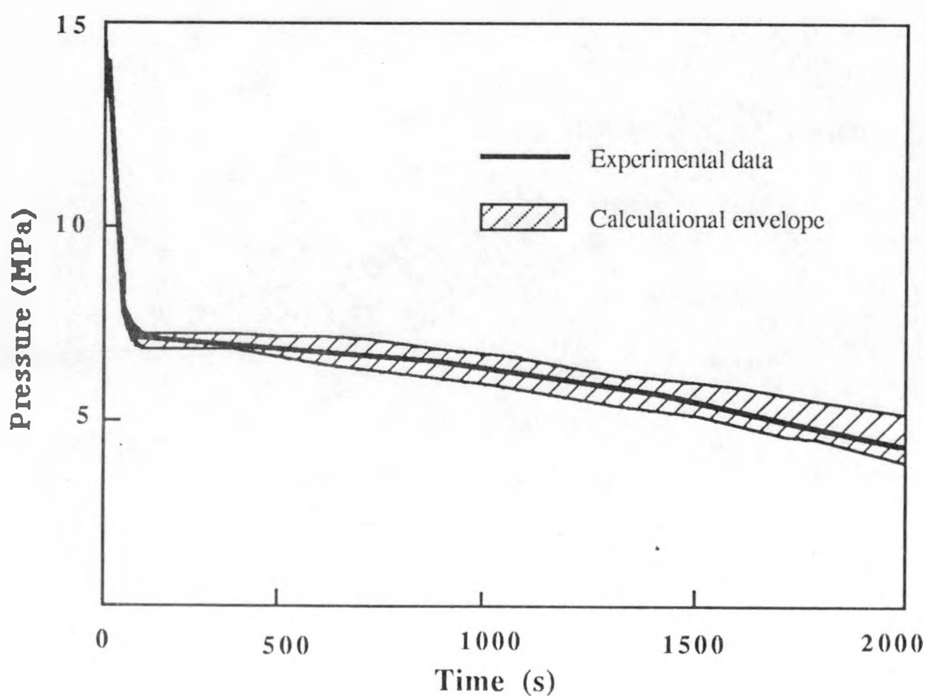

Figure 5.6. Primary system pressure during experiment LP–SB–2.

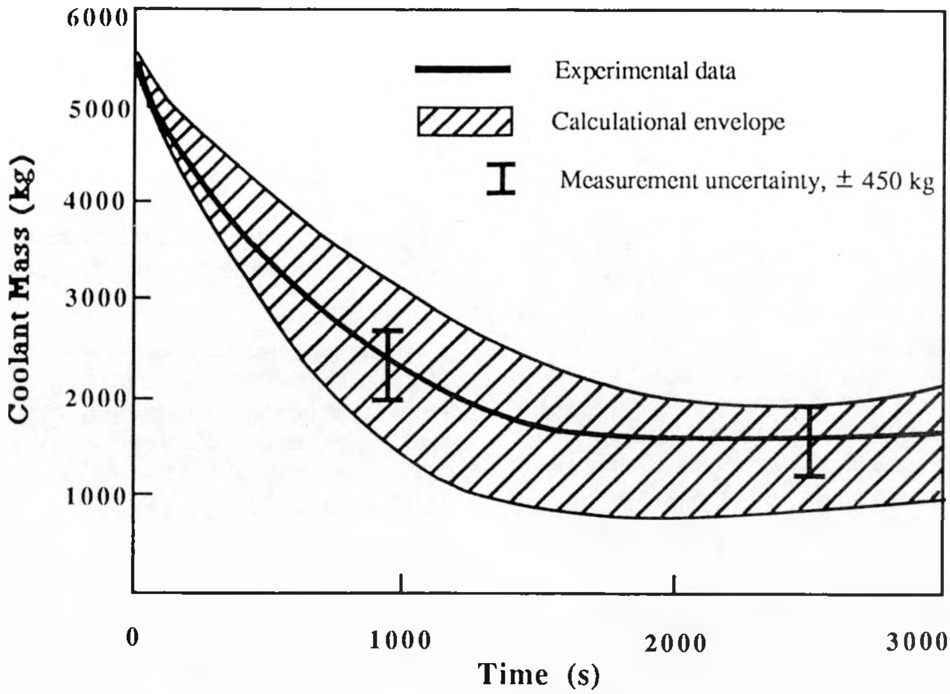

Figure 5.7. Primary coolant inventory during experiment LP–SB–2.

RELAP5/MOD2

The new horizontal stratification model implemented in RELAP5/MOD2 code did not provide significant improvement of the calculations. The code failed, in general, to calculate onset of the stratified flow in the hot leg at the correct conditions. The analyses indicated a need for an improved stratification criteria and for better entrainment and vapor pull–through models for the horizontal branch.

DRUFAN 02

Although the hot leg was modeled with three parallel, vertically stacked volumes (as for LP–SB–1), the code failed to properly calculate onset of stratification due to large interphase slip. As for the RELAP codes, this indicated a need of improved stratification criteria and vapor/liquid entrainment models.

SMABRE

The comparisons indicated that this small, fast running code performed qualitatively similar to the more complex codes.

TRAC–PF1/MOD1

A calculation with a version of TRAC (based on version 12.7) gave only fair agreement with the main system parameters, and predicted mild uncovery of the top of the core. The main source of error was the lack of a model relating quality in the break line to flow conditions in the hot leg. Much better agreement was obtained when TRAC was forced to use the experimental break line quality. The (corrected) Taitel–Dukler transition criterion predicted the transition to full stratification in the hot leg satisfactorily. The steady fall in the loop velocities, seen after the pump head degraded and the cessation of liquid flow after stratification occurred, were not observed. Possible reasons are discussed in but would need to be confirmed by further work.[31]

Experiment LP–SB–3. Experiment LP–SB–3 was designed to provide data on certain phenomena and processes, and did not aim to represent a particular accident sequence. This required careful planning of both the experiment specifications and the operating procedures, which could have been achieved by massive application of computer code. RELAP5/MOD1 was applied for the planning and predictions of this experiment. Parameters such as break size, break isolation time, primary coolant pump operation, criteria for auxiliary feed, initiation of secondary bleed (steam dump) etc., were determined by code calculations which included many sensitivity studies. These experiment design calculations were performed in Italy and by the Project.[23,30]

The final prediction,[33] performed using RELAP5/MOD1 Cycle 18, compared with the experimental data, are reported in the Quick Look Report.[28] All the major phenomena were predicted correctly and in correct order, though the times of the major events were shifted from the observed values. The secondary system pressure following the pump trip decreased more rapidly than predicted because of greater than assumed leakages at the main steam control valve. As a result, the primary and secondary systems remained coupled with heat being transferred from the primary to the secondary side throughout most of the transient. Differences between predicted and measured primary coolant mass inventory were due to underwater saturated break flow and to a greater than predicted amount of primary coolant remaining after the pump trip. The coolant pumps were tripped in the experiment earlier than in the prediction because the online mass inventory measurement system indicated that the system inventory reached the trip setpoint (2000 kg) at 1600 s, compared with the predicted time of 2363 s. Linear extrapolation of the data indicated that the primary system mass inventory at 2363 s would have been approximately 1980 kg, only 20 kg below the predicted value. The measured time to the beginning of core heatup was much longer than the predicted time, mainly because of the overpredicted coolant inventory. The calculated cladding heatup rates were nearly adiabatic, while the measured values were approximately 70% of adiabatic. The lower measured heatup rates indicate the existence of cooling mechanisms which the code did not simulate, such as entrainment of liquid from the lower regions of the core, which was never completely voided in this experiment. The effects of the secondary bleed and feed recovery procedure, which terminated the fuel cladding temperature excursion, but did not have a very strong cooling effect, were overpredicted. The reason of this discrepancy was that the code calculated clearing of the loop seal, which remained filled in the experiment. Calculated accumulator flow oscillations were not observed in the experiment, but, between the experiment and the calculations, the integrated accumulator flow agreed well.

The postexperiment calculations in the framework of the EASR analyses were performed with RELAP5/MOD2 by ANSALDO of Italy. These calculations showed several improvements over the predictions with RELAP5/MOD1, and were able to allow for the discrepancy in pump trip times. The new code version correctly calculated the primary coolant inventory and the initiation of fuel cladding heatup. There were no accumulator injection oscillations in these calculations, indicating that the coupling of the accumulator model with the overall numerical scheme of the code had been improved. Also, it was observed that the new calculations gave a significantly smaller mass error of less than 1.5 kg over the entire transient. The only parameter which did not improve in the new calculations was the core heatup rate during the core uncovery, which still remained higher than measured.

A number of postexperiment code analyses of the LP–SB–3 experiment were performed by different Project members and are summarized and compared in the comparative analyses report prepared by ANSALDO of Italy.[34] The available postexperiments are set out in the following table:

Table 5.5. LP–SB–3 Project Members Analyses

Country	Organization	Code
Italy	ANSALDO	RELAP5/MOD2/ CY36.02
U.K.	CEGB	RELAP5/MOD2/ CY36.02
Finland	VTT	RELAP5/MOD2/ CY36.02
Finland	VTT	SMABRE
Switzerland	EIR[a]	RELAP5/MOD1/ CY18
Switzerland	EIR[a]	RELAP5/MOD2/ CY36.02
Spain	UEFSA	RELAP5/MOD1/ CY18
Japan	JAERI	RELAP5/MOD1/ CY18
U.K.[b]	AEE Winfrith	TRAC–PF1/MOD1

a. Presently Paul Scherrer Institute.
b. Not reported in the Comparative Analysis Report.

The Winfrith Atomic Energy Establishment analyses with TRAC–PF1/MOD1 were made available to the members in a separate report.[30]

The performance of the codes in simulating the LP–SB–3 experiment was, in general, satisfactory, and the calculated parameters bounded the measured data. Figure 5.8 shows the envelope of the primary system pressure calculations compared with the measured pressure.

RELAP5/MOD1

The oldest code among the codes applied for analyses of the LP–SB–3 Experiment performed relatively well. The most significant deficiencies were the overestimated heatup rates and oscillations of the accumulator injection, and overestimated core cooling effects of the feed and bleed operation.

RELAP5/MOD2

The newer version of the RELAP code still overestimated the cladding heatup rates. Some of the core cooling, due to liquid falling back into the reactor vessel from the operating loop hot leg, could not be calculated because of the one dimensional property of the code. The experimental data indicated increase in heat transfer from the uncovered fuel rods after the break isolation. This phenomenon, which is probably related to enhanced droplet entrainment from the water in the lower parts of the core, was not simulated by the code. Similarly to RELAP5/MOD1, this code overestimated the effects of feed and bleed operation on the core cooling.

SMABRE

This simple fast running code developed in Finland performed well compared to the more complex codes. It exhibited a tendency to overpredict the core thermal response, which is rather a positive feature for simple fast running codes.

TRAC–PF1/MOD1

In general, the results of simulations with this code are similar to those with RELAP5/MOD2. The most significant discrepancy is the overestimated core heatup rate, which is related to weaknesses of the interphase drag model. In contrast to the RELAP5 results, this code did not calculate the reflux condensation.

It may be concluded that these results show a satisfactory ability to calculate reactor transients strongly influenced by the phenomena displayed in the LP–SB–3 Experiment and that the codes do not show any major inadequacies in modeling these phenomena.

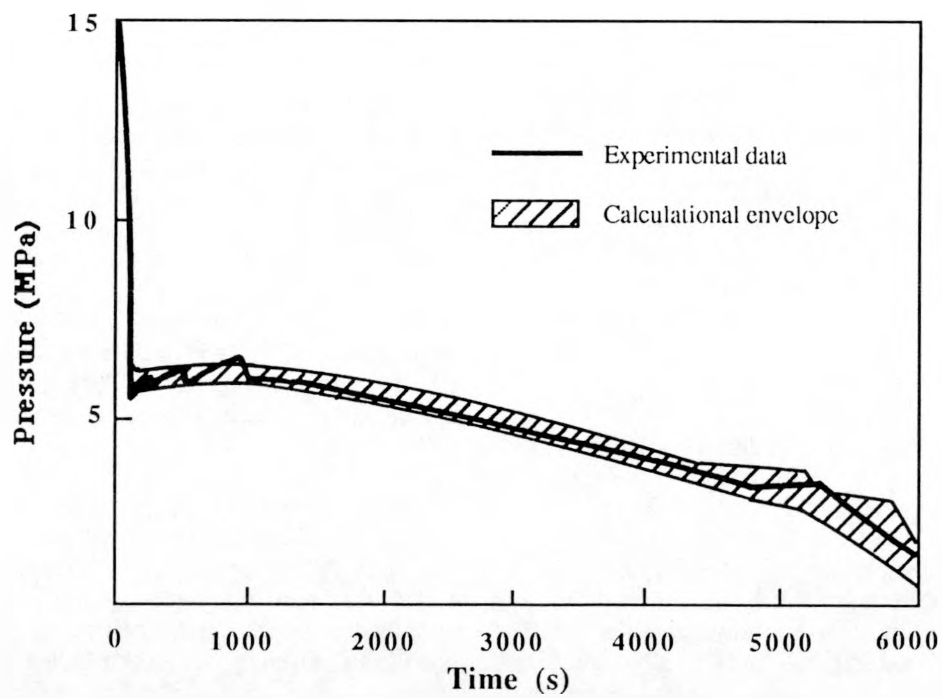

Figure 5.8. Primary system pressure during experiment LP–SB–3.

5.3 Large–Break Experiments

5.3.1 Experiment Objectives and Description. A large–break LOCA is defined as an instantaneous double–ended offset shear of a primary coolant pipe. Such an accident is a very unlikely event in a PWR, but is frequently used as a design basis accident for licensing purposes. This is based on a design philosophy that identifies a large–break LOCA in the cold leg of a PWR as the most severe challenge to the ECCS. The importance of this transient in the OECD LOFT program can be seen from the fact that three of its experiments, LP–02–6, LP–LB–1, and LP–FP–1 addressed specific issues associated with this accident scenario. Investigations addressed large–break LOCA phenomena from the viewpoint of three different countries. The first two addressed issues of concern to the United States and the United Kingdom. The third experiment was based on a design basis accident for a German PWR with the primary objective of obtaining information on fission product release and transport after cladding rupture. This section discusses the thermal–hydraulic issues and phenomena addressed in the first two OECD LOFT large–break experiments. The third experiment is discussed in Section 5.4.

The major features of a large–break LOCA are the very fast primary system depressurization and the consequent loss–of–coolant and core heatup. Because the timescale does not permit operator intervention, there is total reliance on the adequacy of automatic engineered safeguards to provide sufficient emergency core cooling to limit cladding heat up before it results in cladding ballooning and cladding failure. Also, because it is not possible to demonstrate that all the features of the full–scale plant are fully represented in any experiment or sequence of experiments, there is substantial reliance on computer models to extrapolate experimental data in the presentation of the formal safety case. The fact that there has been a consensus agreement to concentrate on a single scenario even if unrealistic severity has at least had the advantage of limiting the number of accident scenarios to be analyzed and in focusing attention on the major issues.

As previously noted, LOFT has a number of features, not matched in any other facility for the study of this accident, in particular its size, its ability to cover the full accident pressure range, and the availability of nuclear and fission product decay heating. However, issues of engineering practicality have meant that it was not possible to achieve a fully consistent set of scaling factors between LOFT and the full–size plant. In addition, the half–height core and the location of the core too low in the vessel make it relatively easy to quench the core in LOFT. Also, two phenomena not envisioned in the Appendix K approach have strongly influenced tests in LOFT, namely the early bottom–up and top–down quenches. The bottom–up quench is driven by continued pump effectiveness as it runs down. The top–down quench is caused by fluid drainback from the hot legs of the steam generator. Differing views on how these effects should be allowed for in extrapolating to the full–sized plant are responsible; For example, the different approaches taken by the US and the UK to the planning of tests LP–02–6 and LP–LB–1.

A further issue not fully resolved is the effect of the external thermocouples fitted to some of the fuel pins that are crucial for the experimental evaluation of peak cladding temperatures. Analyses carried out in the USA have come to the conclusion that although there are some systematic errors because of mounting thermocouples on the fuel cladding external surface, these are not serious and in particular, the quenches recorded by these thermocouples correctly represent the behavior of non–instrumented fuel pins. UK studies that have modelled the conduction path between the thermocouple and the cladding to which it is attached show that, under quench conditions, the thermocouple has poor thermal contact with the cladding and quenches significantly more readily. This has an important influence on the validation of computer models because, if the behavior indicated by the external thermocouples is accepted unmodified, the computer codes can predict quench temperatures in LOFT only by using very high values of T_{min}, which are inconsistent with data from steady state special effects tests.

Experiment LP–02–6. This was the first OECD LOFT large–break experiment and was conducted in October 1983. It was directly sponsored by the USNRC and the initial and boundary conditions were chosen to be representative of USNRC licensing limits for a commercial PWR.[35] This included the loss of offsite power coincident with LOCA initiation. A particular feature of the experiment was the first use in LOFT of prepressurized fuel rods in the center fuel bundle and it was an important objective of the test to demonstrate that this did not lead to fuel rod damage from cladding ballooning.

The experiment was initiated by opening the quick–opening blowdown valves in the broken loop hot and cold legs. The subsequent low–pressure signal resulted in reactor scram and pump trip. The pumps remained connected to their flywheel during pump coastdown. Opening the break resulted in almost instantaneous flow reversal in the core followed by rapid coolant depletion. The fuel rod cladding experienced

DNB within 1 s of experiment initiation and cladding temperatures started to rise. The mass flux instrumentation at the bottom of the core indicated the re–establishment of positive core flow at about 5 s and this resulted in quenching of the lower two–thirds of the core by 10 s after transient initiation. The upper part of the core experienced cooling but not quenching. The re–established positive core flow provided only a limited amount of coolant to the core which then began to heat up again, but at about 15 s, a top–down quench started which extended over the upper third of the core. Five seconds later, this region of the core started to heat up again. About 30 s after transient initiation, the entire core was dry and the cladding temperatures were above saturation. ECCS injection started at 17.5 s and this caused a complete core quench at about 56 s. The maximum peak cladding temperature of 840 K was reached at about 41 s.

Fuel rod plenum pressure measurements indicate that there was no cladding rupture and post–experiment samples taken from the primary coolant system showed that no fission products were present in the coolant. Later analyses also indicated no fuel rod failure or appreciable cladding ballooning.[36]

Experiment LP–LB–1. Experiment LP–LB–1, the second large–break experiment in the OECD LOFT program, also simulated a double–ended offset shear of one inlet pipe in a four–loop PWR and was initiated from full power (50 MW). The boundary conditions for the experiment were chosen to be consistent with the LOCA minimum safeguard assumptions for the proposed UK PWR at Sizewell.[37] These assumptions included the loss of offsite power coincident with LOCA initiation and restricted the quantity and availability of accumulator and pumped ECCS injection. A major objective was to achieve an experiment in LOFT that would simulate the range and balance of the phenomena seen in computer code calculations supporting the UK preconstruction safety report and this further influenced the choice of boundary conditions. As a result, the accumulator volume was only 70% and the pumped low–pressure ECCS injection only 50% of that used in the LP–02–6 experiment. An early rapid primary pump coastdown was included to attain maximum cladding heatup by limiting the early rewet phenomena because the strong early rewet effects seen in the earlier LOFT tests were believed to be a feature of the facility not present to the same extent under full–scale conditions. Both decisions helped to maximize the core stored energy at the end of blowdown.

The experiment was initiated by opening the blowdown valves from a core power level of 49.3 MW (51.7 kW/m maximum heat generation rate).[38] The reactor was scrammed on a low–pressure signal at 0.13 s and the primary pumps were tripped slightly later and disconnected from their flywheels at 0.63 s. The fuel cladding went into DNB in less than 1 s in the high power region of the fuel bundle. The early decoupling of the primary pumps from their flywheels resulted in insufficient flow into the vessel from the intact cold leg to produce the bottom–up flow into the core and consequent early fuel quench that occurred in LP–02–6. The rapid cladding temperature rise stopped at about 13 s because of liquid fallback from the upper plenum. This top–down liquid flow quenched the upper part of the central fuel assembly and caused extensive cooling in the peripheral fuel bundles. The maximum fuel cladding temperature during the blowdown phase reached 1261 K shortly before top–down cooling started. This top–down cooling lasted until about 25 s, when the fuel cladding again started to heatup. ECC injection from the accumulators began at 17 s and from the LPIS at 32 s and resulted in a core quench starting at about 34 s. This core quench propagated both from the bottom and the top of the core, progressing towards the peak power region, and was complete at 72 s. The maximum cladding temperature recorded during the ECC injection phase was 1257 K.

All the fuel rods used in the LP–LB–1 were unpressurized and there was concern that the high temperatures reached in this experiment would weaken the cladding causing cladding collapse onto the fuel pellets. Analysis and coolant samples indicated that the cladding was not ruptured but that there could have been some limited deformation.[38]

5.3.2 Experimental Findings.

ECCS Performance. The LOFT facility, as described in Appendix A, was equipped with an emergency core cooling system (ECCS) that included a High–Pressure Injection System, a Low–Pressure Injection System and Accumulators all installed in the cold leg of the operating loop. This installation was typical of that for a full–scale plant.

Experiments LP–02–6 and LP–LB–1 addressed issues raised in licensing commercial plants in the United States and the United Kingdom respectively.

Both experiments assumed the loss of offsite power coincident with LOCA initiation (which results in delayed HPIS injection). The ECCS injection in LP–LB–1 used only 70% of the accumulator volume and

50% of the pumped injection for LP–02–6. An early rapid pump coastdown was included to attain maximum cladding temperatures by limiting the early rewet phenomena. The hot wall time delay was at most 2 s. The refill phase was therefore very similar in both experiments. However, the reduced amount of water entering the core in LP–LB–1 delays reflood by approximately 10 s, with a consequent increase in peak clad temperatures in reflood. Nevertheless both tests showed that, even when severely degraded, the ECCS systems in these LOFT tests were able to quench the core and to provide core reflood without significant damage to the fuel elements.[36,38] It was also observed that, at these higher cladding temperatures, the reflood front can pass the hotspot without quenching the cladding.

Blowdown Quenches. The LOFT large–break experiments of the previous USNRC program[7,8] indicated early fuel cladding quenches during the blowdown phase. The first of these brief quenches was associated with a front of fluid (sometimes referred to as a density wave) moving up through the core while the second quench resulted from liquid falling from the upper plenum onto the core. It was found that the bottom–up quench is a function of primary coolant pump operation. If there is a long period of pump coastdown this results in a bottom–up quench able to remove most of the thermal energy stored in the fuel rods resulting in significant core cooling. If the pump is tripped early and the flywheels are disconnected, thus producing a rapid pump coastdown down, this bottom–up quench does not occur.

LP–02–6 simulated a large–break LOCA case with the reactor coolant pumps tripped at transient initiation, but the flywheels were not disconnected so the coastdown down was typical for a commercial power plant and relatively long. There was an early bottom–up quench in the lower two–thirds of the core while the upper part of the core was cooled but not quenched. The absence of a bottom–up quench in LP–LB–1, where pump coastdown was inhibited, confirmed that in LOFT, the bottom–up quench is a function of pump operation. In LOFT, an effective bottom–up quench can reduce the maximum cladding temperature during blowdown by as much as 200 K and can accelerate the final quench.

After the bottom–up quench, the cladding temperatures measured in LP–02–6 indicated a top–down quench in the upper part of the core. LP–LB–1 also showed some top–down quenching during blowdown that involved about one–third of the core and lasted about 15 s. These top–down quenches are a result of water draining back from the steam generators and the pressurizer.

Core Thermal Response. The model of core thermal behavior during a large–break LOCA which underlies the Appendix K approach, implies rapid cladding heatup at the start of the transient because of thermal energy stored in the fuel, a period of cooling during blowdown, and then continued heatup at a lower rate from decay heat finally terminated by reflood. The maximum cladding temperature would be expected to occur during the reflood phase when precursory cooling from quenching lower in the core would begin to reduce the hot spot temperature. Finally, the cladding temperature everywhere would fall to the saturation level as a result of the complete reflood with ECC water. This scenario was supported by experimental evidence from facilities such as Semiscale and was the basis of the calculational procedures used in US licensing. The large–break LOCA experiments conducted during the USNRC LOFT program showed, for the first time, that there could be significant quenching of the cladding during blowdown and, as a consequence that the peak cladding temperatures reached during blowdown could be significantly lower than predicted by Appendix K calculation rules.

These early quench effects arise from the combined interaction of a number of factors many of which are strongly influenced by specific features of the LOFT facility, in particular, the short length core and the presence of external thermocouples. This presents the problem, not only of disentangling their effect in LOFT, but also of assessing their importance in the full–scale plant.

As stated above, one of the objectives of LP–02–6 was to determine whether fuel rod damage would occur during a simulated design basis accident with prepressurized fuel rods. To determine this, the rods in the center fuel module were prepressurized to 2.41 MPa. Fluid samples from the primary coolant taken after the experiment gave no indication of fission products and fuel plenum pressure measurements showed that there had been no clad rupture. Analysis also indicated no fuel rod rupture or appreciable cladding ballooning.[36] Cladding damage was also a possibility in LP–LB–1. Although the fuel rods were not prepressurized in this experiment, there was a risk that the cladding could collapse onto the fuel pellets. Analysis and coolant samples indicated that the cladding had not ruptured but could have been deformed.[38]

5.3.3 The Performance of Computer Codes in the Analyses of Large-Break Experiments.

Experiment LP-02-6. Two computer codes were used for pretest predictions: TRAC-PD2/MOD1 and FRAP-T6/MOD1. FRAP applies detailed thermal-hydraulic data from TRAC to calculate thermal conditions in the fuel. The initial cladding heatup was well predicted by TRAC but the calculations indicated a relatively slow cooling of the hot spot after 5 s (the data shows a rapid quench). This lack of a quench in the calculation is responsible for higher peak temperatures after blowdown and helps to delay the final quench. The comparison of measured and calculated reactor vessel cold leg flows also shows some deficiencies in the calculations. The calculated initial coolant flow from the reactor vessel is too low and the reestablishment of the flow to the vessel occurs later and is smaller in magnitude than measured. The core thermal behavior is very sensitive to small changes in flow conditions as the vessel mass balance data indicate. The accuracy of prediction of hydraulic conditions in the core can be expected to improve with the introduction of improved models of heat transfer in low flow film boiling conditions considered important in the early quench period.

Several project participants analyzed the LP-02-6 Experiment as shown in the following table.

Table 5.6 LP-02-6 Project Members Analyses

Country	Organization	Code
Japan	JAERI	RELAP5/MOD2
F.R.G	GRS	DRUFAN 02
U.K.	UKAEA	TRAC-PF1/MOD1
Spain	ETS	TRAC-PD2/MOD1
Switzerland	EIR	RELAP5/MOD2
US[a,b]	LANL	TRAC-PD2/MOD1
US[a]	LANL	TRAC-PF1/MOD1
US[a]	EG&G	RELAP5/MOD2

a. These calculations are not included in the Comparative Analysis Report.

b. These calculations are pre-experiment calculations.

