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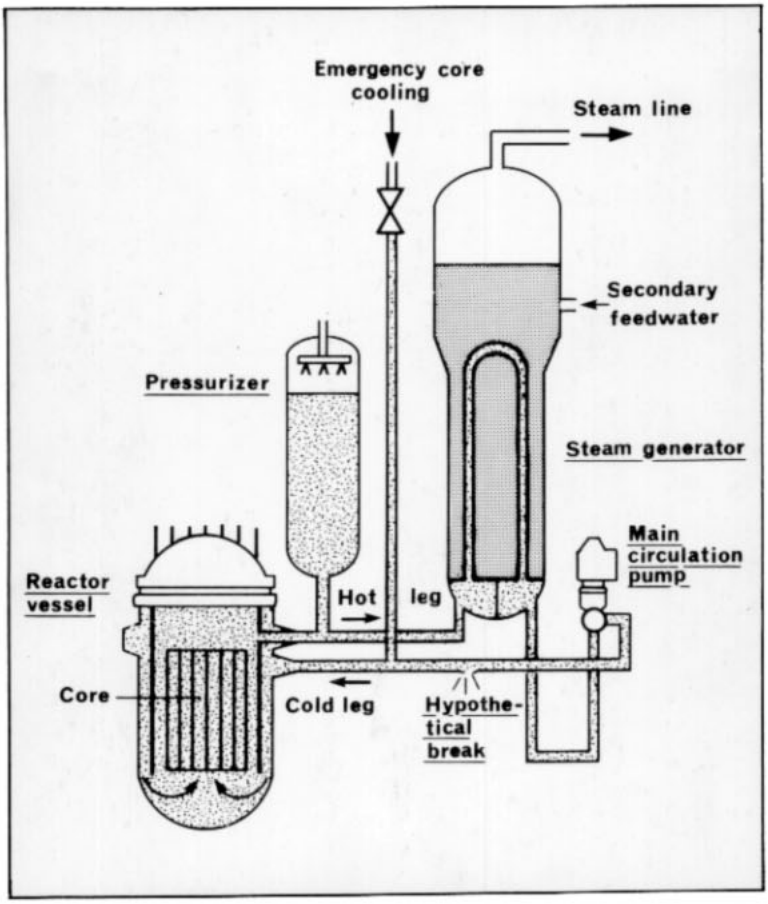
COMPUTER CODES FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENTS

A NORDIC ASSESSMENT



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Computer Codes for Small-Break Loss-of-Coolant Accidents

A Nordic Assessment

Final report of the NKA project SÄK-3

Compiled by

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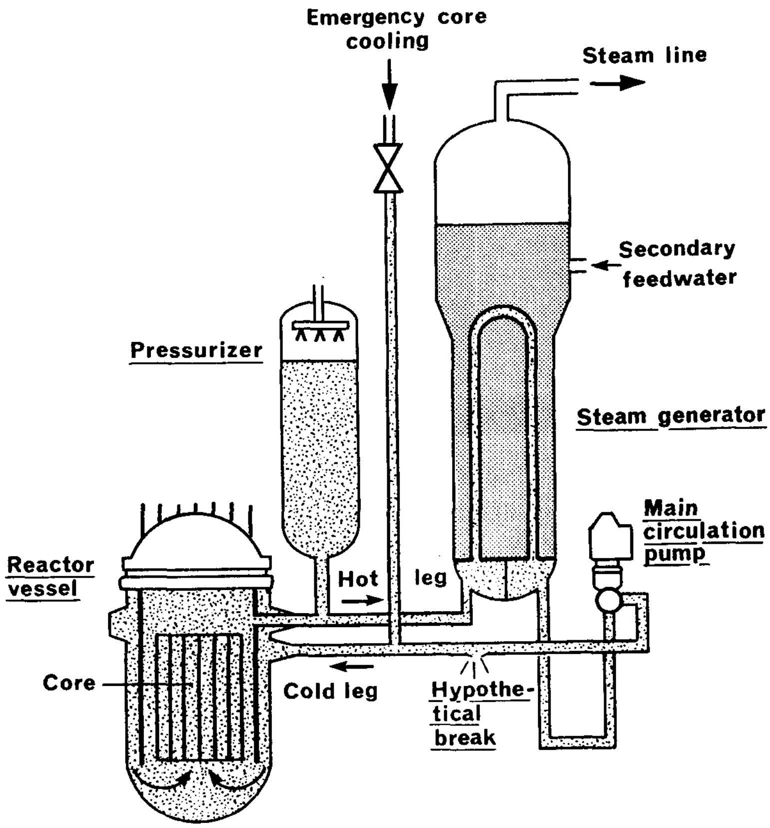

Cover picture: Principal coolant circuit in a pressurized water reactor (PWR).

4 out of 5 test cases in the present study concern a small-break LOCA in a PWR. The assumed break position in these test cases is indicated. During a LOCA emergency core cooling water is supplied from (1) the accumulator (pressurized with nitrogen) and from (2) the high pressure injection system (HPIS).

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During the SÄK-3 project a considerable change of persons has taken place in the working group. The list of participants covers all persons who have actively contributed to the project, i.e. authors of introductory reports, level 0 and level 1 reports, and their work is hereby acknowledged.

Abstract

An assessment of computer codes for analysis of small-break LOCA's is performed.
The work comprises the American systems codes TRAC/PF1, RELAP5/MOD1, RELAP5/MOD2 and the Finnish fast-running code SMABRE which are assessed theoretically and by comparative calculations of five small-break LOCA experiments. On this basis comparisons are made of the advantages and drawbacks of each code in order to conclude which should be selected for small-break LOCA analysis, including a recommandation of necessary modifications.

INIS-Descriptors: COMPARATIVE EVALUATIONS; COMPUTERRIZED SIMULATION; LOSS OF COOLANT; R CODES; S CODES; T CODES; TESTING

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SUMMARY

When assessing the safety of nuclear reactors, so-called Loss-of-Coolant Accidents (LOCA's) are analyzed. In a LOCA the cooling water, which transports the heat from the nuclear core, is lost through a break in the reactor system, with a resulting risk that the nuclear core will be overheated and damaged, perhaps even leading to melt-down of the core. LOCA's are normally classified according to the size of the break relative to the size of the cooling pipes.

Up to about 1980, the main efforts had been concentrated on large-break LOCA's. However, various probabilistic safety studies had indicated that LOCA's caused by small breaks give a considerable contribution to the overall risk spectrum. This fact was highlighted by the Three Mile Island accident in 1979.

Computer models existing at that time to predict consequences of large-break LOCA's could not uncritically be considered suitable also for analysis of small-break LOCA's which have a time scale of hours or perhaps days rather than minutes.

Therefore the present SÄK-3 project was set up early in 1981 with the aim of providing one or more computer codes for small break LOCA analyses.

A small-break LOCA is characterized by a slow decrease in pressure as water or a water/steam mixture escapes through the break, and the decreasing pressure will lead to steam formation through boiling. With running pumps in the cooling circuits the flow velocities may be sufficiently high to maintain the water and steam in a homogenous mixture. However, if the circulation pumps were to stop, the flow velocities would be reduced. This would cause steam to separate from the water and collect in the upper parts of the reactor system; stratified (layered) flow would then be likely to occur in the horizontal pipes.

If no emergency core cooling water were supplied to make up for the liquid escaped through the break the water level in the reactor vessel would continue to drop and eventually uncover the core. This could ultimately result in melt-down of the core.

It was soon recognized that the development from scratch of a dedicated small-break LOCA code would not be possible within the project if it should provide the same degree of detail as existing large-break LOCA codes. Instead, existing codes, which were available in the Nordic countries, have been evaluated regarding their applicability to small break LOCA's.

Thus, three codes developed in the USA, mainly for large-break LOCA's, were studied: TRAC/PF1 (from Los Alamos Scientific Laboratory), RELAP5/MOD1 and RELAP5/MOD2 (from Idaho National Engineering Laboratory). In addition a Finnish, specially developed, simple small-break LOCA code SMABRE was also studied.

The codes were studied in practice by applying them to experimental cases. The code simulations could then be compared with the experimental results and the deviations analyzed. The test cases include small-break experiments performed in the Loss-of-Fluid-Test (LOFT) facility at Idaho National Engineering Laboratory (USA), in the Loop Blowdown Investigation (LOBI) test facility at the EEC Joint Research Centre Ispra (Italy), and in the (Swedish type) BWR integrated test facility FIX-II at Studsvik Energiteknik AB (Sweden).

The LOFT test cases comprise two experiments with a 2.5% break. In one experiment the circulation pump remained running while in the other it was stopped; otherwise conditions were identical. The LOBI test cases comprise a 0.4% break experiment in the original LOBI facility, instrumented for large-break LOCA's, as well as a 1% break experiment in the modified LOBI facility, instrumented for small breaks. This experiment is an international standard problem, ISP18, and especially valuable for code evaluation, as the experimental results are unknown to the participants during their computer calculations.

A single FIX-II experiment with a 48% break was included, which is an "intermediate" break size rather than a "small" one. However, it represents the critical break size for Swedish type BWR'S, i.e. the largest break size not leading to core dryout before a considerable decay of core power has taken place.

A small-break computer code should be able to predict all the major events of a small-break LOCA: pressure decrease, steam formation, separation of steam from water, start/stop/direction of flows in the reactor system, uncovery of the core. The prediction should be qualitatively correct (do the events occur and in what sequence?) and the quantitative predictions (magnitude of pressure, temperatures, velocities; time to uncovery etc.) should not be too far off.

The theoretical study of the codes (review of manuals and code source text) showed that most of the steam/water flow effects occurring in a small-break LOCA are modelled reasonably well, regarding both the purely fluid mechanical effects and representation of the system. One exception is the modelling of the special fluid dynamical effects related to stratified flow, which was found to be either inadequate or missing.

This finding was confirmed by the practical part of the code study - comparison with experimental test cases. The integral properties such as system pressure and temperature were predicted with reasonable accuracy, and in the case where a running circulation pump ensured a nearly homogenous steam/water mixture (the first LOFT-experiment), the prediction of local flow properties such as flow velocity and void fraction (steam volume fraction) was also satisfactory. In all test cases no single code was found to be superior to the others regarding agreement with the experimental results. However, steam/water separation and the stratified flow effects, observed in the test cases with the circulation pumps stopped, were poorly predicted by all the codes studied. Thus, the codes were not able to properly simulate the emptying of a horizontal pipe, connected to the reactor vessel, when the water level in the reactor vessel dropped below the pipe entrance. Large temperature differences were observed in the experiment between steam and water in these pipes, and neither was this predicted by the codes.

The importance of this inadequacy was illustrated by one of the calculations where the complete omittance of stratified flow caused a false core uncovery to be predicted. This resulted in the erroneous prediction of a 400°C temperature rise of the fuel rods to nearly 700°C lasting for about 10 minutes.

Furthermore, it was observed that under conditions simulating boiling water reactors (the FIX-II test case), at least the RELAP5/MOD1 code may be unable to predict rewetting of fuel rods by the droplet spray cooling effect, so that a supplementary, more detailed computer model, dedicated to fuel rod temperatures, may be needed.

Both the theoretical and practical study clearly showed the complexity of the codes, the difficulty in using them and the need to gain experience in using them (especially the "large" American codes). For example, in relation to steam/water separation, an adequate modelling requires that the user specifies a sufficiently fine nodalization (subdivision of flow volumes into computational cells) in the parts of the system where separation may occur. It is therefore also an important result of the SÄK-3 study that the participants, through the running of the test cases, have become familiar with the codes and their limitations.

In general, no significant differences in prediction accuracy have appeared between the three large codes studied. The fast runnning code SMABRE is not directly comparable with these concidering prediction accuracy as it, first of all, should be regarded as a high-speed supporting tool. For this reason, and because the codes are complex and difficult to use, the choice of code for small-break analysis should be made on the basis of its userfriendliness and its degree of detail.

Each of the codes has its advantages which makes it preferable for certain applications.

Thus, SMABRE will be superior for extensive parametric studies, yielding values that are quite approximative, because of its small size and speed. This is obtained by a simplified description of the physical phenomena involved.

For most detailed applications the RELAP5/MOD1 and RELAP5/MOD2 codes were found to be preferable to TRAC/PF1 because they have a more flexible representation of the reactor system. Also, RELAP5/MOD2 is somewhat faster than TRAC/PF1. However, for certain boiling water reactor applications the RELAP5 codes may require a supplementary computer model for a detailed calculation of fuel rod temperature.

TRAC/PF1 should be chosen only in cases with significant multi-dimensional flow.

Finally, the study has shown that in cases where steam/water separation and stratification may occur, all the four codes, as they exist at present, may give wrong results regarding events that will occur and their sequence. Implementation of an adequate model for stratified flow into the code selected for small-break analysis would therefore be desirable.

It must be noted that in the treated test cases, all with functioning emergency core cooling, there was no indication of core uncovery or core dryout which could cause severe "fuel" rod overtemperature. This suggest that small-break LOCA'S of this type imply little danger for severe core melt-down accidents, provided that emergency core cooling is supplied in time. A single test case suggests that even incomplete emergency core cooling may be sufficient.

The present study has demonstrated that computer models for future reactor safety investigations of LOCA's are available in the Nordic countries together with trained personnel to use them. During the study experience has been gained as to the costs of running these codes (50 000 - 100 000 kroner in computer costs for a single small-break analysis with the RELAP5 codes or TRAC), and it has demonstrated how complicated it is to use them and how large and complex they are. Availability of trained people in connection with these computer codes is therefore a m u s t. They should be able to operate the codes properly and keep them updated in accordance with the international development in the field. The recommended implementation of an improved stratified flow model should be regarded as such an updating.

SAMMENDRAG

I vurderingen af nukleare kraftreaktores sikkerhed indgår analyse af såkaldte tab-af-kølemiddel-uheld (LOCA). I et LOCA tabes det kølevand, som skal transportere varmen fra den nukleare kerne, ved at der opstår et brud i reaktorsytemet. Dette medfører en risiko for at kernen bliver overophedet og skadet, måske med en nedsmeltning af kernen til følge. LOCAer klassificeres normalt efter brudstørrelsen i forhold til størrelsen af cirkulationsrørene i kølekredsløbet.

Indtil omkring 1980 var sikkerhedsarbejdet hovedsageligt rettet mod stor-bruds LOCAer. Imidlertid indikerede flere sandsynlighedsbaserede sikkerhedsstudier, at LOCAer forårsaget af små brud giver et betydeligt bidrag til det totale spektrum af ricici. Dette forhold blev understreget af Three Mile Island (TMI) uheldet i 1979, hvor der optrådte en mindre læk i kølesystemet.

De daværende data-modeller til konsekvensberegning af stor-bruds LOCAer kunne ikke uden videre anses for anvendelige også til analyse af små-bruds LOCAer, der har en tidsskala på timer eller måske dage i stedet for minutter.

SAK-3 projektet blev derfor etableret i begyndelsen af 1981 med det formål at fremskaffe en eller flere beregnings-modeller til analyse af små-bruds LOCAer.

Et små-bruds LOCA er karakteriseret ved et langsomt faldende tryk, idet vand eller en vand/damp-blanding undslipper gennem bruddet. Det faldende tryk fører før eller siden til dampdannelse ved kogning. Med aktive pumper i kølekredsløbet vil strømningshastighederne normalt være tilstrækkelige til at vand og damp forbliver i en ensartet blanding. Hvis cirkulationspumperne imidlertid stopper, vil hastighederne blive reduceret. Dette vil få dampen til at udskille sig fra vandet og samle sig i den øverste del af reaktorsystemet, og der er stor sandsynlighed for, at lagdelt strømning vil opstå i de vandrette rør.

Hvis intet nødkølevand tilføres til erstatning af det kølevand som mistes gennem bruddet, vil vandnivauet i reaktoren fortsat falde for til sidst at afdække reaktorkernen. Dette kan i yderste konsekvens føre til nedsmeltning af kernen.

Det blev tidligt erkendt, at udvikling af et helt nyt LOCA-beregnings-program, specielt for små brud, ikke ville være mulig inden for projektet, hvis det skulle være lige så detaljeret som de eksisterende stor-bruds LOCA-programmer. I stedet er eksisterende programmer, der var tilgængelige i Norden, blevet vurderet med henblik på anvendelse til små-bruds LOCAer.

Det drejer sig primært om tre programmer udviklet i USA hovedsageligt til stor-bruds LOCAer: TRAC/PF1 fra Los Alamos Scientific Laboratory og RELAP5/MOD1 og RELAP5/MOD2 fra Idaho National Engineering Laboratory. Herudover er et simplere finsk udviklet program, SMABRE, specielt til små-bruds LOCAer, blevet vurderet.

Den praktiske vurdering af programmerne bestod i med disse at gennemregne et antal test eksempler bestående af udførte forsøg. Beregningerne kunne så sammenlignes med de eksperimentelle resultater og afvigelserne analyseres. Test eksemplerne indeholdt små-bruds forsøg udført i Loss-of-Fluid-Test (LOFT) forsøgsfaciliteten ved Idaho National Engineering Laboratory, i Loop Blowdown Investigation (LOBI) forsøgsfaciliteten ved EF's fælles forskningsanlæg i Ispra (Italien) og i forsøgsfaciliteten FIX-II (for kogende-vands-reaktorer af svensk type) på Studsvik Energiteknik AB (Sverige).

LOFT test eksemplerne bestod af to eksperimenter, hvor bruddet udgjorde 2,5% af strømningsarealet i et cirkulationsrør. I det ene eksperiment blev cirkulationspumpen ved med at køre, mens den i det andet blev standset under i øvrigt uændrede forsøgsbetingelser.

LOBI test eksemplerne bestod både af et eksperiment med 0,4% brud i den oprindelige LOBI facilitet, som var instrumenteret til stor-bruds LOCAer, og af et eksperiment med 1% brud i den *modificerede LOBI facilitet, som nu var instrumenteret til små-bruds LOCAer. Sidstnævnte eksperiment dannede udgangspunkt for et såkaldt internationalt standard problem (en beregningsøvelse) ISP18. Sådanne beregningsøvelser er specielt velegnede til afprøvnig af beregningsmodeller, da de eksperimentelle resultater er ukendte for deltagerne frem til afslutningen.*

Et enkelt FIX-II eksperiment blev medtaget, med et 48% brud, hvilket er et mellemstort brud snarere end et "lille". Denne brudstørrelse betragtes som kritisk for kogende-vands-reaktorer af svensk type, idet den netop ikke fører til udtørring af reaktorkernen, inden kernens varmeproduktion er aftaget væsentligt.

Et små-bruds program må kunne forudsige alle hovedbegivenheder under et små-bruds LOCA : tryksænkning, dampdannelse, adskillelse af damp fra vand, start/stop/retning af strømningen i reaktorsystemet, samt afdækning af kernen. Forudsigelserne må være kvalitativt korrekte (sker begivenhederne og i hvilken rækkefølge ?), og de kvantitative forudsigelser (størrelse af tryk, temperaturer, hastigheder, tidspunkt for afdækning af kernen osv.) må være nogenlunde rigtige.

Det teoretiske studium af programmerne (gennemgang af manualer og programtekst) viste, at de fleste damp/vand strømnings-effekter, der forekommer i et små-bruds LOCA, modelleres rimelig indgående, både hvad angår de strømningsmekaniske effekter og repræsentationen af reaktorsystemet. En undtagelse er modelleringen af de specielle strømningsmekaniske forhold under lagdelt strømning, som blev fundet at være utilstrækkelig eller helt udeladt.

Denne iagttagelse blev bekræftet af den praktiske del af undersøgelserne af programmerne - gennemregning af eksperimentelle test eksempler. Under disse gennemregninger fandtes ingen enkelt kode at være de øvrige overlegen med hensyn til overensstemmelse med de eksperimentelle resultater. De størrelser, der beskriver reaktorsystemet som helhed, såsom systemets tryk og temperatur, blev forudsagt nogenlunde nøjagtigt. I det tilfælde, hvor en *kørende cirkulationspumpe sikrede en næsten jævn damp/vand-blanding (det første LOFT forsøg), var forudsigelsen af lokale hastigheder og dampindhold også tilfredsstilende.*

Imidlertid blev vand/damp adskillelsen og den lagdelte strømning, der forekom i alle test eksemplerne med standsede cirkulations-pumper, dårligt gengivet af alle de undersøgte programmer. Således kunne ingen af dem forudsige tømningen af et vandret rør, forbundet til reaktortanken, når vandnivauet i reaktor-tanken faldt under rørmundingen. Store temperaturforskelle mellem damp og vand blev målt under eksperimentet i disse rør, og dette kunne programmerne heller ikke simulere.

Betydningen af denne mangel blev illustreret i en af beregningerne, hvor fuldstændig udeladelse af lagdelt strømning forårsagede en falsk forudsigelse af afdækning af kernen. Dette ledte til en fejlagtig beregning af en temperaturstigning på 400°C i brændelsstavene til næsten 700°C i næsten 10 minutter.

Endvidere blev det observeret, at under forhold svarende til de herskende i kogende-vands-reaktorer (FIX-II test eksemplet) vil i det mindste RELAP5/MOD1-koden ikke altid kunne forudsige genvædning af brændselsstavene ved køling med vanddråber. Følgelig kan det være nødvendigt at råde over en supplerende, mere detaljeret EDB-model specielt til beregning af brændselsstavstemperaturer.

Både den teoretiske og den praktiske del af studiet har klart vist, at data-programmerne er ganske komplicerede og vanskelige at anvende, samt at erfaring er nødvendig for at kunne bruge dem (specielt de "store" amerikanske koder). For eksempel forudsætter en rimelig modellering af forløb med vand/damp adskillelse, at brugeren specificerer en tilstrækkelig fin inddeling i beregningsmæssige celler i de dele af systemet, hvor en sådan adskillelse kan forekomme. Det er derfor også et vigtigt resultat af SÄK-3 studiet, at deltagerne ved hjælp af test eksempler er blevet fortrolige med programmerne og deres begrænsninger.

Generelt set er der ikke fundet afgørende forskelle mellem de undersøgte programmer hvad angår nøjagtigheden i deres forudsigelser (måske med undtagelse af SMABRE-programmet, der først og fremmest må betragtes som et hurtigt beregningsværktøj). Af denne grund, og fordi programmerne er komplicerede og vanskelige at bruge, må valget af program til analyse af små brud ske på baggrund af brugervenlighed og detaljeringsgrad.

Hvert af programmerne har sine fordele, hvilket gør, at valget af program afhænger af anvendelsens art.

På grund af sin lille størrelse og store beregningshastighed vil SMABRE således være overlegen til omfattende parameterstudier, hvor ret grove resultater er tilstækkelige.

Til de fleste detaljerede anvendelser fandt man, at RELAP5/MOD1 og RELAP5/MOD2 bør foretrækkes fremfor TRAC/PF1, da de førstnævnte indeholder en mere fleksibel repræsentation af reaktorsystemet. Endvidere er RELAP5/MOD2 noget hurtigere end TRAC/PF1. For visse kogende-vands-reaktor anvendelser kan RELAP5-programmerne imidlertid, som nævnt, kræve en supplerende EDB-model til den detaljerede beregning af brændselstemperaturer.

TRAC/PF1 bør kun anvendes, hvis der forekommer betydelig to- eller tre-dimensional strømning.

Som nævnt har studiet vist, at ingen af de fire programmer i deres nuværende form kan behandle tilfælde med vand/damp adskillelse og lagdeling tilfredsstillende. Man kan således risikere fejlagtige resultater hvad angår såvel de indtrufne begivenheder som rækkefølgen af disse. En mere udtømmende model for lagdelt strømning i det EDB-program, der vælges til analyse af små-bruds LOCAer er derfor ønskelig.

I de behandlede test eksempler, som alle havde fungerende nødkøling, var der ingen tegn på afdækning af kernen eller tørkogning af denne, som kunne forårsage farlige overtemperaturer i de elektrisk opvarmede "brændsels" stave. Dette antyder, at små-bruds LOCAer af de undersøgte typer kun frembyder *ringe fare for alvorlige kernenedsmeltninger, forudsat at nødkøling kan etableres i tide. Et enkelt test eksempel antyder endog, at en mindre del af den foreskrevne nødkølingsmængde kan være tilstrækkelig.*

SÅK-3 studiet har vist, at EDB-programmer til fremtidige analyser af LOCAer i forbindelse med undersøgelser af reaktor-sikkerhed, samt uddannet personale til at anvende disse, findes i de nordiske lande. Projektet har også givet erfaring for omkostningerne ved at anvende disse data-programmer (50 000- 100 000 kroner i datamaskine-udgifter for en enkelt gennemregning af et små-bruds LOCA med TRAC eller RELAP5 programmerne), og det har vist, hvor vanskeligt det er at bruge dem, samt hvor store og komplicerede de er. I forbindelse med disse data-programmer er det nødvendigt at råde over erfarent personale både til korrekt anvendelse af programmerne og til løbende ajourføring i overensstemmelse med den internationale udvikling på området. Den anbefalede indbygning af en forbedret model for lagdelt strømning bør ses som en sådan ajourføring.

CONTENTS

1. INTRODUCTION

The SÄK-3 project was initiated at the beginning of 1981 with the aim of providing one or more computer codes for analysis of small-break LOCAs.

The project was formulated at a meeting[1] in Helsinki December 1980 as a proposal to the Nordic Liaison Committee for Atomic Energy. The proposed work schedule was later approved at the NORHAV Program Council Meeting[2] in Stockholm, January 1981. The organization and funding of the project are indicated on pages iii-iv.

The limited resources within the project did not allow development of a detailed small-break LOCA computer code from scratch. However, a number of codes potentially applicable for small-break LOCA analysis were or would soon be available to the participants.

To reach to the aim of the project these computer codes should be studied in theory as well as in practice with regard to analysis of small-break LOCAs.

In other words, the idea was to perform an assessment of the computer codes, individually as well as relative to each other, in order to (conf. Fabic, Ref. 3)

1. ascertain whether the codes can address the geometries, the system configurations, and all the important thermohydraulics and reactor physics-related processes that can occur in those postulated light-water reactor accidents or transients which the codes are supposed to handle,

2. ascertain whether the relevant physical processes are modelled well enough,

3. establish the degree of accuracy with which the codes can predict the key results.

The only source of knowledge of small-break LOCA's in full scale power reactors is a number of small-break experiments which have been performed at facilities in different countries. Therefore the practical study of the computer codes has consisted in comparing code predictions of a number of small break experiments, test cases, with the experimental data.

The codes studied are the three American systems codes TRAC-PF1[4], RELAP5/MOD1 [5], RELAP5/MOD2 [35] and the Finnish small-break code SMABRE[6].

The experiments have been selected so as to cover various break sizes and test facilities. The experiments used as test cases are:

1. LOFT L3-6	2.5 %	cold leg break, PWR, pumps running
2. LOFT L3-5	2.5 %	cold leg break, PWR, pumps stopped
3. LOBI SD-SL-03	0.4 %	cold leg break, PWR, LOBI mod.1
4. FIX-II-3031	48.0 %	recirculation line break, BWR
5. LOBI A2-81/ISP-18	1.0 %	cold leg break, PWR, LOBI mod.2

The LOBI A2-81 is the international standard problem No. 18, ISP18. As the experimental results were kept unknown to the persons performing these test case calculations until they were finished, LOBI A2-81 is especially valuable for code evaluation.

Originally it was the intention, during the study of the codes, to make modifications to submodels when this was motivated by discrepancies between code results and experimental data.
However, limited time and manpower has made it necessary to study and use the codes as "integral calculational tools" especially regarding heat transfer models. Apart from what has been necessary for implementation at the different computers, the codes have been modified only to a very limited extent.

On the other hand, a "sister project" to the present project, SÄK-5, on heat transfer correlations, contains detailed studies of the heat transfer models of the codes treated in SÄK-3, including some modifications and tests of these. For detailed information on this study the reader is referred to the SÄK-5 final report[7].

The SÄK-3 reporting of the code studies by test case calculations has been split in three levels: Level 0 includes the basic reports, each describing the prediction of one code for one test case and Level 1 are the code comparison reports, one for every test case.

The present final report sums up the code evaluations from the Level 1 reports and presents the final conclusion of the project. For more detailed information of the individual test case calculations the reader is therefore referred to the relevant Level 0 and Level 1 reports.

2. SELECTED CODES

The following codes have been selected for study:

Code	Origin	To be assessed by	Remarks
(RAMONA-2-PWR	Norway, IFE	Norway, IFE	Later dropped
RELAP5/MOD1 5)	USA, Idaho	Sweden, Studsvik Finland, VTT	
RELAP5/MOD2 35)	USA, Idaho	Sweden, Studsvik Finland, VTT	Later added
TRAC-PF1 4)	USA, Los Alamos	Denmark, RISØ Norway, IFE	
SMABRE 6)	Finland, VTT	Finland, VTT	Later added

These codes were selected partly because it was known that they were carefully developed thermohydraulic codes, partly because these codes were available to the laboratories participating in the SÄK-3 project. Although these codes are mainly for PWR's they were considered sufficiently flexible to be applied also to small-break LOCA's in BWR's.

The study of RAMONA-2-PWR was dropped in 1982 because necessary modification work exceeded manpower resources.

At about the same time, in accordance with a Finish suggestion, it was decided that a study of the Finnish fast-running code SMABRE, especially developed for small break LOCA analysis, should be included in the project.

The new advanced RELAP5 version, RELAP5/MOD2, has due to its late apperance, only been tested by a single test case.

In addition to the above mentioned selected codes other thermo-hydraulic codes might also have been relevant to study (RETRAN, DRUFAN, CATHARE), but lack of accessibility and of economic and manpower resources for implementation and test made this impossible.

RETRAN (USA) is a commercial code to which none of the project participant have had acess. RETRAN is based on the solution methods of the elder code RELAP4 and thus it has close relationship with RELAP5, too. The RETRAN code has mainly been aimed for plant transient analyses, but in the later development the capability to LOCA analysis has been emphasized too.

DRUFAN is a German code, developed since the middle of 1970 and in parallel with the RELAP4 and RELAP5 codes. The code applies the drift flux method for phase separation and thermal phase equilibrium is assumed. The code has a simplified horizontal stratification model based on an artificial tube inclination, but a more sophisticated stratification model is under development. The developer, GRS (Gesellschaft für Reaktorsicherheit mbH), has reported small-break calculations with the code, but only a large-break LOCA version is publicly accessible (via the NEA data bank).

CATHARE is a so-called two fluid code (with six flow equations), which has been developed in Grenoble, France. The detailed code assessment is under work and the verified code version is believed to be released soon. The developer is starting a user club for CATHARE, where the user experiences are reported, and at the same time the availability of the codes will be enlarged mainly by bilateral contracts on code verification. However, the code has not been accessible to the SÄK-3 project.

2.1 RAMONA-II-PWR.

Among the potentially applicable codes available for the SÄK-3 works the RAMONA-II-PWR code was initially selected for the Norwegian contribution. A description of the code was especially worked out for SÄK-3[8]). The aim of this RAMONA code version was directed to operating transients in PWR's. In order to keep computer costs acceptable the code development has turned to simplicity in modelling by adapting a fixed schematic structure. For the transient analysis the RAMONA-II PWR has a build-in one-dimensional model for the neutron kinetics.

An additional evalution of RAMONA-II PWR brought up the necessity for code modification to permit, for small break LOCA's, a proper modelling in the areas of

- break flow
- high pressure injection system
- pump description (to be more detailed)
- pressurizer surge line

With the preparation of a RAMONA-II PWR input for the LOFT L3-6 test case problems also turned out concerning the steam generator secondary side which so far was modelled as a boiling pool only.

Unfortunately, it soon turned out that the Norwegian resources were unsufficient to convert the RAMONA-II PWR from being a transient analysis code into a small-break analysis code. The work on the code was therefore terminated.

2.2. RELAP5/MOD1

The RELAP5/MOD1-code[5)] was developed at Idaho National Engineering Laboratory in the USA. It is a one-dimensional system analysis code designed for the analysis of LOCA and non-LOCA transients in light water reactors. The RELAP5/MOD1 versions used in the SÄK-3 project are cycles 6, 14, 18 and 19.

RELAP5 is intended for CDC-type computers and is not easily convertible to other computers. The code has mainly been programmed in FORTRAN IV but also COMPASS, the CDC assembler language, is used by some part of the code package.

The code was implemented and run at VTT, Finland and at Studsvik, Sweden on CDC CYBER 173 and 172 computers, respectively. The Studsvik Computer was later exchanged with a Cyber 170/835 to which the RELAP5 code was transferred without trouble. Due to the size of the computers the small core memory (SCM) option of the code was applied in both countries. In the implementation of the earlier version some difficulties were met because of minor errors related to the SCM-version of RELAP5. Apparently, the SCM-option of the code was not as well tested as the large core memory option of the code.

In the code a dynamic storage allocation is applied, which means that the size of the data areas of the code are changed during the calculation. This system decreases the storage requirements of the code, but on the other hand it makes modifications of the code rather complicated.

2.2.1. Physical and numerical modelling.

A reactor system is represented as an input specified hydraulic network consisting of a number of hydraulic control volumes (calculational cells) connected by junctions and heat structures.

The code uses a two-fluid model including five conservation equations: two phasic continuity equations, two phasic momentum equations and one overall energy equation. The least massive phase is assumed to be at saturation. With the use of only one energy equation together with the saturation assumption only two interfacial constitutive equations are needed: one for interfacial drag and the other for interfacial mass exchange. Other constitutive equations, which are included in the RELAP5 model, are those for the calculation of the wall friction and for the wall heat transfer.

In the numerical calculation a staggered spatial mesh is used in deriving the finite difference approximations for the hydrodynamic equations. Continuity and energy difference equations are used as finite-difference approximations for hydraulic volumes and momentum equations are associated with junctions (boundaries between volumes). For the solution a linear, semi-implicit integration scheme is applied.

Pressure and flow boundary conditions are defined by special hydraulic volumes and junctions, so-called time-dependent volumes and junctions which have proved to be very useful. These components can be defined also as functions of other quantities than the transient time only.

The heat transfer and the temperature distributions in solid heat structures are calculated using a one-dimensional form of the transient heat conduction equation in rectangular, cylindrical or spherical coordinates.

The heat transfer between a hydrodynamic volume and a solid heat structure is calculated by the wall heat transfer correlation package. A volume may be connected to several heat structures, but a heat structure must not be connected to more than two volumes, a left side volume and a right side volume.

The control system provides the capability to evaluate new variables by algebraic and ordinary differential equations from the time advanced quantities. These variables can be utilized in the output and in defining trips to control the boundary conditions.

2.2.2. Constitutive Relations

The RELAP5 hydrodynamic model requires four constitutive relations: The vapour generation rate, the interfacial drag, the wall friction, and the wall heat transfer. These relations are empirical correlations and the ability of the hydrodynamic model to calculate physical phenomena accurately and smoothly depends strongly on the accuracy and structure of the correlations and the way they are structured.

The interfacial drag model is based on existing flow regime maps for determining flow pattern and on the associated drag correlations. Depending on the flow conditions the following flow regime maps are used, all depending on mass flux and void fraction (see fig. 2.2.1):

1. Vertical flow regime map for vertical pipes

2. Horizontal flow regime map for horizontal pipes

3. Annular flow regime map for annulus components such as the downcomer.

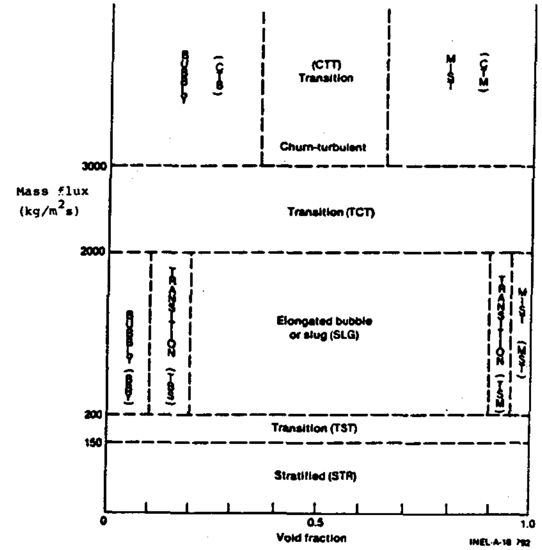
4. High mixing flow regime map for flow in the pump.

As a whole the interfacial drag correlations together with the flow pattern model form a relatively complex system, which has not been sufficiently verified during the code development. During the SÄK-3 project different versions on the RELAP5/MOD1 code were used. The modifications in the code versions concerned the flow pattern model and especially the interfacial drag correlations. The most significant change was the Chow's interfacial drag smoothing update, which was intended for smoothing the transitions between correlations.

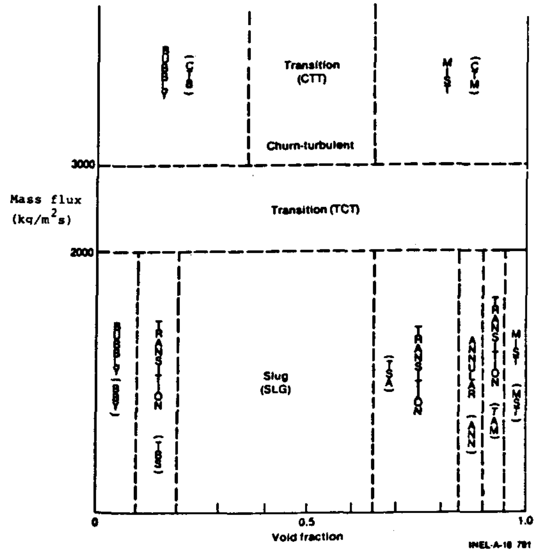
The wall friction model takes into account the wall shear effects. The form losses due to abrupt area changes and other flow geometry effects are calculated by other models. The model consists of a single phase friction model (the Colebrook correlation) times the Baroczy two-phase multiplier. The total wall friction is divided to the gas and liquid phase on basis of special phasial friction models.

The mass transfer model in RELAP5 is a vapor generation model, where the vapor generation is proportional to the amount of non-equilibrium. In principle a similar model is employed for steam condensation. With only one energy equation no interfacial heat transfer model is needed.

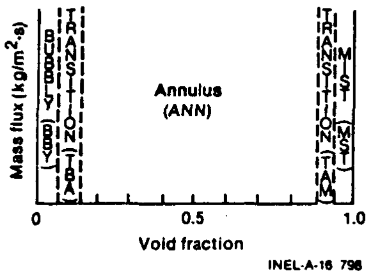
Formation of vapor bubbles is assumed to start as soon as water is superheated, and similarly condensation starts the moment the temperature falls below the saturation temperature.



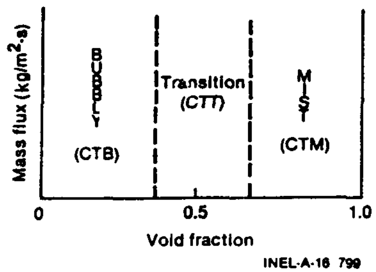
Horizontal flow regime map (HRT).



Vertical flow regime map (VRT).



Annulus flow regime map (ANF).



High mixing flow map (HMF).

Fig. 2.2.1. RELAP5/MOD1 flow regime maps (From ref. 5).

The RELAP5 wall heat transfer correlation package consists of a forced convective group of correlations and a pool boiling/natural circulation group of correlations. The forced convection correlations are used, when the mass flux is higher than 200 kg/m^{2}s. For lower mass flux the heat flux is calculated using correlations from both groups and the maximum value is used. The heat flux is limited by critical heat flux models for saturated conditions and subcooled high flow conditions. Post-critical-heat-flux heat transfer is calculated with low flow correlations for transition boiling and film boiling.

Because in RELAP5 there is only one overall energy equation, the heat from the structure cannot be partionned between the phases on basis of the flow patterns, but must be added to heat up to the whole coolant mass of the volume.

Practical experience has shown that the wall heat transfer model together with the evaporation/condensation model and the single energy equation has servere limitations. Thus, it turns out that essential vapour superheat cannot be predicted - even during steam cooling (steam and droplets) of hot surfaces - except in positions with no liquid at all. Escpecially for simulation of LOCA's in BWR this is a very severe limitation.

For some fast or complex processes special models have been developed. For the choking at a break or at an abrupt area change a quasi-steady model, which employes an analysis of the characteristic velocities of the time-dependent differential equation system, is used. This model assumes thermal equilibrium between the phases. However, there is no justification for this assumption. When this special model is applied it is not necessary to build up a fine nodalization grid at such point where this choking may occur. Other processes, where a special quasi-steady treatment is applied, are abrupt area changes and branching. The hydrodynamic performance of the pumps and valves is simulated using models, which are based on simplified assumptions and experimental data. An accumulator is modelled as a lumped-parameter component.

The steam separation from water can be simulated with a separator component. This component allows only steam to flow through a certain junction unless the volume is completely flooded with liquid.

2.3. RELAP5/MOD2

RELAP5/MOD2[35] is the new and most advanced version of RELAP5. Like MOD1, it has been developed at Idaho Engineering Laboratory, USA. The most significant improvement relative to the MOD1-version is the addition of a second energy equation to give a full nonequilibrium six equation two-fluid model with separate energy equations for gas and liquid instead of a single overall energy equation. Other improvements include a revised interfacial drag formulation, a new wall heat transfer model, a revised wall friction partitioning model, a revised vapor generation model and the addition of several new special process and component models. A totally new constitutive model has been added to RELAP5/MOD2 for the reflood heat transfer.

At Studsvik, Sweden, the code was implemented and run on a CDC CYBER 170/835 computer. In the implementation of the code, some errors were encountered and reported, most of which have been removed in the latest version, cycle 36.01. The code version used in the LOBI-MOD2 calculation at Studsvik was the RELAP5/MOD2 cycle 36 with some code corrections.

At VTT, Finland, the code was implemented and run on a CDC CYBER 173 computer with 256 K of central memory (SCM). The code version used in the LOBI-MOD2 calculation at VTT was the RELAP5/MOD2 cycle 36 with a number of updates introduced by Studsvik.

The present frozen version, cycle 36, is not the final one. However, it will not, apart from corrections of formal code errors, be updated until the end of an assessment period of two years.

A new feature in the RELAP5/MOD2 code structure is a steady state block. If a steady-state problem is specified, the steady state is a acquired by running a modified transient until a convergence criterion is met. The steady state block is physically compatible with the transient block and allows the steady state results to be used for the initial conditions of the transient run. Thus, the code user avoids the laborious work to update the input data.

2.3.1 Physical and numerial modelling.

The reactor system is simulated by a network of hydraulic control volumes, connecting junctions and heat structures. The pressure and flow boundary conditions are defined by so-called time dependent volumes and junctions. A control system and special component models are included to permit modelling of the control system of a power plant and secondary conditioning systems.

The hydrodynamic model of the RELAP5/MOD2 code is of the one-dimensional, transient, two-fluid type. The two-phase steam/ water mixture is allowed to contain a noncondensible component in the steam phase and a nonvolatile component in the water phase. The model consists of six conservation equations, i.e. two phasic continuity equations, two phasic momentum equations and two phasic energy equations. This permits a proper representation of unequal velocities of the steam and water and thermal non-equilibrium between them. However, homogenous flow, thermal equilibrium and frictionless flow models may be specified as options.

Besides the conservation equations, the two-fluid model consists of constitutive relations for interactions with the wall: friction and heat transfer, and for interfacial phenomena: drag, heat transfer and mass transfer. The two energy equations together with the constitutive relations for interfacial heat- and mass transfer allow for even extreme non-equilibrium phenomena, e.g. superheated steam in presence of subcooled water droplets. Finally it includes a thermodynamic model (equation of state) for the water and steam properties.

The transport of the noncondensible in the steam phase is represented by an additional mass conservation equation. The noncondensible component is assumed to be in mechanical and thermal equilibrium with the steam phase and all state properties of the gas phase are mixture properties of the mixture of the steam and the noncondensible.

As the solvent in the water phase is assumed very dilute, a simplified Eulerian numerical scheme is used to represent the transport of it.

The numerical solution of the flow equations is based on a finite difference method, where the tubes are subdivided in a number of control volumes, separated by junctions. A staggered mesh is used where velocities are defined at junctions while the other fluid properties are defined at the control volume centers. The semi-implicit numerical solution scheme uses a direct sparce matrix solution technique for the time step advancement. The user specifies a maximum time step, which is limited by the restrictive material transport Courant stability criterion and by the mass truncation error. However, the Courant time step limit permits violations in single nodes without violating stability.

The thermal behaviour of fuel pins, heater elements, pipe walls, heat exchanger surfaces, and other structural elements is simulated by so-called heat structures. Temperatures and heat transfer rates inside heat structures are computed from the one-dimensional form of the transient heat conduction equation in rectangular, cylindrical or spherical geometries. Electrical, gamma or nuclear heating of the heat structures can be modelled. The heat transfer between heat structure surfaces and the steam/ water mixture is calculated by a wall heat transfer correlation package. It contains correlations for convection, nucleate boiling, transient boiling and film heat transfer.

As compared to RELAP5/MOD1 the heat transfer package has been restructured, including a change in the standard correlations for critical heat flux (CHF). Boundary conditions for heat structure surfaces not in contact with the fluid may be symmetry or insulated conditions, or surface temperature, heat transfer rate or heat transfer coefficient specified in tabular form.

A new feature in the RELAP/MOD2 is the dynamic gap conductance model for nuclear fuel pins simulated by heat structures. The model is a simplified deformation model generated from the FRAP-T6 code and the gap conductance is calculated to be primarily a function of the fuel-cladding gap.

Another new model is the reflood option of the surface heat transfer. The heat transfer correlations of the reflood model are restricted to low pressure and low flow conditions and the model should only be used for pressures less than 1.0 MPa and mass fluxes less than 200 kg/sm^2. When the reflood model is activated a two-dimensional conduction scheme is used for rectangular or cylindrical heat structures. The integrated heat conduction equations written in the finite difference form are solved using the alternative-directions implicit (ADI) method. The reflood option includes a fine mesh-rezoning scheme for efficient solution of the two-dimensional conduction with a large axial variation of wall temperatures and heat fluxes.

Nuclear heating of heat structures is calculated by a space-independent (point) reactor kinetics model. Both the immediate fission power and the power from decay of fission fragments are considered.

The control system is similar to that of the RELAP5/MOD1 and provides the capability to evaluate new variables by solving algebraic and ordinary differential equations. These variables are primarily intended to simulate a typical plant control system but they can also define additional output quantities such as differential pressures.

The trip system is similar to that of RELAP5/MOD1. The decision of trip actions resides within component or boundary condition models.

2.3.2 Constitutive relations.

The constitutive relations include decisions for defining flow regimes and flow regime related models for interfacial drag, wall friction, heat transfer, interfacial heat and mass transfer and reflood heat transfer.