The first five calculations are discussed in the Comparative Analysis Report, prepared by PSI Switzerland,[39] the sixth and seventh are reported in Reference 40, and the last calculation in this table is reported in Reference 41. The review of these calculations shows that all the codes correctly calculated most of the major parameters such as primary system pressure, break flows, and densities. The major problems were associated with simulation of the early bottom-up quench from high (greater than 900 K) cladding temperatures. Figure 5.9 shows the measured cladding temperature at the hot spot elevation and the calculational envelope. In judging code performance it is important to recognize that it depends not only on the models incorporated in the codes but also in the details of the actual implementation and how the system is nodalized. The analyses performed lead to the following general conclusions on individual code performance.

RELAP5/MOD2

The break flow is quite well calculated by this code, particularly in the early stages of the transient. The calculated intact loop flow also compares quite well with the experimental data. However, in the later part of the transient, the densities calculated for the hot leg are too high. The bottom-up surge of water through the core just after blowdown is calculated to be 40% liquid. The resulting blowdown quench at the cladding hot spot elevation is calculated correctly in some of the submissions but in others this quench is not predicted at all, or the initial heatup is underpredicted. The time of the reflood quench is usually calculated correctly. RELAP5/MOD2 tends to overpredict the water holdup in the upper plenum and therefore the cladding heatup measured in the upper part of the center fuel module is either not calculated at all or is significantly underpredicted.

DRUFAN 02

Calculations with this code show a very good simulation of the initial core heatup, but the code fails to calculate the bottom-up quench at the peak cladding temperature elevation. The peak cladding temperatures are slightly overpredicted and the reflood quench slightly delayed. The heatup of the upper part of the center fuel module is less well calculated but the top-down quench is simulated.

TRAC-PD2/MOD1

Problems with the critical flow model in this code were probably responsible for the significant underprediction of the break flow in the early part of the blowdown phase. The code calculated a bottom–up surge of water (80% liquid) into the core during blowdown. However, the dryout criterion in the code gave a poor prediction of the cladding quench associated with this water surge. Calculations using a higher minimum film boiling temperature give a better calculation of the blowdown high temperature quench. Without this modification, the maximum cladding temperature is calculated during reflood and the reflood quench is delayed by about 50 s.

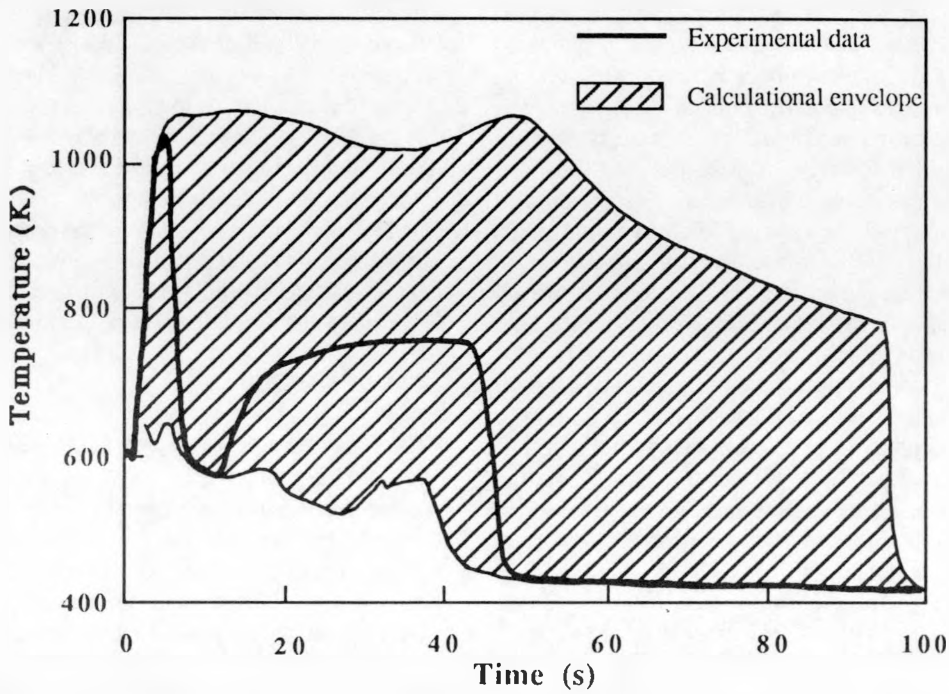

Figure 5.9. Peak fuel cladding temperature during experiment LP–02–06.

TRAC–PF1/MOD1

The break mass flow rate is well simulated. The code calculated the blowdown bottom–up water surge as 70% liquid. In the calculations, this surge of water provides significant cooling at the hot spot elevation but no complete quench. For the top of the core, the code calculates complete bottom–up quench but fails to simulate succeeding heatup. The code slightly overestimates the peak clad temperature. The calculated reflood quench is delayed by about 40 s in comparison to the measured quench. In the upper part of the center fuel module, the peak cladding temperatures are slightly underpredicted for the blowdown and overpredicted for the reflood period. The code simulates the top–down quench.

Experiment LP–LB–1. TRAC–PD2/MOD1 code was used to predict this experiment. The version of the code that was used contained an error in the gap conductance model believed to affect significantly the calculated cladding temperatures for the early part of the transient. Calculations were also carried out immediately after the experiment using a corrected version of the code and measured initial and boundary conditions. The results of both calculations and comparison to experimental data are reported in Reference 38. The predictions showed the initial heatup rate in agreement with experiment data but after 4 s the calculations deviate significantly from the data. These deviations are a direct result of the code errors. There were also significant differences between the calculated and measured cooling attributed to the reflood. In the experiment, the cooling rate increased as the reflood progressed, while in the calculations, the cooling rate was reduced as the temperature difference between cladding and coolant reduced.

The postexperiment calculations provided correct simulation of the initial heatup. The time of the peak cladding temperature was calculated correctly through the calculated temperature at approximately 60 K lower than the measured temperature (1261 K). The post–experiment calculated thermal response during reflood was in relatively good agreement with measured data, however, the reflood quench phase was not included in the simulations. For the upper part of the CFM, the peak cladding temperatures were underpredicted by about 100 K, and the top–down quench for that part of the core was not calculated. The code calculated only some cladding cooling. The strong, hydraulically controlled azimuthal asymmetry measured in the thermal response of the peripheral bundles was also partially calculated. The major differences between the experiment and the calculations were in the temperatures at the peak power location. The code did not correctly calculate the initial cooling during blowdown, the peak cladding temperature, and the cooling during reflood. It was judged that these deficiencies in the calculations indicated or revealed limitations of the post heat flux models used in the TRAC code.

The code calculated, in general, the hydraulic conditions quite well with the exception of the underpredicted depressurization rate during accumulator injection. The code calculated the top–down quench during the blowdown, but underpredicted the extent. The code also calculated properly the simultaneous bottom–up and top–down quench during reflood.

Five Project members analyzed the LP–LB–1 experiment using four different computer codes as shown in the following table:

Table 5.7. LP–LB–1 Project Members Analyses

Country	Organization	Code
Italy	University of Bologna	RELAP5/MOD1
F.R.G.	GRS	DRUFAN 02/FLUT
Finland	VTT	RELAP5/MOD2
U.K.	UKAEA	TRAC–PF1/MOD1
Switzerland[a]	EIR[b]	RELAP5/MOD2

a. EIR submitted two calculations, the second calculation included simplified nodalization.

b. Presently Paul Scherrer Institute

These calculations are discussed in the Comparative Analyses Report, prepared by the UKAEA.[42] The following are the major conclusions of this analysis:

1. All the calculations are sensitive to the detailed choice of noding. Much of this sensitivity could probably be removed by an overall refinement of the noding.

2. Loop flows are generally well calculated.

3. Flows in and out of the core are poorly predicted and there are complex three–dimensional flow patterns in the core. These cannot be predicted by one–dimensional codes but there are also important inadequacies in the TRAC three–dimensional predictions.

4. All the one–dimensional codes predicted an early bottom–up flow into the core leading to early rewetting but no early top–down flow while the experiment showed a significant top–down flow but only very limited early bottom–up rewet. All the codes poorly predicted the core flow from the downcomer in blowdown.

5. Peak cladding temperatures are relatively well predicted, (Figure 5.10) though this was affected by the early bottom–up flow into the core noted above. Temperatures in the core periphery, which are responsible for most of the steam generated, are poorly predicted in all the calculations.

6. There was very little direct bypass of accumulator ECCS around the top of the downcomer and this behavior is generally well predicted by the codes.

7. Rewetting times are well predicted by the RELAP5/MOD2 and DRUFAN/FLUT calculations but are about 25 s late in TRAC. The reflood model in RELAP/MOD1 is known to be inadequate. It is still a matter of controversy as to what extent the experimental evidence on rewet times and conditions is influenced by the performance of the external thermocouples and this may be a possible explanation of the TRAC discrepancy.

Separate comments on the performance of the individual codes are given below.

RELAP5/MOD1

Only limited information was provided on these calculations and this did not include any information on vessel fluid flows. The calculated pressure was somewhat above the data for 10 s, then it decreased rapidly, but it did not model the expected increase in the pressure decay rate because of condensation on injected accumulator water. The predicted break mass flow rate was in reasonable agreement with the experiment. The peak cladding temperature in blowdown was quite well predicted and there is some evidence of top–down cooling at 12 s though this lasts longer and is more extensive than in the experiment. After about 20 s, the cladding dries out at the top of the fuel but the rest of the fuel cools substantially (200–300 K) below the experimental values. There is no modelling of the reflood phase and the calculation was terminated at 120s.

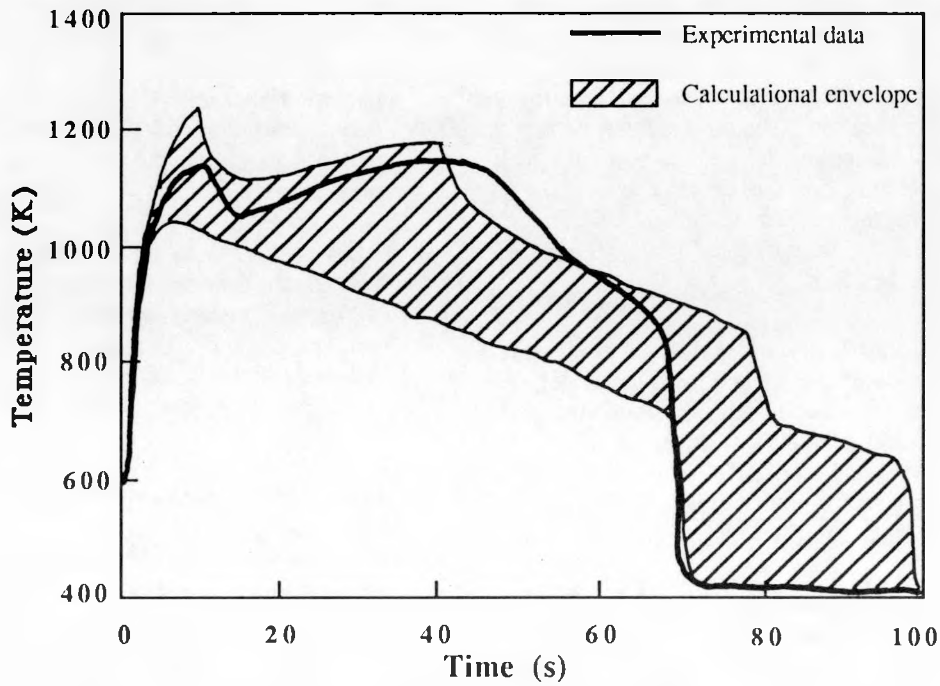

Figure 5.10. Peak fuel cladding temperature during experiment LP–LB–1.

RELAP5/MOD2

The submitted calculations differ significantly, almost certainly because of differences in nodalization. All the calculations underestimated the effect of condensation on the ECCS water from the accumulator on the pressure transient though this was compensated to some extent in the calculations with simplified nodalization by the absence of heat input from the reactor vessel metal work. All calculations produced a strong bottom–up flow into the reactor vessel at about 5 s. The simplified nodalization calculations showed a limited top–down flow at 16–23 s. The calculated peak cladding temperatures during the blowdown were low by about 100 to 200 K primarily because of the early bottom–up quench.

DRUFAN 02/FLUT

This code calculated a strong bottom–up flow into the core at 6.5 s. though this had only a small effect on cladding temperatures. The central fuel module started to cool when the core began to refill at 30 s. The final quench occurred at 650 K but there was an earlier front of rapid cooling from about 900 K. The peripheral fuel went into dryout during blowdown but did not subsequently quench. The top–down flow at 13 s was not predicted. Some ECCS bypass was calculated at 16 s. When FLUT was used for calculations after 16 s, it appeared to provide much greater heat transfer from the core than DRUFAN for the same fluid conditions. A pressurization rate increase was attributed to steam condensation on ECCS injection.

TRAC–PF1/MOD1

The pressure variation, including the effect of the condensation on ECCS liquid and of the final nitrogen purge as the accumulator is exhausted is well predicted. The code correctly predicts the almost total absence of bottom–up flow at 5 s and the observed top–down flow at 13 s. There is no significant ECCS bypass and the lower plenum starts to fill as the ECCS begins. The peak clad temperatures in both blowdown and reflood are very well predicted. The early steam cooling at 25 s is not predicted nor the blowdown quench in the peripheral fuel elements. The final quench is late by about 25 s and appears to be from a temperature about 150 K too low.

5.4 Fission Product Release Experiment LP–FP–1

5.4.1 Experiment Description and Conduct. Experiment LP–FP–1 was the first LOFT fission product release and transport experiment. Its objectives were to obtain data on fission product release from the fuel cladding gap into vapor and reflood water and to collect data on transport of these fission products through and out of the reactor coolant system. The experiment was developed and planned under the guidance of the Federal Republic of Germany. Section 4.7.1 provided the detailed objectives of this experiment and a summary of the planning evolution.

The first attempt to conduct Experiment LP–FP–1 was carried out on December 12, 1984. As the experiment was initiated, the position indicator of the Quick Opening Blowdown Valves (QOBV) showed the cold leg QOBV open but the hot leg QOBV closed. At about 10 s, the Plant Protection System (PPS) was actuated with the following ECCS realignment: HPIS pump A, accumulator A, and LPIS pump A were aligned to inject in the lower plenum; while HPIS pump B, Accumulator B and LPIS pump B, were aligned to inject into the downcomer. HPIS flow started at 15 s and achieved full capacity of 1.95 l/s at about 17 s. Accumulators began to inject at about 19 s. At about 30 s, the LPIS pump started to inject and the core was completely quenched at about 35 s. Most of the fluid thermocouples just below the core have shown superheating starting at 23 s with a subsequent quench at 31 s. This indicates that the refill phase was completed at 31 s. Post–test analysis and comparison with earlier double ended break tests have shown that the hot leg QOBV opened sufficiently to allow maximum flow. Only the position indicator of the QOBV did not operate properly. The data gathered during this experiment attempt were found to be very valuable for evaluation of the completed experiment LP–FP–1 and for general understanding of large–break LOCA processes. Therefore, a decision was made to archive the data, and the aborted experiment was designated LP–FP–1A. The experiment LP–FP–1 was repeated successfully on December 19, 1984.

Experiment LP–FP–1 was the first in the series of two experiments to be conducted in the LOFT facility with intentional release of fission products. This resulted in a set of requirements for the LP–FP–1 Experiment from the facility recovery and readiness standpoint for the second fission product experiment:[43]

1. Experiment LP–FP–1 must be conducted with fuel damage limited to the center fuel assembly

2. The structural integrity of the center fuel assembly must be maintained to facilitate removal from the reactor vessel

3. Peripheral assembly fuel rod cladding temperatures will be limited to prevent damage to the peripheral fuel rods.

To meet these requirements for the LP–FP–1 Experiment, the reactor core was equipped with a 15 x 15 center fuel assembly with a thin zircaloy shroud, which enclosed the inner 11 x 11 fuel rod array where 24 of the fuel rods were enriched to 6–wt%. Twenty–two of these fuel rods were also prepressurized (2.41 MPa). The experiment was designed to cause cladding ballooning and rupture on these rods within 60–90 s after experiment initiation. The remaining two higher enriched, but unpressurized rods, were prepared for post-irradiation examination after the experiment. The positions of the higher enriched rods were selected so that the power levels of all 24 rods were within 1 to 2% of each other to ensure uniform gap fission product inventories, and reduce uncertainty in experiment control, including time of rod rupture.

The experiment was specified to be conducted with minimum burnup of 1175 MWD/MTU in the higher enriched rods.[43] The required fuel burnup was achieved through pre–experiment power operation. Abortion of the first experiment attempt and resumption of the experiment a week later resulted in additional power operation and total burnup of 1417 MWD/MTU. A short power operation interval just before initiation of the transient established minimum decay heat level and the initial conditions to conduct the experiment including the inventory of short–lived fission products.

The experiment was initiated by a reactor scram with a one second delayed opening of the quick–opening blowdown valves. This sequence should remove sufficient stored heat to cause a delay in reaching high temperatures. This delay was necessary to provide a well–defined set of boundary conditions for fission product release and transport. The primary coolant pumps were tripped and disconnected from their flywheels 1 s after QOBV opening to prevent blowdown fuel cladding quenches. This provided conditions similar to experiment L2–5 and LP–LB–1. The broken loop cold leg QOBV was closed at 68 s to ensure that positive core vapor flow existed for the transport of fission products released from the fuel rod gap, along the intended path for fission product measurements.

Figure 5.11 shows that the core thermal behavior was quite different from the behavior observed in the previous large–break experiments. The almost immediate temperature rises observed in other large–break LOCA experiments were not measured in LP–FP–1, because of reactor scram one second before transient initiation. The other major characteristic of the core temperature transient is that the expected early cladding temperature rise was prevented by several quenches, and the actual core heatup started very late in the transient.

The first core heatup began at about 3 s and continued to about 6 s when the first quench occurred (Figure 5.11). This was a bottom–up quench influencing only the lower half of the core. This quench was quite similar to the early quenches observed in the experiments L2–2, L2–3, and LP–02–6. The attempt to eliminate this early quench by tripping the pumps and disconnecting the flywheels failed in this experiment. There are two reasons for this. First, the primary coolant pumps were operated initially at higher speeds than in experiments L2–5 and LP–LB–1. This resulted in higher initial mass flow rate and fluid inertia. As a result of higher mass and inertia, more coolant was delivered from the intact loop to the downcomer than in the other experiments. Secondly, the reactor was scrammed before blowdown (intentionally) to remove some of the initial stored heat.

The following quenches were top–down quenches and were the result of unplanned injection of water in the upper plenum from the Accumulator B injection line. Detailed analyses of this injection indicated that some amount of nitrogen remained in the injection line after Accumulator B was activated to terminate the LP–FP–1A experiment.[44] This gas was then prepressurized in the injection line to primary system pressure during pretransient phase of the LP–FP–1. The blowdown and primary system pressure decrease triggered expansion of the nitrogen in the accumulator piping and injection of water into the upper plenum. It is estimated that approximately 425 kg of water was injected, resulting in a series of fuel cladding quenches and in delayed and unsymmetrical core heatup. The main cladding heatup in the central fuel assembly started at about 80 s and progressed from the bottom up. The heatup was not uniform radially; fuel rods closer to the broken loop heated up later and quenched earlier. The expected burst of the prepressurized fuel pins was delayed by more than 200 s. Post–test analysis revealed that cladding had ruptured in only 8 of the prepressurized 22 fuel rods.

At 344 s, the experiment was terminated on a high–temperature limit for the peripheral bundles by ECC injection into the upper plenum and the intact loop cold leg. The experimental ECC operation was scaled to simulate a commercial PWR of KWU type with effective injection into seven LPIS lines (out of eight), from seven accumulators (out of eight). The quench began at the top of the core, to the core bottom, and finally to the high–power region. The quench was not uniform through the core. The CFM was quenched at 370 s. Fuel assemblies 2 and 6 did not quench until 380 s. The maximum cladding temperature recorded in the CFM was 1210 K and occurred at 347 s. This temperature exceeds the temperature required for cladding ballooning and rupture.

In summary, because of the unexpected thermal–hydraulic conduct of the experiment, the thermal–hydraulic conditions in the core and in the fission product transport path were far less definite than expected.

5.4.2 Fission Product Release and Transport Analyses. An elaborate fission product measurement system (FPMS) was provided for this experiment to collect data on fission product release and transport. This system consisted of:

- A steam sampling system, which operated during the dry steam phase of the experiment.

- A gamma spectrometer system, which was operated during the twelve hours of the post–transient phase (the post–transient phase is the time period from the closure of the broken loop hot leg QOBV to twelve hours later.

- Deposition coupons, which collected samples during the transient and post–transient phases.

A detailed description of the system can be found in Appendix A.

The cladding temperature distribution and the vapor flow patterns in the CFM during the burst release phase were very inhomogeneous because of the unintentional water injection into the upper plenum. This resulted in only eight of the 22 pressurized rods bursting. Six of the burst rods were located in the intact loop hot leg corner of the CFM. Because of the flow patterns in the CFM and the upper plenum, the sample line that should collect the effluent from the CFM (including fission products) sampled only steam. Attempts were made to analyze the possible flow patterns using the

TRAC–PF1/MOD1 code.[45,46] These analyses have shown that the vapor flow in the CFM could have been downwards below the maximum power level and upwards above it. It is not known if the pin ruptures occurred relative to the elevation where the steam flow direction may have changed. The fission product release at the time of burst was most probably into a steam environment; however, small quantities of water could have been present on some core surfaces.

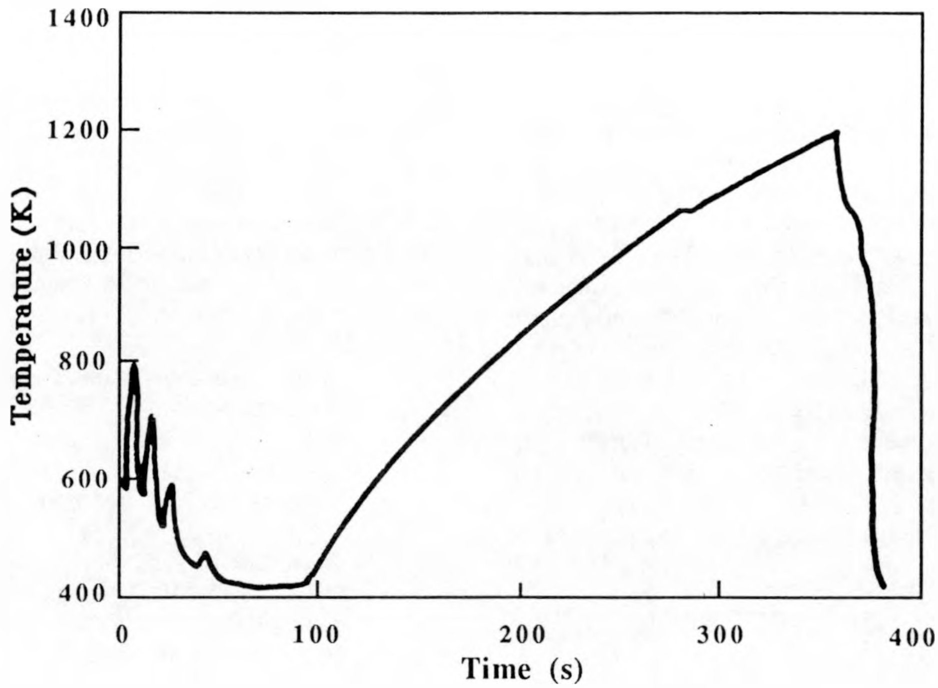

Figure 5.11. Peak fuel cladding temperature during experiment LP–FP–1.

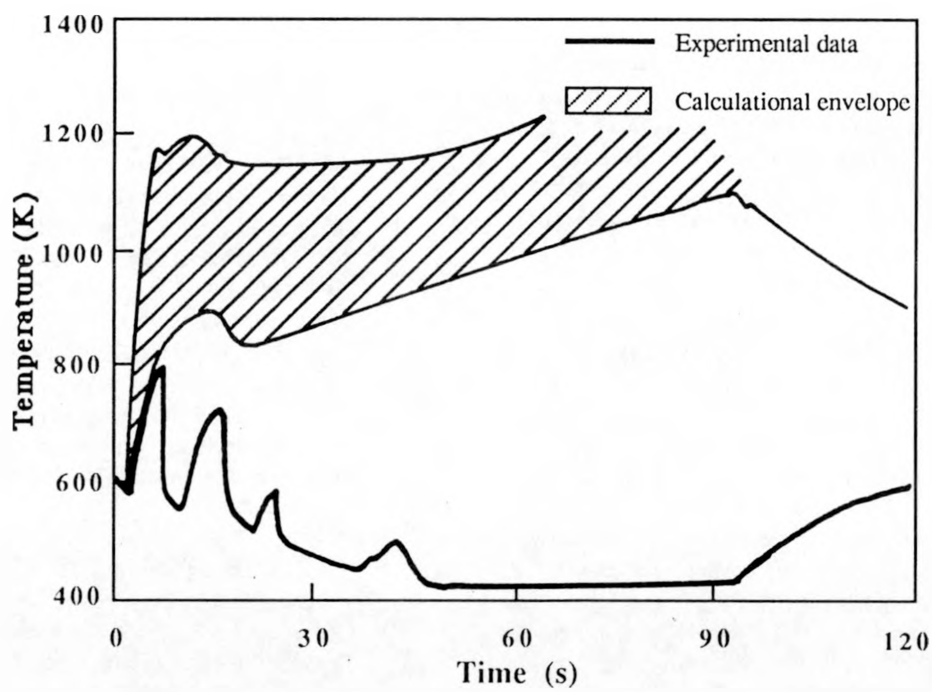

Figure 5.12. Peak fuel cladding temperature during experiment LP–FP–1 (including calculational envelope).

Rupture of the fuel rods was first indicated at 324.5 s. The fission products released at the burst were detected in the broken loop hot leg at 328.3 s. The steam sample system and one-half of the deposition coupons were isolated just before initiation of the ECC system at 344 s. The PCS was isolated from the BST at 530 s. There were three fission product release phases to be distinguished in this experiment. The first is the burst release, defined as an indication of the fuel rod rupture, to the start of the reflood quench. It is believed to be the "dry" release period, lasting 24.5 s. The second release period is the reflood release. It lasted 186 s, beginning with the first indication of final quench, and ending with closing of the hot leg quick opening blowdown valve (QOBV). This reflood phase includes a 92 s period of subcooled fluid transport to the BST that was found to have negligible influence on the fission product release fraction during this phase. The third phase of fission product release in this experiment is the leaching phase where water hydrolyzes nonvolatile iodine and the cesium compounds deposited on the inner surface of the cladding and outer surfaces of the fuel pellets. The dissolved iodine and cesium ions are transported from the fuel gap into the PCS liquid.

The fission product gap inventory was determined by two independent methods including measurements and experimental analyses of samples from the two unpressurized higher enriched fuel rods, and application of computer codes such as FRAPCON2 and ORIGEN2.