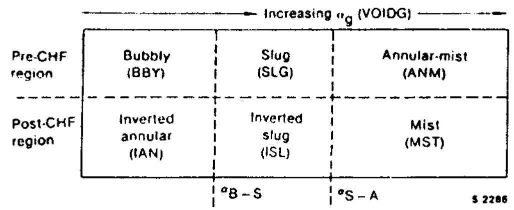
In RELAP5/MOD2 the constitutive relations include flow regime effects for which simplified mapping techniques have been developed to control the use of constitutive relations. Three flow regime maps are utilized: vertical and horizontal maps for flow in pipes and a high mixing map for flow in pumps (see fig. 2.3.1).

The vertical flow regime map contains a number of regimes including the post-CHF region and vertical stratification. The criteria for defining the boundaries for transition from one regime to another are based on the work of Taitel and Dukler, and of Ishii. The main difference as compared to the RELAP5/MOD1 flow regime map is the more advanced criteria for regime transitions, not just based on fixed void fraction boundaries as in RELAP5/MOD1.

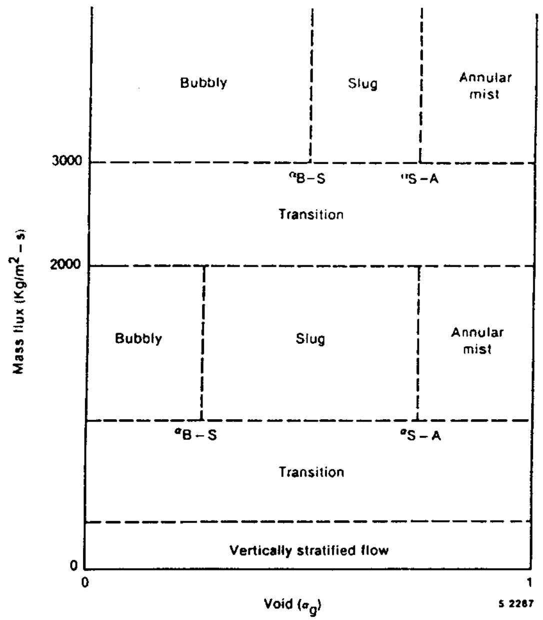
For horizontal flow a criterion developed by Taitler and Dukler is used for transition to horizontally stratified flow. If this condition is not satisfied the flow is classified as bubbly, slug or annular-mist similar to the vertical map. Post-CHF flow regimes are not included in the horizontal flow map.

The high mixing flow regime map for pumps is based on the vapor void fraction and consists of a bubbly, transition and mist regime.

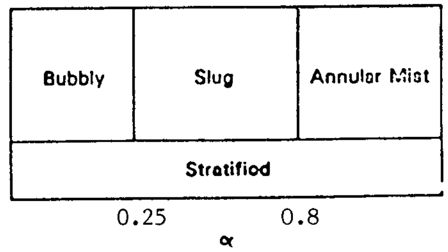
In RELAP5/MOD2 a revision of the interfacial drag formulation has been undertaken in order to remove the limitations experienced also during the SÄK-3 project with RELAP5/MOD1. The



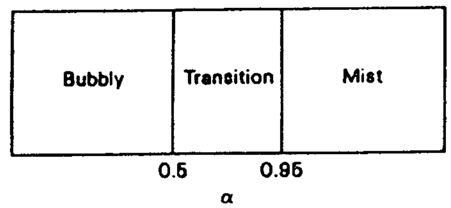
Sketch of vertical flow regime map for pre-CHF and post-CHF regions.



Vertical flow regime map including the vertically stratified regime.



Horizontal flow regime map.



High mixing flow regime map.

Fig. 2.3.1. RELAP5/MOD2 flow regime maps (From ref. 35).

interfacial drag force is computed as a product of two parameters, drag coefficient and interfacial area. Both parameters are dependent on flow regimes. The approach and correlations used to compute these parameters are based mainly on the work of Ishii. Especially discontinuities of the interfacial drag at flow regime transitions are avoided to the degree possible.

The wall friction force terms include only wall shear effects. Losses due to abrupt area change are calculated using mechanistic form loss models. Other losses due to geometry are modelled using energy loss coefficients which are input by the user. The wall friction model is based on a two-phase multiplier approach in which the two-phase multiplier is calculated from a modified Baroczy correlation. The individual phasic wall friction components are calculated by partitioning the two-phase friction between the phases using a technique derived from the Lockhart-Martinelli model. The single phase friction factor is computed using a high calculational speed version of the Colebrook correlation.

A boiling curve is used to govern the selection of heat transfer correlations. In particular the wall to fluid heat transfer regimes are classified as pre-CHF, CHF and post-CHF regimes. Condensation heat transfer is also modelled and the effect of a noncondensible is taken into account. In the pre-CHF regime either single phase liquid convective heat transfer, subcooled nucleate boiling or saturated nucleate boiling may take place, and it is assumed that the wall is totally wetted by liquid and all the heat is transferred to the liquid. In the post-CHF regime either transition film boiling, film boiling or single phase vapor convection may take place. In this case a mechanistic model is partitioning the heat between the liquid and vapor phases. In the condensation regime, heat transfer to the wall from liquid and vapor is dependent on the flow regime.

The interfacial mass transfer model includes, besides the thermodynamic process, the interfacial heat transfer regime and interfacial area, both determined from the regime map. The model includes mass transfer at the wall as well as in the bulk.

The RELAP5 codes include special process models to simulate fast or complex processes. RELAP5/MOD2 includes all the models of the MOD1- version, as listed in chapter 2.2.2. The new process models are described briefly.

The horizontal stratification entrainment model is used in junctions downstream of horizontal components under stratified conditions. The model permits extrodinary vapor pull-through or liquid entrainment depending on geometric conditions and liquid level. The user can specify the location of the junction as input value. This gives the user a choice to prefer one of the two phases in the junction downstream of a horizontal component.

The crossflow junction model is included in RELAP5/MOD2 for use in applications where an approximate treatment of two-dimensional flow improves the physical simulation. A typical region is the reactor core where the crossflow junction can be used to couple the hot channel to the average channel.

The vertical stratification model has been installed in the vertical flow map, see Figure 2.3.1, so that the nonequilibrium modelling capability can include repressurization transients in which subcooled liquid and superheated vapor may coexist. Horizontal stratification implies modified interfacial mass and heat transfer, wall heat transfer, and interfacial drag, with a criterion based on the difference between the void fraction in the volume above and below and on the Taylor bubble velocity. This should remove the stepwise behaviour of the primary pressure experienced in repressurization transients with RELAP5/MOD1 when the water level rises in the pressurizer volumes.

A numerical water packing mitigating scheme has been added in RELAP5/ MOD2 to damp large pressure spikes. The model is similar to the method used in the TRAC code.

RELAP5/MOD2 contains also models for subsystem components such as branches, seperators, jetmixers, pumps, turbines, different types of valves, and accumulators.

2.4. TRAC-PF1

TRAC-PF1[4] is a code for thermohydraulic simulation of a (PWR) reactor system. The code has been developed at Los Alamos National Laboratory, USA, written in FORTRAN IV, and is specially designed to fit on a CDC 7600 computer by use of an overlay structure. The code version used in this study is the TRAC-PF1 7.0/EXTUPD7.6 with a number of general code corrections.

At Risø, Denmark the code was run on Risø's Burroughs B7800 computer with virtual memory, which required a number of special modifications (the CDC 7600 has no virtual memory). At the B7800 computer a typical storage requirement is 240 K words.

At IFE, Norway, the code was run on a CDC CYBER 170/835 computer with a typical storage requirement of 84 K words of central memory plus 128 K words of extended core storage.

TRAC-PF1 is built up modularly, both with respect to code functions and with respect to components in the reactor system. This makes corrections and modifications rather simple as long as they do not interfere with the basic numerical method.

2.4.1. Physical and Numerical Modelling

The reactor system is represented as an input-specified hydraulic network, built up of a number of components: PIPE, BREAK, FILL, CORE (1-D reactor vessel model), VESSEL (3-D reactor vessel model), ACCUM (accumulator), PRIZER (Pressurizer), PUMP, STGEN (steam generator), TEE, VALVE. Hydraulic boundary conditions are provided by the components BREAK (prescibed pressure) and FILL (prescribed flow rate).

Beside hydraulic models, models for heat conduction in walls etc. are provided. Unfortunately, TRAC-PF1 is inflexible with respect to combination of hydraulic and heat conduction models as a certain hydraulic component is furnished with adapted heat structure model(s). E.g. a pipe may only have a pipe wall while the vessel component in addition may have a number of fuel rods.

The hydraulics is described by a two-fluid model with separate flow equations for the gas and the liquid. All components of a reactor system are modelled one-dimensionally with the exception of the reactor vessel which may be modelled in one or three dimensions by the components CORE and VESSEL, respectively. However, in the present study, only one-dimensional modelling was used in order to avoid excessive computer time and because 3-D effects were expected less important in small-break LOCA's and in scaled experimental facilities. That is, six flow differential (conservation) equations are used: 2 mass conservation equations, 2 momentum equations and 2 energy equations. This allows for description of unequal velocities and thermal non-equilibrium.

The numerical method for the one-dimensional hydraulics is a finite difference method, semi-implicit in time and upstream-weighted (donor-cell) in space with a staggered mesh (velocities defined at calculational cell boundaries while other variables are defined at cell centres). The method is a so-called two-step method, enough implicit to allow Courant-unlimited time step size, i.e. unlimited by fluid transit time through a cell.

The heat conduction model for pipe walls and constructional elements is one dimensional (only radial conduction is considered) using a finite difference method. However, the heat conduction model for fuel rods is two-dimensional, using a finite difference method, implicit in the radial (or transverse) direction and semi-implicit in the axial direction. This implies a time step limit depending on the thermal diffusion number. The fuel rod heat conduction model allows calculation of quench front progession during fast quenching by means of a "moving fine mesh", but this option is hardly relevant for small-break LOCA's.

The code includes an extensive trip logic and possibilities for simulating control of valves, pumps, power etc. and for specifying boundary conditions by so-called signal variables, i.e. certain system parameters (time, temperatures, pressures etc.). However, this control system is somewhat inflexible in that there is not generally a free choice of signal variable. Furthermore, only in the connection with trips it is possible to use derived signal variables defined as expressions in system parameters.

In addition to the trip and control system the core power may be calculated from reactor points kinetics.

2.4.2. Constitutive relations

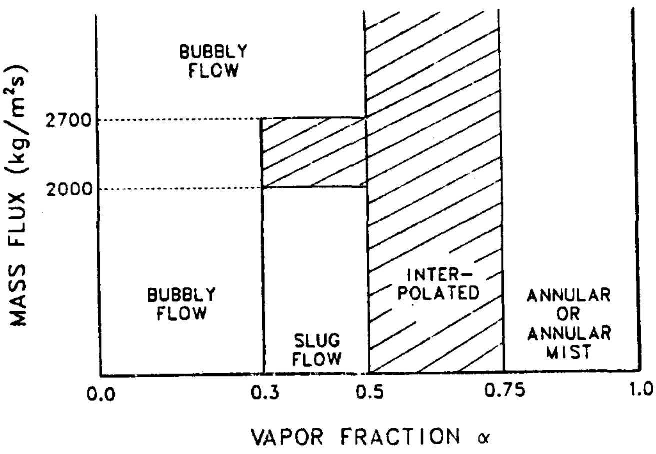
Wall- and interfacial friction, wall- and interfacial heat transfer are calculated by a package of so-called constitutive equations or models. In contrast to nearly all wall models the interfacial models are flow regime dependent.

The flow regimes considered are bubble, slug, annular mist and stratified flow. As standard, a simple flow regime map based on void fraction and mass flux is used, see fig. 2.4.1. However, this standard flow regime map is overruled if a stratified flow criterion is fulfilled.

The interfacial friction models for bubbles and droplets are based on standard models for spherical particles where the bubble and droplet diameter is based on a critical Weber number. However,in slug flow the bubble diameter is allowed to increase up to the hydraulic diameter. Annular mist flow has in addition a liquid film/gas interfacial friction model based on the Reynolds numbers for gas and liquid film respectively.

For non-stratified flow two wall friction options are available (to be input-selected at each cell boundary): a homogenous model and an annular flow model. Both use a single phase friction factor, modified by using a mixture viscosity, times a two-phase multiplier. The single phase friction factor is for the homogeneous model a Blasius-like friction factor while for the annular flow model Woods equation is used. Depending on the void fraction the wall friction is partitionned between liquid and gas. For stratified flow the Blasius correlation is used both for interfacial and wall friction.

The interfacial heat transfer models are quite analogous to the interface friction models: standard models for spherical particles are used in connection with bubbles and droplets while the liquid film/gas interfacial heat transfer model is based on the relevant Reynolds numbers.



Flow-regime map for three-dimensional hydrodynamics.
(Cross-hatched regions are transition zones.)

(From ref.4)

Figure 2.4.1

The wall heat transfer is calculated from a compound model covering the main heat transfer regimes: liquid convection/boiling, transition boiling and film boiling/gas convection, where the regime is determined from the wall temperature relative to the critical heat flux point and the min. film boiling point.

As an alternative one may per input select (for each component) a simpler two-phase mixture heat transfer model: natural convection/Dittus-Boelter forced convection.

Special models apply at certain geometries and components. At abrupt area changes loss coefficient for singular friction (in addition to normal wall friction) may be specified. By input it is furthermore possible to specify phase separation (neglection of interfacial friction) at a given pipe cross-section, which may be used to model steam separation. In an ACCUMulator component this is automatically performed for all cross-sections to maintain a sharp liquid-gas interface.

In the PUMP component equal gas- and liquid velocities are assumed, and instead of a wall friction model a pump head correlation is used. This pump head correlation is based on the single phase homologous pump head curve, which in case of two-phase operation is modified by a "fully degradated" pump curve and a degradation multiplier.

In the PRIZER (pressurizer) component, within certain limits, heat is added/removed to/from the liquid in proportion to the pressure deviation from a specified set point.

May be the most important special model is the choked flow model which can be specified in connection with BREAK components. The choked flow is determined from a characteristic velocity model for the flow at the break plane, allowing a reasonable prediction without using very fine calculational cells up to the break plane.

The package of constitutive models has two obvious deficiencies. First, it has no models for subcooled boiling, i.e. it does not allow steam generation until the liquid temperature has reached saturation. Second, the stratified flow model is inadequate with respect to representing the equilization of liquid level heights in horizontal pipes. A term representing this gravitational head effect is included in the momentum equations but in order to avoid numeric instabilities the term is practically inactive for large time steps.

Another deficiency found during this study is poor prediction of pressure distribution when using large computational cells (differs from Bernouilli pressure distribution) and at TEE's (pipe branches).

2.5. SMABRE

The SMABRE code[6] has been developed at the Technical Research Centre of Finland. The code is a one-dimensional system analysis code designed for the analysis of small break LOCA and some operational transients in PWR type of light water reactors and in integral test facilities designed for studies of PWR plants.

The SMABRE code is programmed in FORTRAN IV and thus the code is easily implemented at different computers. The code has been installed in CDC Cyber-173, PDP 11/70, PDP 11/34, VAX-11 and HP-3000 computers. As a special application the code has been installated at the full scope training simulator of the Loviisa PWR plant, for simulation of two-phase conditions in the primary side during small-break LOCA type transients. The computers in the simulator for the process physics are two parallel PDP 11/70's

In the PDP 11/70 and Cyber 173 with a typical nodalization of 50 to 100 nodes, the code can run in a computer time to real time ratio of 1:1. This feature is significant in simulator training applications.

2.5.1. Physical and numerical modelling

A reactor system is modelled as an input defined hydraulic network consisting of a number of control volumes (calculational cells) and connecting junctions plus heat structures.

The code uses a drift flux model including four conservation equations for local fluid variables defined as mixture mass, steam mass, mixture velocity and steam or water enthalpy. The pressure is calculated as an integrated variable over regions including several mesh points using the integrated mixture momentum equation. The steam velocity is obtained from the calculated local mixture velocity distribution using the drift flux approximation.

The energy equations are solved for both the steam and water phases, but either phase is always saturated. The drift flux velocity defining the phase separation is calculated as a function of pressure and void fraction.

In the numerical calculation a staggered spatial mesh is used in deriving the finite difference approximations for the hydrodynamic equations. Continuity and energy difference equations are used as finite difference approximations for hydraulic volumes using upstream weighted (donor-cell) discretization. The momentum equations are solved for junction flow rates in an integrated form over closed loops. In the solution a linear, semi-implicit integration scheme is applied. This method allows the resulting system of equations to be solved by an iterational procedure without use of matrix operations resulting in the large calculational speed of the code.

The heat transfer and the temperature distributions calculation is simplified, as the radial temperature distribution in structures is calculated only for steam generator tubes and fuel rods. For other structures a lumped parameter approximation is used and the wall heat flux is limited by a factor depending on wall thickness and heat conduction.

The control system of a real plant is partially included in the code, but it is most practical for the user himself to add the control logic in the specific subroutines.

2.5.2. Constitutive relations

The SMABRE hydrodynamic model requires four constitutive relations: the vapour generation rate, the drift flux velocity, the wall friction, and the wall heat transfer. These relations are empirical correlations and the ability of the hydrodynamic model to calculate physical phenomena accurately and smoothly depends strongly on the accuracy and structure of the correlation system.

Having only a single (mixture) momentum equation SMABRE has no model for the interfacial friction. Instead the liquid-gas velocity slip is calculated from the drift flux model, i.e. based on correlations for the so-called drift flux velocity (the local difference between gas velocity and the average volumetric flux) and on correlations for the radial void fraction and velocity distributions.

The drift flux velocity correlation is a function of pressure and void fraction. Different drift flux options may be chosen for a junction (boundary between two volumes) as follows:

1. Vertical drift separation
2. Horizontal drift
3. No drift separation

The wall friction model takes into account the wall shear effects. The form losses due to abrupt area changes and other flow geometry effects are given as an input. The wall friction model consists of a single phase friction model (a simplified Colebrook correlation) times a two-phase multiplier.

Instead of an interfacial heat transfer model SMABRE has a model for vapor generation/condensation where the mass transfer rate is proportional to the extent of the non-equilibrium. The model contains no minimum superheat/subcooling to initiate evaporation/condensation.

The SMABRE wall heat transfer correlation package consists of forced convection, nucleate boiling, transition boiling and film boiling correlations. The Biasi correlation is used for the critical heat flux (i.e. the maximum nucleate boiling heat flux at the switch-over point to the transition boiling).

For some subprocesses special models have been developed. For the choking at a break the Moody-model has been included in a correlation form, calculated as a numerical fitting to data points, and two coefficients, one for subcooled regime and another for saturated regime has been used. The hydrodynamic performance of the pumps and valves is simulated using models which are based on simplified assumptions and experimental data. An accumulator is modelled as a lumped-parameter component.

The operation of these components may be specified in terms of the time but other quantities may be used as well.

The most serious limitation of the SMABRE model system is, like RELAP5/MOD1 (see sect. 2.2.2), the unability to represent thermal non-equilibrium properly, especially steam superheat in presence of liquid. However, also the the need for user-supplied routines for representation of the control/trip system of a reactor limits the ease with which the SMABRE code can be applied to different nuclear reactor systems.

3. TEST CASE CALCULATIONS

For test cases only so-called "Integral Systems Tests" experiments, simulating an entire reactor system, have been selected (as apposed to "Seperate Effects Tests" for special components or phenomena or "Basic Tests" for study of basic thermohydraulic interactions), and only in a limited number.
This choice was made due to the limited resources available in the project, manpower as well as computer costs, allowing only an overview-like (not complete) assessment and mutual comparison of the codes (among other things because the necessary number of test cases for a complete assessment would be prohibitive considering the large work to prepare the input for description of a reactor system). It was therefore important that the test cases finally selected covered as many phenomena as possible.

Integral small break tests for PWR's have been conducted in the LOFT, Semiscale (Idaho, USA), PKL (W. Germany) and LOBI (J.R.C. Italy) experimental facilities. Small-break tests for BWR's have been performed at the FIX-II facility (Studsvik, Sweden). A number of items should be considered when selecting out test cases:

- System geometry: facility scale, reactor type (PWR/BWR) and configuration.
- Break size and location.
- Kind of heating (Nuclear/electric).
- Boundary conditions.
- Accident phases and events.
- Important hydraulic, heat transfer and other phenomena.
- Instrumentation: kind, accuracy.
- Special national interests: Since there are BWR as well as PWR power plants in Sweden and Finland, test cases covering both reactor types were desired.
- Is the test case an international standard problem (ISP) which allows "blind" calculation ?

A blind calculation offers a specially valuable code assessment as only intial and boundary conditions, not the real experimental results, are known to the persons performing the calculation. Thus code "tuning",.i.e. modification to get closest possible agreement with experimental results, is ruled out.

Based on the above considerations the following 5 test cases were selected:

1. LOFT L3-6	2.5%	cold leg break, PWR, pumps runing
2. LOFT L3-5	2.5%	cold leg break, PWR, pumps stopped
3. LOBI SD-SL-03	0.4%	cold leg break, PWR, LOBI mod. 1
4. FIX-II-3031	48.0%	recirculation line split break, BWR
5. LOBI A2-81/ISP-18	1.0%	cold leg break, PWR, LOBI mod. 2

The last two test cases require special comments.

The FIX-II-3031 experiment is not really a small-break experiment. It should rather be termed an intermidiate break size experiment. However, it was included in this project because it represents the largest break size not leading to an early dryout in the reference plant (the Swedish Oskarshamn 2 reactor). The term "early" refers to a dryout before any considerable decay of the power (about 10s after scram), as would occur during large-break LOCA's. A true small-break experiment (in the order of 1%) performed in the FIX-II facility would not produce any useful information because of the relatively large heat losses from the facility to the surroundings.

The LOBI-A2-81/ISP18 test case is the first experiment in the LOBI mod.2 facility, especially designed for small-break experiments. Used as an ISP it is very valuable for code assessment. Descriptions of the individual test facilities are given in Appendix 1-4.

The remainder of this chapter is structured according to the test cases: for each test case the characteristic features are considered, and it is judged how well the codes predict each of these features.

Table 3.0 gives an overview of the performed test case calculations, including references to the relevant level 0 and level 1 reports.

Table 3.0 Performed test case calculations with references to the corresponding level 0 and level 1 reports, the level 0 reports together with the indication for each test case calculation.

Code, institution / Test case	Level 1 report	RELAP5/MOD 1 VTT	RELAP5/MOD 1 Studsv.	TRAC-PF1 Risø	TRAC-PF1 IFE	SMABRE VTT
LOFT L3-6	(15)	X (19,20)	X (24,25)	X (27)		X (31)
LOFT L3-5	(16)	X (21)		X (28)		X (31)
LOBI SD-SL-03	(17)	X (22)		X (29)		X (32)
FIX-II 3031	(18)	X (23)	X (26)		X (30)	
LOBI A2-81/ISP18		X	X	X		X

Due to the limited time no level 0 and level 1 reports have been prepared for the LOBI A2-81/ISP18 test case.

3.1. LOFT L3-5 and L3-6.

3.1.1. Objective of the experiments.

The LOFT-experiment series L3 was designed to provide large-scale blowdown system data for PWR small break transients The objectives of the two experiments L3-5[9] and L3-6[10] selected for the test cases in the SÄK-3 project were to examine the effect of the primary coolant pump operation on plant response during a small break LOCA. In the experiment L3-6 the pumps were kept running at normal speed throughout the test in order to provide data to analyze the pump operation in the two-phase flow. In the experiment L3-5 the pumps were stopped just after the break was opened. The break size in the both experiments was the same, 16.9 mm in diameter, corresponding to a break of a 100 mm diameter pipe in a large commercial PWR.

3.1.2. Emperimental results.

The initial steady state conditions are given in table 3.1.1

With the reservations given in Appendix 1 the transient behaviour was represented well by the experimental data. The most serious deficiency in the primary side measurements was the lack of discharge flow rate data during approximately the first 50 s in the both experiments. Thereafter the data were valid but the uncertainty limits were 15 % for L3-6 and 25 % for L3-5.

The main events during the transients are listed in table 3.1.2 The aim was that the experiments should be as identical as possible excluding the behaviour of the pumps. However, there were some minor deviations both in the initial conditions of the experiments and in the starting times of some of the main events as can be seen in tables 3.1.1 and 3.1.2. In both experiments the main steam isolation valve never closed totally, as there was an unmeasured leakage through the valve.

In the L3-6 experiment, with the pumps running, the distribution of water and steam remained close to homogeneous during the depressurization. The behaviour of the secondary side was strongly affected by the steam flow through the main steam isolation valve. The valve was closed at the beginning of the transient and was reopened after about 90 seconds, for about 10 seconds. Unfortunately, the valve was not totally closed neither before the reopening nor after closing. The leakage through the valve and the flow during the valve reopening were not measured, which makes the comparison of the calculated and experimental results difficult.

In the experiment L3-5 the pumps were switched off at the beginning of the transient. The coast down rate of the pumps had an abrupt change at about 20 s, caused by the disconnection of the pump flywheel, whereafter the speed of the pumps very quickly decreased to zero. At that time the flow in the intact loop was very slow, and according to the gamma beam densitometer measurements, considerable stratification occured in the horizontal part of the pipes. Also in other parts of the facility water and steam was strongly separated. As in the L3-5 experiment, the uncertainty of the leakage rate through the main steam isolation valve makes it difficult to compare measured data with calculated results.

Table 3.1.1 Initial conditions for L3-5 and L3-6.

	L3-5	L3-6
Power (MW)	50	50
Intact loop		
Mass flow rate (kg/s)	476.4	483.3
Hot leg pressure (MPa)	14.86	14.87
Hot leg temperature (K)	576.0	577.1
Cold leg temperature (K)	558.0	557.9
Broken loop		
Cold leg temperature (K)	556.0	557.6
Hot leg temperature (K)	562.0	561.4
Steam generator Secondary side		
Water temperature (K)	543.0	542.8
Pressure (MPa)	5.58	5.57
Mass flow rate (kg/s)	26.4	27.8

Table 3.1.2 Main events in Experimets L3-5 and L3-6 (seconds after LOCA initiation).

Event	L3-5	L3-6
Reactor scrammed	-4.8	-5.8
Break opened	0.0	0.0
Pumps switched off	0.8	no
HPIS on	4.8	3.6
Pressurizer emptied	22.2	20.2
Upper plenum reached saturation	28.4	28.5
End of subcooled break flow	92.9	44.2
Main steam control valve cycling	82...92	89...99
Primary pressure reached secondary pressure	745	930

3.1.3 Comparison of LOFT L3-6 with code predictions.

An overview at how the tested codes predict the charateristic features of the experiment is given in table 3.1.3.1.

In the experiment L3-6 the pumps were kept running at normal speed throughout the test in order to provide data to analyze the pump operation in two-phase condition. Because of the running pumps the distribution of water and steam remained near to homogeneous.

The primary side pressure was calculated quite well by all three codes, see fig. 3.1.3.1. The temperatures in the primary side were close to saturation and therefore both liquid- and cladding temperatures were also well predicted, see figs. 3.1.3.2 and 3.1.3.3. The secondary side pressure predictions of the codes differed somewhat from the measured values. The possible reasons for the defective prediction were the uncertainty of the leakage rate through the main steam isolation valve, as explained in sect. 3.1.2. Also the poor heat transfer modelling of the condensation heat transfer and probable wrong mass and temperature distributions in the steam generator secondary side had an impact. Definite conclusions about the secondary side predictions can not be drawn because of the inadequate instrumentation in the steam generator secondary side.

The comparison of the calculated and measured pressure differences indicated that the codes could not predict the pump two-phase behaviour correctly as can be seen in fig. 3.1.3.4. Especially the sudden drop in measured differential pressure at 30s, presumably caused by pump two-phase degradation, is not seen in the code prediction. The input to the pump models were based on data for the Semiscale pumps and is not directly applicable to the LOFT-pump.

The calculated densities corresponded quite well to the measured data as seen in fig. 3.1.3.5. However, in this experiment the water and steam distribution was near to homogenous and the good density results in this test case comparison do not necessarily prove that the codes are always able to calculate the mass distribution correctly.

As the break mass flow rate was not measured in the first 50 s of the transient the accuracy of the code predicted break mass flows in this time interval cannot be judged. However, for the rest of transient the code predictions were close to the measured data as seen in fig. 3.1.3.6. This was partly obtained by tuning the discharge coefficiens (i.e. changing them to obtaine best agreement with experimental data). Unfortunately, the lack of experimental break mass flow data for the start of the transient, where it is rather high, makes it impossible to draw any definite conclusions regarding the accuracy of the liquid mass in the entire reactor system as predicted by the codes.

As a whole the code predictions were very similar for this test case and clear differences in the ability of the codes to calculate the small-break transient could not be observed. The computer time consumptions are listed in table 3.1.3.2.

Table 3.1.3.1 Code performance on characteristic features of LOFT L3-6 experiment.

Feature	TRAC-PF1	RELAP5 (Finland)	RELAP5 (Sweden)	SMABRE
Sec.Pressure	Too high	Good up to 900 sec., too low thereafter	Good up to 800 sec.	Good up to 900 sec., too low thereafte
Primary pressure	Well	Well	Well	Well
Primary densities	Intact loop: fairly well, broken loop: poorly	Intact loop: fairly well, broken loop: poorly	Intact loop: fairly well, broken loop: poorly	Intact loop: too high, broken loop: satisfactorily
Disharge flow	Fairly well (choice of disc.coeff.)	Fairly well (choice of disc.coeff.)	Fairly well (choice of disc.coeff.)	Fairly well (choice of disc.coeff.)

Table 3.1.3.2 Computer time consumptions (LOFT L3-6).

Code	TRAC/PF1	RELAP5(Finland)	RELAP5(Sweden)	SMABRE
Computer	B 7800	CDC 173	CDC 172	PDP 11/34
Storage requirement (k words)	140	81	-	-
Number of control volumes	131	59	64	44
Number of junctions	135	63	65	47
Transient run to (s)	1400	1300	500	2500
Number of time steps	2292	9987	-	16417
Mean time step (s)	0.48	0.135	-	0.152...0.5
CP-time consumptions (s)	18781	9725	11000	40860
CP-time/time step (S)	6.43	0.974	-	2.43
CP-/real -time ratio	13.42	7.2	22.0	16.344
CP/real, reduced	0.1386*	0.0346**	0.1797***	0.00654****

* Burroughs CP/real-time/96.8
** CDC 173 " /208.1
*** CDC 172 CP/real-time/122.4
**** PDP 11/34 " /2500

Reduction according to the speed of the computers as estimated by a matrix inversion test but for PDP 11/34 by best guess

3.1.4 Comparison of LOFT L3-5 with code predictions.

An overview of how the tested codes predict the characteristic features of the experiment is given in table 3.1.4.1.

In this experiment the pumps were switched of at the begining of the transient and according to the gamma beam densitometer measurements considerable separation of steam and water occurred.

The primary pressure was fairly well predicted by all the codes as seen in fig. 3.1.4.1. Towards the end of the transient the RELAP5 calculations gave too high pressure, while TRAC-PF1 and SMABRE tended to underestimate the pressure. In the RELAP5 calculations the primary pressure followed quite closely the secondary pressure, while the primary pressure computed by TRAC-PF1 was almost totally insensitive to the variations of the secondary pressure. The coolant temperatures followed closely the saturation line and the temperatures were as well calculated as the pressure as seen in fig. 3.1.4.2. Only the hot leg temperature measurements indicated somewhat superheating which the codes could not predict. The cladding temperature, fig. 3.1.4.3. was also fairly well predicted.

In the calculations of the secondary side the codes had similar difficulties as in the experiment L3-6. The reasons for the defective predictions are expected to be the same as in L3-6, namely, the uncertainty about the leakage rate through the main steam isolation valve, poor modelling of the condensation heat transfer and the probable wrong mass and temperature distribution in the steam generator secondary side. However, poor instrumentation in the secondary side make the comparison between the data and calculated results uncertain.

In the primary side the major difficulties were related to the calculation of the coolant mass distribution as can be seen e.g. in the intact loop cold leg close to the break, fig. 3.1.4.4. None of the codes could predict the right densities.

RELAP5 had a tendency to give too high density values, while the calculated results of SMABRE and TRAC varied above and below the measured data. The assessed codes are obviously not able to predict the low flow two-phase conditions using nodalization where the number of the mesh cells is relatively small.

The differential pressure measurements and calculated results differed much from each other as can be seen e.g. from the pressure vessel data in fig. 3.1.4.5. However, the differential pressure measurements did not seem very reliable for the low values of the pressure losses, because the differential pressure instrumentation was obviously tuned to measure the higher differential pressures during pump operation. Therefore, very profound conclusions about the differential pressure calculations can not be drawn.

Experimental data for the break mass flow was not available for the first 50s of the trasient. By "tuning" the discharge coefficient (i.e. changing it to obtain best agreement with experimental data) the discharge mass flow was quite well calculated by all the codes, see fig. 3.1.4.6. TRAC/PF1 predicted too high mass flow for the first 500 s. In the RELAP5 calculation the discharge flow experienced a sudden jump at about 500 s, which was caused by a discontinuity in the flow pattern model.

As a whole the calculation of the experiment L3-5 turned out to be useful, because this test case revealed much more deficiences in the codes than the first test case L3-6.

The computer time consumptions for the test case are listed in table 3.1.4.2

Table 3.1.4.1 Code performance on charateristic feature of LOFT L3-5 experiment.

Feature	TRAC-PF1	RELAP5	SMABRE
Sec. pressure	Much too high	Too high	Too high
Prim. pressure	Too low towards end	Too high towards end	Too low towards end
Primary respect. secondary pressure	Too far from each other	Too close to each other	Too far from each other
Primary densities	Poorly predicted	Poorly predicted	Poorly predicted
Disharge flow	Too high before 500 S	Quite well, sudden pump at 500 S	Quite well

Table 3.1.4.2 Computer time consumptions (LOFT L3-5).

Code	TRAC-PF1	RELAP5	SMABRE
Computer	B 7800	CDC 173	PDP 11/34
Storage requirement (k words)	130	81	22
Number of control volumes	131	58	44
Number of junctions	135	61	47
Transient run to (s)	2300	1299	2030
Number of time steps	4827	7980	12830
Mean time step (s)	0.48	0.1692	0.158
CP-time consumptions (s)	30620	8616	30775
CP-time/time step (s)	6.34	1.08	2.40
CP-/real	13.31	6.38	15.16
CP/real, reduced*)	0.1375*	0.0307**	0.0061***

* Burroughs CP/real-time/96.8
** CDC 173 " /208.1 *** PDP 11/34 CP/real-time/2500

Reduction according to the speed of the computers as estimated by a matrix inversion test but for PDP 11/34 by best guess