Experimentally determined iodine gap inventories agree reasonably with FASTGRASS code calculations, while remarkable differences are found when comparing the calculated release fraction of iodine with the experimentally determined value. Presently, no model exists that correctly considers essential parameters like heat generation rate, burnup, and others in the gap inventory calculation.

The BST, including the header, (see Appendix A) simulated the containment of a power plant. The fission products released from the gap of the burst rods were distributed between the BST and the primary system. Based on measurements of the total release of fission products, it was assessed that more than 90% of the noble gases released from the fuel gap were transported to the BST. Volatile isotopes such as ^{131}I and ^{133}I were partially retained in the primary system so that only about 60% of the gap release reached the BST. About 12% of ^{137}Cs of the total release was found in BST and also only a small fraction (7 to 12 %) of the non volatile isotopes ^{140}Ba and ^{90}Sr released as aerosols were transported to the BST. This result indicates that about 40% of the iodine and most of the cesium was retained in the primary system during the burst, reflood, and leaching periods.

The LP-FP-1 experiment extended the fission product database for the assessment of loss-of-coolant accidents to longer fuel rods (1.68 m). However, data use is limited due to the low burnup of the fuel (1417 MWD/MTU, partly compensated by the high average linear heat generation rate in the pre-experiment irradiation phase).

5.4.3 Code Analyses. A number of codes were used in the analysis of LP-FP-1. The results are presented in some detail in the four volumes of the EASR and are summarized below:

- A number of calculations were carried out using the thermal-hydraulic code DRUFAN-02, but with a reduced number of nodes in order to set limits to the magnitude of the unintentional ECC injection.

- A number of studies were carried out with TRAC-PF1/MOD1 to investigate the hypothesis that the unintentional ECC injection was driven by the expansion of a trapped nitrogen bubble in the ECCS line. These calculations were shown to be consistent with experimental data from flowmeter FT-P120-31 (EASR Vol. 3)[44] and were then combined with this experimental data to provide a best estimate prediction of the time variation of the injection flow.

- During the investigation, it became apparent that there were important three-dimensional flow effects in the vessel and that one-dimensional codes such as DRUFAN-02 and RELAP5/MOD2 would not be able to predict these effects. As a result, DRUFAN-02 was only used to provide a qualitative analysis. The RELAP5/MOD2 calculations described in the EASR show that this code also was not able to model the strong spatial variations in fluid flows seen in this experiment. The core thermal response was however well predicted.

- It was concluded that TRAC-PF1/MOD1 was the best code available for post-test analysis because it provided three-dimensional modelling of pressure vessel flows. Two sets of calculations were carried out, the first by the FRG in collaboration with the UK and the second by Spain. The Spanish calculation used a standard version of TRAC, but the FRG/UK made a number of modelling changes, the most important being an increase in the minimum film boiling temperature. The flow patterns obtained from both calculations were closely similar and this gave some confidence in their accuracy. This was important because much of the flow data needed to support the fission transport analysis e.g., the flows in the center fuel module, are not measured and must be obtained by calculation. A major result is that the vapor flow in the CFM where the ruptured fuel rods are located (0.66 m) is vertically upwards above the point of maximum cladding temperature and vertically downwards below this point. Because this point is probably close to the point of clad rupture, this means that the released fission products travelled partly up the CFM and partly down. Those travelling downwards were transported by the steam flow to the peripheral region below the core and then rose through this periphery and out through the broken loop hot leg. This may be the reason why the released fission products were not detected at one of the steam sample measurement stations and on the deposition coupons which are situated either in the upper part of the CFM or above this in the upper plenum.

- Burst calculations were carried out using BALO-2A as described in EASR Vol. 3.[44] Although only a single rod is modelled, these calculations can be used to study the influence of such factors as temperature fluctuations and radial power distribution.

- Calculations to determine the formation of the most important chemical species during the dry phase of the experiment, their transport and deposition, and the implications of any aerosol formation were carried out by Spain and are described in the EASR Vol. 4.[44] The calculation methodology is based on the sequential use of a number of different codes

  — ORIGEN-2
  — FASTGRASS—VPF
  — SOLGASMIX-PV
  — TRAP-MELT-2.2

  using flow and temperature values inferred both from experimental data and from calculations.

- FASTGRASS gap inventory calculations for iodine are in reasonable agreement with measurements on intact pins, but those for cesium are a factor 2 to 3 lower. The code is not well validated for such low fuel irradiations.

5.5 Fission Product Release Experiment LP–FP–2

5.5.1 Experiment Description and Conduct. Experiment LP–FP–2, performed on July 9, 1985, was the second fission product release and transport experiment of the OECD LOFT Project, and the last experiment conducted in the LOFT facility. The principal objectives of the experiment were to determine the fission product release from the fuel during a severe fuel damage scenario and the subsequent transport of these fission products in a predominantly vapor/aerosol environment.[47]

The fission test was the largest severe fuel damage experiment ever conducted, and serves as an important benchmark between smaller scale tests and the TMI–2 accident. Many similarities have been identified in the materials behavior observed in the LP–FP–2 test and those reported for both the smaller experiments and the TMI–2 accident. Together, these data provide an important link between the chemical and physical models of material behavior and the detailed description of severe core damage events. Appendix B provides a comparison between the LP–FP–2 experiment results and the TMI–2 accident.

The complete documentation of the OECD LOFT LP–FP–2 Experiment, including interpretation of the measured results and comparisons to the best available computer code calculations, will serve as an important code assessment basis and accident evaluation reference in the coming years. The conclusions drawn from this analysis highlight the lessons learned from the experiment and uncertain areas needing further investigation. The results of this experiment and the results of the detailed analysis are presented in four sections: Experiment Conduct, Fission Product Analyses, Results of the CFM Examination, and Computer Code Assessment.

The thermal–hydraulic conditions for the experiment were generated by a simulated interfacing systems LOCA, a hypothetical event first postulated in the Reactor Safety Study, WASH–1400 and labeled as a V–sequence. A V–sequence accident is defined as a break of a LPIS pipe outside the containment with simultaneous failure to isolate the system. This accident scenario allows discharge of the primary coolant outside of the containment and therefore potential transport of fission products directly to the environment. PRA studies have shown that this interfacing system LOCA represents a significant contribution to the risk associated with PWR operation, and consequently, a dominant accident sequence was selected as the thermal–hydraulic event for the LP–FP–2 experiment. The piping that simulated the LPIS line was attached to the broken loop in the hot leg. This unusual location for the simulated LPIS line (ECCS lines are usually attached to the cold legs of the primary system loop piping) was required because of difficulties with definable fission product transport geometries and instrumentation problems with the cold leg piping. The simulated LPIS line was opened later in the experiment (221 s) after significant amounts of coolant were removed and the system pressure decreased to the operational range of the LPIS line instrumentation. The initial depressurization and coolant removal was achieved via a break located in the cold leg of the operating loop. This allowed faster coolant depletion, maximizing the use of the decay heat difference between the CFM and the peripheral fuel modules for faster fuel cladding heatup in the CFM. Also, this experiment procedure provided thermal–hydraulic processes within the PCS and coolant distribution similar to processes occurring during a LPIS line break in a commercial power plant.

For the experiment, a special center fuel module (CFM) was designed and fabricated (see Appendix A). This CFM had a geometry typical of all other LOFT CFM's, except that the two outer rows of fuel rods were replaced with a 2.54–cm thick thermal insulation shroud. This design was necessary to enable the CFM fuel rods to heat above 2100 K while maintaining the peripheral fuel rods below 1390 K preventing their damage. The module consisted of 100 prepressurized (2.41 MPa) fuel rods enriched to 9.744 wt% ^{235}U, and 21 zircaloy guide tubes, of which 11 contained stainless steel clad control rods.

Measurements were made during the test to monitor the thermal–hydraulic and fission product behavior. The specially designed fission product measurement system (FPMS) consisted of four major types of measurement devices:

- Three steam sample systems.
- Four gamma spectrometers and one gross gamma detector.
- Six deposition coupons and two deposition spool pieces.
- An aerosol collection filter on the LPIS line.

A more detailed description of this system, and of the CFM and break line piping is provided in Appendix A.

The experiment consisted of four distinct phases:

- Fuel preconditioning
- Pretransient
- Transient
- Post–transient.

The purpose of the fuel preconditioning phase, in conjunction with the pretransient phase, was to subject the CFM fuel rods to a minimum burnup of 325 MWD/MTU. The actual burnup achieved during these phases was 448 MWD/MTU. The pretransient phase was designed for instrumentation checkout and preparation, for establishment of the initial experiment conditions, and for completing the burnup. The transient phase is defined as the period between the reactor scram (time 0) and reflood quench of the core at 1795 s.[48] The post–transient phase consisted of an interval of 44 days during which the redistribution of fission products in the gas and liquid volumes in the blowdown suppression tank and the leaching of fission products from the damaged fuel rods in the CFM were measured.

The thermal–hydraulic conditions of the LP–FP–2 experiment were achieved by utilizing two main break lines [Intact Loop Cold Leg (ILCL) and the LPIS], as well as a minor break pathway from the PORV. During the fission product release and transport portions of the experiment, only the LPIS line was open. Figure 5.13 shows the measured PCS pressure overlayed with the primary sequence of events.

The LP–FP–2 transient was initiated by scramming the reactor with the peripheral module control rods. The primary coolant pumps were then turned off and the PCS flow decreased to a point where the CFM control rods could be dropped. At 24 s, the CFM control rods were fully inserted in the core. The ILCL break was then opened at 33 s, and the LPIS line was opened at 221 s. The peripheral core heatup began at 662 s, and the CFM heatup began at 689 s. The ILCL break was then closed at 736 s; however, the PCS depressurization rate did not decrease as rapidly as planned and it became necessary to reopen the ILCL break line at 878 s, in addition to initiating the PORV break at 882 s. After the PCS pressure dropped below the designed operating pressure of the Fission Product Measurement System (1.38 MPa or 200 psi), the ILCL and the PORV lines were closed at 1022 and 1162 s, respectively.

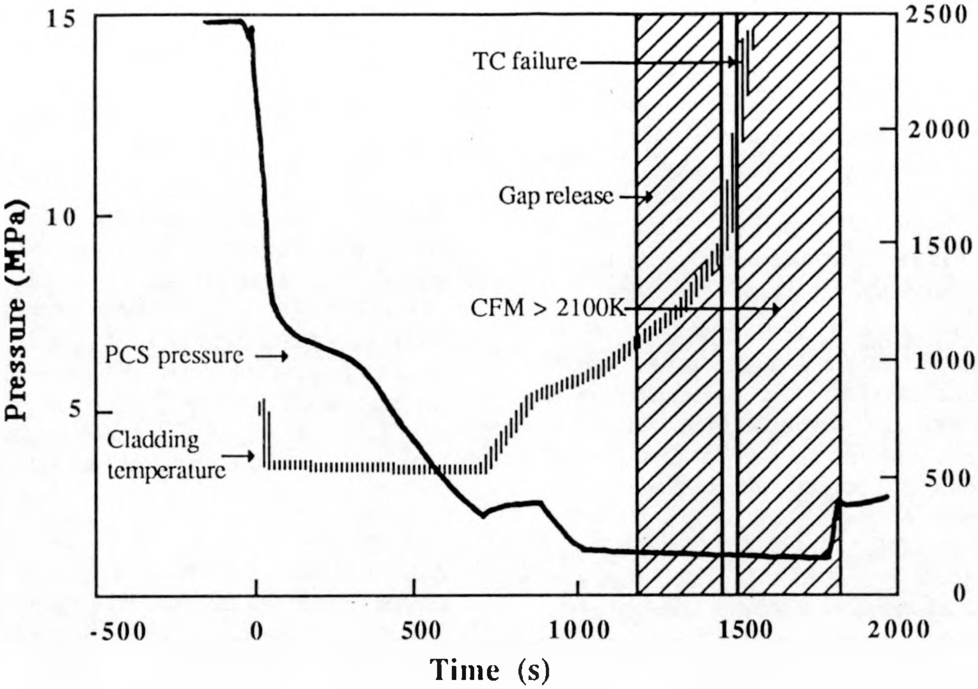

Figure 5.13. Measured primary system pressure and peak cladding temperature during experiment LP–FP–2.

From 1163 s to the end of the transient (1778 s), the LPIS line provided the only pathway for fission products to be transported outside the PCS. Measured cladding temperatures exceeded 2100 K at 1504 s and the peak outer shroud wall temperature reached the test limitation condition of 1517 K at 1766 s. The LPIS and FPMS lines were then isolated at 1778 s, and the transient was terminated at 1783 s with the injection of water from the ECCS. For more than 270 s, the CFM experienced temperatures in excess of 2100 K. Thermocouples used in the CFM were calibrated as high as 2100 K. However, many of the CFM temperature measurements were affected by thermocouple cable shunting effects before the temperature at the thermocouple location reached 2100 K. Cable shunting is defined as the formation of a new thermocouple junction (virtual or real) on the thermocouple cable resulting from exposure of the cable to high temperature. Detailed assessment of the thermocouple behavior in this experiment is provided in Appendix C of the EASR.[49]

The highest measured temperatures in the CFM were initially reached at the 42–in. elevation. Later, the highest measured temperatures were reached at the 27–in. elevation (maximum power density). These maximum measured temperatures were reached early in the 270 s high–temperature interval. Consequently, actual peak temperatures were probably several hundred K higher than 2100 K. This conclusion is confirmed by the CFM postirradiation examination that shows material formations consistent with temperatures in the range of 2800 K and localized regions above 3000 K. The SCDAP/RELAP5 code calculated peak temperatures for the hot region (corresponding to the 42–in. elevation) to be 2800 K.

The temperatures measured axially through the CFM upper structure showed that during the fast heatup of the CFM the flow through these structures was oriented upward. Also, these temperature measurements indicated some mixing flow from the peripheral fuel modules entering the CFM upper structures. All of the CFM outflow went through the CFM upper structure to the nozzle level where it mixed with flow through the corner modules and entered the broken loop hot leg (BLHL). The SCDAP/RELAP5 code also calculated this behavior.

During the 270 s of high temperatures in the CFM, steam flow paths existed through the fuel module eventhough it was undergoing structural failure and relocation as evidenced by the CFM sectioning and by the measured temperatures at the 10–in. elevation (which survived the transient). The postexperiment configuration of the CFM was highly nonuniform at all elevations. As a result of the nonuniformity of the relocated materials, there were steam flow paths through the CFM that persisted throughout the transient. This finding indicates that ECCS operation in a severe accident will remain effective in rapidly cooling a PWR core. This was also demonstrated by the rapid reflood and cooldown of the LOFT core and CFM. The large debris volumes in the CFM cooled at slower rates and underwent cracking from thermal gradient stresses.

5.5.2 Fission Product Analyses. This section describes the analyses of the experiment carried out for the EASR[49] which was prepared as the Option 5 contribution by the United States. To assist the analysis of the experimental data, several state–of–the–art computer codes were utilized. Figure 5.14 shows a flow chart of computer codes and models used in these analyses. Best estimate and sensitivity calculations were conducted and compared with the measured data. The calculation results were used to gain insights into the dominant thermal–hydraulic and fission product phenomena and to form the basis for the detailed description of the test and the conclusions reached.

The release of fission products from the cladding gap of ruptured fuel rods was first detected in the steam sample lines at 1200 $\pm$ 20 s, and in the LPIS line by 1249 $\pm$ 60 s. Fission product release from the fuel was first detected at 1500 $\pm$ 10 s. A gross gamma detector was used to provide general timing and magnitude information on the fission products in the F1 line, which used the space immediately above the CFM for the sample. A gamma spectrometer in the LPIS line (G5) measured both volatile and low volatile fission products in this line, but almost no noble gases were identified because of higher than anticipated detection threshold limits. The detection threshold limits were increased by deposition of other fission products near the spectrometer, especially iodine. During the post–transient phase, spectrometers on sample lines from the lower plenum (G1) and from the liquid space of the BST (G3), measured the concentrations of noble gases, volatile fission products, and activation products in the PCS and BST liquid.

The postexperiment examinations of the FPMS components indicate that major fission products such as ^{103}Ru, ^{131}I, ^{137}Cs, ^{140}Ba, ^{141}Ce, and ^{144}Ce were deposited at several locations and measurements were made to quantify these fission products throughout the LOFT system. Results of these measurements are provided in detail in the Data Report.[50] Data analyses and fission product inventory calculations showed that transient releases were limited to 2 to 5% of the initial CFM inventory. The primary reason for the small releases was the initially large grain structure of the fuel (~12 μm). The major release of volatile fission products occurred during reflood and amounted to about 14% of the initial CFM inventory.

Analyses of the postexperiment, online data, and computer code analyses indicate that the primary fission product chemical forms during the LP–FP–2 transient were CsOH and AgI. CsI was not present in significant concentrations as shown by the upper plenum protected coupon data (these coupons were available for deposition only during the dry phase of the transient). Large quantities of Ag and I, but very small quantities of Cs were deposited on these coupons during the transient phase.

Analyses indicated that fission product deposition in the upper plenum was dominated by AgI condensation on surfaces. About half of the AgI condensed and deposited low in the upper plenum and small aerosol particles transported most of the remaining AgI through the system piping to the collection tank. CsOH is more volatile than AgI and did not condense on surfaces in the lower sections of the upper plenum because of temperature and concentration conditions. The two protected upper plenum coupons were hot enough that CsOH did not condense on either coupon; however, the small amounts of Cs that were detected on the unprotected and protected coupons is characteristic of CsOH reacting with stainless steel.

Iodine detected in a section of the LPIS line that was analyzed, had characteristics of an insoluble species (e.g., AgI). This piping section was washed by an alkaline solution of potassium permanganate ($KMnO^4$) with NaOH, and then by a solution of nitric and oxalic acids. Only 41% of the iodine that was removed by the two wash process was removed with the alkaline solution. Had the iodine been present as a soluble compound (e.g., CsI), then nearly all of it would have been removed with the first washing.

The large quantities of Ag detected in the upper plenum indicate that the control rods failed because of high–pressure bursting, which sprayed a fine material molten alloy out into the CFM on adjacent rods and spacer grids.

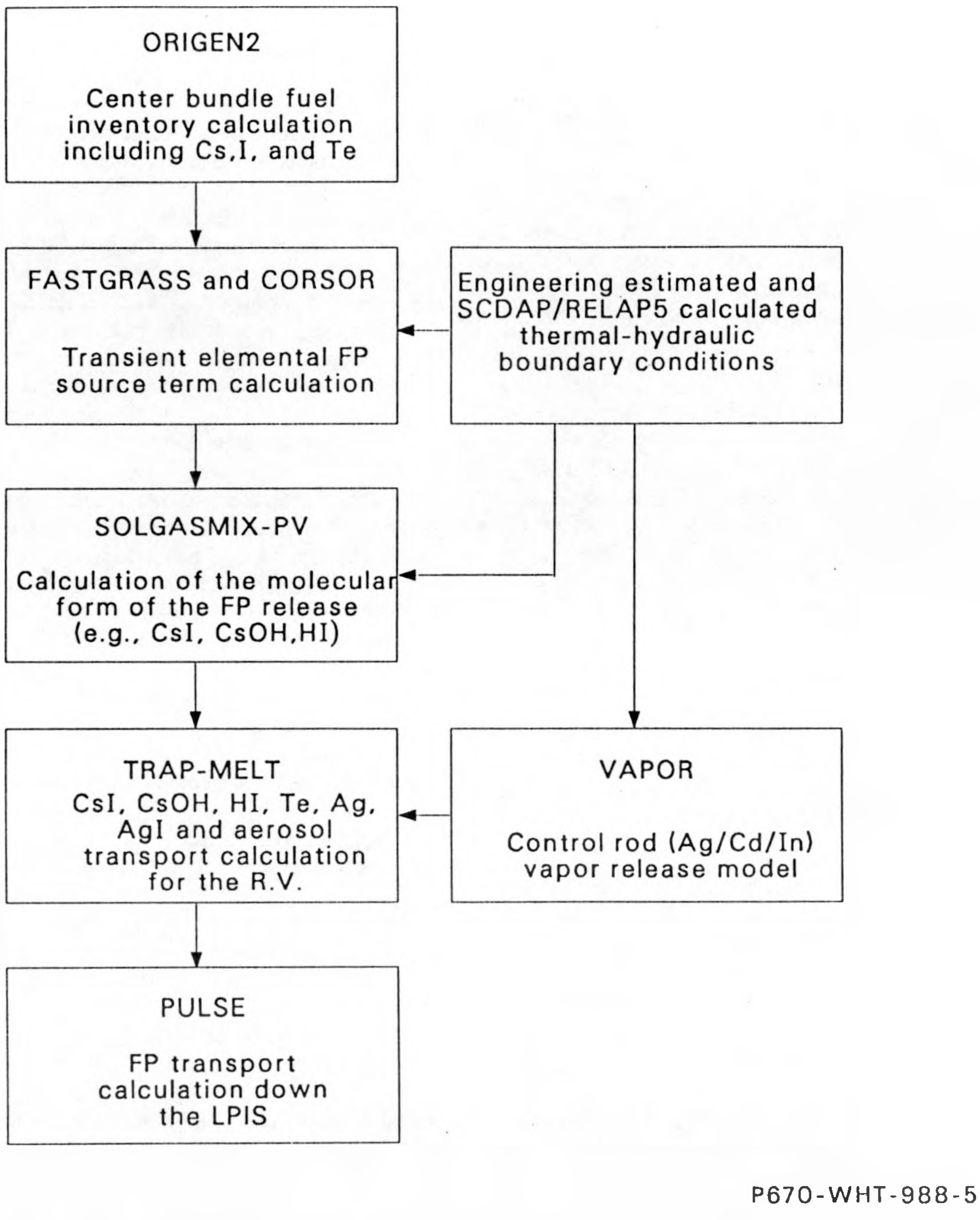

Figure 5.14. Analysis Flowchart.

Some sprayed material was probably transported immediately to the upper plenum as aerosol material Ag deposited in the CFM evaporated when core temperatures exceeded 2400 ~ 2500 K. During the high temperature portions of the experiment the vaporized Ag provided the source material for gas phase reactions with released iodine.

Transport of fission products in the BLHL and LPIS was dominated by condensed species on small, (~0.3 μm) probably liquid Sn aerosols. Only a small fraction (~25%) of the fission products that entered the LPIS were deposited in this line. Most of these fission products and Sn aerosol material were transported to the BST. The BST inventory of fission products accounted for: 2% Kr, 1.7% Xe, 0.9% I, and 0.23% Cs. The release fractions to the LOFT containment vessel amounted to 1.3% for Xe, 0.068% for I, and 0.035% for Cs.

A large quantity of H_2 gas was generated during the LP–FP–2 experiment, equivalent to the oxidation of ~58% of the zircaloy. Approximately 1024 ± 364 grams were produced and 205 ± 11 grams were detected in the BST. The majority of released H_2 was trapped in gas bubbles in the PCS. Either the H_2 was released during the transient and then trapped in the PCS, or a large fraction was generated during reflood.

Most of the noble gases released during the transient were transported through the LPIS to the BST. However, the concentration levels in the LPIS were too small to detect these nuclides with the G5 gamma spectrometer because of the presence of I and Cs deposition in the viewed section of the piping.

The primary release pathway for noble gases to the containment was through the G2/BST sampling line. The decrease in BST gas inventory can be correlated with increased containment activities. The primary release pathway for I and Cs to the containment vessel appears to take place in liquid leaks around the traversing incore probe (TIPs) tubes.

5.5.3 Results of the CFM Examination. Following the successful completion of experiment LP–FP–2 it was decided to perform postirradiation examinations of the fuel bundle to provide additional information determining the factors that may have influenced the thermal–hydraulic and fission product behavior. These postirradiation examinations also expanded the scope of this experiment to include material behavior and interactions occurring within a fuel bundle during a severe core accident. This section presents the results obtained from the postirradiation examinations of the LP–FP–2 center fuel module.[51]

During the experiment, the peak temperatures exceeded the goal of 2100 K for approximately 4.5 minutes, with localized peak temperatures exceeding the melting point of the UO_2 fuel (3120 K). Both nondestructive and destructive examinations of the LP–FP–2 fuel module were performed. The nondestructive examinations included visual examinations of the exterior surface of the fuel bundle, gross and isotopic gamma scans of the overall fuel bundle, and neutron radiographs at two perpendicular orientations through the fuel bundle. The destructive examinations entailed sectioning the fuel module to provide 21 transverse cross sectional surfaces for metallographic examination. Figure 5.15 provides a typical cross sectional sample as prepared for the examination. Small core bore samples obtained in specific areas from these metallographic samples were selected for scanning electron microscope wavelength dispersive spectroscopic examination, as well as radiochemical analyses for fission product retention.

The postirradiation examinations revealed that the relocation of material within the fuel module resulted in the formation of distinctive regions very similar to those observed in smaller scale severe fuel damage experiments, as well as in the TMI–2 accident. Near the bottom of the fuel module was an accumulation of relocated metallic melt material and fuel debris. The metallic melts consisted of low melting eutectic phases, primarily composed of silver and zirconium, with iron, chromium, and nickel. This material was the first material to relocate during the experiment, and occurred after failure of the control rods released the (Ag, In, Cd) control material alloy in the form of aerosols and liquids. The silver was able to interact with the zircaloy cladding and cause it to liquefy at temperatures well below its melting point of 2030 K. The zirconium in the resulting Ag–Zr melt was subsequently able to interact with the stainless steel cladding on the control rods, and the Inconel spacer grids, causing these components to also liquefy at temperatures below their melting point (~1725 K).

In the central portion of the fuel bundle was a large blockage of previously molten (U, Zr) O_2. This material formed as a result of interactions between molten zircaloy and the UO_2 fuel, along with oxidation of the resulting (U, Zr, O) by steam passing through the bundle. This material relocated later in the test than did the metallic melt material in the lower blockage region, with significant amounts apparently relocating and oxidizing after the onset of reflood when large quantities of steam were available, and the thermocouples indicated large temperature excursions. Temperatures exceeded fuel melting (>3120 K) in the center of this blockage region, with the molten liquid phase contained within a shell of solidified $(U,Zr)O_2$. Thermocouple data indicates that temperatures remained hot in the central portion of this melt region for a few hundred seconds following reflood. This is very similar to the behavior that occurred in the TMI–2 accident.

In the upper portion of the fuel bundle was a debris bed that was primarily composed of fuel pellets without any intact zircaloy cladding to restrain them. Small amounts of ceramic melt material between the fuel pellets held this debris bed together. This debris bed rested on top of intact rod stubs that protruded from the $(U, Zr) O_2$ blockage region. The formation of similar debris beds has also been observed in smaller scale severe fuel damage experiments, and in the TMI–2 accident.

Extensive liquefaction and oxidation of the stainless steel upper tie plate in the upper end box of the fuel module occurred. Fuel fragments and molten materials relocated upward from the fuel module and deposited in this region, interacting with the upper tie plate. Thermocouple data indicates that the only time in the experiment when temperatures exceeded 1000 K, and in which this liquefaction and oxidation could have occurred, was after the onset of reflood. The LP–FP–2 fuel module was reflooded from the bottom up over a period of about 13 seconds, but the thermocouple data from several locations throughout the bundle indicates that temperatures actually increased after the onset of reflood. This suggests that the large amounts of steam that became available after the onset of reflood resulted in extensive zircaloy oxidation, in turn, generating large amounts of heat. The fuel fragments and molten materials which deposited on the upper tie plate would have relocated during this reflood period.