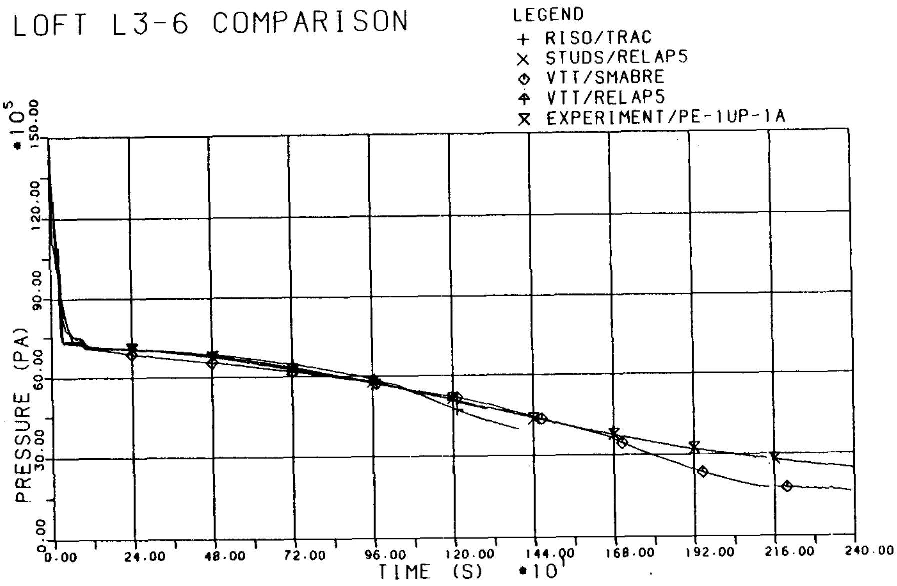


Fig. 3.1.3.1 Pressure in reactor vessel upper plenum

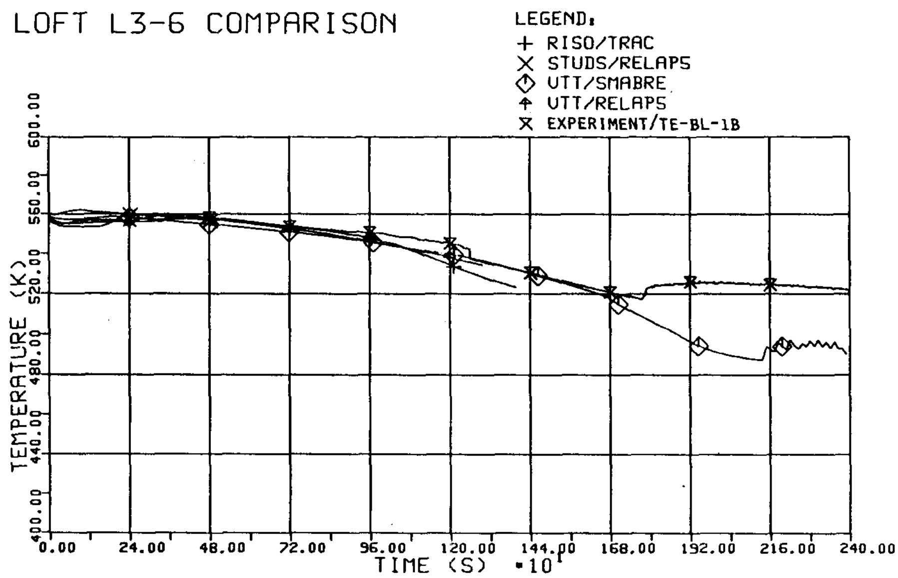


Fig. 3.1.3.2 Temperature in broken loop cold leg

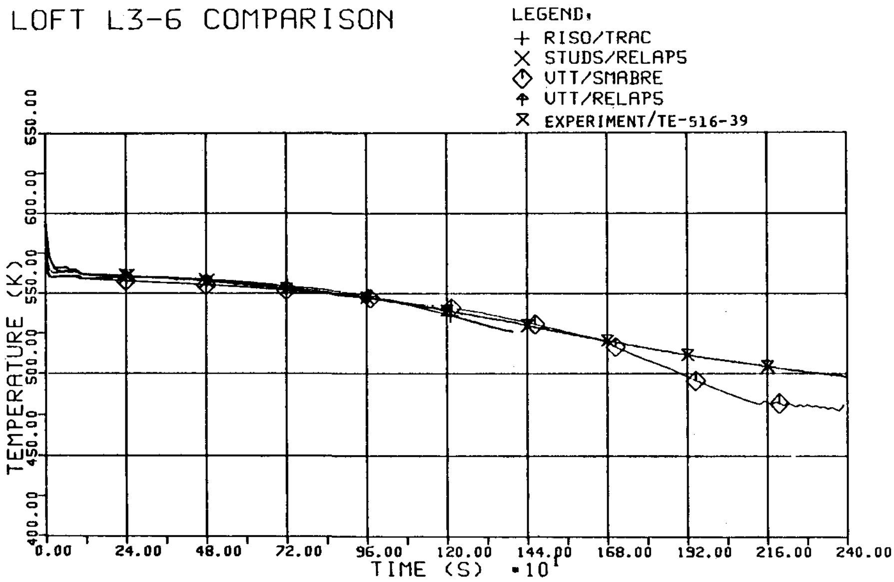


Fig. 3.1.3.3 Cladding temperature, average rod at 39 inch elevation

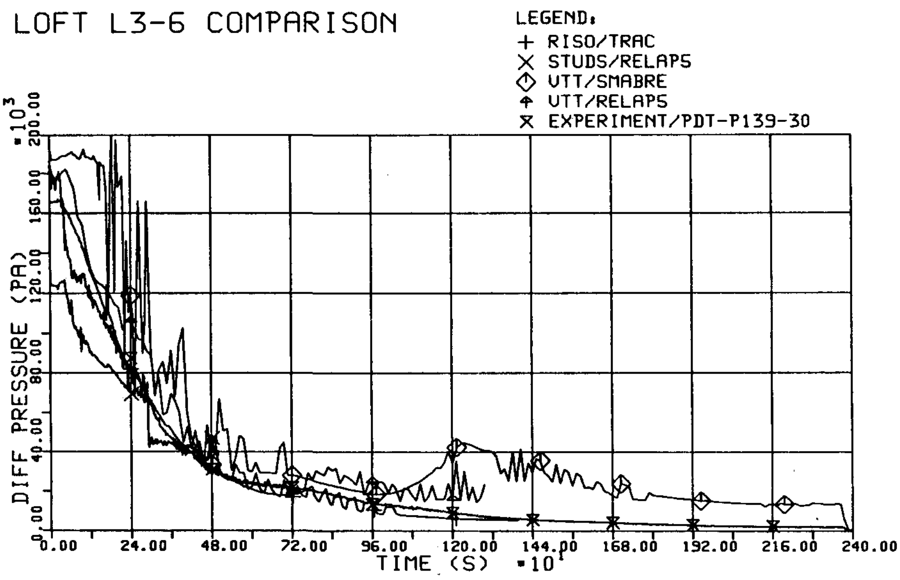


Fig. 3.1.3.4 Differential pressure across reactor vessel

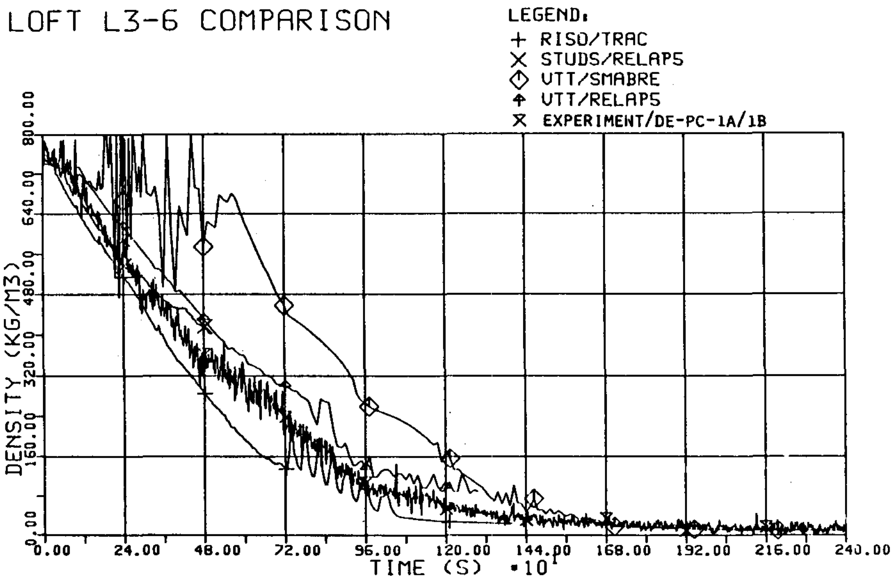


Fig. 3.1.3.5 Density in intact loop cold leg near break

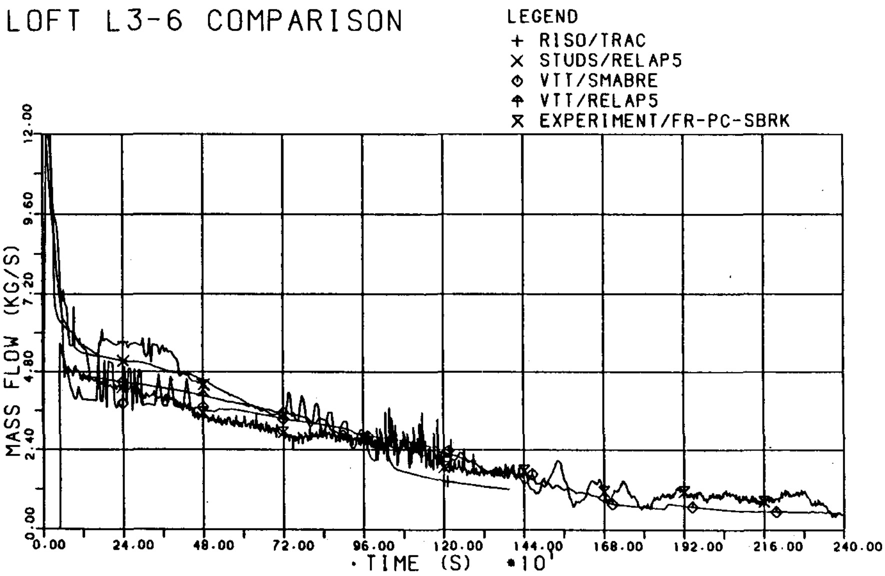


Fig. 3.1.3.6 Break mass flow

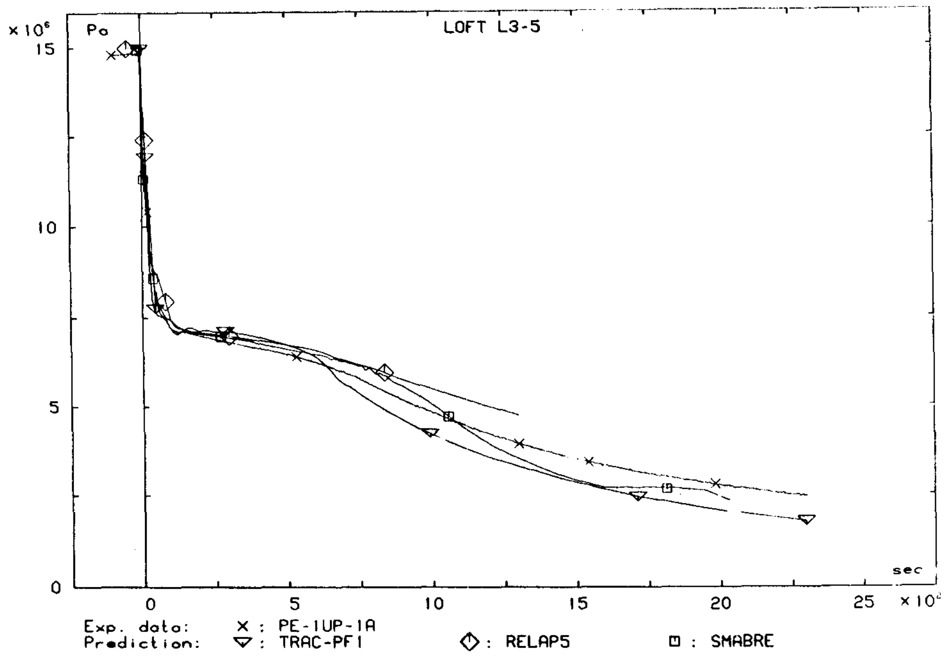


Fig. 3.1.4.1 Pressure in upper plenum

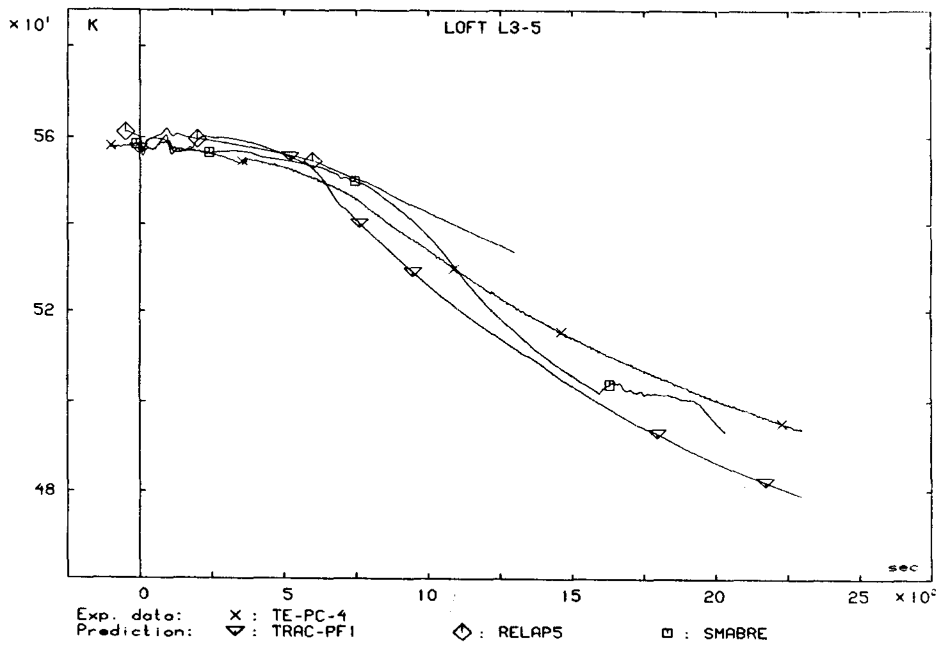


Fig. 3.1.4.2 Fluid temperature in intack loop cold leg

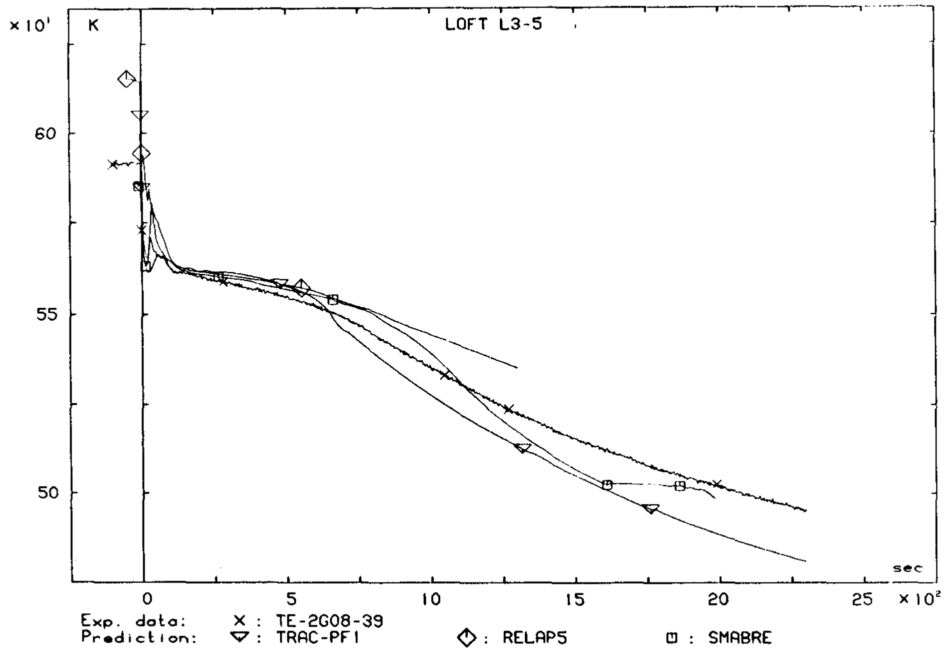


Fig. 3.1.4.3 Cladding temperature, average rod at 39 inch elevation

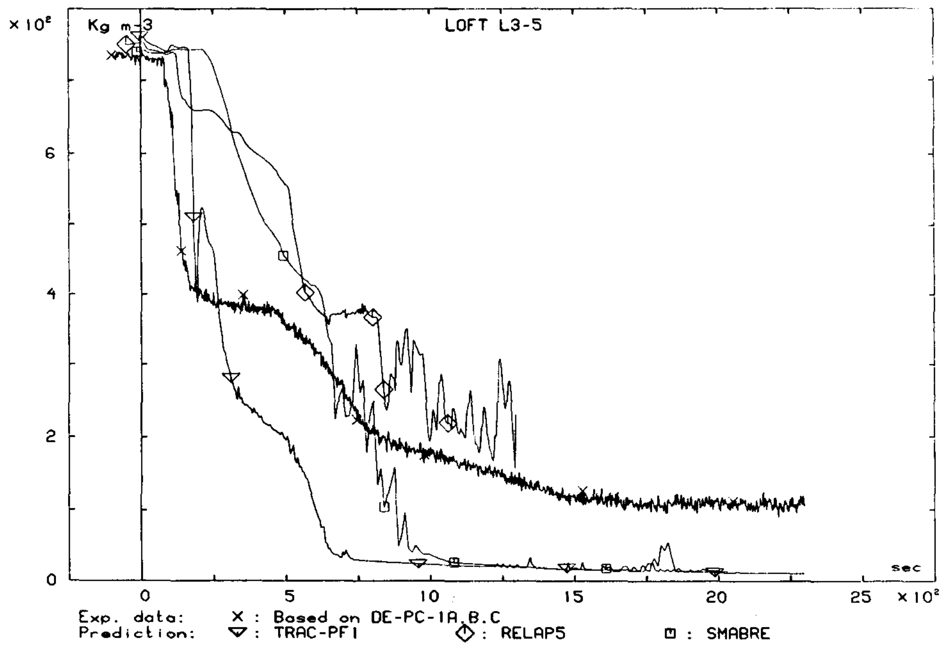


Fig. 3.1.4.4 Mixture density in intact loop cold leg

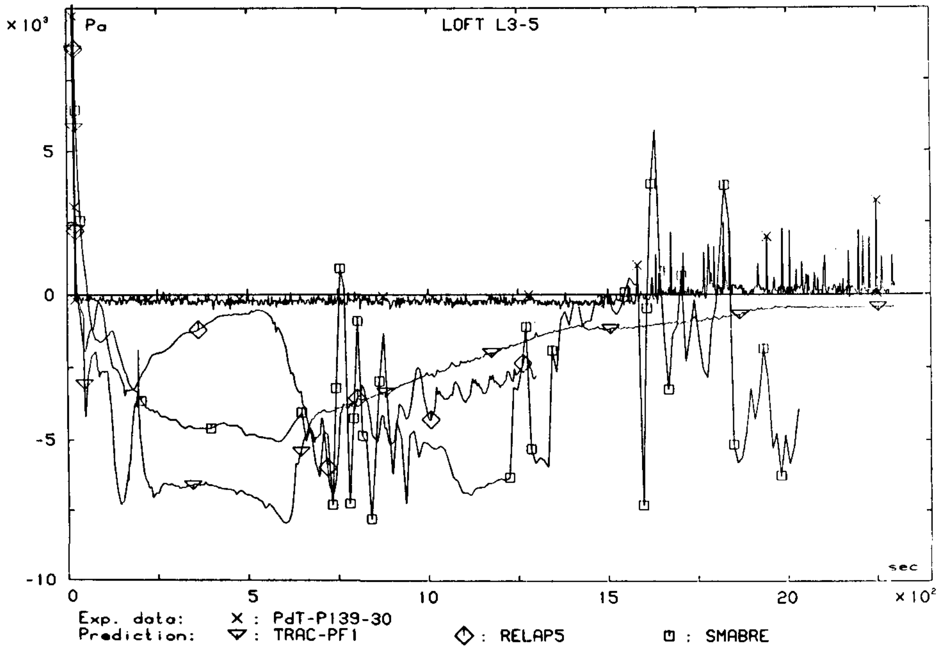


Fig. 3.1.4.5 Different pressure across reactor vessel

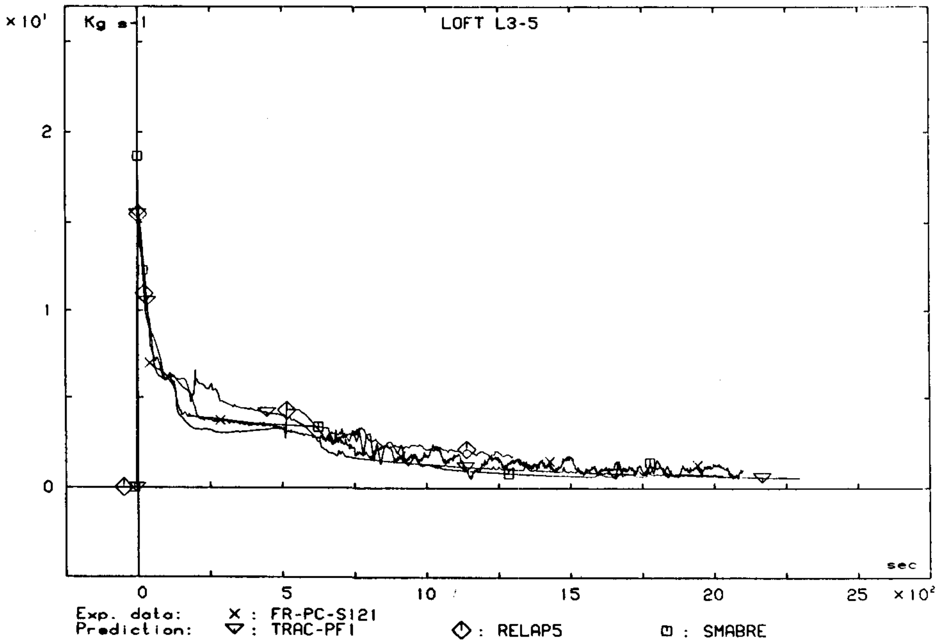


Fig. 3.1.4.6 Break mass flow

3.2. LOBI (mod. 1) SD-SL-03.

3.2.1. Objective of the experiment.

The SD-SL-03 experiment[11] was run September 24th, 1980 as the third and last of the SD-SL (Shake Down Small Leak) series, which aimed at investigating if the loop temperature could be controlled down with 100 Kh^{-1} by steam release from the secondary side, plus at getting some experience for designing the LOBI mod. 2 facility (ecpecially for small break tests). The experiment simulated a 0.4% cold leg break between the pump and the reactor vessel.

3.2.2 Experimental result.

Unfortunately, the flow rates were too low to be measured by the flow instrumentation, drag bodies, because they were designed for the high flow velocities during large-LOCA experiments. The temperature, the pressure and the density measurements are however valid information, and it is possible to deduce the progression of the transient from these data. The main events of the transient are given in table 3.2.2.

After a steady state was achieved (see table 3.2.1) the break was opened at time = 0s. After a 20s delay the pump speed was controlled down to zero and the power to rest power. For nearly 200s the core was cooled by natural circulation. However, steam formed in the upper plenum, departed through the 8 holes in the core barrel into the downcomer top and pushed down the downcomer liquid level. As this level dropped below the cold leg studs the cold legs emptied into the downcomer and the hot legs were nearly filled with water from the steam generator, leaving liquid in the pump surge loop (loop seal). Thereby a situation was established with no natural circulation but instead with the steam generators working in reflux mode (steam flowing to, condensed liquid flowing from the steam generator through the hot leg in counterflow with the steam). In this situation the steam could escape rather unrestricted from the upper plenum to the break through the cold leg, but not through the hot leg due to the loop seal.

From now on the transient progressed with a continous decrease in pressure and temperature. In the uninsolated pipe sections the liquid got rather subcooled, probably due to near-stagnation. However, two times during the remainder of the transient the liquid was redistributed (most pronounced in the intact loop): at about 1100s because the secondary feed water was erroneously switched on, resulting in a short period with natural circulation in the intact loop, and at about 3600s due to a core heating drop out.

The situation observed in the SD-SL-03 experiment with steam escape directly to the break through the cold leg may not be the most probable way a small break LOCA in a real power plant develops because the bypass between the upper head and the upper downcomer (the 8 holes through the core barrel) are too large in comparison with full scale reactor leakages.

Table 3.2.1 Initial steady state

Reactor power	(MW)	5.22
Intact loop mass flow	(kg/s)	21
Broken loop mass flow	(kg/s)	7
Upper plenum pressure	(MPa)	15.44
Intact loop fluid temperature	(K)	
at vessel outlet		594
at vessel inlet		563
Broken loop fluid temperature	(K)	
at vessel outlet		600
at vessel inlet		564
Pressurizer fluid temperature	(K)	617
Secondary side pressure	(MPa)	5.74

Table 3.2.2 Main events in the experiment LOBI SD-SL-03 (seconds after LOCA initiation).

Event	Time (s)
Break opened	0
Pump speed controlled down to 0	20....
Core power controlled down to "rest power" = 2% of full power	20....
Start of flashing in upper plenum	100
Natural circulation stops because downcomer liquid level gets below cold leg studs	300
Natural circulation in intact loop reestablished for short time due to second feed water switched on	1100
Redistribution of liquid in loops due to core power drop out	3600

3.2.3 Comparison with code predictions.

An overview of how the tested codes predict the characteristic features of the experiment is given in table 3.2.3.

The very small-break size in this experiment (0.4%) means that the primary pressure, through the saturation temperature, is nearly completely determined by the secondary pressure and only to a smaller extent by the loss of mass and energy through the break. The secondary pressure was specified as input for the codes, and it is therefore not surprising that the primary pressure level was well predicted by all three codes (see fig.3.2.1) An overprediction by TRAC-PF1 of the primary pressure in the beginning of the transient may be due to a lacking model for delayed flashing. In the last part of the transient RELAP5 and SMABRE underpredicted the primary-secondary pressure difference, probably because the steam generator heat transfer was overestimated.
Like the primary pressure, the fluid temperatures settled along saturation line, were also well predicted, although TRAC/PF1 and RELAP5, apparently exaggerating the effect of the injected cold pump-bearing-seal-water, got too large subcoolings at the end of the transient.

Both measured and calculated fuel simulator cladding temperatures, fig. 3.2.2., were close to saturation and neither of them indicated core uncovery. Thus the present comparison cannot indicate the ability of the codes to predict core uncovery.

The cold leg clearance and hot leg refill at about 300s were not predicted by any of the codes as can be seen from the comparisons of cold leg average densities in fig. 3.2.3.

The clearly higher average fluid density when measured diametrically (DD26HDIA) as compared to measured periferically (DD26HPER) do indicate the stratification of water and steam in the horizontal cold leg.

The physics behind the cold leg clearance is explained the previous section (3.2.2) and the reason for the poor prediction is probably that the code models for horizontal stratified flow are either inadequate or missing. However, a too coarse nodalization of opper plenum cannot be excluded as a reason. As it can be understood from section 3.2.2, the cold leg clearence is conditional on clearence of the 8 holes from upper plenum to downcomer for steam escape. It is possible that the code simulations of the experiment were performed with too coarse nodalization in the upper plenum to allow simulation of steam pocket formation in the top of the upper plenum, where the 8 holes are situated. (Too coarse nodalization will result in prediction of a two-phase mixture in the entire upper plenum).

As a consequence of not predicting cold leg clearance none of the codes were able to predict the observed stop in natural circulation (see fig. 3.2.4.). TRAC/PF1 predicted the natural circulation in the broken loop to stop at 700s, much (400s) too late. Again, a too coarse nodalization of the steam generator may be an additional reason. It may have resulted in prediction of too homogenized flow and thereby in too high steam generator heat transfer, leading to overstimation of the driving force for natural circulation.

From the measured data a negative steam generator differential pressure seems to be to related to natural circulation.
Consideration of liquid-steam distribution in a steam generator in normal operation mode and in reflux condensation mode makes this plausible. TRAC/PF1 and RELAP5 reflects this phenomenum.

The existence of loop seals, pure liquid in the pump surge loops, may be inferred from the average density measurements. Apart from some large downward oscillations in the TRAC/PF1 predictions of the average density at the broken loop pump inlet, this is reasonably well predicted by the codes.

The liquid content in the downcomer may be derived from the differential pressure (mainly hydrostatic head), fig. 3.2.5.

It is seen that the (collapsed) liquid level is steadily falling from 1100s to 3600s. The codes underpredict the liquid content, especially RELAP5 and SMABRE (up to 30%), probably because cold leg clearence with cold leg water emptying into the downcomer was not predicted.

The break mass flow rate was unfortunately not measured so that no information on code prediction accuracies of this property and of the total fluid content in the system can be gained from this experiment. The code predictions of the break flow rate is shown in fig. 3.2.6. SMABRE, which uses another break model (Moody) than the "characteristic velocity model" used in TRAC/PF1 and RELAP5, gives a considerably smaller break flow than these.

A comparison of computer time consumption for the LOBI SD-SL-03 test case is given in table 3.2.4.

Table 3.2.3.

Code performance on characteristic features of LOBI-SD-SL-03 experiment.

Feature	TRAC-PF1	RELAP5	SMABRE
Sec. pressure	Input	Input	Input
Prim. pressure	+ Good, except overprediction 100-500s	+ Good up to 1500s, thereafter too close to sec. press	- Slight overprediction (max. 5 bar) up to 1600s, thereafter too close to sec. press.
Prim. fluid temperatures	+ Close to sat., but too low in later part of transient	+ Close to sat., but too low in later part of transient	+ Close to sat.
Cladding temperature (no core uncovery observed)	+ Close to sat.	+ Close to sat.	+ Close to sat.
Stratification, cold leq clearance	o No cold leg clearance	o No cold leq clearance	o No cold leq clearance
Natural circulation stop	o I.L. never, B.L. at 400s too late	o Never	o Never
Steam generator diff. press. in relation to nat. circulation	+	+	o Both pos. and neg. during natural circulation
Loop seal formation	- Large downward oscillations in aver. density at B.L. pump inlet	+ Small oscillations in aver. density at I.L. pump inlet	+
Downcomer liquid content	- Too little	o Much too little	o Much too little

+ Good ; - Fair ; o Poor ; I.L.: Intact loop ; B.L.: Broken loop

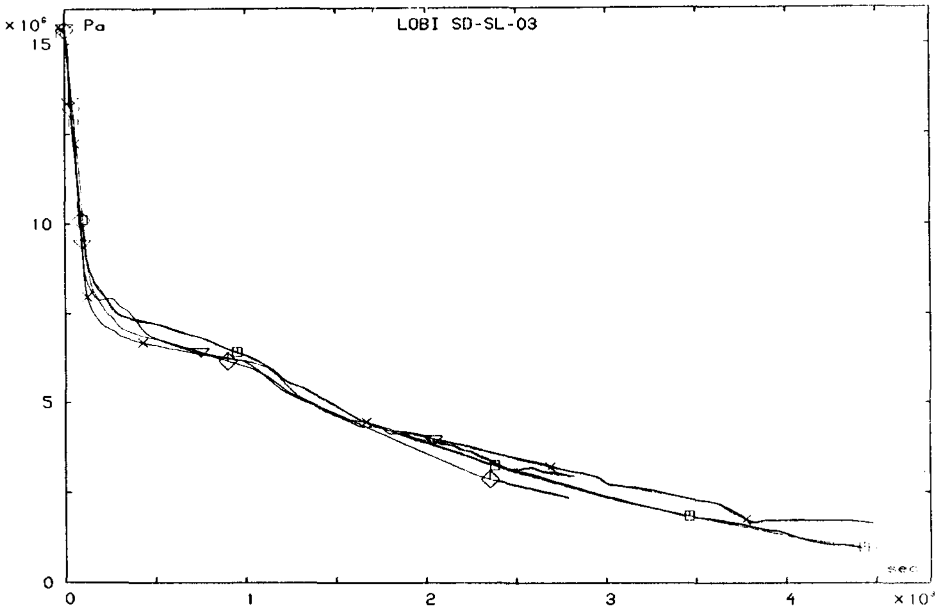
Table 3.2.4

Computer time consumption (LOBI SD-SL-03)

Code	TRAC-PF1	RELAP5	SMABRE
Computer	B7800	CDC 173	PDP 11/34
Storage requirements (K words)	241	80	22
Number of control volumes	219	75	63
Number of junctions	219	80	66
Transient run to (s)	2822	2750	4500
Number of timesteps	10909	25000	18126
Mean time step (s)	0.26	0.11	0.25
CP consumptions (s)	107873	25000	69423
CP/timestep (s)	9.89	1.0	3.83
CP/real	38.23	9.09	15.43
CP/real, reduced	0.3949*	0.0437**	0.0062***

* Burroughs CP/real time/96.8
** CDC 173 " /208.1 *** PDP 11/34 CP real-time/2500

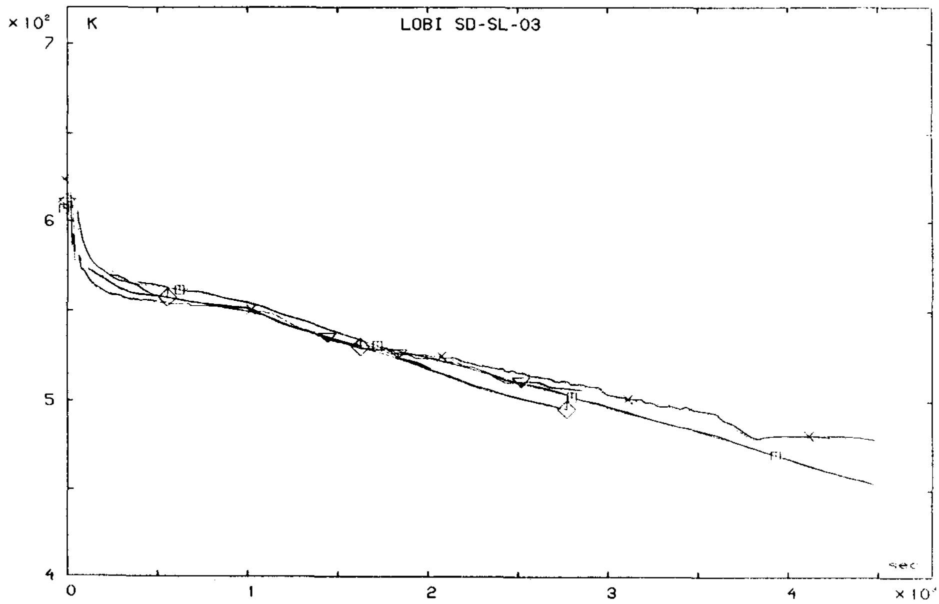
Reduction accordinq to the speed of the computers as astimated by a matrix inversion test but for PDP 11/34 by best quess