Of the five Inconel spacer grids in LP–FP–2, only the bottom one remained completely intact, and the next highest one was only partially intact. The three uppermost spacer grids were completely liquefied during the experiment. Material interactions observed at the partially intact second spacer grid indicated that Zr–Ni eutectic interactions resulted in liquefaction of the spacer grids below the 1725 K melting point of the Inconel 718 spacer grid, probably around 1400–1500 K. Relocating material accumulated on both of the remaining spacer grids, which indicates that spacer grids tend to impede material relocation until they ultimately fail. The greatest flow blockages found in the LP–FP–2, 78–86% of the available flow area, were located through, or just above, the remaining spacer grids.

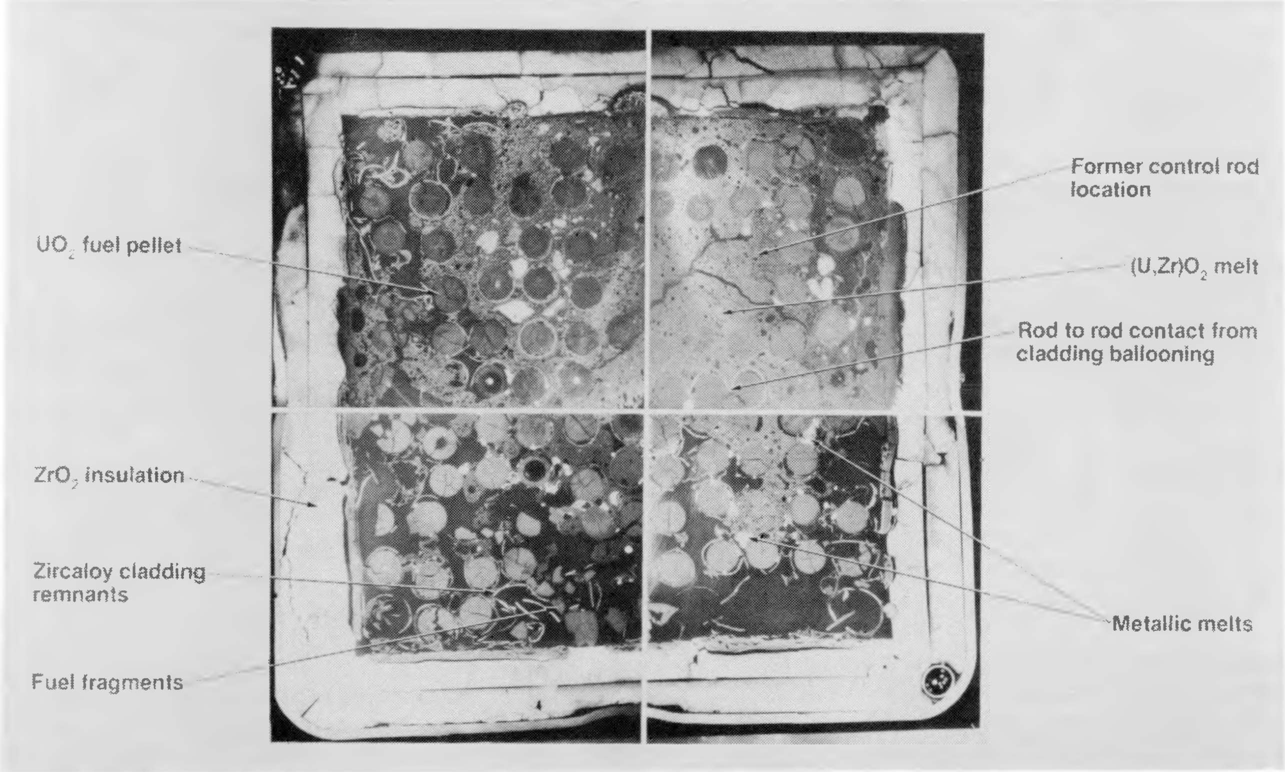

Figure 5.15. Cross–sectional view of the LP–FP–2 center fuel module showing typical postexperiment configuration.

Approximately 70% (10 kg) of the available (Ag, In, Cd) control rod material alloy was released to the bundle either as an aerosol or as a molten liquid. The lack of zircaloy in the upper portion of the bundle, and the accumulation of Ag–Zr metallic material in the lower portion of the bundle, indicates that much of the silver interacted with unoxidized zircaloy cladding and liquefied it at temperatures above 1400 K. The stripping of the zircaloy cladding in the upper portion of the bundle contributed to the formation of the fuel pellet debris bed in this region.

Cladding ballooning and rupture occurred throughout the central portion of the fuel bundle, resulting in rod– to–rod contact and fusion of the oxidized cladding remnants. Throughout this region only a fully oxidized outer shell of the original zircaloy cladding was still intact. The unoxidized inner surface of the zircaloy had melted and relocated.

Fuel grain boundary separation occurred throughout the higher temperature portions of the fuel bundle. Pore migration, fission gas diffusion, and grain boundary coalescence contributed to this phenomenon, but the separation was enhanced in fuel adjacent to metallic melts. This suggests that reduction of the fuel near these metallic melts resulted in the formation of hypostochiometric UO_2 and liquid metallic uranium on these grain boundaries. The liquid phase would tend to weaken the grain boundaries and cause grain boundary separation. Fuel fragmentation and powdering were also observed throughout the highly damaged regions of the fuel bundle. This evolved as a result of the factors causing grain boundary separation, but was also probably enhanced as a result of thermal shock at reflood.

Fuel liquefaction occurred as a result of interactions between molten zircaloy and the fuel at temperatures above approximately 2200 K. Interactions between some fuel pellets and molten iron oxides also resulted in the liquefaction of some fuel, and the formation of foamy fuel structures with large amounts of porosity. Porous fuel structures indicative of temperatures near fuel melting were observed in the high–temperature regions of the bundle where molten (U, Zr) O_2 surrounded the fuel pellets, and in the core of this high–temperature region the fuel pellets were completely gone, suggesting temperatures in excess of the 3120 K melting point of UO_2.

Measurements from the metallographic cross sections indicated that less than 20% of the fuel had liquefied, but that approximately 65% of the zircaloy cladding and inner liner had liquefied. Most of the liquefied zircaloy and fuel were contained in the high–temperature ceramic melt region. Material balances based on density measurements and elemental analysis of core bore samples could account for all the fuel and zircaloy in the fuel bundle.

The amount of hydrogen generated as a result of zircaloy oxidation was estimated to be 862 g, with lower & upper limits of 575 & 1069 g. The nominal value corresponds to 49% of the available zircaloy. An additional 163 ± 83 g of hydrogen could also have been generated as a result of oxidation of stainless steel and Inconel components in the fuel bundle, bringing the total to 1025 g, with lower & upper limits of 655 & 1310 g of hydrogen. This total amount agreed very well with the 1024 ± 364 g estimated from grab samples taken from the blowdown suppression tank, and estimates of the amount of hydrogen in the primary coolant system. Those estimates included 205 ± 11 g in the blowdown suppression tank, and 819 ± 364 g in the primary coolant system. The amount in the blowdown suppression tank is indicative of the amount of hydrogen generated during the transient phase, and the amount in the primary coolant system is indicative of the amount released upon reflood. This also agrees well with the minimum amount of 181 g (+109, –91) g H_2 that was generated as a result of oxidation of the intact cladding shells and the material in the lower blockage region, and hence can be assumed to have been generated during the transient. The remainder of the oxidized zirconium in the molten regions in the upper portions of the fuel bundle indicates that the oxidation probably occurred during the transient, with large amounts apparently occurring after the onset of reflood when large amounts of steam and water were available.

The as–fabricated fuel grain size was 14 μm, and most of the intact fuel at the end of the experiment had this same grain size. Grain growth was observed in the center of some fuel pellets in the high temperature portion of the fuel bundle, with an average grain size of 27 μm and a 2–sigma standard deviation of 17 μm.

The results from these postirradiation examinations provide additional insight into the factors influencing the thermal–hydraulic and fission product behavior in the LP–FP–2 experiment, as well as providing data on material behavior occurring during a severe core accident. The data from the LP–FP–2 experiment provide a valuable link between smaller scale experiments and the TMI–2 accident.

In fact, based on these examinations and results of the fission product analyses,[49] INEL proposed that much of the CFM damage occurred during the reflood portion of the experiment and was associated with a rapid temperature excursion caused by enhanced metal–water reaction.

This thinking was stimulated by the fact that the EASR amount of generated hydrogen was in total about 1 kg but only 200 g of this amount was found in the BST. The primary system was isolated from the BST prior to reflood, therefore, the hydrogen in the BST was generated during the transient portion and most of the hydrogen trapped in the primary system had to be generated later because there is no particular reason for large hydrogen retention in the primary system when the LPIS line was open.

The steam mass flow rates through the CFM during the transient were assessed to be in order of 10 to 15 g/s. If all steam would react at these flow rates with the CFM metal, only 400 ± 150 g of hydrogen would be produced during the transient. Therefore, the amounts of hydrogen in the primary system could be produced only during the reflood when the flow rate increased to approximately 5 kg/s.

If these large amounts of hydrogen (larger than during the transient portion) were generated during reflood, then an association had to exist with higher CFM temperatures than the temperatures occurring during the transient. Centerline thermocouples indicated that during the transient, the fuel temperatures did no exceed 2400 K, but after the reflood front entered the CFM, the same thermocouples indicated rapid temperature increases up to 2970 K before failure.

The Postirradiaton Examination (PIE) of the center fuel module indicated significant thermal damage of the upper–tie plate. The thermocouples at this plate indicated temperatures less than 1000 K during the transient, and rapid temperature increase after reflood initiation, with temperatures in excess of the stainless steel melting point of 1700 K. The fuel centerline measurements and the upper tie plate thermocouples both indicate that peak temperatures within the CFM were reached during the reflood period and not during the transient portion of the experiment. These high temperatures in the bundle could be only the result of the reflood water reacting violently with the hot zircaloy. This exothermal reaction, releasing a large amount of hydrogen, elevated the temperatures within the bundle to a degree that the escaping hot gases caused significant melting of the massive upper–tie plate. At the same time, the steam flow rates and temperatures in the bundle were sufficient to relocate fuel pellets and melts upward to the upper–tie plate. The temperatures within the bundle exceeded locally, as the PIE indicated, fuel melting temperatures (3120 K). The PIE also indicated that the oxide shells and the lower blockage (mainly metallic melt) released approximately 180 g of hydrogen, a measurement close to the 200 g measured within the BST. The oxide layers and the lower blockage certainly occurred during the transient phase of the core heatup. The ceramic melts, melt in insulation, the stainless steel, and Inconel yielded approximately 818 g of hydrogen, a measurement close to the 819 g of hydrogen assessed for the primary system. Assessment of fission product releases and the amounts found in BST also indicate that most of the release happened during the reflood phase when the fuel went through enhanced heatup.

5.5.4 Code Analyses.

SCDAP/RELAP5

The SCDAP/RELAP5 computer code was used to calculate the overall thermal–hydraulic boundary conditions required in the fission product release and transport analysis. In particular, the SCDAP/RELAP5 code provided center bundle and upper plenum flow rates, and control rod and fuel temperatures in the Center Fuel Module (CFM).

The results of the SCDAP/RELAP5 calculations generally agree with the experimental data. Despite the complexity of the LP–FP–2 experiment, the calculations provided good resolution of the thermal–hydraulic processes during this experiment within the limits of the nodalization. All collected experimental data indicate complex flow patterns within the core and upper plenum structures. Because of the relatively coarse nodalization (compared to the characteristic dimensions of the flow patterns) it is impossible to reproduce all measured data, but the SCDAP/RELAP5 code does closely approximate most of the measured temperatures in the upper plenum.

The calculated flows everywhere in the core during the high temperature transient are upward. The flows calculated by the SCDAP/RELAP5 code revealed a very high sensitivity to the definition of the LOFT system model as evidenced by calculation of upflow/downflow loops (involving the upper plenum and core) that were dependent on definition of resistances between vertical and horizontal volumes. The measured temperature data was used as the reference base for improvement of the LOFT model. The reported SCDAP/RELAP5 calculation of the LP–FP–2 transient is in good agreement with the reference database.

The calculated hydrogen production was less than half of the ~ 1 Kg determined from measured data. Possible reasons for the low results are: hydrogen was generated during reflood, but the reflood phase was not included in the calculations; hydrogen was produced during and following melt relocation, but this phenomena was not modelled; or the calculated CFM flows were too low, maintaining steam starved conditions for too long a time period.

ORIGEN2

The calculated Cs to I ratio for the CFM was 4.2 and the measured ratio for the PCS was 5.2 ± 16%. Except for a few isolated species like ^{134}Cs, the overall uncertainty in the ORIGEN2 results, for key isotopes, is ± 10%. ORIGEN2 calculations for the BST indicate that a best–fit to the measured data can be made assuming release fractions of 2% (Kr), 1.7% (Xe), 0.9% (I), and 0.23% (Cs).

CORSOR

Using the SCDAP/RELAP5 CFM temperatures, CORSOR overpredicts the fission products release prior to reflood by a factor of ~15, and the total release by a factor of ~4.

FASTGRASS

Based on several sensitivity calculations made by FASTGRASS for different liquefaction/dissolution assumptions and different grain sizes, and utilizing the SCDAP/RELAP5 CFM temperatures as input to the code, the best–estimate FASTGRASS calculated transient release fractions are 1.5 to 3.5%. These results generally agree with the measured release fractions based on the BST data.

VAPOR

VAPOR calculations, assuming SCDAP/RELAP5 boundary conditions, yielded a Cd release equivalent to prior measurement, but underestimated the Ag release. The total silver release was probably affected by control material spraying instead of vaporization of relocating and relocated control rod alloy. Note: no control rod spray model exists for VAPOR.

SOLGASMIX–PV

Based on the Gibbs free energies of formation where temperature conditions are <1700 K, where Cs/I ratios are low (~4) and Ag/I ratios moderate to high (>3), the dominant chemical species for I is AgI. CsI is calculated to exist in only very small concentrations.

At higher temperatures, HI and I are the dominant I species. At conditions where little Ag is present, CsI appears to be the dominant I species. CsOH is the dominant Cs species under all cases.

TRAP–MELT2

Consistent agreement was obtained between the observed upper plenum deposition data and the code calculated result when a release fraction of 2% was assumed for I and Cs, and the chemical form of the release is AgI and CsOH. The upper plenum coupon data are not consistent with calculations made with HI, I_2, CsI, or $CsBO_2$.

The primary deposition phenomena observed for the TRAP–MELT study was: (a) condensation of AgI, (b) reaction of CsOH with structural surfaces occurring at higher and cooler locations in the upper plenum, (c) reaction of CsOH with metal in the CFM, (d) condensation of CsOH and AgI onto Sn aerosol material between the upper plenum and BLHL, and (e) aerosol deposition in the CFM upper structures (namely on the two protected coupons) by thermophoresis. Gravitational settling is the dominant aerosol deposition mechanism high in the upper plenum.

PULSE

Aerosol deposition in the LPIS is dominated by turbulent deposition phenomena. Because of the high velocities (Re > 300,000), and relatively high temperature (~500 K), most of the FPs and aerosol material that entered the LPIS line passed through and were not deposited in the pipe or the F3 filter.

In addition to the EASR analyses, a number of thermal–hydraulic calculations were performed by the project members. The results of these calculations are summarized in Volume 1 of the Comparative Analysis Report, Reference 51. The following table shows the organizations and the computer codes used in this post-experiment analyses:

Table 5.8 LP–FP–2 Project Members Analyses

Country	Organization	Code
Finland	VTT	RELAP5/MOD2
Italy	Pisa University	SCDAP/RELAP5
Spain	ENUSA	RELAP5/MOD2
Switzerland	PSI	RELAP5/MOD2
USA	EPRI	MAAP

The comparison of these calculations showed highly developed capabilities to determine the thermal–hydraulic conditions during the early stages of a severe core damage accident. The overall system response was well calculated. The major uncertainties and differences between the calculations were the result of the break flow. Experiment LP–FP–2 had three discharge paths which were operated at different times. The critical geometry of these paths was basically unknown, leading to variable assumptions in modelling the breaks. This resulted in differences in parameters such as calculated coolant inventory, or the time of the core heatup. It was concluded that the ability of the codes to calculate the thermal–hydraulic conditions is sufficient for use in fission product transport calculations. The variations in the conditions could affect the fission product transport through the hottest upper plenum region. However, it was judged that this would have negligible influence on the overall transport through the upper plenum.

Figure 5.14 shows the measured cladding temperature in the CFM at 27–in. elevation and the calculational envelope. Despite variation in the heatup timing, all the calculations produced good heatup rates. Heatup associated with rapid cladding oxidation was calculated only by SCDAP/RELAP or MAAP, because the other codes of this comparison do not contain models for oxidation heating.

The second volume of the comparison report discusses fission product calculations. Only two calculations are compared: the EASR calculations discussed previously, and calculations performed by Spain. In the Spanish calculation, the set of codes used in the analysis was: ORIGEN2, CORSOR, FASTGRASS–VFP, FRAPCON, FRAP–T6, SOLGASMIX–PV, and TRAP–MELT2. Where possible, measured thermal–hydraulic data were used in these calculations.

The comparisons show several areas of agreement and also demonstrate some of the more difficult problems facing the analyst of severe core damage events. The timing of clad rupture was calculated accurately. Hydrogen generation calculations agreed within about 25%, but were less than the measured data. The agreement is even better when one considers that the calculations suggest about one–third of the materials released remained suspended in the primary system at the end of the experiment. Control rod failure was quite different from the previously existing models. Large releases of silver liquid spray were not expected and are only modelled in one of the calculations because of the observation of the phenomena in the experiment. The data base for such a model is very limited. This effect can completely change the chemistry and subsequent transport processes.

Prediction of the initial fission product inventory appears to be acceptable for both calculations. The two calculations are within 5% of each other. The PIE retained fission product work indicates that the ORIGEN2 results are very good. The calculations of release fractions show larger differences. Older methods (CORSOR) using empirical release rates are more conservative than more mechanistic methods (FASTGRASS). The calculated fission product release fractions were generally conservative. There was a factor of two difference between results using the same code and a factor of ten between different codes.

Upper plenum transport and deposition comparisons show differences in retention of fission products by a factor of two or three. This is not a large difference, considering the two calculations used different areas and chemical species (in the Spanish calculations, AgI was excluded as chemical specie). In both calculations, the dominant deposition process in the UP was identified as condensation.

Comparison of calculated depositions in the LPIS line also showed a difference of several factors; however, the EASR calculation showed excellent agreement with the measured data. Both calculations indicated that turbulent and inertial impaction were the dominant deposition processes for the LPIS line. Generally the PULSE code provided better deposition calculations than TRAP–MELT2.

The two calculations do not confirm much about chemistry. It seems most likely that AgI was a dominant iodine specie, but no hard evidence exists. The chemistry issue is further complicated by the fact that the iodine could have reacted with silver either in suspension or after the silver plated on upper plenum surfaces.

In conclusion, the calculations have shown agreement in terms of dominant deposition processes. Disagreement is larger in the release fractions. Fission product chemistry remains an important issue with little data available for comparison.

6. ASSESSMENT OF BENEFITS BY INDIVIDUAL COUNTRIES

AUSTRIA

The unique importance of the LOFT facility for nuclear reactor safety research was recognized in Austria in the early days of the USNRC LOFT Program. Even after Austria decided not to use nuclear power for its own electricity generation, neighboring countries continued to install new, light water nuclear power stations. It is, therefore, of utmost importance that Austria maintains sufficient expertise to assess the safety of nuclear power plants close to its borders and participation in the LOFT Program offers direct and indirect opportunity to maintain expertise in the operational safety of light water reactors.

Austria joined this program in 1977, and after its conclusion, became an active supporter of continuing research in LOFT under the auspices of OECD. The continuation of experiments in LOFT and conducting the associated analyses would aid the qualitative and quantitative understanding of physical phenomena during accidents in light water reactors and support the refinement and assessment of computational techniques and system codes that are used as predictive tools in analyzing nuclear power plant safety.

LOFT provided, as no other experimental facility could, the opportunity for the most realistic experiments in an integral system environment. Austria expressed strong interest in the broad experimental program of the OECD LOFT that addressed a wide spectrum of safety–related events ranging from design basis accidents and associated phenomena to events beyond design basis accidents that might potentially lead to large activity releases into the environment. In the Austrian view, such broad programs would be beneficial for the following reasons:

1 Such a program, conducted under wide international scrutiny could be seen as a reflection of reactor safety technology worldwide. This factor is particularly because severe accidents with releases of radioactive substances can have consequences beyond boundaries of the country where they occur.

2 Austria would gain access and participate in a broad and up–to–date program of research on reactor safety and would maintain the experience needed if the Austrian public demand a deeper assessment of the safety of light water reactors abroad.

The eight experiments of the OECD LOFT Program provided important insights in reactor behavior during various accident conditions. Experiment LP–FW–1 provided data on long–term transients, associated heat transfer mechanisms, and phenomena within the pressurizer during discharge through the power–operated relief valve. The small–break experiments LP–SB–1 and LP–SB–2 showed that independent of the pump operation, similar minimum coolant mass inventory is reached for hot leg breaks. Analyses of these experiments indicated inadequacies in modelling flow stratification in large diameter pipes and in the simulation of branch flow under two–phase conditions.

Experiment LP–SB–3, which simulated a cold leg small–break, provided an important data point on accident management strategies by using secondary feed–and–bleed. This experiment also provided important data on core uncovery transients for code assessment. The large–break experiments LP–02–6 and LP–LB–1, which addressed licensing concerns in the United States and United Kingdom, showed that even when severely degraded, the emergency core cooling systems in these LOFT tests were able to quench the core and provide core reflood without damage to fuel elements. Analyses indicated no fuel rod rupture or appreciable cladding ballooning even when peak cladding temperatures reached 1261 K. The fission product experiments increased the data base and understanding of fission product transport from fuel gap (LP–FP–1) and from fuel matrix of a severely degraded core (LP–FP–2). Particularly important is the LP–FP–2 experiment in which local temperatures of 3000 K were reached.

The extended OECD LOFT program provided detailed chemical and metallurgical analyses of the damaged center fuel module extending the general knowledge of core melt progression. It was found that no complete blockage occurred during this transient to simulate an early part of a core melt accident and that the degraded core was easily reflooded and quenched. The distribution of the materials in the damaged bundle had similarities to the TMI–2 postaccident core configuration. This experiment will play an important role, together with evaluation of the TMI–2 data, in the understanding of severe accident progressions and evaluation of source terms for commercial nuclear power plants.

In summary, it is believed that the OECD LOFT Project significantly contributed to reactor safety research and therefore, generally contributed to the safety of nuclear power plants by aiding understanding of a broad range of accident conditions and by extending the data base for testing and validating the analytical methods used in reactor safety assessment. Analyses conducted by the Project members indicate increasing maturity of safety analysis methods and predictability of accidents. In the Austrian view, the most significant value of the OECD LOFT Project is the successful international collaboration on the program definition, analysis, and management.

FEDERAL REPUBLIC OF GERMANY

The analyses of the LOFT experiment in Germany started in 1976 and was parallel to the experiment conduction in Idaho, USA. The first generation thermal–hydraulic codes like RELAP4/GRS, BRUCH–D–06 and DRUFAN, used for licensing of commercial pressurized water reactors, were assessed during the NRC–LOFT project through pre– and post–test calculations of different experiments. After the TMI–2 accident, further development of DRUFAN to extend its application field was carried out. The new version, DRUFAN–02, was then used to analyze large and small–break LOCA as well as selected operational transients with phase separation in the primary coolant system.

Germany encouraged the OECD LOFT initiative in order to close gaps left from the thermal–hydraulic NRC–LOFT Program and to start in–pile fission product release experiments.

Phenomenological analyses of the thermal–hydraulic experiments have made significant contributions towards a better understanding of the physical phenomena occurring during PWR transients. Special interest was given to break flow, core thermal behavior, component behavior, residual water in the pressure vessel, transport, and distribution of ECC water. Also, the appearance of particular phenomena like stratified flow, natural circulation, and reflux condensation were observed in LOFT. Stratified flow in the case of small–break LOCAs was found to decrease mass losses out of the system and reduce the potential of core uncovery. Different plant recovery methods, like feed and bleed in the primary or secondary systems operated successfully and were found to be reliable.

Results from LOFT experiments cannot be directly applied to German PWR's with U–tube steam generators mainly because geometrical similarity was not preserved. Nevertheless, results from phenomenological analyses and code assessment were remarkable. As stated before, two generations of thermal–hydraulic codes were also assessed during the last 13 years through the analysis of LOFT experiments. Most of the experimental data are stored in the GRS data library and will be used to assess the new code generation such as ATHLET and ATHLET–SA.

The first fission product experiment, LP–FP–1, was specified in cooperation with experts from Germany. The emergency core cooling operation was scaled to simulate a commercial PWR of KWU type analyses. Analysis of the experimental results has shown that the fission product release fractions are independent of decay constants for stable, long lived isotopes (e.g. ^{85}Kr), and lambda dependent for short lived isotopes (e.g. ^{131}I). Fission product results of LP–FP–1 extended the data base for the assessment of LOCA release to longer fuel rods and more probable reflood conditions. Limitations made concerning the low burnup of the fuel were partly compensated by high average linear heat generation rates in the pre–test irradiation. A comparison of the LP–FP–1 results with release values of German calculation fundamentals for Design Basis Accidents demonstrates that the LP–FP–1 experiment provided valid proof for the German calculation fundamentals. Also, the data from the core damage experiment LP–FP–2 contributes to the assessment of the GRS code system for severe accidents ATHLET–SA.

FINLAND

Finland joined the NRC LOFT program in 1976 and immediately sent its first representative to INEL to work on the project. Since then, four other Finnish research scientists from the Technical Research Center (VTT) have done the same. Only the last one, however, has worked in the successor OECD LOFT program, the object of this report. Their work mainly concerned various pre– and post–test analyses with computer codes.

The importance and uniqueness of the LOFT facility, especially its considerable size and nuclear core, was recognized from the very beginning. Because the Finnish PWRs differ from the type of reactor LOFT has simulated, the emphasis in applying the results of the experiments was heavily on phenomenological analyses and code assessment. Many of the NRC LOFT tests and most of the OECD LOFT tests have been analyzed with different versions of the RELAP4 and RELAP5 codes, as well as the fast running Finnish SMABRE code. All the analyses so far have been thermal–hydraulic, and the fission product behavior and other aspects of the severe accident phase of LP–FP–2 (or LP–FP–1) have not yet been analyzed with any code. Those analyses will be done in the future, but already the knowledge of severe accident phenomena has been greatly enhanced as a result of the fission product measurements and the chemical and metallurgical investigations of the LP–FP–2 center bundle.