Exp. data: × : PA38-REF
Prediction: ▽ : TRAC-PF1 ◇ : RELAP5 ⊡ : SMABRE

Plot 1: Pressure in upper plenum

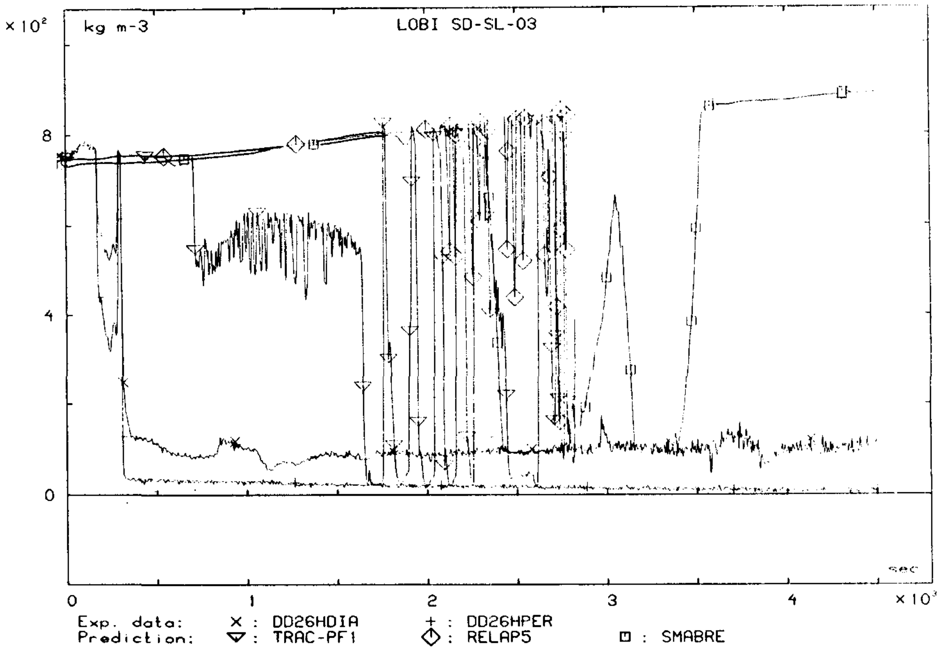
Fig. 3.2.1



Exp. data: × : TH34D210
Prediction: ▽ : TRAC-PF1 ◇ : RELAP5 ⊡ : SMABRE

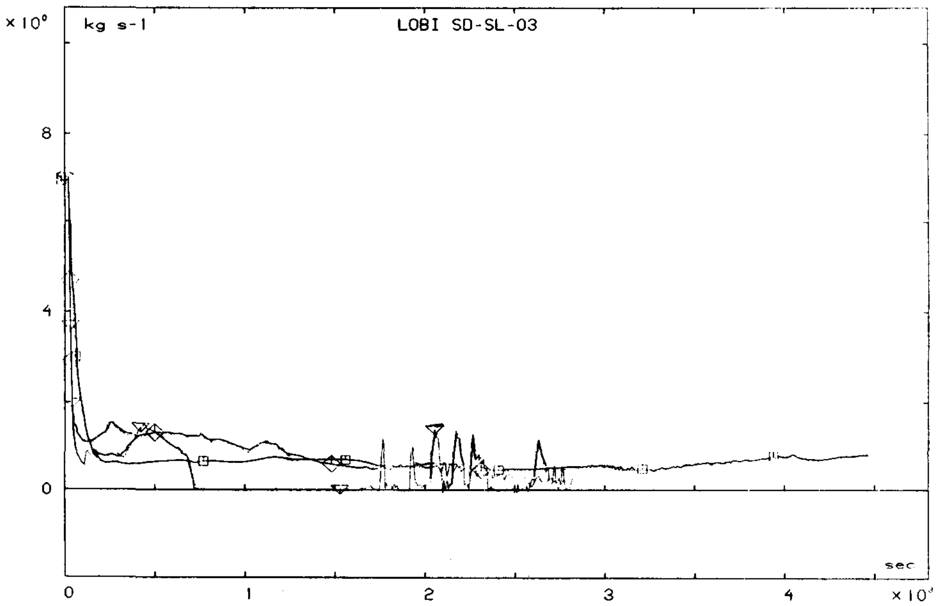
Plot 21: Cladding temperature at level 10 (3.42m), central zone

Fig. 3.2.2



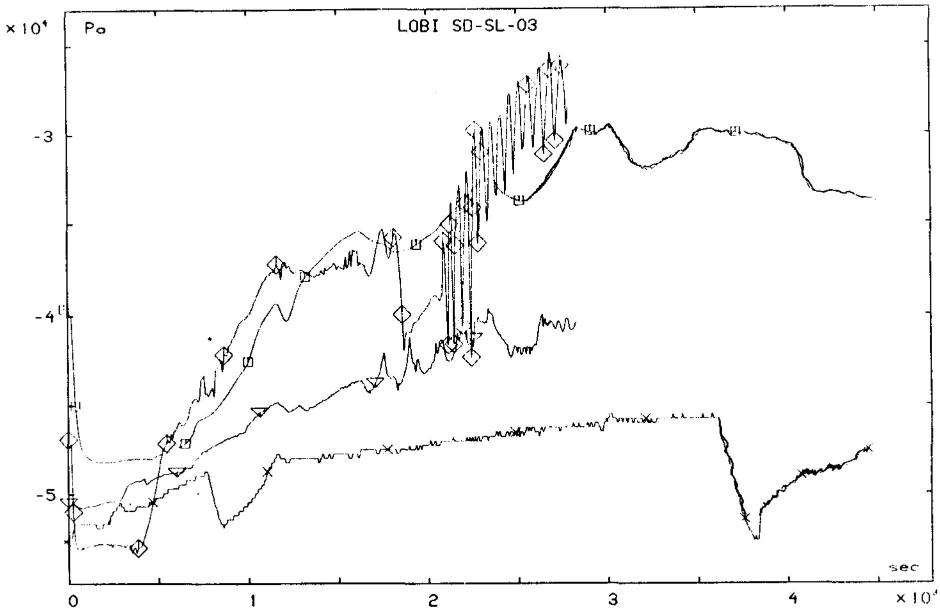
Plot 15: Fluid density in broken loop cold leg at RPV inlet

Fig. 3.2.3



Plot 24: Mass flow rate in broken loop at steam generator inlet

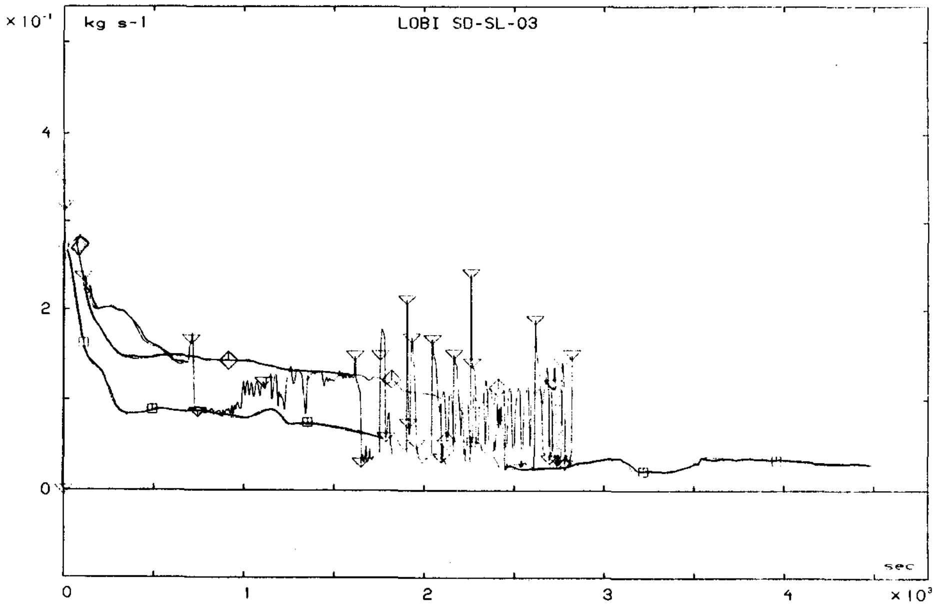
Fig. 3.2.4



Exp. data: X : PD313333
Prediction: ▽ : TRAC-PF1 ◇ : RELAP5 ⊡ : SMABRE

Plot 8: Differential pressure over the down-comer

Fig. 3.2.5



Exp. data: Not existing
Prediction: ▽ : TRAC-PF1 ◇ : RELAP5 ⊡ : SMABRE

Plot 25: Break mass flow rate

Fig. 3.2.6

3.3 FIX-II 3031.

3.3.1 Objectives of the experiment.

A preceeding experiment series on the FIX test aparatus, the FIX-I setup, had been aimed at investigating the time from opening the break and until the first dryout at various test conditions. The FIX-I therefore had particular dryout detectors and additionally a high power rod simulator. The experiments with FIX-I showed that the minimum break size on a cold leg of the Swedish Oskarshamn 2 reactor which could cause an early dryout would be about 900 cm^2. One of the main objectives of the reconstructed FIX facility, The FIX-II setup, has been to conduct experiments for the same type of breaks but now also beyond the dryout time and until end of the blow down.

3.3.2 Experimental result.

The details of the experiment is found in ref. 13.

The experiment was started from an initial steady state as given in table 3.3.1. The major events of the experiment after the break was opened at time = 0s are listed in table 3.3.2.

The split break occurring on the recirculation line between a pump and the vessel inlet causes a rapid opening of the Steam Relief Valve (SRV). A fast depressurization drop starts whereby the increased core void formation would secure a fast shut off of the fission power generation in the reference plant. On a low pressure signal the SRV is the again closed at 2.1 s after break to increase the core cooling capability. However, since the operation signal of the low pressure injection is slowly attained with the present break size the SRV is once more opened at about 12 s for all the rest of the blow down period.

In the experiment the SRV cycling is forced to take place at predetermined time set-points, see table 3.3.2. Also the isolation of the feed water and the condenser spray water are established at given times.

In the experiment the intact line pump (Pump 1) had a speed regulation according to a specified speed down curve from the break time on. The broken loop pump (Pump 2), however, was not controlled by a speed down regulating equipment and so, for the present test, the pump remained running at about constant speed during the test.

Table 3.3.1. Initial conditions, measured and computed for test No 3031

Quantity		Measured	RELAP5 Studsvik	RELAP5 VTT	TRAC-PF1 IFE
Pressure in the steam dome	(MPa)	6.86	6.88	6.86	6.88
Power to the 36-rod bundle	(MW)	3.337	3.337	3.337	3.337
Power to the bypass heaters	(kW)	53.0	53.0	53.0	53.0
Cooling power in the filler body space	(kW)	198	208.9	204.6	195.8
Mass flow rate, pump P1	(kg/s)	4.56	4.55	4.56	4.59
Mass flow rate, pump P2	(kg/s)	1.56	1.54	1.56	1.57
Mass flow rate, bypass	(kg/s)	0.58	0.52	0.58	0.59
Mass flow rate, 36-rod bundle	(kg/s)	5.53	5.57	5.54	5.58
Mass flow rate, spray line	(kg/s)	5.33	5.29	5.10	5.33
Mass flow rate, feed water line	(kg/s)	2.21	2.19	2.41	2.21
Temperature, feed- and spray- water	(K)	453.15	452.15	452.45	452.50
Temperature, water at bundle inlet	(K)	543.25	542.38	543.25	540.4
Water level in the spray condenser	(m)	6.24	6.25	6.24	6.25
Rotational speed, pump P1	(rad/s)	159.7	158.7	159.6	159.7
Rotational speed, pump P2	(rad/s)	211.12	216.4	211.1	211.2
Heading, pump 1	(kPa)	125.0	121.7	125.3	127.5
Heading, pump 2	(kPa)	111.1	108.5	111.4	107.3
Δp, core inlet	(kPa)	29.5	29.7	38.6	30.3
Δp, steam separator	(kPa)	27.6	34.8	31.2	27.5

Table 3.3.2 Main events in the experiment FIX-II 3031 (seconds after LOCA initiation).

Break valve starts to open	0.0 s
Break valve fully open	0.08 s
SRV(Steam Relief Valve) starts to open	0.2 s
SRV fully open	0.7 s
SRV starts to close	1.3 s
Spray flow is closed	1.8 s
Feed water flow is closed	1.9 s
SRV is closed	2.1 s
SRV starts to open *	11.8 s
SRV fully open *	12.3 s
Experiment finished	70.0 s

---

* SRV cycling period to attain the pressure level, at which the low pressure coolant injection is activated, faster.

3.3.3 Comparison with code predictions.

Predictions for the FIX-II test case were obtained by VTT and and by Studsvik (two calculations with different nodalization) using the RELAP5/MOD1 code and by IFE using the TRAC/PF1 code. An overview of the performance of the codes is given in table 3.3.3. The computer resources used in the calculations are given in the table 3.3.4.

All the calculations succeded to get the initial steady state conditions in good agreement with data from the experiment, table 3.3.1. However, in the TRAC/PF1 calculation it was necessary to apply a fictive, high condenser spray velocity in order to get the system pressure correct. Also, in that calculation, the loss coefficients specified for in-loop restrictions by the facility description were rather much adjusted to obtain the measured differential pressure. In the RELAP5 calculations adjustment of the feedwater temperature and the feedwater/spraywater flow ratio was necessary.

No really significant problems were encountered in carrying through the transient calculations, however, some problems to mention are

- An unrealistic counter current flow occured between two volumes having large differences in the void fractions. A program update was undertaken. (Studsvik RELAP5/MOD1)

- In the previous FIX-II/ISP15 calculation quite short time steps occasionally had occured due to difficulties in the static quality in a volume as encounted by the time step control. To prevent that difficulty in the present test case the maximum time step given by input was reduced. The CPU/real time ratio could eventually, as a consequence, be too large (VTT/RELAP5).

- The break mass flow during the initial subcooled period turned out too small due to an incorrect Bernoulli pressure reduction used in the code. The calculation was repeated using a fictive 50 % too large break area during the initial 10 s of the transient (IFE/TRAC/PF1). In the RELAP5 calculations a value of the break discharge coefficient based on experience was sufficient.

The system pressure decay, fig. 3.3.1, shows good over all comparisons between the calculations and the experiment. Apart from a strong dependence on the break mass flow and also the on/off operation of the steam relief valve the supply of heat from structures is important to the pressure. All the calculations involved modelling of virtually all passive structures in contact with the system fluid.

Comparisons of the loop flows and coolant distribution are of a primary interest when judging the prediction capabillity of the codes. Unfortunately, the FIX-II facility involves no in-loop measurements of void and moreover the mass flow measurements by orifice pressure losses are only applicable in single-phase situations. Now, from the hydraulic heads of the two main pumps, particularly the broken loop pump operated at constant speed, a sudden degradation seen at 12 - 13 s indicates the onset of voiding, fig. 3.3.2. The delayed voiding seen for the Studsvik/ RELAP5 28-volume calculation is an effect of too few volumes being used.
Fig.3.3.3 shows apparent differences in the lower plenum void in the calculations. Noteworthy are the high voids predicted by VTT/RELAP5 and IFE/TRAC/PF1 flow from about 35s on. This means that coolant must be distributed to somewhere else in the loop. Actually fig. 3.3.4 shows a high water content on top of the core, particularly in the VTT/RELAP5 calculation.

The figs. 3.3.5 and 3.3.6 compare the rod clad temperatures at levels 5 and 10 of the experiment, i.e. at 36 % and 63 % of heated length. At the level 5 the VTT/RELAP5 calculation does predict the time of the first dryout well while the two Studsvik/RELAP5 calculations show a too early CHF (critical heat flux) initiated already from the decreasing core coolant flow upon closure of the steam relief valve. Then, at the level 10 all the predictions together with the experiment show the early CHF, but as a general comment, the IFE/TRAC/PF1 calculation came up with a too slow temperature rise. The experience from the quite similar FIX-II experiment 3025 (ISP15) showed that the use of the Biasi CHF-correlation instead of the standard high flow CHF-correlation (Hsu-Becker) of RELAP5 improved the prediction of the early CHF in the intermediate size break test of a BWR. Typically, no code succeded in predicting the cooling down to about saturation temperature of the rod cladding at all levels. The reason is the absence of models for rod quenching and radiative heat transfer to a wetted box wall.

Table 3.3.3 Code performance on characteristic features of FIX-II experiment 3031.

Feature:	RELAP5 (Studsvik)	RELAP5 (VTT)	TRAC/PF1 (IFE)
System pressure	Slight underestimate during SRV closure 2-12 s, when slightly overestimated	Good over all	Good over all
Break mass flow	Good over all (interference from a code error is seen)	Slightly low	Good over all
Flow rates in return legs	Comparison good until voiding starts in the experiment	Comparison good until voiding starts in the experiment	A fictive too large break area was necessary to compensate for a model error in the code
Coolant distribution*	The 58-volumes and the 29-volumes calculations show apparent difference	An accumulation of water in the upper plenum is judged not realistic. Lower plenum void goes high	Very high void in the lower plenum from 35 s on.
Pump performance	The degradation in the BL pump heading due to voiding starts late. Otherwise the comparison is good	Comparison is good	The IL pump shows no sudden degradation due to voiding. Otherwise the comparison is good
Rod clad temperatures	The mid-core DNB initiates too early and no quenching occurs later for 58 volumes. Temperatures are overestimated for the most.	The onset of DNB is well predicted at all levels. Then the temperatures are overpredicted particularly in the upper core region.	Temperatures at levels except in the lower core are overpredicted. Slower changes with the time compared to RELAP5 are displayed.

* No experimental data are available.

Table 3.3.4 Computer time consumption (FIX-II 3031)

Computer code	RELAP5/MOD1 Cycle 19 (Swedish version)	RELAP5/MOD1 Cycle 19 (Swedish version)	RELAP5/MOD1 Cycle 18 (Finnish version)	TRAC/PF1 7.0/EXTUPD7.6
Computer used	CDC 170/835	CDC 170/835	CDC 173	CDC 170/835
Numbers of volumes	58	29	72	77
Numbers of junctions	60	31	75	31
Numbers of heat slabs	63	28	56	32
Transient run to (s)	70	70	70	70
Numbers of time steps	2881	2433	4301	985
Mean time step (s)	0.024	0.029	0.016	0.071
CP consumptions (s)	2174	762	5458	2359
CP/timestep (s)	0.75	0.31	1.27	2.39
CP/real	31.1	10.9	66.7	33.7
CP/real, reduced	0.74*	0.26*	0.32**	0.80*

* CDC 170/835 CP/real-time/42.2 ** CDC 173 CP/real-time/208.1

Reduction according to the speed of the computers as estimated by a matrix inversion test.