VTT has been the participating institute in Finland, but it has kept the Finnish Center for Radiation and Nuclear Safety (STUK) as well as the power companies Imatran Voima Oy (IVO) and Teollisuuden Voima Oy (TVO) well informed of the LOFT results, which, have formed the central part of the experimental data base used in Finland. The analyses of the experiments have provided invaluable experience and knowledge of the capabilities of the codes to model the many phenomena involved. This has led to the further development and more reliable use of the codes. The test results, together with their analyses, have also served to confirm that current DBA safety margins are adequate and efficient plant recovery methods are available. The last tests indicate that the severely degraded core was not completely blocked and could be effectively reflooded and quenched.

In summary, both the NRC and the OECD LOFT Programs are considered as invaluable, because they have produced unique experimental data for code assessment and greatly enhanced understanding of the physical phenomena that occur during LWR accidents. The Finnish staff attached to the programs gained special experience from direct participation in a large experimental program using advanced technology. Also, participation in the collective planning and management of the OECD LOFT Program in the Program Review Group and Management Board turned out to be rewarding.

ITALY

The Italian participation in the OECD LOFT Project has involved the various national organizations, each charged with the following specific responsibilities:

a. Nuclear reactor regulation (ENEA/DISP)

b. Nuclear safety research exploitation (ENEA–SIET–Universities etc.)

c. NPP ownership and operation (ENEL)

d. NPP construction – license administration (ANSALDO/NIRA).

It was of particular benefit for ENEA/DISP to obtain, through the Project, a direct experimental confirmation of the large safety margins characterizing the industrial nuclear reactors presently adopted in the OECD Members–States for various DBA conditions (particular reference can be made, in this respect, to the limited fuel temperature increase and the fuel integrity observed even in case of core uncovery).

Important information provided by the fission product tests are also considered to be of great interest to the Regulatory Body because they relate to the severe accident analysis area, which is still, in many aspects, an open issue.

The important contribution provided by the Project was the realistic evaluation and assessment of reactor response to various operating procedures such as secondary feed and bleed procedures.

LOFT test planning, experiment execution, pre– and post–test analysis, and test results evaluation have provided an undoubted contribution to the Italian Research Organizations in planning, exploiting, and utilizing national integral experimental programs such as SPES.

All of the above mentioned organizations involved in the OECD LOFT Project had the opportunity of direct contact and collaboration with an high–level international team of researchers and of experiencing the most advanced instrumentation techniques and analytical tools.

JAPAN

The wide variety of accident simulation experiments conducted within the OECD LOFT Project were utilized in Japan primarily for the improvement and assessment of computer codes; but these results also gave us a good understanding of the physical phenomena important in each accident scenario. These experiments also provided a valuable input to the operational procedures and regulatory guides used in Japan.

LP–FW–1 showed the importance of the behavior of the pressurizer and steam generators in abnormal operational transients. Large–break LOCA experiments showed a large safety margin attributed to early rewetting of the core. The small–break LOCA experiments confirmed the effectiveness of the current guidance to operators in Japan to shut down the primary pump early during a small–break LOCA. The fission product release experiments helped our understanding of the important phenomena, provided data for the development of computer codes, and assisted in establishing regulatory criteria for fission product release and severe accident conditions. In developing regulatory criteria, we also have to consider data and analyses from other experiments.

The use of OECD LOFT data in improved computer modelling and in general support of the nuclear safety research program in Japan is described in more detail in the following sections.

1. The Loss–of–Feedwater Experiment

The experimental results from the loss–of–feedwater experiment LP–FW–1 were analyzed with the RELAP5/MOD1 code. The analysis focused on an assessment of the calculation of mass and heat transfer between the two phases as RELAP5/MOD1 uses a five equation model that leads to some ambiguity in thermal non–equilibrium calculations.

During the experiment there was an early swell of the liquid level in the pressurizer because of the thermal expansion in the primary system. The temperature transients in the hot and cold legs were well predicted and in consequence, it is believed that the calculation of the thermal expansion in the primary system was also correct. However, the calculated level swell in the pressurizer was less than the data prediction and this was due to an underestimate of the steam condensation at the liquid surface in the pressurizer. This suggests that the RELAP5/MOD1 assumption that one phase is always at saturation is not applicable to this situation. It was concluded that this problem could be resolved only by the development of a full thermal non–equilibrium six equation model.

Other findings of the analysis concluded that the interfacial drag calculation was poor and that the calculation of the flow discharge from the power–operated relief valve (PORV) was inconsistent. Code modifications were recommended.

Full use was made of this analysis in the experimental and analytical program of the Japan Atomic Energy Institute.

2. Small–Break Experiments

The three small–break tests performed in the OECD LOFT program were analyzed using RELAP5/MOD1.

In LP–SB–1 and LP–SB–2, the break flow quality and the break uncovery timing were strongly influenced by phase separation in the hot leg. The analysis therefore concentrated on this issue. Because RELAP5 is a one–dimensional code, special coding techniques were used in both the upper plenum and the hot leg to simulate the essentially three–dimensional behavior of the flow in the hot leg. The prediction of phase separation in the hot leg was however still unsatisfactory for both tests. It was concluded that the flow stratification criteria in the code were incorrect for large diameter pipes at high pressures and that there was a need for new data in this area. This was addressed by recent experiments in a large diameter horizontal pipe at pressures up to 12 MPa carried out by the Japan Atomic Research Institute.

LP–SB–3 provided data on core heatup and recovery procedures following a cold leg small–break LOCA. The analysis for LP–SB–3 focused on assessing core heat transfer under slow coolant boil–off conditions. In the analysis, it was found necessary to use the experimental values for the timing of key event trip signals and for the depressurization rate in the steam generator secondary. The RELAP5/MOD2 code predicted well the core liquid level depression and the fuel cladding transients. However, very small time steps between 0.0005 and 0.05 s were needed to minimize the error in the primary coolant inventory.

These small–break experiments provided helpful support to the experimental and analytical program of JAERI. They also confirmed the current guidance to operators in Japan that early pump shutdown is effective in minimizing the loss of primary coolant inventory during a small–break LOCA.

3. Large–Break Experiments

The analysis of LP–02–6 was carried out with RELAP5/MOD2 and focused on the core wide rewet phenomena observed early in the transient caused by a temporary surge of low quality coolant into the core from the lower plenum. RELAP5/MOD2 was able to predict this early surge of fluid into the core but not the consequent temporary quench of the fuel pins. It was suspected that the heat transfer calculation was in error. In particular, the minimum film boiling temperature correlation used in the code was not well supported by experimental evidence. Because this temporary rewetting gives a substantial reduction in the peak clad temperature in blowdown, the ability of codes to calculate this correctly was considered important for reactor safety analysis.

JAERI therefore conducted experiments to obtain data on the minimum film boiling temperature in rod bundles at high pressures (12 MPa) as only limited data were available for these conditions. The minimum film boiling temperatures measured in these experiments were considerably higher than the values predicted by RELAP5/MOD2 but were consistent with data from similar high pressure rod bundle experiments from the THTF facility at ORNL. This new data were also consistent with the Groenevelt–Stewart correlation developed using measurements on small size tubes at high pressures.

By using a version of the RELAP5/MOD2 code that incorporated the Groenevelt–Stewart correlation and a boiling curve modified to be consistent with the physical phenomena, the early core–wide rewet was satisfactorily predicted.

The data from LP–02–6 have beenf beneficial to the work on code development and thermal–hydraulic research at the Japan Atomic Energy Research Institute.

4. Fission Product Release Experiments

LOFT tests LP–FP–1 and LP–FP–2 simulated an accident leading to fuel damage, limited in the case of LP–FP–1 but severe for LP–FP–2. Both tests provided useful evidence on fuel and fission product behavior after fuel damage. Work in Japan has concentrated on LP–FP–2 because it provides severe accident data while LP–FP–1 is closer to the conditions of a design basis accident.

Data from the LP–FP–2 Extended Program are of particular value for the investigation of fuel behavior under severe accident conditions in an LWR. The PIE results show many similarities to those from PBF SFD tests and from the TMI–2 core, supporting the assumption that information from LOFT on damage progression and material interaction can be applied to a wide range of severe accident conditions. The program has developed techniques for the PIE of severely damaged large size rod bundles and the metallurgical examination data and methods will be helpful for the TMI–2 debris examination program at JAERI.

4.1 Verification of Computer Codes

The computer codes TRAP/MELT, HORN, and SHAPE were used in the analysis of the experiment. TRAP/MELT was applied to the analysis of fission product behavior in the LPIS line. HORN was used to analyze the gas–phase transport of fission products in reactor cooling systems under severe accident conditions. SHAPE was used to evaluate the core heatup and the fission product source.

Results from these analyses have been applied in the assessment and development of the codes.

4.2 Benefits Derived From the PIE of LP–FP–2

The PIE of LP–FP–2 produced valuable information on fuel behavior under severe accident conditions in an LWR. The results showed that the test bundle experienced very high temperatures (up to about 3000 K) leading to severe fuel damage. The metallurgical examination showed a wide variety of interactions involving fuel, fuel cladding, and other core materials that are of general interest in identifying the phenomena occurring under severe accident conditions. There is also information on material fragmentation, blockage formation, and debris accumulation at the spacer grid positions. The information on material liquefaction and relocation has helped our understanding of melt progression modeling. The general similarity with other in–pile data, such as that from the PBF SFD tests and from TMI–2 sample examination, supports the view that the interactions between material seen in LP–FP–2 can be applied to a wide range of severe accident conditions.

The PIE data, used in conjunction with thermocouple readings, has provided evidence on the maximum temperature experienced and a vertical temperature profile over the fuel bundle. This is essential information for computer code evaluation.

The retained fission product analyses and the SEM/EDS/WMS examinations are expected to provide both a qualitative and quantitative understanding of material and fission product behavior.

4.3 Summary Conclusion on the Fission Product Tests

The OECD LOFT fission product tests have increased our understanding of the phenomenology of fuel behavior during a severe accident and provided unique source term data for a large size fuel bundle.

Studies carried out in Japan using this data have emphasized the verification of computer codes developed for severe accident studies. As a result of this work it is believed that future versions of these codes can be expected to provide a better simulation of fuel damage progression and fission product behavior.

The work at JAERI on the investigation of TMI–2 debris samples and other fuel damage experiments will make full use of the experience and technology resulting from these LOFT studies.

It is expected that the LOFT data, together with that from TMI–2 and elsewhere, will play an important role in understanding accident progression and in source term evaluation for commercial reactors. Integration of this world-wide data on fuel damage will be one of the major concerns of the program of severe accident research at JAERI.

SPAIN

The participation of Spain in the OECD LOFT Project has been organized to obtain the maximum transfer of knowledge and technology. With this in mind, a number of Spanish organizations signed an agreement, dated November 4, 1984, stating the contribution they would each make, their responsibilities and representation on the management committees of the International Project.

The contributing organizations include: Consejo de Seguridad Nuclear, Empressa Nacional del Uranio, S.A., Unidad Eléctrica, S.A., Junta de Energía Nuclear, now CIEMAT, and the Universidad Politécnica de Madrid represented by Cátedra de Tecnología Nuclear in the Escuela Técnica Superior de Ingenieros Industriales. When the Project was extended, a further protocol was signed which included the participation of the Empresa Nacional de Residuos, S.A.

The Agreement set up a Steering Committee with the specific responsibility of organizing and managing the Project and of disseminating the benefits of participation throughout the Spanish Nuclear community. It also appointed a Project manager, with executive responsibility, and a Scientific Advisor. The Steering Committee laid down its own terms of reference and issued the administrative procedures needed to organize and carry out the various tasks within its work program. Practical experience has shown that this organization has worked well.

The benefits obtained for Spain from participation in the OECD LOFT Project can be grouped as: (a) direct, and (b) indirect.

Direct benefits include the following:

1. Information obtained by participants on the physical phenomena arising in light water reactors during thermal-hydraulic transients. This includes both those transients in which the fuel geometry remains intact and those involving significant fuel damage.

2. Access to complex computer codes which have been written to analyze these transients.

3. The transfer to Spanish scientists and engineers of expertise in the postirradiation analysis of fuel which has suffered severe damage during the transient.

Indirect benefits include the following:

1. Close collaboration with scientific and technical organizations in other countries which have experience in these advanced technologies.

2. The establishment of improved relationships between the organizations in Spain participating in the Project. This has permitted a clearer definition of the responsibilities and goals of such organizations.

3. The information obtained from this Project, and the experience obtained by so many experts working closely together, has been of substantial benefit in planning further work programs.

As the Project comes to an end, one could conclude that:

1. There is a nucleus of about 20 thermal-hydraulic specialists, who are now transferring their knowledge to a widening circle of younger scientists and engineers.

2. There is now considerable expertise in setting up and using complex computer codes.

3. Contacts between organizations in Spain and those in other countries have been consolidated and will continue after the termination of the project.

4. The expertise obtained in the OECD LOFT Project will continue in other international projects, e.g., ICAP and PHEBUS CSD.

The success of this transfer of expertise can be measured by a number of criteria, such as the ability to explain discrepancies in the analysis, the creativity shown in carrying out the work, and the application of the acquired skills to study other cases of interest where the new LOFT data can be used to support and interpret such cases.

For example, it can be said that Spanish researchers now have a good understanding of the physical phenomena and associated mathematical models, and they have been able to explain some of the discrepancies between calculations and experimental data. In this way, it has been possible to document the limits of the validity of different codes, when assessed against LOFT data.

The creativity of Spanish work has also been demonstrated in studies carried out to simplify the nodalization used in the codes for particular experiments, by the support given to chemical analysis techniques and in the development of computer graphics techniques for data presentation.

From a wide point of view, it can be seen that the essential benefit from the Project is in supporting the further development of nuclear power, and in the influence it has had on the organizations which have contributed to the program. In detail one may note:

1. A better assessment of the design margin and development potential of Spanish nuclear stations.

2. The improved potential of Spanish industry in the design and fabrication of the hydraulic systems for future nuclear stations.

3. An increased participation in the design of fuel elements and recharge loadings with improvements in safety and economics.

4. The implementation of scientific and development programs, for example, in the fields of thermal–hydraulics, fuel element performance, and decontamination technology.

All these topics are of value for the forward program, and our internal Steering Committee has continued to study and analyze these aspects in much more detail. In the same way, the organization participating in the Project, and the nuclear industry, are giving serious consideration to issues which may be of increasing importance, such as the advanced light water reactors, and the new version of Appendix K of 10 CFR Part 50, recently issued by the USNRC.

SWEDEN

One major Swedish motive for joining the OECD LOFT project was the project's commitment to address phenomenology associated with core degradation accidents. A decision had been made in Sweden that, within a specified timeframe, measures should be installed in order to mitigate environmental consequences of core melt accidents. These measures, which included both procedures and equipment, were developed based on code calculations and engineering judgement of core melt progression scenarios. Performance of the fission product transport tests, and in particular the LP–FP–2 with post–test examinations of the fuel bundle, provided important data for assessment of uncertainties in calculations and judgements, and the results were used as part of the licensing documentation for the mitigation measures. Moreover, the experiment enhanced understanding of the core degradation processes that occurred in the TMI–2 accident, which provides another important basis for judgement. The LP–FP–2 data will decisively impact the assessment of accident management measures, particularly in the prediction of system behavior and phenomenology during reflood of a partly degraded core. A resident engineer was sent from Sweden to participate in the Project and in the evaluation of data observed in the experiment.

The thermal–hydraulic experiments produced data for development and improvement of computer codes for analysis of design basis LOCAs and transients. The LOFT experiments, being able to simulate the thermal–hydraulic behavior at a reasonable scale with neutronic feedback, are particularly suitable for code assessment. The experiments are of great importance in Sweden in the ongoing development of licensing methodology based on best estimate calculations and quantified uncertainties.

The OECD LOFT Project successfully demonstrated the value of international collaboration. The Project management and staff also encouraged active participation by staff from the signatory countries and international commitments in technical support of the project, a fact which deepened the engagement in the project and contributed to the positive outcome.

SWITZERLAND

The LOFT was a unique facility which provided data on thermal–hydraulic and fission product with a nuclear fuel in a simulated PWR primary system environment. Switzerland joined the USNRC program in the late 1970's. They participated in the experimental program and also initiated a separate national experimental program. One of the aims of this national program was to provide data to understand the effect of external LOFT fuel thermocouples on the thermal–hydraulics. Additional efforts were made to analyze and simulate the LOFT experiments with computer programs. When the international consortium was formed in 1982 to continue the USNRC LOFT program, Switzerland joined at the beginning. The purpose of the international LOFT Program was further extended to provide thermal–hydraulic data in plant transients to address national needs. It also provided other data which helped to improve the understanding and predictability of transient behavior and to enhance the reliability, economics and safety of nuclear reactors.

Switzerland actively participated in the international LOFT program in various aspects of the thermal–hydraulic and fission products experiments. A delegate was sent to the program to participate in planning and postexperiment activities, especially of the fission product experiments. Extensive efforts were spent in reviewing the program, planning, and postexperiment analyses. Extensive code calculations were performed to understand thermal–hydraulic and fission product behavior in almost all OECD LOFT experiments. The results of these analyses were incorporated in various OECD LOFT reports. A comparison report was prepared to compare the results of code calculations performed by member countries for experiment LP–02–6 (one of the OECD LOFT large–break experiments). The RELAP5/MOD2 computer program was further assessed and a model was developed to predict the swell level during the boil–off phase of large–break LOCA and incorporated in the official version of the code. An international workshop was organized in Grindenwald (Switzerland) to discuss the merits of LOFT large–break LOCA experiments, associated analytical work, and simulations of large–break LOCAs in large plants. Further efforts were spent to extrapolate the understanding gained in the OECD LOFT small–break experiment LP–SB–3 to real PWR plant conditions. More activities are underway to understand and simulate fuel behavior, Zr/steam oxidation, H_2 production, and fission product distribution, release and retention.

UNITED KINGDOM

The UK was involved from the beginning in the discussions that led to the formation of the OECD LOFT Project. The final negotiations took place, against the background of a Government decision early in 1979 to permit the CEGB to proceed with plans for the construction of a 1200 MWe PWR at Sizewell on the east coast of England. These plans were subject to the satisfactory outcome of a Public Enquiry into the safety, environmental, and economic implications of the proposal and license approval by the Nuclear Installations Inspectorate. Information from the LOFT Project was not expected to form part of the licensing documentation for the reactor safety case, but there was a view widely held in the UK by designers, operators, and regulators that the arguments of the formal safety case should be buttressed by a wide ranging program of realistic experiments and associated calculation techniques that could provide a quantitative understanding of the physical phenomena that arise in safety case scenarios. For this reason, the UK contribution to the OECD LOFT program has directly involved all these interests through the participation of the UKAEA, CEGB, NNC, HSE, and various academic institutions.

For the thermal–hydraulic experiments, essentially in the field of loss–of–coolant accidents (LOCAs), the major objective was to support the development of advanced computer codes, in particular TRAC–PWR and RELAP5/MOD2, and the validation of proprietary licensing codes, by providing integral data for code validation. Indeed LOFT was seen as the only integral facility available that could provide a full simulation of large–break LOCA transients, in spite of nontypicalities (e.g. the half–length core) and was therefore of vital importance. The fact that all six of these tests are now included in the CSNI code validation matrix is evidence that this first objective was achieved. However, in planning the test program there were other detailed objectives that were considered important by the UK.

For large–break LOCA's it was felt that, in addition to cladding temperature limits set by the need for a margin against the zircaloy–steam reaction (nominally taken the 1473 K), it was also necessary to consider the problem of cladding ballooning which can occur at much lower temperatures (1020 to 1120 K). For Sizewell, the safety case was based on the theory that even with pessimistic thermal–hydraulic calculations, coolant blockages large enough to present a safety hazard could not be envisaged. However, it was recognized that a more satisfactory approach would be to use best estimate calculations based on advanced codes to improve the margin against the onset of cladding ballooning. In the past, attempts to model design basis large–break LOCAs by direct scaling resulted in LOCA transients that were overly dominated by early rewet effects. The UK believed that a more representative transient could be achieved by suitable modification of the test conditions. The UK was also not convinced that a satisfactory allowance for the effect of the external thermocouples had been made in previous analyses, and felt that if this could be done the results would be in better agreement with analysis and would reinforce the value of further large–break LOCA tests in LOFT. According to the UK, it was possible to make test proposals for LP–LB–1 that would add significantly to the LB/LOCA integral test database and would be directly relevant to Sizewell. They concluded that the two tests, L2–6 and LP–LB–1, substantially increased the database provided by L2–3 and L2–5 and are essential support to the ICAP program of thermal–hydraulic code improvement. They note five issues where the new data has been of value in clarifying difficulties:

1. The balance of flows through the steam generator loops and in and out of the primary coolant vessel is now well predicted.

2. There is now a better understanding of the problems in predicting early rewet phenomena (both bottom–up and top–down), though all these experiments were only partially successful in showing a representative reflood of a hot core because of the low power in the peripheral rods. The four tests provide a wide range of data and enhance confidence in predictions for full–scale plants.

3. The tests have demonstrated the importance of three–dimensional flow patterns in the vessel. Nevertheless, the database is limited and needs to be supplemented by results from other tests (e.g., UPTF).

4. The OECD LOFT program has been an important incentive to understand the role of external thermocouples and has encouraged new analysis that is now able to reconcile the results from LOFT with those from separate effects tests.

5. The objective of demonstrating better margins against the onset of cladding ballooning has been supported.

The set of small–break LOCA tests were seen as less critical for the Sizewell reactor because it had been designed with an increased capacity for the high pressure emergency coolant injection system (HPIS). On the other hand, it was also recognized that it was more difficult to present a consistent safety analysis approach based on pessimistic calculations for this class of accidents and that there was already a strong emphasis on correct phemenological modeling and best estimate calculations. The UK therefore accepted that the test program set out by EG&G for LP–FW–1, LP–SB–1, and LP–SB–2, and the Italian proposal for LP–SB–3, would add usefully to the existing database. From the subsequent analysis they note the following points:

1. It is clear from LP–SB–1 and LP–SB–2 that there were inadequacies in the modeling of flow stratification effects in large diameter pipes and, in particular, in the calculation of fluid conditions in off–take junctions.

2. LP–SB–3 demonstrated that secondary feed–and–bleed could be a useful core recovery technique and provided integral experiment data on core uncovery in a slow boil–down.

3. LP–FW–1 provided valuable experience in analyzing primary feed–and–bleed in a long–term transient and provided useful support to other evidence on code inadequacies.

The inadvertent early release of ECCS water in the LP–FP–1 experiment meant that the conditions under which fission products were released and transported were uncertain and probably not consistent with intentions. However, the analysis of the fission product release data and the assessment of the amount and effectiveness of the early ECCS release presented a searching challenge to our own understanding of both of these topics. The UK regards its participation in the analysis of these issues as a major contribution towards understanding of the processes involved.

LP–FP–2 was originally proposed as a fission product release and transport experiment and was of limited interest to the UK. The successful achievement of a high temperature transient with a significant heat input from the exothermic steam/zircaloy reaction and the extension of the Project to provide detailed sectioning and a chemical and metallurgical analysis of the center fuel bundle has meant that the experiment is now seen as a major database for severe core damage effects in a large fuel bundle provided with a full complement of control rods. Because the complete data from the experiment are only now becoming available and analysis capabilities in this field are under rapid development, it is too early to give any detailed assessment based on a full analysis. However, the existence of two melt zones, the lower being dominated by the presence of control rod material has clear parallels with the data from TMI–2. The analysis of this test can be expected to continue over a period of a number of years.

Finally, the UK would draw attention to the value of the OECD LOFT Project as a successful example of international collaboration. This was seen in the Project management, the joint approach to experiment selection and analysis and in the substantial support to the team at EG&G by staff attached from signatory countries.

UNITED STATES OF AMERICA

The OECD LOFT Project provided the United States with a unique opportunity to collaborate with other nations to help resolve nuclear reactor safety issues. This collaboration has led to the recognition of specific problems and concerns of other nations, and has enabled ideas to be freely transferred for the benefit of all participating members. The principal benefit from the experimental data comes from providing more operational margin for US plants. The major benefits to the United States of each of the eight experiments conducted in the program are summarized in the following paragraphs.

The Loss-of-Feedwater Experiment LP-FW-1.

A strong scientific base was obtained for development of operator recovery procedures for a loss-of-feedwater event. This experiment was initiated at nominal PWR operating conditions with a complete loss of secondary system feedwater. The experiment provided data to assess the effectiveness of using primary system feed (with the high pressure injection system) and bleed (with pilot operated relief valve flow) to remove energy from the primary coolant system. The transient data also allowed for systems code assessment and identification of transient characteristics in a loss-of-feedwater condition.

Small-Break Experiments LP-SB-1 and LP-SB-2.

These experiments, when combined with other LOFT experiments, provide a complete set of transient small-break data for systems code assessment. Small-break issues (primarily break size, location, and primary coolant pump operation) that resulted from the TMI-2 accident required experimental data for resolution. Vendor calculations, summarized in NUREG-0623, showed significant differences in reactor system behavior for these parameters. The differences were attributed to systems codes differences and uncertainties in modeling phenomena such as phase separation, reflux condensation, and pump head degradation. These two experiments were designed to simulate small-breaks in the hot leg of a PWR operating loop for the two cases of early pump trip and continuously operating pumps. They would complement two other experiments for a cold leg break with the same two pump operating conditions (LOFT L3-5 and L3-6). Together, the four experiments provided a complete set of transient data for systems code assessment.

Small-Break Experiment LP-SB-3.

This experiment was designed to provide information on a small-break transient for conditions which had not previously been simulated in the LOFT facility. The transient simulation addresses (a) phenomena associated with slow coolant boil-off leading to core uncovery at high system pressures, (b) the effectiveness of steam generator feed and bleed as a means of plant recovery from degraded core cooling conditions, and (c) the effectiveness of accumulator injection when a low pressure differential exists between the accumulator and the primary system. The experiment provided a sound data base on which to develop and assess the calculational capability of these phenomena in commercial PWRs.

Large-Break Experiment LP-02-6.

This experiment was the first large-break simulation starting from conditions representative of NRC design basis boundary conditions. The design of the experiment also included coincident loss of offsite power, minimum US ECCS capacity, pressurized fuel in the center assembly, and pump coastdown representative of the normal pump coastdown in commercial PWRs. The latter was intended to provide conditions most representative in a PWR in order to determine the degree of cooling that would take place during the blowdown phase. The transient results were very beneficial in all regards, an example of which was the degree of blowdown cooling. The normal pump coastdown results came in between the previous results for rapid pump coastdown (small cooling) and continuous pump operation (large cooling), with significant cooling extending through all but the upper one-third of the core. The transient is an excellent reference for licensing calculations in the US.

Large-Break Experiment LP-LB-1.

This experiment was conducted from initial conditions similar to LP-02-6. Operationally, the transient also included coincident loss of offsite power. The pumps, however, underwent a rapid coastdown leading to the maximum cladding thermal excursion. The ECCS was adjusted to be equivalent to the UK minimum safeguard condition. Relative to LP-02-6, the accumulator injection was 70% and the pumped injection flowrate was 50%. This transient is beneficial to the US by providing data to complement the LP-02-6 transient in the areas of blowdown cooling in the core and the effectiveness of the ECCS operating in a more degraded condition.

Fission Product Experiment LP-FP-1.

The benefit of this experiment to the US was the information obtained on the gap release to a high temperature vapor environment, the gap release that occurred during and as a result of reflood, and the fission product transport through and out of the primary coolant system undergoing refill by an ECCS design equivalent to the Federal Republic of Germany PWR (hot and cold leg injection) nominal ECCS operation. This experiment was the first in which fuel damage was allowed to occur. Fuel damage was limited to gap fission product release under transient conditions of a large–break accident with delayed ECCS operation. The transient is similar to the LP–LB–1 transient except for the delay in ECCS operation.

Fission Product Experiment LP–FP–2.

Data from this experiment (when taken with data from TMI–2, the Power Burst Facility Severe Fuel Damage Test series, and other small–scale tests) provide a wealth of information on severe accident phenomenology and a basis for assessing severe accident computer codes. The experiment was conducted under a V–sequence scenario with operating conditions selected to lead to an early core uncovery. The resulting transient was allowed to progress through approximately 4.5 minutes of severe fuel damage before a rapid reflood was initiated.

The results provided important data on early phase in–vessel behavior to assist in resolving questions relating to core melt progression, hydrogen generation, fission product behavior, composition of melts that might participate in core–concrete interactions, and the effects of reflood on a severely damaged core. The experiment also provided a unique basis of comparison among severe fuel damage tests in that actual fission–product decay heating of the core was used. This verified other test results that used fission heating because of their low specific power from decay heating.

The experiment was particularly important in that it was a large–scale integral experiment that provided a valuable link between smaller–scale severe fuel damage experiments and the TMI–2 accident. The experiment exhibited distinct candeling and flow blockage formation (both metallic and ceramic) that is consistent with TMI–2 behavior and has shown that the complex phenomena that occur during a severe accident can not only be understood but also can be modeled. Data was obtained on the effect of control rods on material relocation, the melt sequence and axial stratification of materials, the lack of complete blockage formation and resulting steam starvation, and the effects of reflood on core damage and hydrogen generation. The test has shown that the metallic relocation process, which may later lead to blockage formation as at TMI–2, is definitely a non–coplanar, non–coherent process involving rivulet flow. This explains why the complete blockage of the steam flow to terminate steam oxidation and hydrogen generation does not occur until very late in melt progression, if at all. This absence of flow blockage, also observed in previous severe–fuel–damage tests in other facilities, was therefore not the result of a low by–pass flow area. This experiment will continue to yield new findings and understanding for years to come. These, plus the information the project has yielded to date, are and will continue to be important for the development and assessment of computer models that are used to predict the behavior and consequences of severe fuel damage in commercial PWRs.

In summary, the OECD LOFT experiments provided not only an opportunity to collaborate with an international group of scientists, but also provided and will continue to provide understanding of severe accident phenomena to aid in promoting nuclear safety. The experiments also provide data to assess and improve best estimate analytical capabilities used in safety analysis, especially with respect to 10 CFR 50 Appendix K requirements. These data therefore provide more operational margin for all US plants. All plants operate with the intention of avoiding accidents, knowing that any nuclear reactor accident has a negative impact on the nuclear industry worldwide.

7. CONCLUSIONS

All nuclear power stations are designed to provide an in–depth defense against a number of major plant failure scenarios and to make the most effective use of the engineered safety features built into them. Among the most important of these are loss–of–coolant accidents (LOCAs) where a major or minor breach of the primary coolant envelope leads to depressurization and a loss–of–coolant. In order to satisfy regulators that such accidents are fully covered by design features, it is necessary to identify a sufficiently comprehensive set of accident scenarios, to understand fully the physical phenomena involved and to be able to set quantitative limits to those key parameters of the transient such as the peak fuel cladding temperatures, which provide a guarantee against core failure and fission product release.

There is now increasing international support for the view that, for a wide range of accidents, it is important to support this regulatory process by the development of advanced computer codes that model phenomena realistically and which can be used to make quantitative predictions. The development of these codes, as has already been noted, requires the fitting together of a number of detailed numerical models that have been based on a thorough physical understanding of the phenomena involved, on laboratory experiments, on large–scale experiments on components, and on integral tests. Integral tests, which have not been used for code development, are then needed to validate the performance of these computer codes. This is a brief account of a technique defined more fully in the introduction.

This approach now has an important and essential international dimension and not only because neither nuclear incidents nor public opinion on nuclear reactor safety can be confined within national frontiers. The background to this approach is as follows:

1. There is now a substantial research program in a number of countries that supports these objectives. Because the physical phenomena involved are universal, the sensible scientific approach is to collaborate in understanding them. This makes full use of resources and expertise that, even on a world–wide basis, are limited.

2. Only a limited number of advanced computer codes are being developed and nearly all of these developments are either supported or organized in collaboration between many countries and organizations.

3. Integral facilities are expensive. Although there are now a number of examples of international collaboration on programs of LOCA research, the construction of individual facilities and their major financial support has always depended on a strong initiative from a single sponsor. There are now, however, a number of cases where fuller and longer deployment of an existing facility has been made possible through a shared program.

4. There is a clearly identified risk that facilities and expertise in particular areas may disappear as individual national programs come to an end. International programs have achieved some success in ameliorating this problem.

5. There is an identifiable audience, those scientists and engineers with a good general understanding of the techniques and scientific issues involved, but without a detailed knowledge of nuclear reactor design and operation, that the nuclear community should be attempting to address. This audience and its lines of communication are, by their general nature, international.

It is useful to examine the history of the OECD LOFT Project against this background.

1. The OECD LOFT Project provided a valuable extension of the USNRC Program. Nevertheless, the construction of such a facility could not have been supported solely by international funding. Also, as the program continued, there was a consensus view that further thermal–hydraulic tests could be usefully carried out in LOFT, and in particular, further fission product and severe accident tests would make valuable use of a unique facility. There was no prospect of funding these experiments.

2. Although the first three tests LP–FW–1, LP–SB–1, and LP–SB–2 were essentially fully defined at the start of the program, members through the Program Review Group, were able to make a substantial contribution to the detailed definition of the remaining tests.

3. The Option 5 decision, which closely involved members in the analysis and assessment of the tests, although originally the product of financial stringency, was in fact one of the major successes of the Project. It helped to identify more sharply the role of post–test computer analyses both as an aid in understanding the test phenomena and in providing an invaluable forum for a comparative analysis of these calculations and the associated computer codes. It can be seen as part of the climate of opinion also expressed in the NEA Specialist Committee on code validation and in the setting up of the International Code Analysis and Assessment Program (ICAP).

4. From the management side, the first important task was to secure funding. Then came the recognition that the agreed technical program could not be met within the committed resources. There was a need to redefine the program and to mobilize extra resources under the Option 5 proposal. Finally, recognition that the major value of LP–FP–2 was likely to be as a severe core damage experiment on a large fuel bundle led to the funding of the LOFT Extension Agreement. The Management Board also gave careful consideration to the archiving and publication of the experimental and test assessment data.

Members undoubtedly found it valuable to obtain access to the advanced technology being developed within the Project by attaching staff from whose expertise the Project itself also benefitted. The community of interest that this and the common program of computer analysis also encouraged was an important contribution to the success of the Project.

In summary, the Project can claim the following achievements:

1. It successfully ran an international project in which both managerial decisions and the detailed planning of the program were organized by the collective decisions of members. The lessons learned from this aspect of the Project should be of permanent value for future international initiatives.

2. The detailed experimental results provided valuable new evidence on thermal–hydraulic issues and an important international data base for computer code verification.

3. It provided a valuable forum for the exchange of specialist views and for computer code comparisons.

4. The two tests, LP–FP–1 and LP–FP–2, extended the use of LOFT to provide data on fission product release and transport from failed fuel. LP–FP–2 is also a major data source on severe core damage phenomena in a large fuel bundle and work on the assessment of data from this test can be expected to continue over a number of years.

5. Effective measures were taken to make the data of long–term value by archiving in the NEA data bank, by making the data available as part of the CSNI Code Validation Matrix, and by linking further work based on the LOFT data with current international programs such as the CSNI Specialist Committees, the ICAP program, and the USNRC Severe Fuel Damage Program.

6. There is general agreement that there are problems in retaining facilities and expertise in a number of areas of reactor safety and that the facilities offered by LOFT are irreplaceable. The OECD LOFT program was able to successfully make use of LOFT but was not able to provide a route for its further retention.

8. ACKNOWLEDGMENTS

The OECD LOFT Project lasted for seven years and involved the total expenditure of over $100M. The program was completed on schedule and within the agreed budget. The major technical objectives were achieved and the information obtained is now available and accessible to the international community. Such a result could only be achieved by the close and effective cooperation of many people covering project management and administration, technical support, and the direct efforts of the team at Idaho. This support, at all levels, is warmly acknowledged but, as in any major enterprise, a few individuals can be seen to have made a key contribution and a recognition of this is an appropriate conclusion to this report.

The initial stages of a project, where there is a need to form a viable concept and to establish the broad technical scope and budget, are perhaps the most difficult. Progress can indeed be very elusive until the final stages are reached. For all this, the Project is particularly indebted to Mr. K. Stadie who mobilized the resources and experience in such international projects of the NEA. The final agreement, after many difficulties, owes much to his diplomatic skills. He also provided continuing support, helped to ensure that all the information arising from the Project would be properly archived and available, and ensured that within the NEA, there would be a forum for continuing analysis of the data.

Dr. D. Hicks was Chairman of the Management Board for the first three years of the Project and it owes much to his wise guidance through these early years. Under Dr. K. Sato, the LOFT Extension Agreement was negotiated, which substantially increased the data obtained from LP–FP–2.

Prof. E. F. Hicken was Chairman of the Program Review Group for the major part of the program. Under his leadership, the initial broad program proposals were refined into specific experiments. Effective collaboration on post–test analysis was greatly increased after accepting the Option 5 proposals and the program of post–test analysis for LP–FP–2 was established. Dr. O. Mercier and Mr. J. Fell followed the same demanding pattern in completing the work on the fission product release tests.

Dr. G. D. McPherson was the USDOE LOFT Project Manager until the final months of the Project and his previous experience as the USNRC Project Manager was a valuable link to the earlier program. He provided a smooth liaison between the team at Idaho and the various managerial and technical committees. He was always keen to ensure that Project members had an effective influence on the detailed planning of the program and were able to receive maximum benefit from their participation in it. He took a particular responsibility for providing USDOE support on licensing and safety issues.

Dr. P. North, as Onsite Contract Manager, is commended both as the leader of the EG&G team at Idaho and for his own personal contribution. He deserves credit for completing a demanding sequence of experiments, some of them breaking new ground for the LOFT Facility, within budget and on time. He also gave strong support to the change in the detailed objectives for LP–FP–2, which ensured that it would be of exceptional interest in providing severe core damage data for a large fuel bundle.

Finally, the particular thanks of the Project are due to the many scientists and engineers from Project member countries who joined the team at Idaho as attached staff. They contributed both to the success of the experimental program and to the major objective of bringing technical expertise and experience back to their own countries.

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APPENDIX A

THE LOFT FACILITY

APPENDIX A

THE LOFT FACILITY

Facility Description

The LOFT experimental facility was a 50 MW(t), volumetrically scaled, pressurized water reactor (PWR) system. The LOFT facility was designed to study the engineered safety features (ESF) in commercial PWR systems as to their response to the postulated loss–of–coolant accident (LOCA). With recognition of the differences in commercial PWR designs and inherent distortions in reduced scale systems, the design objective for the LOFT facility was to produce the significant thermal–hydraulic phenomena that would occur in commercial PWR systems in the same sequence and with approximately the same time frames and magnitudes. Experiments conducted in the LOFT facility provided "integral" system data for assessment of analytical licensing techniques and for identification of unexpected thresholds or events that may occur during a LOCA. The term integral implies that the entire system is modeled and the entire LOCA sequence is carried out as opposed to separate effects tests in which specific phenomena, components or single systems are studied during a particular phase of the LOCA.

Figure A.1 shows the LOFT facility in comparison with the ZION commercial nuclear reactor and the Semiscale experimental facility.

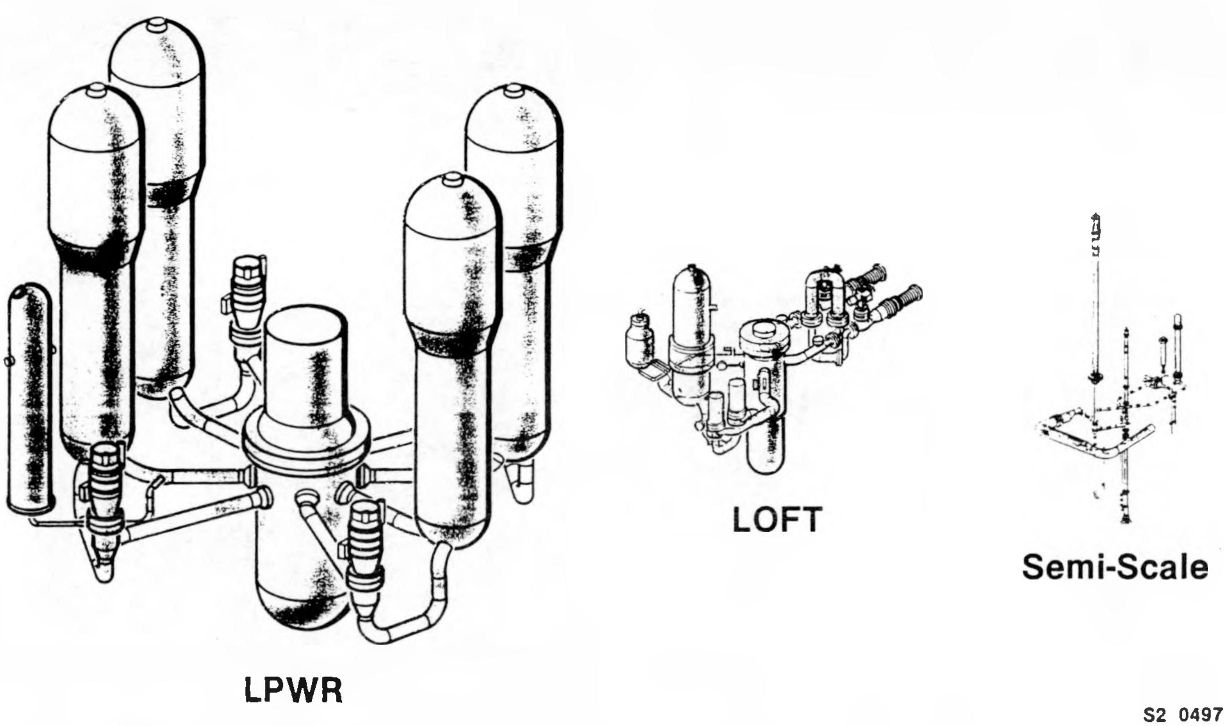

Figure A.1 Scale comparison of LOFT with the Semiscale experimental facility and ZION commercial nuclear reactor.

The LOFT facility was also intended for experiments and acquisition of data on operational transients that may occur in a commercial or generic reactor. Such transients as loss of feedwater, loss of primary coolant flow, and loss of steam load may lead to pressure relief valve setpoints being exceeded. Relief valves then actuate and vent primary system coolant. Improper relief valve operation can lead to loss-of-coolant transients as occurred at Three Mile Island.

The LOFT Experimental Facility shown in Figure A.2 is described in detail in Reference A-1. The facility consisted of five major systems:

1. Primary Coolant System.
2. The Reactor System that contained the 1.68-m high nuclear core.
3. Blowdown Suppression System.
4. Emergency Core Cooling System.
5. Secondary Coolant System.

These systems were instrumented extensively to measure the the system parameters.

The LOFT Primary Coolant System, shown in Figure A.2, consisted of an intact loop containing active components to simulate three unbroken loops of a four-loop PWR, a reactor vessel containing a nuclear core, and a broken loop to simulate the single broken loop of a PWR. The broken loop contained passive steam generator and pump components (simulators) and did not have appreciable flow prior to loss-of-coolant experiment (LOCE) initiation. The pump and steam generator simulators contained orifice plates to simulate the pressure drops of their counterparts. The broken loop terminates in two quick-opening blowdown valves which simulate the pipe break. The break area was sized with orifice plates located at the break planes.

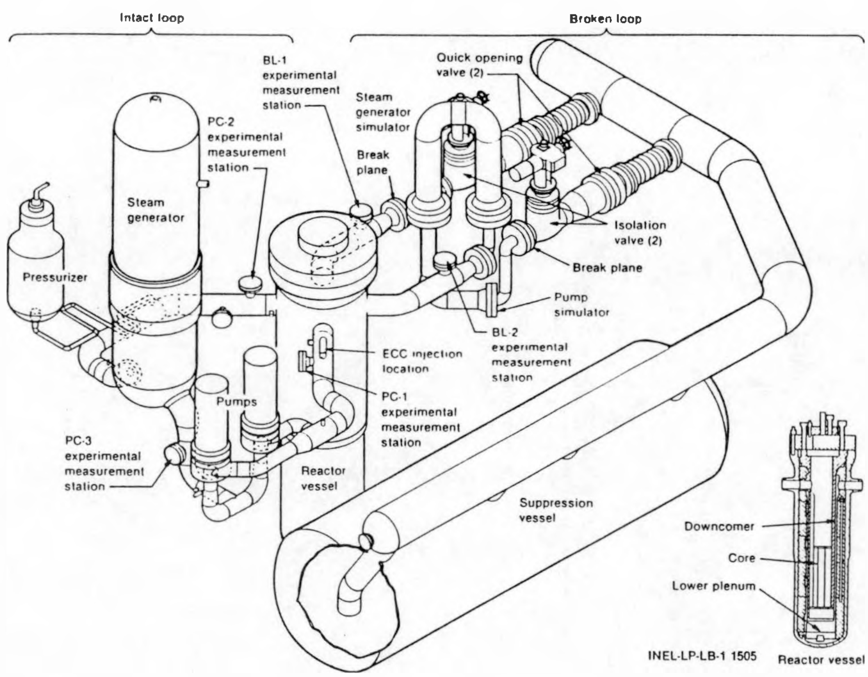

Figure A.2. Axonometric projection of the LOFT system.

The reactor system (Figure A.3) contained a 1.68-m nuclear core that was about one-half the length of typical reactor cores (3.7 m long) in commercial plants. However, this was the only compromise made in the nuclear fuel for the LOFT core. PWR fuel rod assemblies were used in the geometry shown in Figure A.4. The triangular corner assemblies were partial square assemblies and had reactor control rods in the guide tubes. The center fuel assembly was the most heavily instrumented assembly with instruments placed in the vacant guide tubes as well as on the fuel rods. The LOFT fuel assemblies were complete with upper and lower end boxes and fuel rod spacer grids at five elevations. More specific detail of the LOFT core design is contained in Reference A-1.

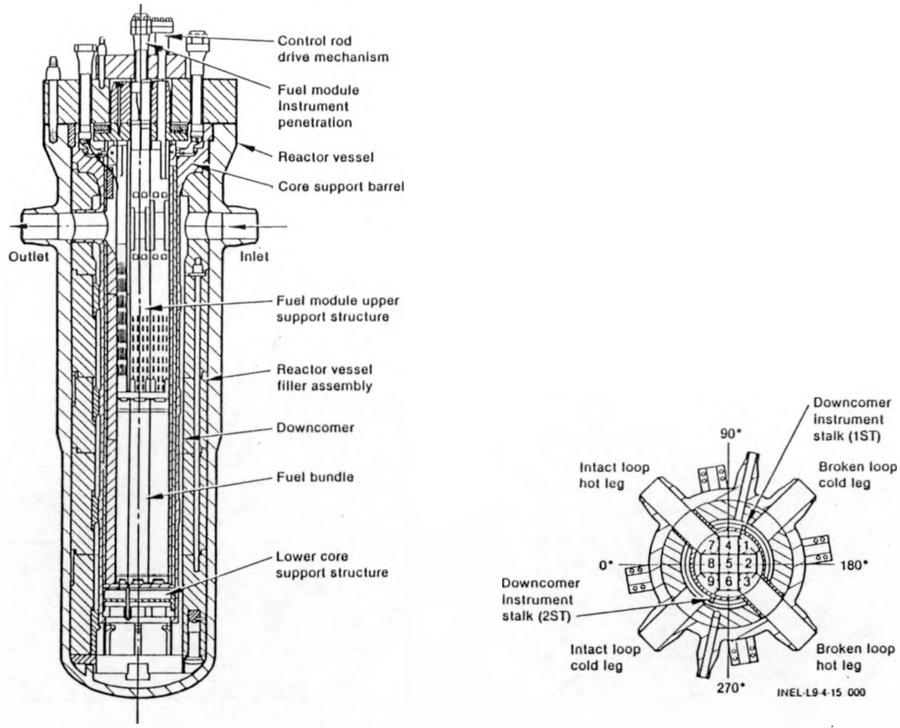

Figure A.3. LOFT reactor vessel assembly.

The LOFT nuclear core can be considered a segment of a generic PWR core which is subjected to the same transient or off–normal conditions that a generic PWR would undergo in the event of a LOCA or operational transient. Thus, the core geometric size, peaking factors, and power generation lead to primary coolant system volumes via the criteria of maintaining, as close as possible, the coolant volume–to–total core power ratio in order to create the same transient and off–normal conditions that a generic PWR core would be subjected to. This view of the LOFT model was explicit in the early planning and design.

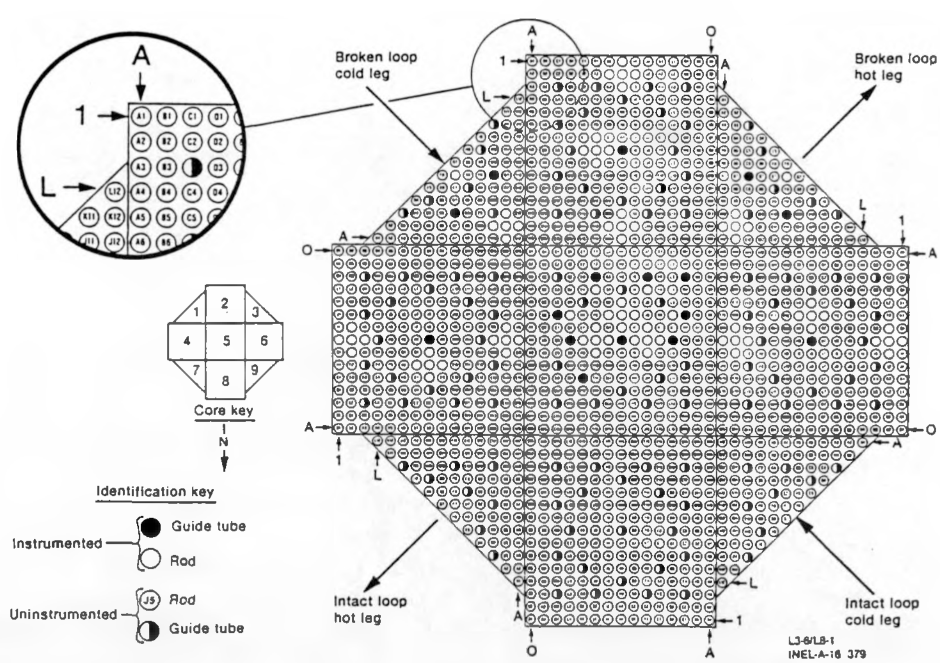

Figure A.4. LOFT core arrangement.

The Blowdown Suppression System was designed to simulate the containment back pressure in large PWRs during LOCA events. It consisted of a large pressure suppression vessel, downcomers and a header connected to the primary system via the quick–opening blowdown valves (see Figure A.2).

The Emergency Core Cooling System (ECCS) consisted of the same three systems currently in commercial PWRs—the high pressure injection system (HPIS), the accumulators, and the low pressure injection system (LPIS). The systems were actuated similar to their generic counterparts and inject scaled amounts of emergency core coolant (ECC) typical of the ECC delivery behavior in commercial PWRs. The LOFT ECCS has the capability of injecting ECC to any of several locations including the intact loop hot or cold legs, and the reactor vessel downcomer, lower plenum, or upper plenum. An identical backup ECCS is also available which functions separately from the ECCS used in a LOCE. Figure A.5 schematically shows the LOFT system including ECCS.

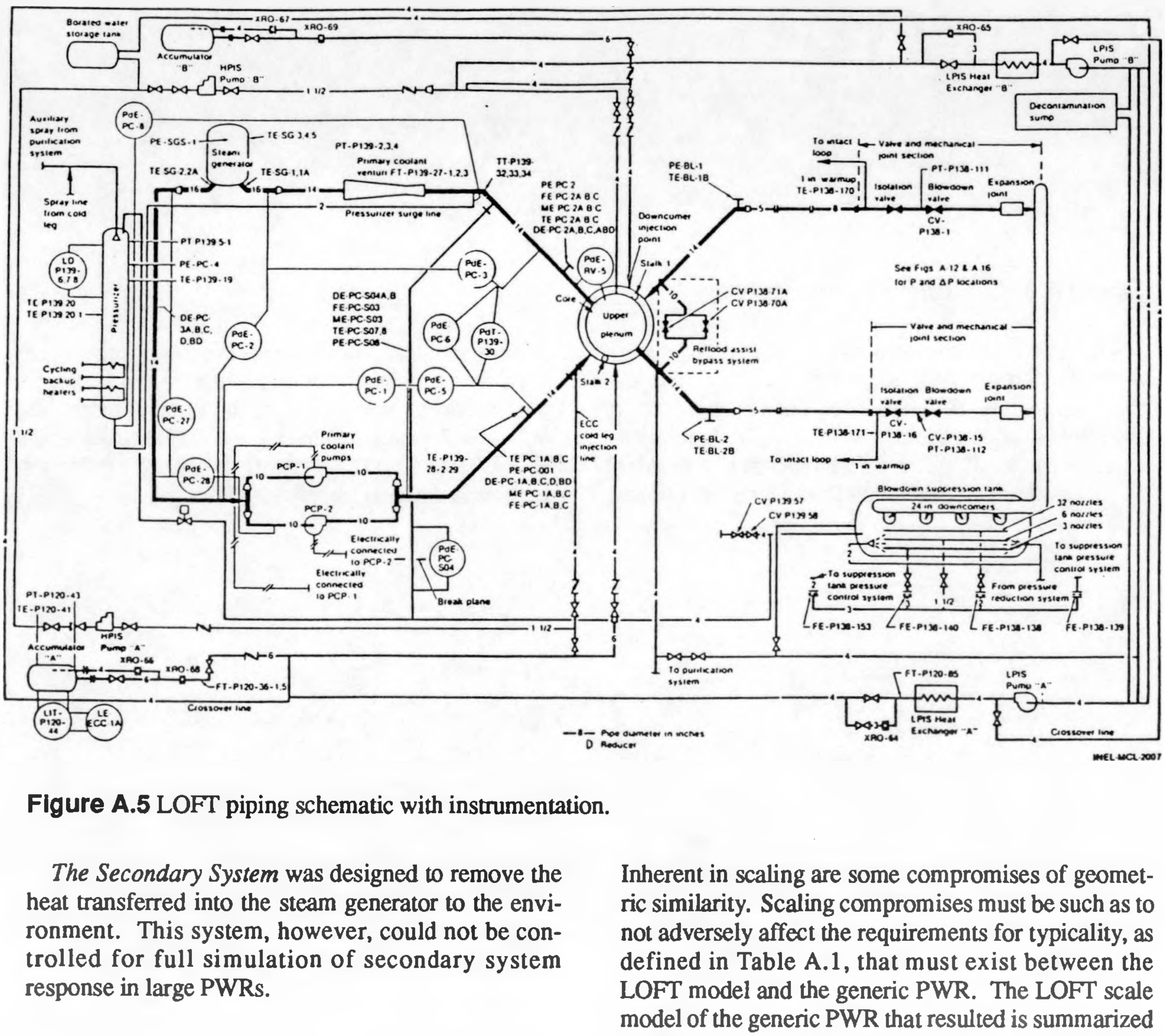

Figure A.5 LOFT piping schematic with instrumentation.