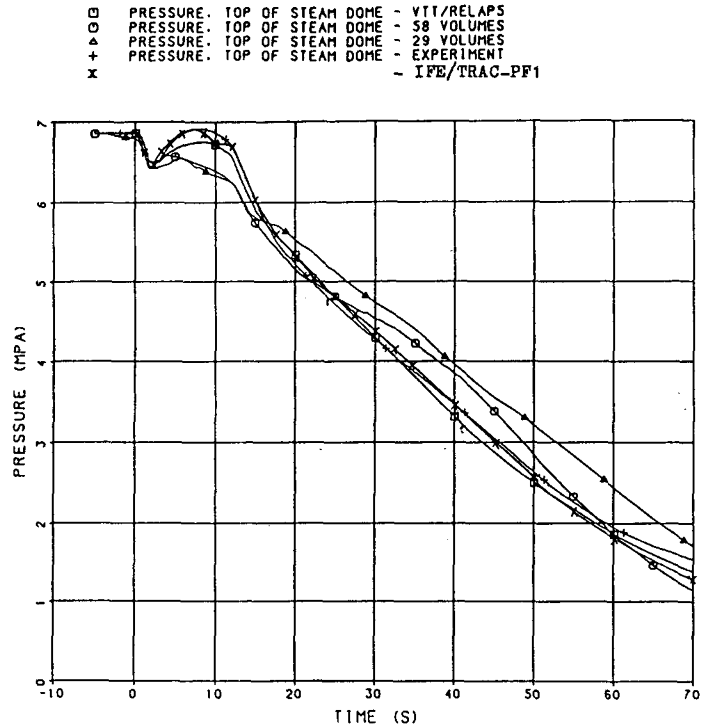


Fig. 3.3.1

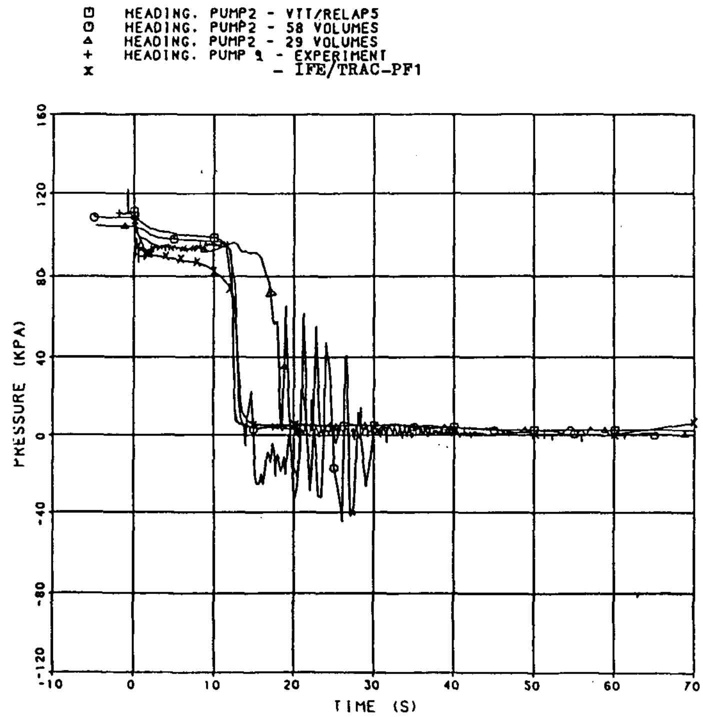


Fig. 3.3.2

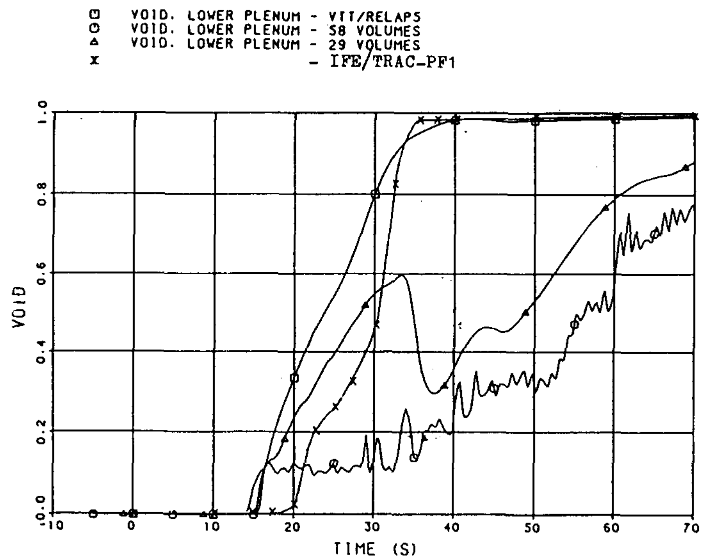


Fig. 3.3.3

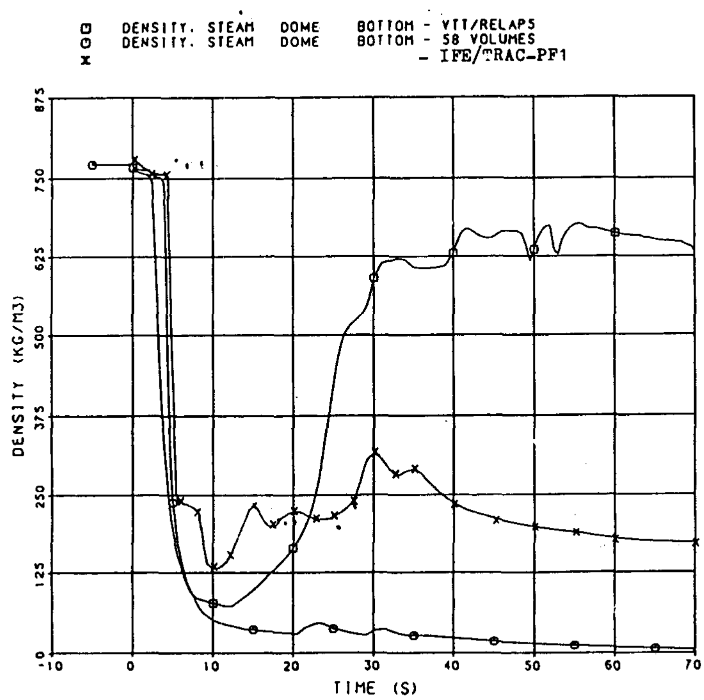


Fig. 3.3.4

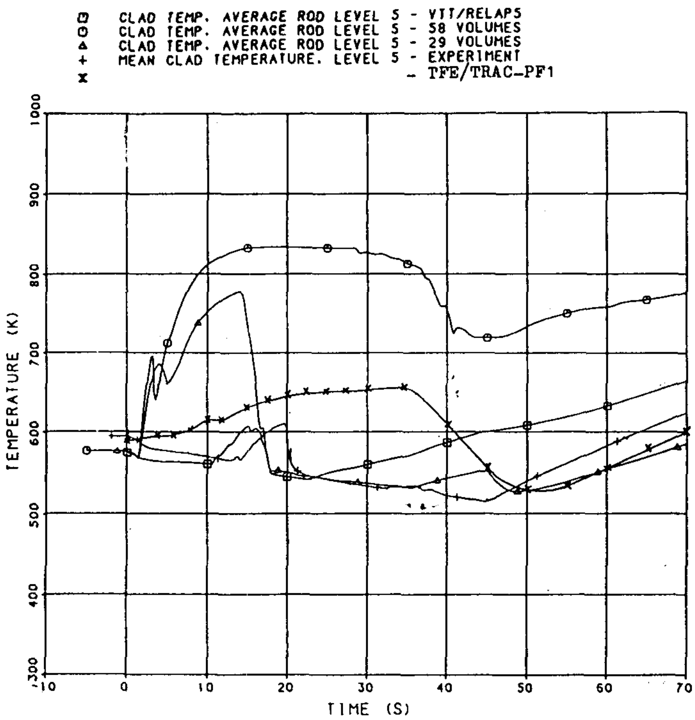


Fig. 3.3.5

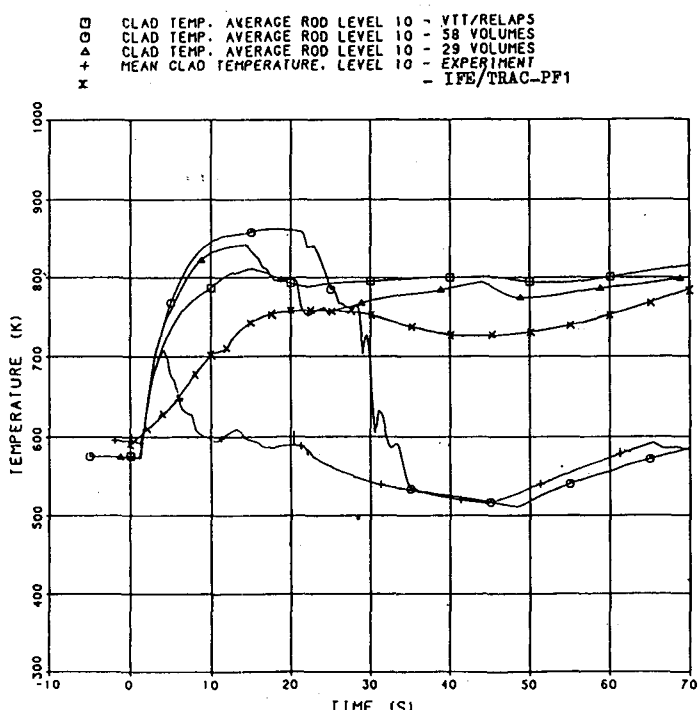


Fig. 3.3.6

3.4 LOBI (Mod.2) A2-81/ISP18

3.4.1 Objective of the experiment

The A2-81 experiment[34)] was run September 27 1984 as the first experiment in the LOBI/Mod.2 facility, which models a 1300 MWe PWR of KWU-design. As such it served as "International Standard Problem No.18" (ISP18). In the present study the ISP18 calculation was "double blind" in the sense, that not only were the experimental results unknown during the calculation but no data from earlier experiments in the test facility were available either.

More specifically, the experiment simulated a 1% cold leg break LOCA (break between pump and pressure vessel), without accumulator action but with the High Pressure Injection System (HPIS) for emergency core cooling running at 50%. The aim was to see if the reactor temperature could be brought safely down with a rate of 100°C per hour by a secondary side steam release.

3.4.2 Experimental results

All instrumentation worked well with the break flow meter as the single exception. Being a turbine flow meter, it was unable to measure the two-phase flow which developed after the break.

Although no other mass flow measurements were available in the primary system, except for the HPIS mass flow, the measured pressures, temperatures and mixture densities show quite clearly how the experiment progressed. The main events of the transient are given in table 3.4.2.

From a steady state situation (see table 3.4.1) the transient was started at time = 0 with the opening of a 3 mm diameter break on the broken loop cold leg (between pump and pressure vessel), causing a rather rapid initial pressure decrease. Triggered at 13.2 MPa by the falling primary pressure through a trip system (see table 3.4.2), at time = 32s the power started to drop along a predetermined decay curve and the secondary systems, feedwater and steam lines, were switched into a pressure-controlled-down mode. Between 45s and 117s the pump speeds of the circulation pumps were regulated down to zero. At time = 74s (primary pressure = 9 MPa) the HPIS started, which slowed down the depressurization rate considerably. From thereon the primary pressure (fig. 3.4.1) was smoothly falling from about 9 MPa to 1 MPa, which it reached at about 4500s, where the experiment was ended.

At about 200s the falling primary pressure initiated steam formation in the primary system leading to stratified flow in the hot and cold legs. Between 500s and 2000s steam formation in the primary sides of the steam generators, the U-tubes, gradually cleared these from water, but with the loop seals still water filled. Hence somewhere between 200s and 500s natural circulation probably stopped, thereby changing the operation of the steam generators into reflux mode (steam flow through hot leg to steam generator, condensed liquid flowing back to the reactor vessel through the hot leg). Thus a part of the core power was removed by the steam generators (in reflux mode) while the rest was removed with the steam escaping through the bypass from upper plenum to upper downcomer and further through the broken loop cold leg out through the break. The differential pressure readings show that the pressure levelling out by the steam bypass between upper-plenum and reactor downcomer top was sufficient to give no significant tendency to clear the loop seals or to uncover the core. Also the reactor downcomer was always water filled. After steam formation in the primary system had started both fluid and heater rod temperatures were very close to saturation, whereas significant liquid subcooling was found in the hot and cold legs, just another indication of stratified flow.

The behaviour of the secondary system was characterized by close to saturation conditions and by a smooth decrease of the collapsed liquid level in the steam generator downcomers.

Table 3.4.1 Initial Steady State.

Reactor power	(MW)	5.21
Intact loop mass flow	(kg/s)	21.0
Broken loop mass flow	(ks/s)	6.7
Upper plenum pressure	(MPa)	15.8
Fluid temperature	(K)	
intact loop vessel inlet		566.9
broken loop vessel inlet		566.5
Secondary pressure	(MPa)	6.54
Feedwater flow	(kg/s)	
intact loop		2.0
broken loop		0.72
Feedwater temperatures	(K)	486.1
Recirculation ratio (downcomer-to-feedwater)		
intact loop		~ 6.5
broken loop		~ 4.5

Table 3.4.2 Main events in Experiment LOBI A2-81/ISP18 (seconds efter LOCA initiation).

Event	Trigged by P_{prim} (MPa)=	Time (s)
Break opened		0
Core power controlled down to "rest power" ~ 1% of full power	13.2	32.......
Second. feedwater closed	13.2	32
Second. steam line pressure controlled down along a 100°C/hour cooling curve, which starts at 83 bar at time = 0.	13.2	32.......
Pump speed controlled down to zero	11.0	45...117
HPIS start	11.7 + delay 35s	74
Broken loop locked rotor resistance simulator in position resistance (4s after pumps speed reaches zero)		~ 121s
Start of flashing in top of upper plenum and formation of stratification in hot and cold legs; presumable stop of natur. circulation		~ 200s
Steam formation and emptying of SGEN U-tubes		500-2000s
End of experiment	~ 1.0	~ 4500s

3.4.3 Comparison with code predictions.

This test case was calculated using TRAC/PF1, RELAP5/MOD2 (Swedish and Finnish versions) and SMABRE. An overview of the performance of the different codes regarding the characteristic features of the experiment is given in table 3.4.3.

Due to a program error the Swedish RELAP5/MOD2 was unable to treat stratified flow in horizontal pipes, but would treat this as bubbly or slug flow (a flow pattern map for vertical pipes was erronously used), overestimating the interface flow resistance between steam and water. The error was not discovered until the end of the calculation, and economics did not permit a repeated calculation with the corrected program version.

Similar to the SD-SL-03 experiment the break size (1%) was moderate. Thus, after the initial depressurization, the primary pressure, through the thermal coupling in the steam generators, was closely coupled to the secondary pressure and only to a smaller extent determined by the loss of mass through the break. As the secondary pressure decrease followed a programmed table from the time on where the primary pressure reached 13.2 MPa, the primary pressure as well as the secondary pressure were rather well predicted in all calculations, apart from a small time shift, as seen in figs. 3.4.2 and 3.4.9. It is noted that from 700s on, RELAP5/MOD2 (Finnish) underpredicts the primary pressure somewhat, apparently caused by a too large drop in heat-exchange through the steam generators once steam formation started in the primary side of these.

All calculations predicted a break mass flow (see fig. 3.4.1) starting at about 0.4 kg/s and decreasing towards about 0.25 kg/s. Unfortunately, no experimental values were available for comparison, but the initial decrease rate of the primary pressure is an indirect measure of the break flow. During the first 100s, before the HPIS water was supplied, RELAP5/MOD2 (Swedish) and especially TRAC/PF1 gave a somewheat too slow primary pressure decrease. Thus TRAC/PF1 predicted HPIS to start (prim.

pressure = 11.7 MPa + 35s delay) at 92s, i.e. almost 20s too late. This too slow pressure decrease indicates an underprediction of the break mass flow. The late predicted HPIS start gave rise to the above mentioned time shift in the pressure predictions.

The fluid temperatures at core outlet and in the hot legs as well as the heater rod surface temperatures were, not surprisingly, rather well predicted as they are close to the saturation temperature, although the predicted core inlet fluid temperatures and rod temperatures were somewhat high, see fig. 3.4.8. This inlet temperature is a consequence of an unability in the calculations to represent the mixing of the cold HPIS-water (from the intact loop cold leg) with the downcomer water. The mixing takes place as the cold and heavier HPIS-water sinks down in the downcomer before it continues through the broken loop cold leg to the break. In the experiment this mixing leads to colder downcomer water entering the core as well as to warmer water flowing into the cold leg of the broken loop. Obviously this is a multidimensional flow process, which could not be predicted in the calculations which all used a one-dimensional downcomer representation allowing the HPIS water to bypass the downcomer and flow into the broken loop cold leg. (However, in all of the codes a pseudo two-dimensional downcomer representation would have been possible.) Fig. 3.4.8 also shows the high fuel rod overtemperature by almost 400°C erronously predicted by RELAP5/MOD2 (Swedish version) for about 10 minutes of a false core uncovery (due to lacking stratified flow model) to be discussed in more detail below.

The discontinuation of natural circulation forcing a redistribution of steam and water in the primary circuit after initiation of steam formation was not so well predicted in any of the calculations. All calculations allowed for the bypass of steam, formed mainly in the core, from upper plenum into the downcomer top, so that the decrease of downcomer liquid level to the cold leg entrance was predicted. However, as seen from the fluid temperatures, fig. 3.4.3, and the mixture densities, fig. 3.4.5, no code was able to predict the stable stratified flow in the cold legs, by which, in the experiment, the steam actually made its way to the break. In fact, according to the calculations, the cold legs were almost waterfilled or heavy oscillations in the water content came up (RELAP5/MOD2, Finnish version). Also, none of the codes were able to predict the large observed difference in water- and steam-temperature because of a too high heat transfer.

On the other hand, all calculations but TRAC/PF1 did overpredict the steam formation in the hot legs and the primary side of the steam generators as indicated by the hot leg mixture density, fig. 3.4.4, and the U-tube differential pressures. With large content of steam in the hot legs and the steam generators the calculations did correctly predict reflux condensation to occur, with the condensate flowing partly down into the loop seals, partly back through the hot legs.

All calculations but TRAC/PF1 gave false predictions, at various times, of loop seal clearance due to a certain overpressure. This can be seen from the differential pressure over the descending leg of a loop seal, fig. 3.4.6. In the Swedish RELAP5/MOD2 calculation, due to the complete omitance of stratified flow, this overpressure was large enough also to force down the liquid level in the core completely (core uncovery) causing the earlier mentioned 400°C overtemperature in the heater rods.

Contrary to the other calculations and the experimental results TRAC/PF1 predicted refilling of the broken loop steam generator, as seen from the differential pressure over the steam generator U-tube, fig. 3.4.7. TRAC then predicted, falsely, natural circulation in the broken loop to start again at about 1600s, at which time both hot legs and steam generators had been sufficiently emptied to stop natural circulation. The intermediate stop in natural circulation from about 800s to 1600s caused a very high fluid density to be predicted by TRAC/PF1 in the broken loop cold leg due to very cold water caused by mixing in of pump bearing injection water (≈ 30°C).

Thus all calculations did overpredict the energy release through the steam generator by reflux condensation. Furthermore, in the TRAC/PF1 calculation of the broken loop the energy release was also accomplished by natural circulation.

The decrease of the collapsed liquid level in the secondary side of the steam generators, which is a measure of the energy released through the steam generators through boiling-off secondary water, was relatively well predicted by all calculations, see fig. 3.4.10. However, some deviations were seen in the broken loop steam generator, where the Finnish RELAP5/MOD2 predicted a too slow decrease while the others predicted a too fast decrease of the masss content.

Summing up the results of the A2-81/ISP18 calculations it is seen that integral properties such as pressure, heater rod surface temperatures and most fluid temperatures are relatively well predicted while phenomena connected with steam formation and distribution of water and steam in the primary system is poorly, for some aspects wrongly, predicted. This is first of all due to an inadequate representation of stratified flow. As a consequence, all calculations did overpredict the energy loss through the steam generators, by reflux condensation and by natural circulation (the latter only in the TRAC broken loop prediction), as opposed to energy release by steam escape through the break.

Also, this deficiency led to predictions of various non-observed flow phenomena: natural circulation restart, loop seal clearance and core uncovery.

Regarding the prediction accuracy none of the codes were significantly superiour to the others.

Table 3.4.4 contains relevant information regarding computer time consumption for this test case. Even when taking the number of control volumes into account RELAP5/MOD2 is seen to be evidently faster than TRAC/PF1, whereas SMABRE is considerably faster than both.

Table 3.4.3. Code performance on characteristic features of LOBI/A2-81/ISP18 experiment.

Feature	TRAC-PF1	RELAP5/MOD2 Swedish version	RELAP5/MOD2 Finnish version	SMABRE
Sec. pressure	- Input, time shift	- Input, time shift	+ Input	+ Input
Prim. pressure	+ Good, time shift	+ Good, time shift	- Good, but somewhat too low after 700s	+ Good
Prim. fluid temperatures				
hot legs	+ Close to sat.	+ Close to sat.	+ Close to sat.	+ Close to sat.
cold legs	- No non-equilibrium	- No non-equilibrium	- No non-equilibrium	- No non-equilibrium
Cladding temperature (no core uncovery observed)	+ Close to sat.	- Close to sat. but false overtemperature due to core uncovery	+ Close to sat.	+ Close to sat.
Stratification	o No cold leg strat.	o No strat. at all (code error)	- Unstable cold leg strat.	o No cold leg strat.
Loop Seal (no clearance observed)	- Tendency to loop seal clearancete	o Loop seal clearance	o Loop seal clerance	o Loop seal clearance
Steam generator U-tube clearance	+ B.L. st. gen. refill	- Somewhat too fast	- Somewhat too fast	- Somewhat too fast
Nat. circulation stop	o Restart of circulation in B.L. at 1600s	- Fair	- Fair	- Fair
Downcomer and core liquid level	- Fair	o False core uncovery	- Fair	- Fair

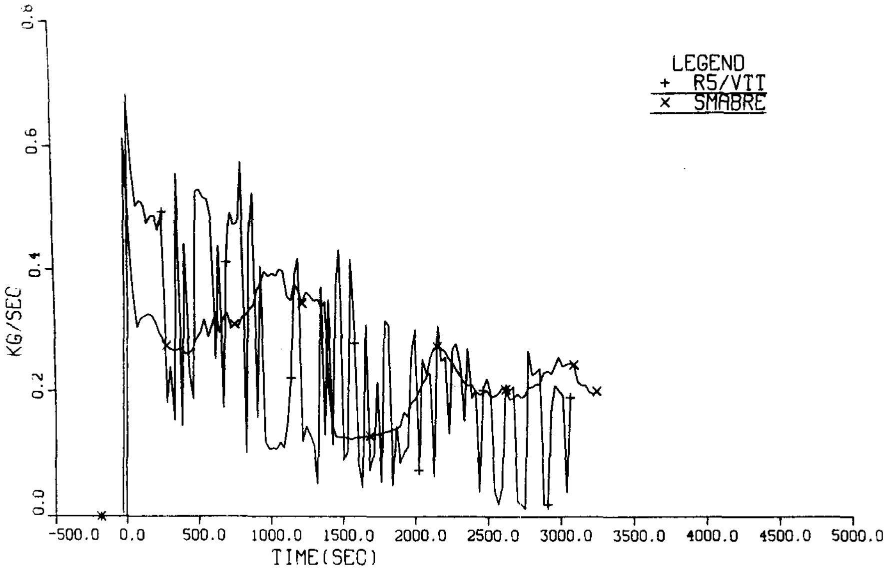
+ Good ; - Fair ; o Poor ; I.L.: Intact loop ; B.L.: Broken loop

Table 3.4.4 Computer time consumption (LOBI A2-81/ISP18).

Code	TRAC-PF1	RELAP5/MOD2 (S)	RELAP5/MOD2 (F)	SMABRE
Computer	B7800	CDC 180/835	CDC 173	CDC 173
Storage requirements (K words)	241	99.4	106	44.5
Number of control volumes	206	106	145	129
Number of junctions	211	112	151	134
Transient run to (s)	2949	3553	3065	3260
Number of timesteps	19024	11900	19728	14780
Mean time step (s)	0.16	0.30	0.16	0.22
CP consumptions (s)	201636	15278	58625	7000
CP/timestep (s)	10.6	1.28	2.97	0.47
CP/real	68.4	4.3	19.1	2.15
CP/real, reduced	0.71*	0.10**	0.09***	0.010***

* Burroughs CP/real time/96.8
** CDC 180/835 CP/real time/42.2
*** CDC 173 CP/real time/208.1

Reduction according to the speed of the computers as astimated by a matrix inversion test.



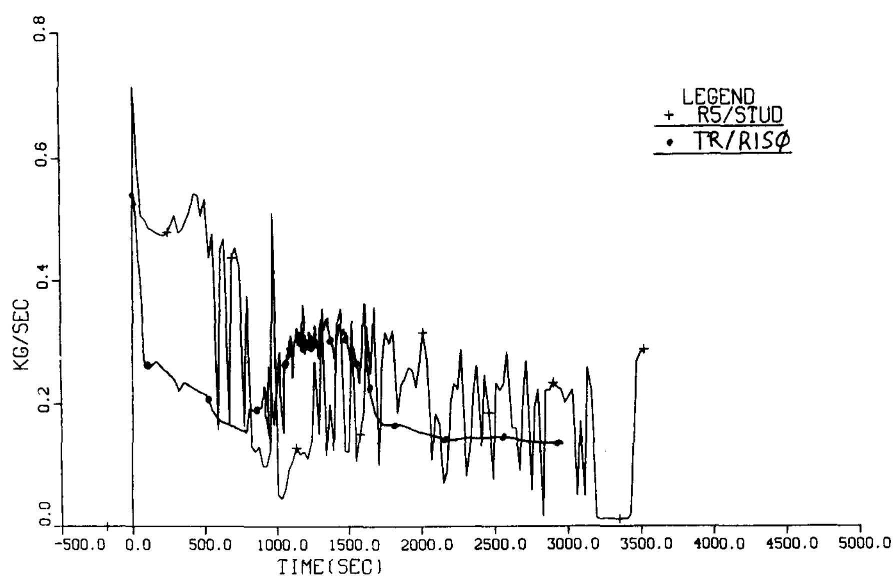


Figure 3.4.1 Break mass flow rate.

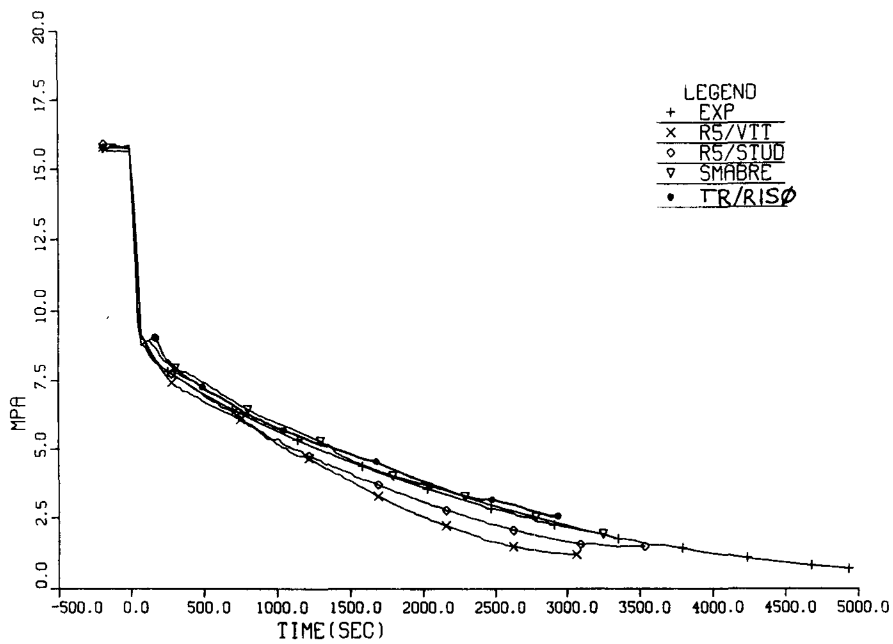


Figure 3.4.2 Pressure in upper plenum.