The Secondary System was designed to remove the heat transferred into the steam generator to the environment. This system, however, could not be controlled for full simulation of secondary system response in large PWRs.

LOFT Facility Scaling

The LOFT facility was scaled to generic PWRs by maintaining the system and component coolant volume–to–total–power ratio whenever possible.[A–2] Inherent in scaling are some compromises of geometric similarity. Scaling compromises must be such as to not adversely affect the requirements for typicality, as defined in Table A.1, that must exist between the LOFT model and the generic PWR. The LOFT scale model of the generic PWR that resulted is summarized in Table A.2, which contains comparisons of geometric and physical parameters between LOFT and commercial PWRs. The physical parameters listed are for nominal operating conditions in the Westinghouse four–loop ZION PWR and in the LOFT model prior to the LOCE designated L2–3.

Table A.1 Typicality requirements for the LOFT model design

Item	Reason
System volume to core power ratio	Distribution of energy
Break area to system volume ratio	Depressurization of event time similarities
Length–to–diameter ratios (system resistance)	Pressure drop balance
Elevation	Pressure distribution
Surface area to volume ratios	Heat transfer distribution
Core power distribution	Thermal response

The values listed in Table A.2 indicate that the coolant volume–to–total core power ratio is not exactly the same between LOFT and TROJAN and LOFT and ZION. The differences are due to the design compromises that were made.

Table A.2 LOFT – commercial PWR comparisons

Item	LOFT		TROJAN	
	Volume (m^3)	Total (%)	Total (%)	Volume (m^3)
Reactor Vessel				
Outlet Plenum	0.95	12.51	15.95	55.47
Core and Bypass	0.31	4.12	7.50	26.05
Lower Plenum	0.71	9.32	8.58	29.73
Downcomer and Inlet Annulus	0.69	9.00	5.89	s20.42
Subtotal		34.95	37.95	
Intact Loop[a]				
Hot Leg Pipe	0.35	4.60	1.94	6.71
Cold Leg Pipe	0.37	4.85	2.08	7.22
Pump Suction Pipe	0.33	4.38	3.09	1 0.70
Steam Generator	1.45	18.97	26.40	91.49
Pump	0.20	2.60	1.96	6.80
Subtotal		35.40	35.47	
Broken Loop				
Cold Leg to Break[b]	0.16	2.16	1.72	5.97
Vessel to Steam Generator	0.15	1.98	0.65	2.24

Table A.2 (continued)

Item	LOFT		TROJAN	
Steam Generator	0.52	6.88	8.80	30.50
Pump	0.05	0.72	0.65	2.27
Additional Volume				
Part of Outlet Plenum	0.19	2.46	N/A	N/A
Additional Volume Part of Inlet Plenum	0.22	2.83	N/A	N/A
Pressurizer	0.96	12.62	14.7	50.97
Total	7.63	100.00	100.00	346.60

Item	LOFT	ZION
Core (LOFT L2–3, ZION nominal conditions included)		
Fuel rod number	1300	39372
Length (m)	1.68	3.68
Inlet flow area (m^2)	0.16	4.96
Coolant volume (m^3)	0.295	20.227
Maximum linear heat generation rate (KW/m)	39.4	39.4
Coolant temperature rise (K)	32.2	32.2
Power (MW)	36.7	3540.5
Peaking Factor	2.34	1.60
Power/coolant volume (MW/m^3)	124.4	175.0
Core volume/system volume	.038	.057
Mass flux ($Kg/s\text{–}m^2$)	1248.8	3707.3
Core mass flow/system volume ($Kg/s\text{–}m^3$)	25.6	51.7

a. TROJAN values are for three loops combined.
b. Includes pump suction piping.

Instrumentation

The LOFT facility was augmented with an extensive "experimental" measurements system[A–1] in addition to the normal PWR instrument systems for reactor operation and control. The following parameters were measured with the experimental instrumentation: temperature, pressure, differential pressures, density, coolant velocity, coolant momentum flux, liquid levels, pump speed, and neutron flux.

State measurements of the coolant in the primary system provided the capability of following the redistribution of mass and energy in the primary coolant system following the initiation of a transient. Extensive thermal measurements in the nuclear core provided detailed information on the thermal response of the fuel cladding and fuel centerline temperatures. Nuclear measurements in the core assisted in determining the initial or steady state energy distribution. The philosophy followed on measurement locations in the nuclear core, as shown in Figure A.4, was to instrument one–half of the core on a circular symmetry basis with emphasis on the center fuel assembly. The intent was to permit determination of the thermal and mechanical effects of instrumentation on the fuel rods during post–irradiation analysis. Utilizing circular symmetry simplified the core structure by permitting identical fuel assemblies to be used in core locations 2, 4, 6, in locations 1 and 3, and in locations 7 and 9.

Experimental measurements were also located on the ECC systems, the secondary coolant system, the pressure suppression system, and on components such as pumps, valves, and control rod drive mechanisms for mechanical operation measurements during a transient. Location of the major experimental instrumentation is indicated in Figure A.5. The nomenclature for the LOFT instruments is presented in Table A.3.

Table A.3 Nomenclature for LOFT instrumentation

Te	–	Temperature element
TT	–	Temperature transmitter
PE	–	Pressure element
PT	–	Absolute pressure transmitter
PdE	–	Differential pressure element
LT	–	Coolant level transmitter
FE	–	Coolant flow element
FT	–	Coolant flow transmitter
AE	–	Acclerometer
RPE	–	Pump speed element
DIE	–	Displacement element
ME	–	Momentum flux detector
NE	–	Neutron detector
PNE	–	Pulse neutron detector

Temperatures were measured in LOFT using three types of thermocouples: Type K – chromel versus alumel; Type S – platinum versus platinum 10% rhodium; Type T – copper versus constantan. There were two groups of mechanical design of the thermocouples: the grounded spade junction and the grounded weld junction. The spade junctions were used as metal surface temperature measuring devices and the grounded weld junctions thermocouples were primarily used as coolant temperature measuring devices.

Pressure measurements were made by two type of transducers: free field and standoff absolute. The free field transducers were used for the subcooled portion of the blowdown. This type of transducers is characterized with very fast response time but they are also sensitive to temperature changes. The standoff transducers were used for pressure measurements during the two–phase part of the transient, they are slower in response but less temperature sensitive.

Differential pressure was measured using transducers similar to the standoff absolute pressure transducers with the diaphragm separating the high and low pressure fields.

Coolant density was measured in the hot and cold legs of the primary system using three beam gamma densitometers. A 22–Ci, Co–60, source was used. The source was collimated into three beams as indicated in Figure A.6. NaI scintillation cells with photomultiplier tubes were used as detectors. There was also a fourth detector used to measure the background radiation for subtracting it from the actual measurement. The density measurement with the three beams allowed recognition of flow regimes in the piping and could be used to infer liquid level measurements.

Coolant velocity, momentum flux and flow direction was measured using drag disk–turbine (DDT) assemblies. Such assemblies located in the hot and cold legs of the primary coolant piping consisted of three drag disk and turbine groups and a thermocouple as shown in Figure A.7. The drag disk device consisted of cylindrical drag body and a linear variable differential transformer to detect motion. The drag disk measured coolant momentum flux and indicated flow direction. The turbine was a six bladed turbine with graphite bearings and eddy current coil to pick up blade movement. Additional drag disk turbine assemblies were installed at the inlet and outlet to the core. The DDT turbine was calibrated to measure velocity in either direction.

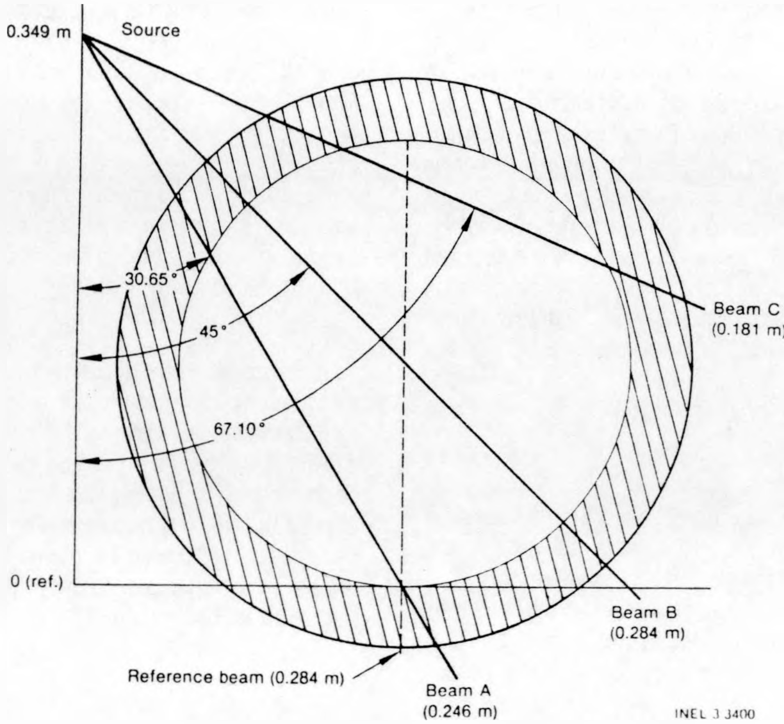

Figure A.6 Gamma Densitometer arrangement in the hot leg of the intact loop.

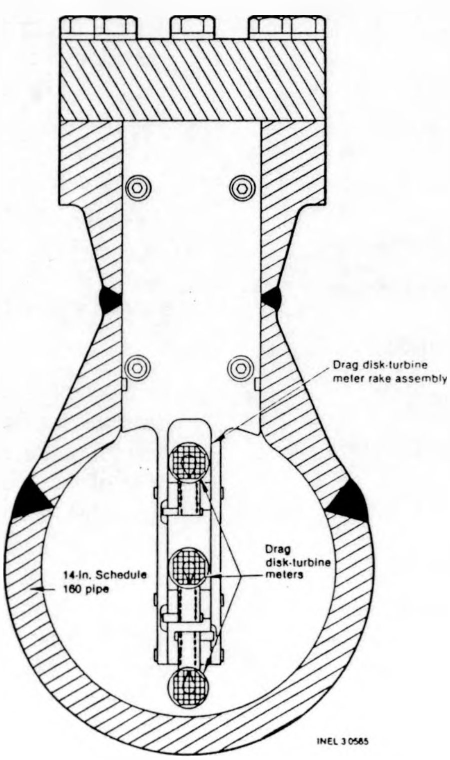

Figure A.7 Drag and Disk–Turbine (DDT) assembly.

Liquid level in the reactor vessel was measured at several locations using electrical conductivity probes consisting of several electrodes at various intervals in a tube. The tube was perforated at each electrode to provide good communication between the electrode and surrounding fluid. The presence of absence of liquid was determined by measuring the electrical conductivity of the surrounding fluid.

The primary coolant pump speed was monitored with a transducer consisting of an eddy current pickup coil mounted in the pump bearing cavity on the elevation of a tachometer plate mounted to the shaft with radial slots. The direct current readout was converted to revolutions per minute.

Displacement transducers were used in LOFT to measure the dynamic vertical motion and thermal displacement of the central fuel assembly in the core. The device used a linear variable differential transformer with two coils and a floating core attached to the upper core support structure sleeve. The transducer core was attached to the upper core support structure.

The neutron flux was measured with two types of transducers: scanning and fixed location detectors. The scanning detector, a traversing incore probe, provided graphs of the axial flux distribution at four different locations in the core. The fixed detectors used Co–60 neutron flux detectors for fast response. These detectors are called self–powered gamma detectors, because they use the current generated by decay of cobalt to indicate power level.

OECD LOFT Experimental Configuration

For each experiment, the facility was configured according to the experiment objectives. The following indicates system changes that were made for individual experiments. A standard LOFT large–break LOCA and anticipated transients configuration is assumed. Also, special instrumentation used in fission product experiments LP–FP–1 and LP–FP–2 is presented.

1. Experiment LP–FW–1:

Standard LOFT configuration, Figure A.2.

2. Experiment LP–SB–1:

The configuration of the LOFT primary system for Experiment LP–SB1 is shown in Figure A.8. The break location was in the hot leg of the intact loop between the steam generator and the reactor vessel. The break nozzle was in the break piping connecting the midplane of the intact loop hot leg to the blowdown suppression tank. The break piping and the relative location of the instrumentation in the line are shown in Figure A.9.

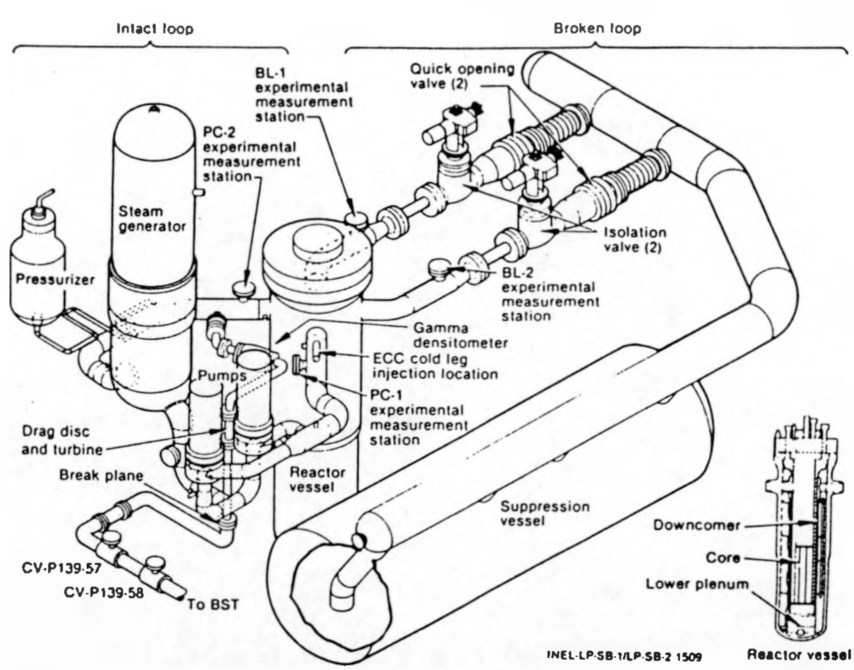

Figure A.8 Axonometric projection of the LOFT system configuration for Experiments LP–SB–1 and LP–SB–2.

3. Experiment LP–SB–2:

Same as Experiment LP–SB–1.

4. Experiment LP–SB–3:

Configuration of the LOFT Facility for Experiment LP–SB–3 is shown in Figure A.10. The break location was in the cold leg of the intact loop between the primary coolant pumps and the reactor vessel. The break nozzle was in a pipe that connected the intact loop cold leg to the blowdown suppression tank. Figure A.11 shows the configuration of the break piping and the relative location of the experiment instrumentation.

5. Experiment LP–02–6:

Standard LOFT configuration, Figure A.2. New center fuel module provided with prepressurized fuel pins.

6. Experiment LP–LB–1:

Standard LOFT configuration, Figure A.2. Center fuel module with unprepressurized pins.

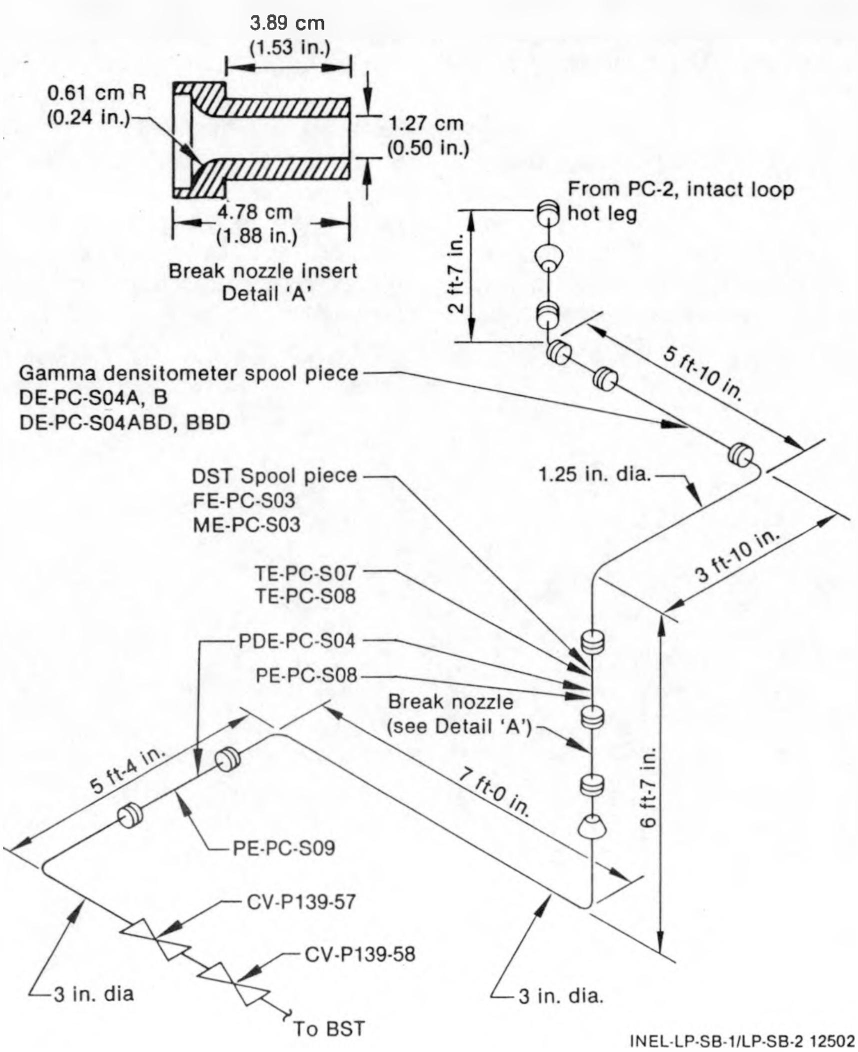

Figure A.9 Experimental spool piece configuration for hot leg break piping.

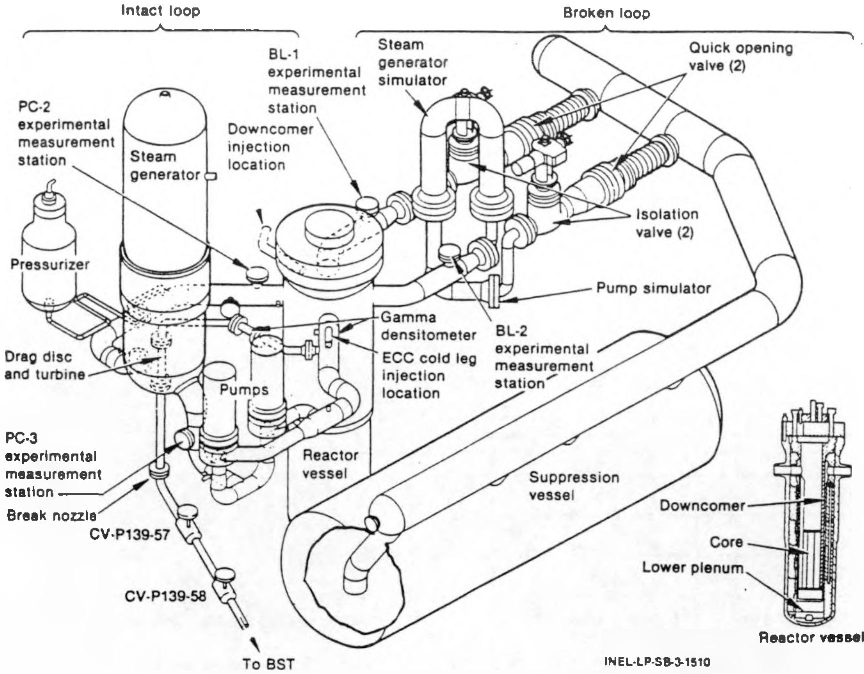

Figure A.10 LOFT configuration for cold leg intact loop small–break experiment LP–SB–3.

7. Experiment LP–FP–1:

Standard LOFT configuration, Figure A.2.For this experiment special center fuel module was manufactured with a zircaloy shroud. This fuel module included 24 enriched to 6–wt% ^{235}U (regular fuel enrichment in LOFT was 4–wt% ^{235}U). Figure A.12 shows a cross-section of the center fuel module and indicates the instrumentation in the module. Twenty–two of these were prepressurized at cold conditions to 2.41 MPa.

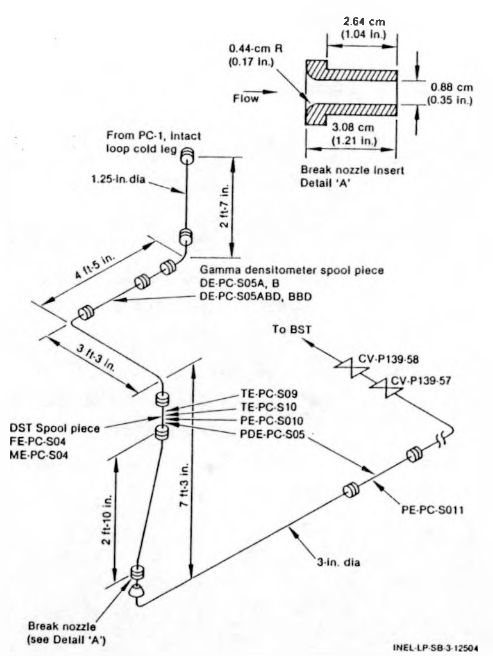

Figure A.11 Break piping configuration.

A Special fission product measurement system (FPMS) was designed for this experiment. The FPMS consisted of three basic systems: the steam sample system, which was operated during the transient phase of the experiment; the gamma detection system, which was operated during the 12 hour post-transient phase, and the deposition coupons, which collected samples during both phases. Figure A.13 shows the FPMS system schematically.

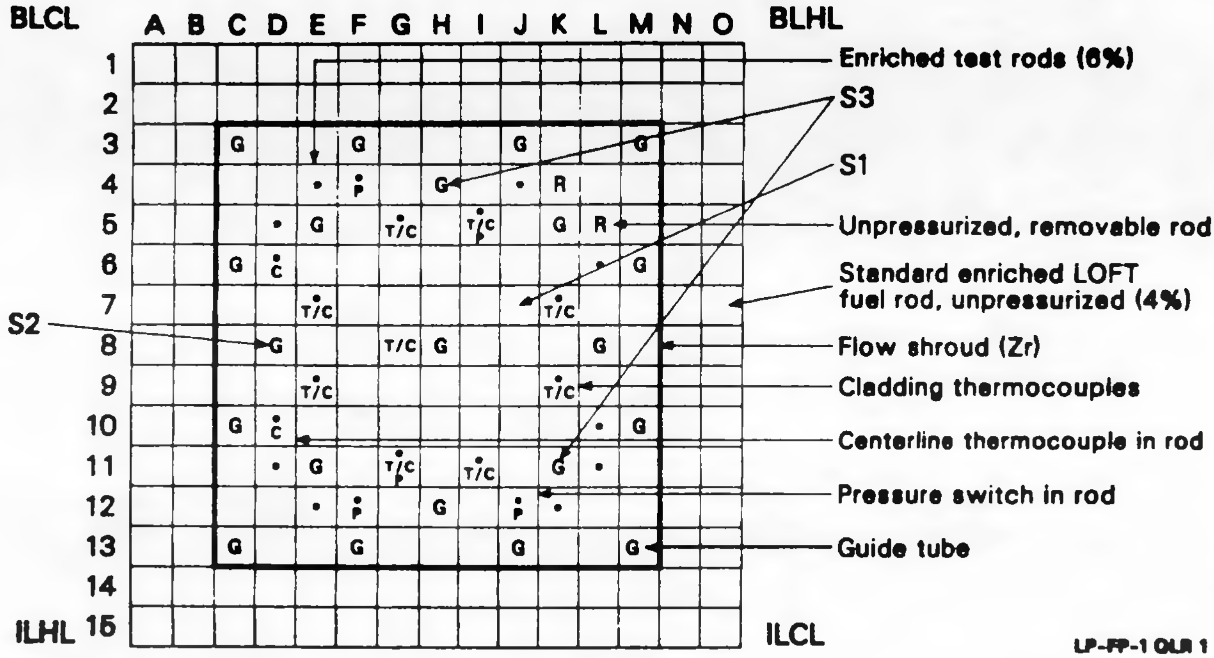

Figure A.12 Central fuel assembly instrumentation locations.

The steam sample system had four sampling locations:

S1 – in upper plenum about 7 cm above the center fuel module upper tie plate and directly below the upper plenum ECC injection port.

S2 – in the center fuel module, 171 cm above the top of the lower tie plate.

S3 – two samples drawn from center fuel module 115 cm above the lower tie plate.

S4 – broken loop hot leg upstream of the steam generator simulator.

The sample lines were routed from the sample points to the instrumentation and processing equipment mounted on a movable skid. The lines were heat traced and kept to a minimum length to maximize the fission product transport to the instrumentation. The instrumentation on the skid included gross gamma detectors, flow meters, iodine species samplers, steam condensers, liquid traps and temperature and pressure measurements.

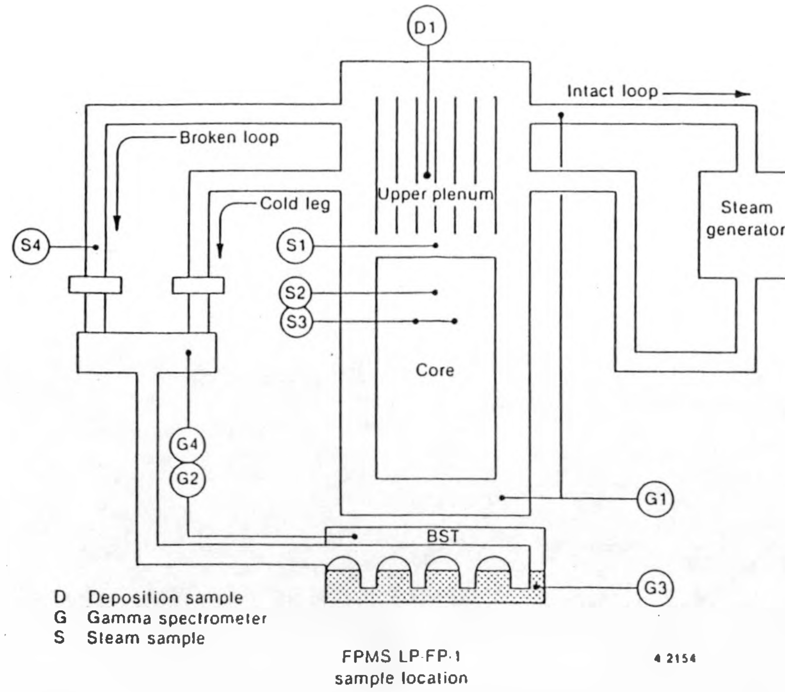

Figure A.13 FPMS schematic.

The deposition coupons were located in the reactor vessel upper plenum on three elevations: 15, 61, and 165 cm above the upper tie plate. On each elevation were two coupons and both were exposed to the reactor environment during the heatup phase. One coupon at each elevation was isolated and sealed prior to initiation of reflood, while the second coupon remained exposed.

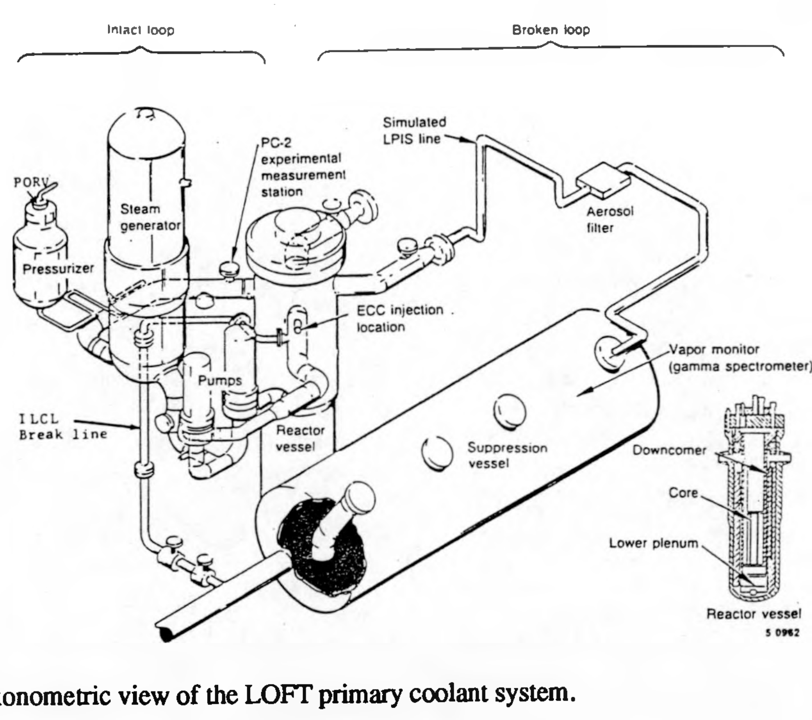

Figure A.14 Axonometric view of the LOFT primary coolant system.

Three gamma spectrometers were used in the experiment to provide a real time quantitative measurement of the radio isotopes present in the LOFT system during the 12 hour post-transient sampling period. The sample points are shown in Figure A.13.

8. Experiment LP–FP–2:

The configuration of the LOFT facility for Experiment LP–FP–2 is shown in Figure A.14. Important changes were made to the LOFT facility in order to conduct the LP–FP–2 experiment. These changes included removal of the broken loop cold leg piping and the simulated steam generator, removal of the blowdown valves and the blowdown header, installation of a simulated LPIS line at the broken loop hot leg, a special center fuel module and addition of the fission product measurement system (FPMS).

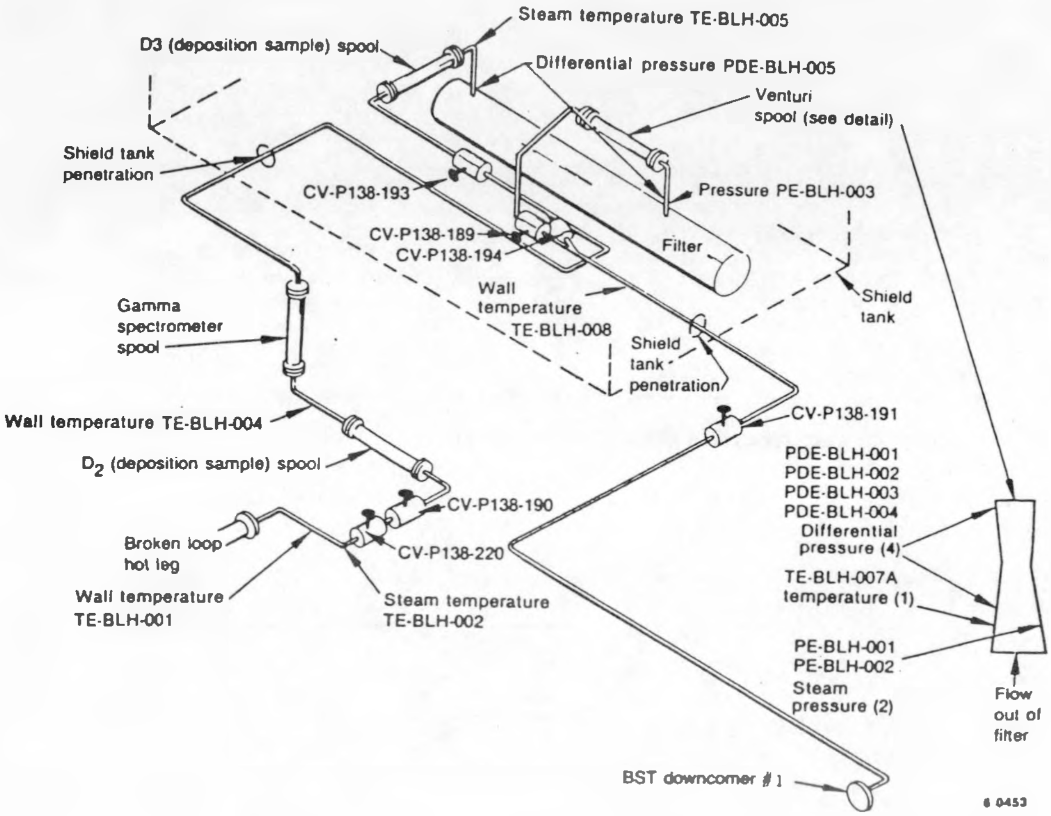

Figure A.15 LOFT simulated LPIS line.

The simulated LPIS line (shown in Figure A.15) was scaled to represent correctly a LPIS line of a commercial Power plant. The required scaling parameters included the break path flow area and LPIS line length. Break area scaling provided representative thermal–hydraulics, and specifically, similar coolant velocities for transport of fission products and aerosols. LPIS pipe length scaling was necessary to provide similar residence times for transport and retention phenomena in the LPIS piping. The scaling rationale is described in Appendix A of the EASR.[A–3]

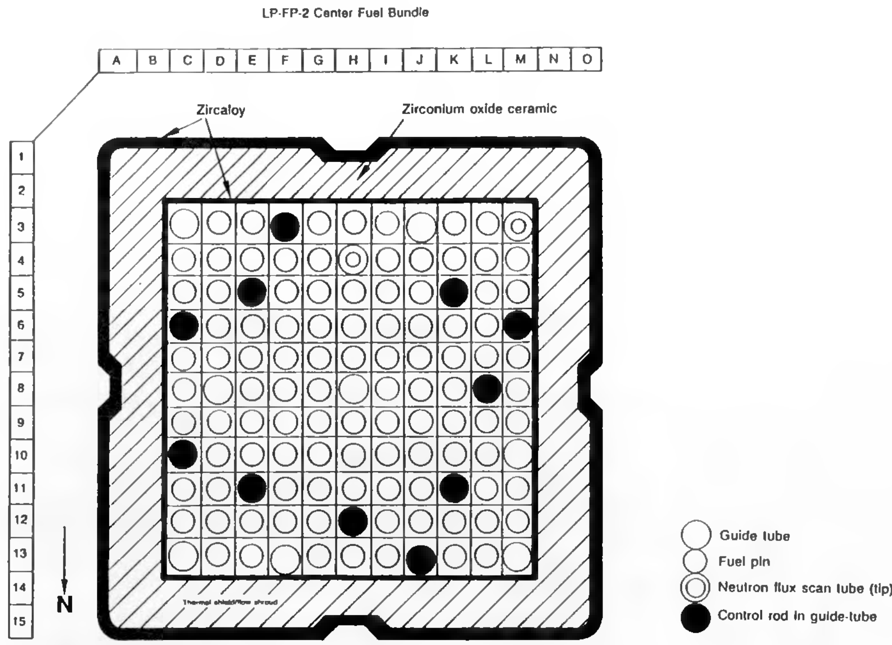

Figure A.16 LOFT core central fuel module (CFM) design.

Design of the center fuel module for experiment LP–FP–2 is shown in Figure A.16. The outer two rows of fuel rods in the standard 15 x 15 array were replaced with a shroud that provided thermal insulation and hydraulic separation of the remaining 11 x 11 array of fuel rods from the peripheral modules. The shroud consisted of zircaloy walls with zirconium oxide ceramic internal insulation. The fuel rods within the CFM were enriched to 9.74 wt% ^{235}U. The purpose of increasing the enrichment was to provide at least three minutes of cladding temperatures above 2100 K before the peripheral fuel rods reach the damage limit of 1462 K.

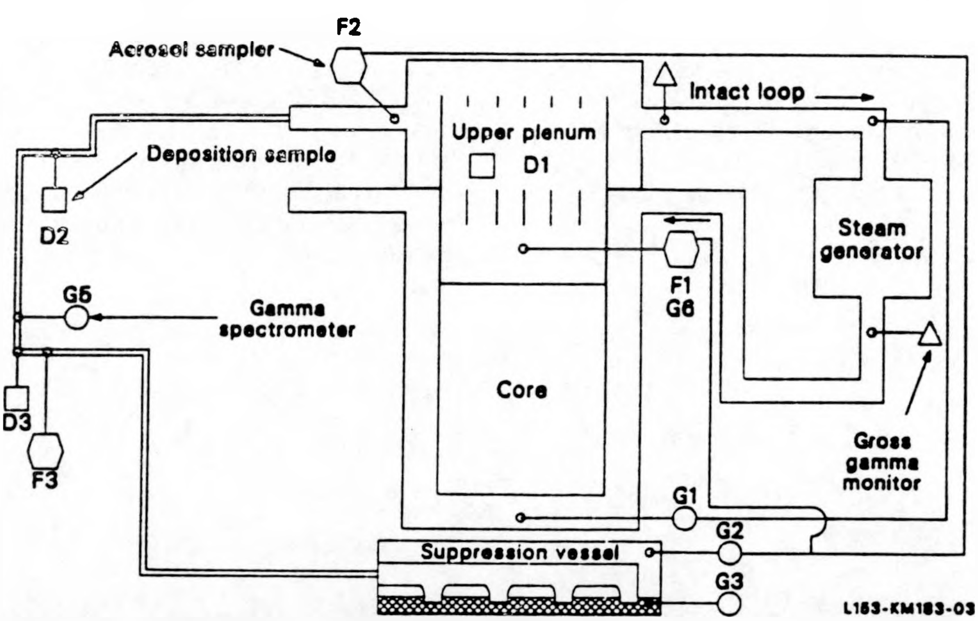

Figure A.17 LOFT LP–FP–2 FPMS instrumentation.

The fission product measurement system, illustrated in Figure A.17, consisted of three basic subsystems: (a) four gamma spectrometer systems and one gross gamma detector, (b) a deposition sampling system, and (c) filter sample systems.

The four on-line gamma spectrometers and the G6 gross gamma monitor were located at five different sample locations: (a) G1 sampled from the reactor vessel lower plenum or, alternatively, from the intact loop hot let; (b) G2 sampled from the blowdown suppression tank vapor spaces during the post-transient, and from the combined F1+F2 sample lines during the transient phase of the experiment; (c) G3 sampled from the blowdown suppression tank liquid space; (d) G5 sampled from the simulated LPIS line during the transient and post-transient; and (e) G6 sampled the F1 line at the top of the reactor vessel. The G4 detector was used during Experiment LP-FP-1 and was not used in this experiment. Each gamma spectrometer was designed to operate remotely and could be calibrated using a ^{228}Th source mounted on a collimator wheel. With the exception of G5 and G6, this system operated only during the post-transient phase.

The deposition sampling system consisted of six stainless steel coupons and two deposition spool pieces. Two coupons were located at each of three elevations above the central fuel module (for a total of six coupons, collectively designated D1). At each elevation, both coupons were exposed to the fluid stream during the transient. One coupon at each elevation was to be isolated from the PCS prior to initiation of reflood while the other coupon remained exposed to the fluid. However, the protective cover did not seal around the lowest level coupon and contact with reflood water occurred. The other coupons functioned as planned. The two deposition spool pieces, located near the inlet and outlet of the simulated LPIS Figure A.15. Schematic of the LOFT system showing the relative position of the FPMS instrumentation line header, designated D2 and D3 respectively. These spool pieces were designed to provide a measurement of the primary coolant system surface deposition of volatile fission products during the heatup or transient. Since this line was isolated prior to reflood, these coupons were protected from the reflood water and therefore did not experience postexperiment deposition, leaching, or removal of reversibly plated fission products.

The final FPMS subsystems consisted of two aerosol/steam sampling lines with corresponding equipment and an aerosol filter system on the LPIS line. These sample lines were designed to provide a continuous sample of the vapor and aerosols generated during the heatup phase of the experiment. The F1 sampling line consisted of the following major components:

1. Sample line probe placed above the CFM.

2. Argon dilution gas supply.

3. Dual cyclone separator/isolation valves.

4. Dilution filter.

5. Virtual impactor.

6. Collection filters.

7. Infrared moisture detectors.

8. Hydrogen recombiner.

The F2 sampling line was similar to the F1 line, except that there were no dilution gas supply and moisture detectors. The F3 filter sampling system consisted of the D2 and D3 deposition spool pieces, a filter, and a flow venturi. The three sample line locations are: F1, 180 cm) above the top of the lower tie plate and located directly above the center fuel module; F2, the broken loop hot leg spool piece just outside of the upper plenum, and F3, the exit of the simulated LPIS line header.

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APPENDIX B

COMPARISON OF LOFT LP–FP–2 WITH TMI–2 AND PBF–SFD–ST

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A comparison is made between principal features of the results of the LOFT LP–FP–2 experiment and results from the examination of the damaged core of the Three Mile Island Unit–2 (TMI–2) reactor. Observations from other severe fuel damage tests, such as those carried out in the Power Burst Facility (PBF), are also utilized. Melt progression, the effect of spacer grids, damage to upper core support structures, energy and hydrogen generation upon reflood, and the retention of volatile fission products in high temperature melts are discussed.

First, it should be mentioned that there are both similarities and large differences between the LOFT and TMI–2 reactors and between the conditions of the LP–FP–2 experiment and the TMI–2 accident. Some of this information is presented in Table B.1 for these two reactors, along with data on the conditions during the PBF Severe Fuel Damage Scoping Test (PBF–SFD–ST). The test bundles are very small scale relative to the TMI–2 reactor (<0.2 % by volume). In the TMI–2 accident, and in both tests, steam was supplied to the bundle by boiloff of coolant within the reactor pressure vessel, and the excursion was halted by reflooding. The LOFT LP–FP–2 experiment was powered by decay heat as was the TMI–2 accident, whereas, the SFD–ST experiment used fission heat. Fission heat is generated in the uranium fuel, whereas, decay heat is associated with fission products that may relocate to some extent from the fuel during the course of a severe accident. A large flow bypass (~80% of the LOFT core flow area) surrounding the central fuel module that was undergoing severe damage existed in the LOFT LP–FP–2 test. The annular bypass surrounding the TMI–2 core has a flow area of about 1.5% of the core flow area. In the PBF–SFD tests, a small zirconia tube provided a bypass representing about 0.5% of the test bundle flow area. The coolant in both the FP–2 test and the TMI–2 accident contained boric acid which is typical for pressurized water reactors, whereas, the coolant used in the PBF–SFD tests did not. Research at Winfrith in the UK has shown that boric acid can affect the chemical form of the fission products iodine and cesium.[B–1] Finally, it should be noted that the LP–FP–2 experiment simulated a V sequence, which is a large–break LOCA that depressurized the LOFT core to 1.4 MPa during the high temperature portion of the transient, whereas, the TMI–2 accident[B–2] was a small–break LOCA involving system pressures greater than 5 MPa. The PBF–SFD tests[B–3, B–4, B–5, B–6] were designed to operate at a constant pressure of 6.9 MPa to simulate the TMI–2 accident.

Table B.1 Comparison of LOFT LP–FP–2 with the TMI–2 accident and PBF–SFD–ST

	PBF–SFD–ST	LOFT LP–FP–2	TMI–2
Geometry			
Bundle size	32	121	36,816
Bundle length (m)	0.91	1.67	4
Bypass (%)	0.5	80	1.5
Pressure (MPa)	6.9	1.4	5+
Power	Fission	Decay	Decay
Steam supply	Boiloff	Boiloff	Boiloff
Cooldown	Reflood	Reflood	Reflood

The mode of in-vessel core melt progression can have important consequences for later stages of an accident, particularly the timing and, perhaps, the mode of lower head failure and subsequent ex-vessel processes, such as direct containment heating and core/concrete interaction. Core melt progression in the TMI-2 accident is briefly described and comparisons with the LOFT LP-FP-2 test follow. The origins of the metallic crust, the molten ceramic pool, the ceramic melt relocation to the lower plenum, and the upper debris bed, illustrated in the diagram of the postaccident configuration within the TMI-2 reactor pressure vessel presented in Figure B.1, are described.

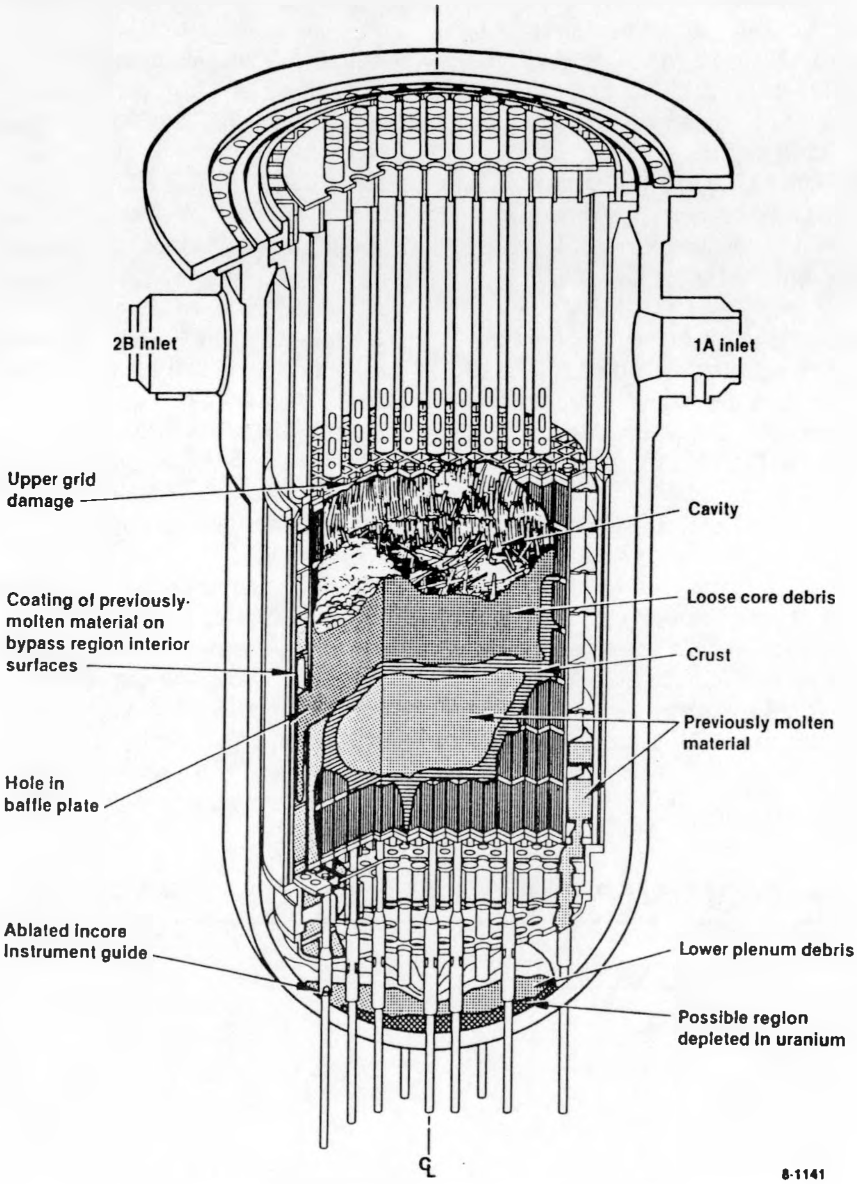

Figure B.1 TMI-2 Core End-State Configuration.

In the TMI-2 accident, metallic core materials (principally silver-indium-cadmium control rod alloy, iron, nickel, and chromium form control rod stainless steel cladding), and zircaloy from control rod guide tubes and fuel rod cladding, relocated early in the core heatup to lower, cooler portions of the core and froze forming a crust upon which subsequent ceramic melts froze and accumulated later in the accident sequence. This ceramic melt continued to relocate, freeze, and collect until the mass became uncoolable and it remelted in the center, forming a large pool surrounded by a ceramic shell. Natural heat convection within the pool caused a thinning and eventual failure of the ceramic crust caused by thermal–mechanical forces high on one side of the pool. This allowed a relocation of about half of the ceramic melt to the lower plenum of the reactor within a one to two minute period. This melt relocation to the lower plenum was through water which caused fragmentation of the melt into debris of varying sizes which was eventually coolable by a continued supply of coolant from the high–pressure injection system, which prevented failure of the bottom head. The remaining melt in the core was small enough to eventually freeze without relocationg to the lower plenum. It is conceivable that, without the formation of an initial metallic crust, a considerably larger fraction of molten core debris could relocate directly to the lower plenum, providing a different, and perhaps stronger, challenge to the lower head.

The LP–FP–2 test was run at high temperature for a much shorter time than was experienced in the TMI–2 accident, so one can only expect to see evidence of the initial formations of various material relocations and accumulations. The upper region of the LP–FP–2 bundle consists of a debris bed of fuel pellet fragments similar in appearance to the upper debris bed formed in the TMI–2 accident. This debris bed could have formed either by the shattering of oxidized fuel rods as a result of the reflood, as is thought to have occurred in the TMI–2 accident during the B–Loop pump transient and in the PBF–ST, or by cladding melting and relocation (leaving fuel pellet fragments without restraint) during either the transient or reflood phases of the experiment, as occurred during the transient of the PBF–SFD 1–4 test.[B–6]

The lower blockage in the LP–FP–2 test is made up primarily of molten metals (silver and indium from control rods, zirconium from guide tubes and fuel rod cladding, and iron, chromium and nickel from stainless steel control rod cladding and from Inconel spacer grids) that relocated downward and froze at the lower spacer grid location early in the transient. This material is analogous to the primarily metallic lower crust that formed in the TMI–2 accident and the metallic lower blockage that formed in Test PBF–SFD 1–4.[B–6] The ceramic blockage that formed above the second spacer grid in the LP–FP–2 test represents the beginnings of a ceramic pool such as formed above the lower crust in the TMI–2 accident and is similar to the ceramic blockages that formed in all four of the PBF SFD tests.[B–3, B–4, B–5, B–6]

The two blockages that formed in the LP–FP–2 test occurred at spacer grid locations. Similarly, the lower crust blockage that occurred in the TMI–2 core was located at the lowest spacer grid and the blockages that ocurred provided an impediment to the flow of melt and additional surface area for heat transfer. Both of these factors tend to promote freezing of relocating melts at spacer grids in regions of the core that are at temperatures below the freezing point of the melt.

Retention of significant fractions of the fission products, iodine and cesium (normally considered to be highly volatile) were found in ceramic fuel–containing debris that had been molten during the LP–FP–2 experiment. Similar results have been obtained in ceramic debris that relocated to the lower plenum in the TMI–2 accident.[B–7] Measurements from micro–gamma scanning, chemical element distribution, and scanning electron microscopy on ceramic debris removed from the lower plenum of the TMI–2 reactor pressure vessel suggest that cesium may be retained in an oxide form associated with another metallic ion. Forms such as Cs_2CrO_4 and $CsFeO_2$ are possible. Other possibilities for cesium compounds stable at high temperatures with low vapor pressures may include Cs_2MoO_4, Cs_2ZrO_3, and $CsBO_2$. Thermodynamic data required to evaluate the high temperature (~ 3000 K) stability of these compounds are not available.

The stainless steel upper tie plate at the top of the center fuel module in the LP–FP–2 test sustained severe localized damage due to melting and oxidation. Similar localized damage was observed in the stainless steel upper core support plate in the TMI–2 core.[B–2] In the evaluation fo the LP–FP–2 test, it was concluded that this damage occurred during reflood as a result of the generation of high temperature effluent (steam and hydrogen) due to the exothermic zirconium–steam reaction in the center fuel module. An assessment of possible mechanisms and the energetics associated with the damage to the upper core support plate in the TMI–2 core suggests that the damage likely occurred during the B–loop pump transient as a result of high temperature steam and hydrogen generated by zircaloy–steam reaction in the degraded core.[B–8] These evaluations indicate that similar mechanisms are responsible for damage to stainless steel upper core support structures in both the LP–FP–2 experiment and the TMI–2 accident.

An evaluation of the thermocouple responses in the bundle and within the upper tie plate region and the hydrogen measured in the primary coolant system following the LP–FP–2 experiment suggests that considerable energy and hydrogen (80% of the hydrogen produced during the test) were produced in the center fuel module during reflood. An analysis of the increase in system pressure measured during the B–loop transient in the TMI–2 accident and the associated energetics suggests that up to 32% of the hydrogen generated in the accident could have been produced during this event. It should be noted that hydrogen peaks following reflood have also been measured during the PBF–SFD–ST[B–3] and the CORA–12 out–of–pile severe fuel damage test (in which temperature escalation was also measured) at

KfK.[B-9] These experiences with reflooding high temperature damaged fuel bundles indicate that considerable energy and hydrogen production can be anticipated. Therefore, in terms of accident management, it is important to supply enough coolant to remove this newly created energy as well as the sensible heat already present in the core, and to accommodate the hydrogen produced in the reflood process.

This comparison of principal features of the LP–FP–2 experiment and the TMI–2 accident reveals a number of important core damage phenomena that are common to both the accident and small scale severe fuel damage experiments that span a variety of scales and conditions. The extent to which these phenomena are present depend not only on the thermal–hydraulic conditions such as steaming rate, system pressure, and reflooding, but also on the key parameters of steam availability and time at temperature.

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APPENDIX C

OECD LOFT MEMBERS STAFF ATTACHED TO THE PROJECT AT THE INEL

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