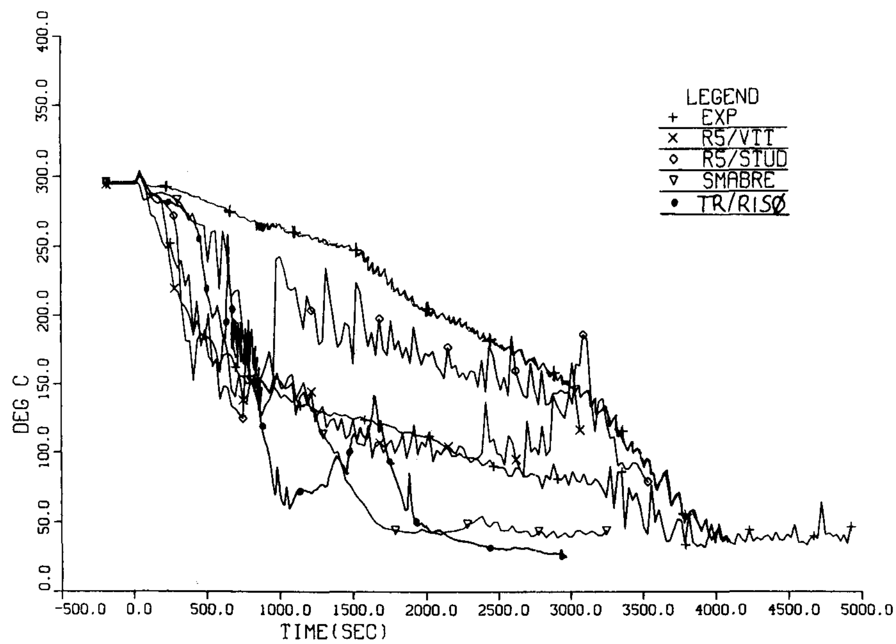


Figure 3.4.3 Fluid temperature intact loop cold leg (vessel inlet). The high experimental value is the steam temperature in the upper part of the pipe, the low value is the water temperature in the lower part.

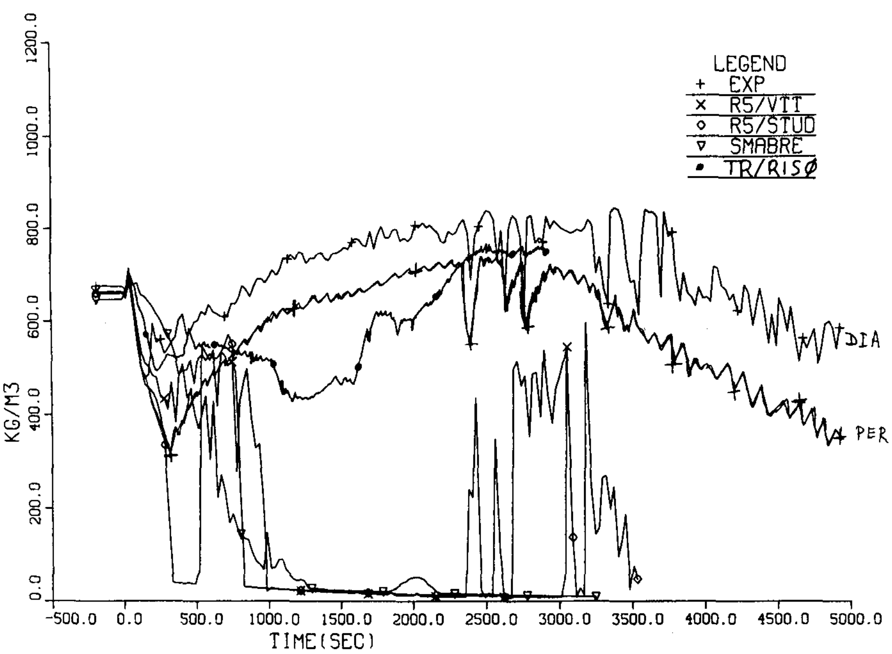


Figure 3.4.4 Mixture density in intact loop hot leg (vessel outlet).

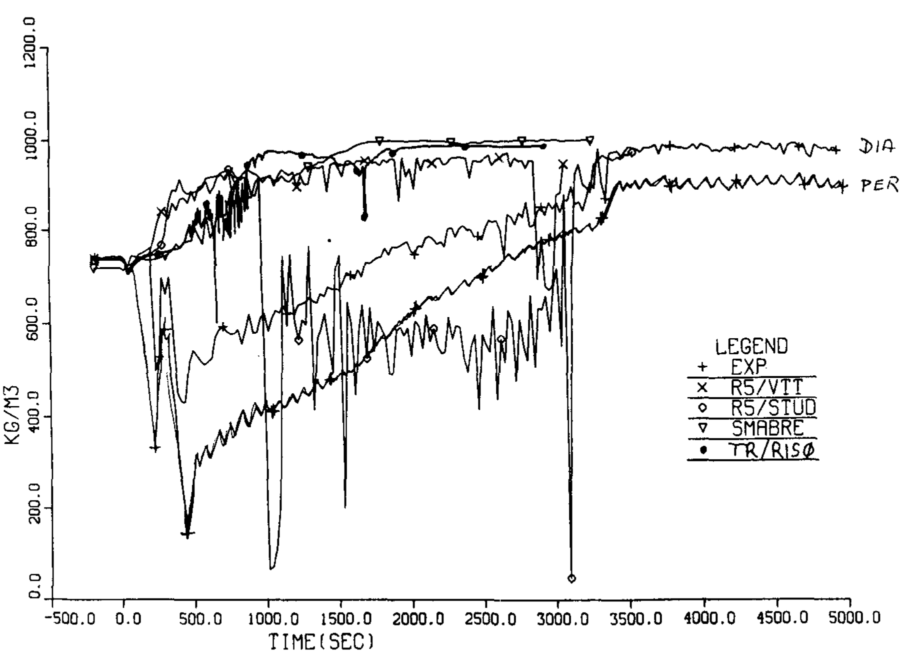


Figure 3.4.5 Mixture density in intact loop cold leg (vessel inlet). The clear difference between the diametrical and the peripheral measurement indicates clearly stratified flow.

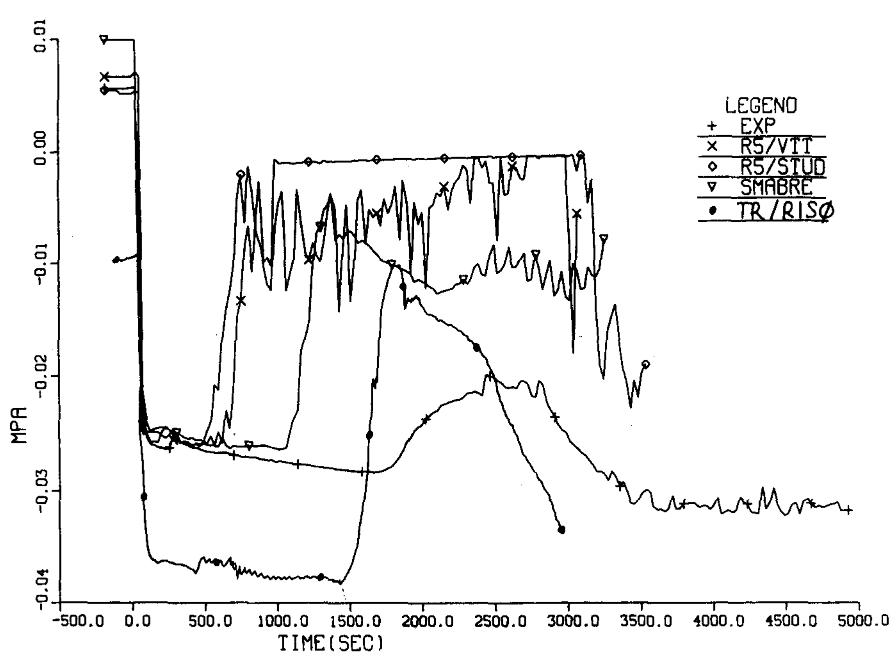


Figure 3.4.6 Differential pressure over descending leg of the loop seal of intact loop.

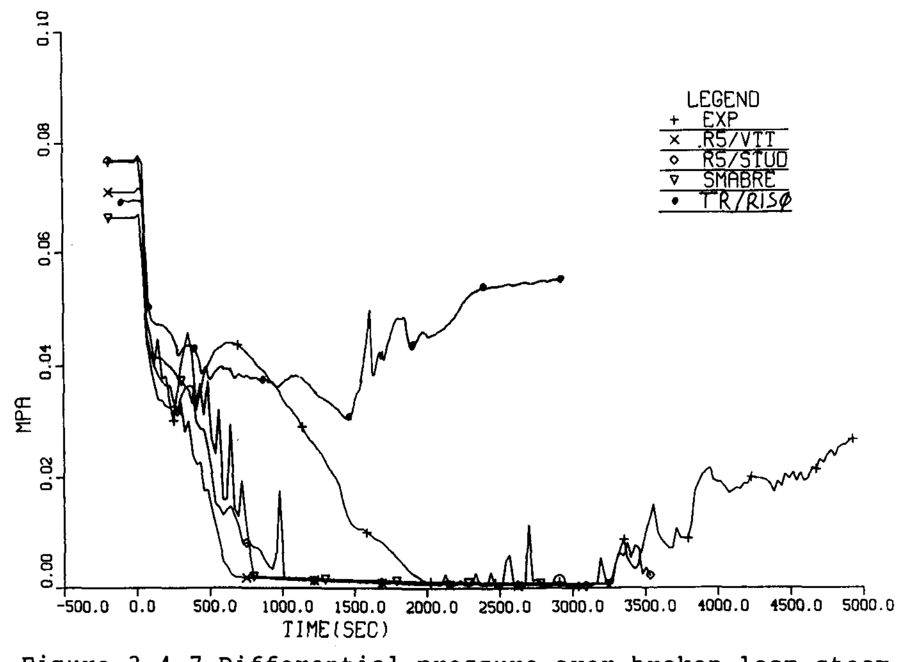


Figure 3.4.7 Differential pressure over broken loop steam generator ascending U-tube leg.

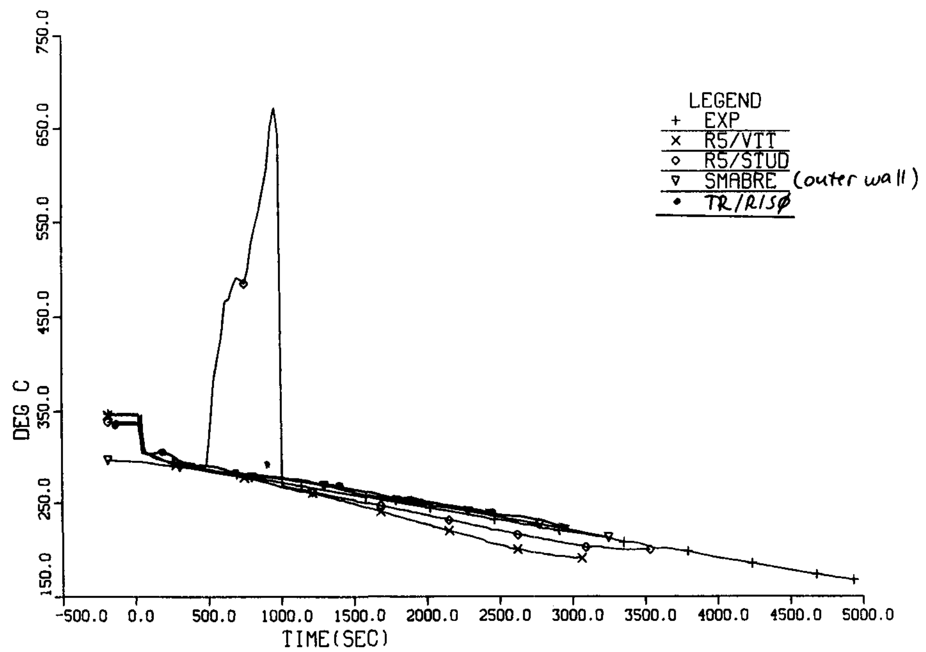


Figure 3.4.8 Rod temperature in middle core region (level 7).

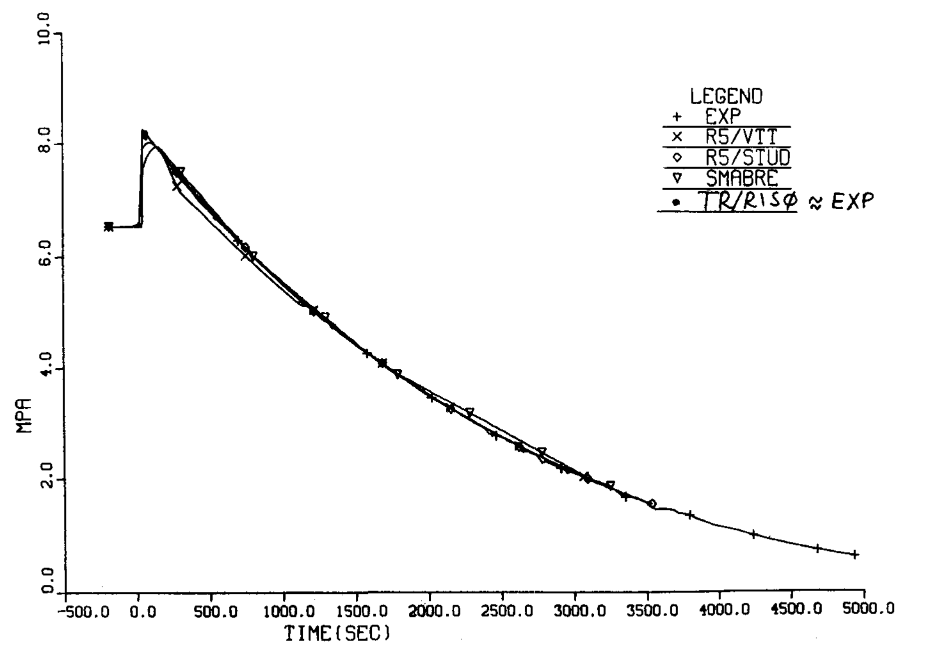


Figure 3.4.9 Secondary pressure in steam dome of intact loop steam generator.

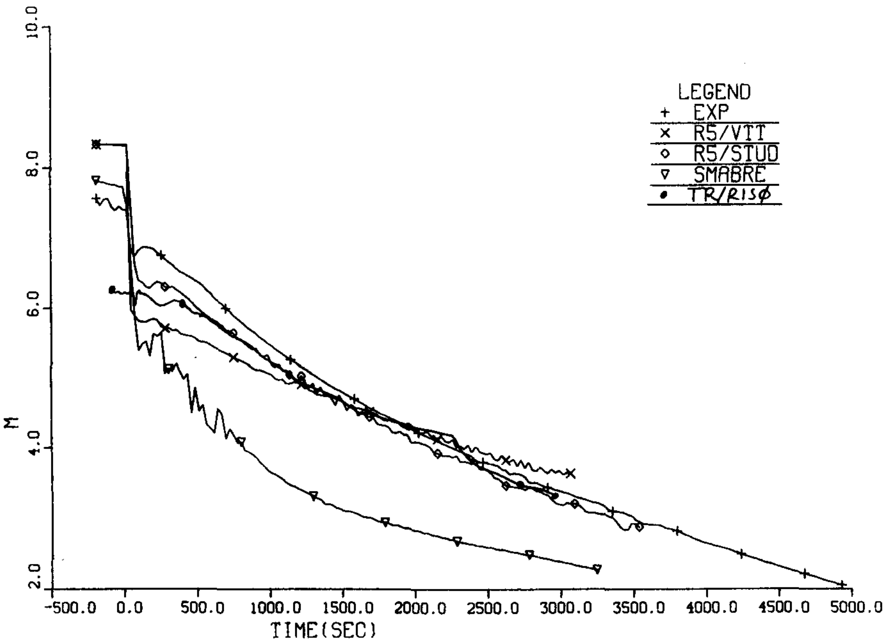


Figure 3.4.10 Collapsed liquid level in downcomer of intact loop steam generator.

4 DISCUSSION AND TECHNICAL CONCLUSIONS

The computer code studies performed theoretically as well as by means of the comparative test case calculations, have given useful, but incomplete, information of the applicability of the studied codes, TRAC/PF1, RELAP5/MOD1, RELAP5/MOD2 and SMABRE, to small-break LOCA analysis.

First, it is observed that SMABRE is not directly comparable with the two others. It is, so to speak, in a class by itself as it is first of all a fast small-break LOCA code and, in order to fulfill this requirement, it is much more simplified than TRAC/PF1 and the RELAP5 codes, disregarding effects considered unimportant to small-break LOCA's.

Input data

A very large amount of input data is needed both for the TRAC and RELAP5 codes, requiring a rather large amount of work, and consequently there are many possibilities for making errors. Due to the smaller degree of detail of SMABRE it is much simpler to prepare input for this code. Comparing the two large codes it is judged that input preparation is easiest for the RELAP5 codes because of their better modelling facilities: multiple hydraulic branching, more flexible coupling of hydraulic channels with heat structures, as well as a more extensive and better formulated trip logic and possibility of defining control variables in terms of state variables by arithmetic expressions or differential equations. It would be desirable, both for the TRAC and RELAP5 codes, to have an interactive input pre-processor to help the user prepare the input data. Such an input pre-processor is known to exist for the German DRUFAN code. It would reduce both the amount of input preparation work and the error possibilities.

Verification

Both the RELAP5 and TRAC codes have been used and verified extensively internationally, although mainly for transients other than small-break LOCA's. SMABRE has been tested only in Finland and of course less extensively.

Speed.

During the test case calculations no significant difference in calculational speed has been observed between TRAC and the RELAP5/MOD1 code, taking into account the different numbers of volumes (calculational cells) used. However, with RELAP5/MOD1 difficulties with excessive computer time have been observed as use of very fine calculational cells may press the calculational time step down to a very low value. Thus, very fine calculational cells may be practically impossible to use in RELAP5/MOD1. In contrast TRAC/PF1 has a more advanced numeric solution method, for which the time step is not Courant limited (i.e., it is not limited to a small value by small calculational cells and/or high fluid velocities). However, RELAP5/MOD2 showed out to be clearly faster that TRAC/PF1 in the single test case where they could be compared. From experience SMABRE is 5 - 10 times faster than the RELAP5 codes - and thus also faster than TRAC/PF1.

Accuracy.

In the investigated test cases the over-all system parameters such as primary and secondary system pressure, fluid temperatures and fuel rod temperatures were predicted reasonably well. This also applies to the question of whether core uncovery occurs or not.

However, regarding the distribution of steam and water in the systems, clear deviations were seen between experimentel results and computer code predictions for most of the test cases, and in a single test case the tested codes failed to predict rewetting of fuel rods due to a non-existing or non-activated fuel rod quenching model.

When judging the accuracy of the code predictions one must destinguish between the two sources of inaccuracy: numerics and physical modelling. Looking first at the mathematical/numerical inaccuracy, it must be admitted that the number of hydraulic control volumes used in this study to represent the test facilities (50-200) give results which are certainly not "converged solutions". This can be seen from the two test case calculations where the effect of a reduced number of control volumes has been investigated, LOFT L3-6 and FIX-II 3031. American studies have indicated that for representation of a reactor system at least 500 control volumes would be necessary to be close to a converged solution.

Physical modelling.

However, by far the largest inacuracy contribution comes from inadequate physical modelling, leading not only to quantitive errors but also to qualitatively wrong predictions of the thermohydraulic behaviour. This applies to all the codes studied. For analysing small-break LOCA's the most serious deficiency of the physical modelling for all of the studied codes is the lack of adequate models for stratified flow, including level gradient driven flow in horizontal pipes as well as low "stratified" heat transfer between steam and water, allowing large liquid subcoolings. This was clearly seen in the relatively poor code predictions of the three pumps-off test cases LOFT L3-5, LOBI SD-SL-03 and LOBI A2-81/ISP18 with pronounced stratification in contrast to the good code predictions of the pumps-on test case LOFT L3-6 with mainly homogenous flow. Furthermore, the Swedish calculation of the LOBI A2-81/ISP18 experiment with RELAP5/MOD2 indicated that a proper modelling of stratified flow may be crucial for the correct prediction of important phenomina such as core uncovery and the resulting dangerous overheating of fuel rods.

In the present case, the result was too pessimistic: a false 400 ^{o}C overheating of the fuel rods (up to about 680 ^{o}C) for about 10 minutes. However, there is no guarantee that under different conditions inadequate representation of stratification may not lead to over-optimistic predictions.

The BWR-test case, FIX-II 3031, showed that without a quenching and radiation model, the prediction of rewetting of fuel rods by spray cooling may fail, resulting in considerable rod overtemperatures. For TRAC/PF1, the quench option was simply unactivated while for RELAP5/MOD1 it was non-existent. In this test case the false overtemperatures were not dangerous (~ 560 ^{o}C), but again, this may depend on the specific conditions. A special purpose computer code for fuel rod temperatures may be used to solve this problem.

The comparative small-break LOCA test case calculations performed have not revealed any clear difference in prediction accuracy between the codes. As small-break LOCA's are normally slow transients, SMABRE with its very simplified modelling, especially the use of a drift-flux model (1 momentum equation) instead of interfacial friction models (2 momentum equations), has not produced significantly poorer predictions than the two other, much more detailed, codes.

Nevertheless, with their two energy equations TRAC and RELAP5/MOD2 have a principally better physical modelling of thermal non-equilibrium than both SMABRE and RELAP5/MOD1 for which the use of a single energy equation prevents simulation of any significant steam superheat in the presence of liquid droplets. In the present analysis no effect of this has been observed, as for the small-break LOCA test cases studied the temperatures were always close to saturation (probably applicable to small-break LOCA's in general). Consequently, the single energy equation approach may be adequate for small-break LOCA's. If, on the other hand, it is also important to be able to analyze BWR emergency core cooling by water spraying, where considerable steam superheat may occur, TRAC/PF1 or RELAP5/MOD2 should be preferred.

The 3-D reactor vessel option of TRAC/PF1 is hardly relevant for small-break LOCA's.

A deficiency observed in the physical modelling of the codes, but which seems to have had a limited effect in the test case calculations, is the necessity of the correlations of both the TRAC/PF1 wall friction and SMABRE drift-flux model to be user specified rather than be determined from the flow pattern. Furthermore, the question still seems to be open, at least for RELAP5/MOD1, as whether the constitutive models go continuously and smoothly from one correlation to another (at flow pattern or regime transitions).

A more general observation concerns the interaction of the physical models and the nodalization in control volumes. For all the codes studied some of the physical models do not work properly with coarse nodalization and they will have to be modified if coarse nodalization is needed (e.g. for computer time reasons).

The preceding review of the study of TRAC/PF1, RELAP5/MOD1, RELAP5/MOD2 and SMABRE for analysis of small-break LOCA's may be summarized as follows:

- The codes have been shown to work reasonably well regarding over-all parameters such as system pressure etc. However, it has been found that the codes fail to predict the distribution of water and steam in the reactor system correctly when stratification occurs,due to inadequate models for stratified flow. Depending on the specific conditions this may lead to overoptimistic as well as overpessimistic predictions (in the present study only the latter was observed). Hence, the codes would be considerably improved if a proper model for stratified flow were implemented so that a best estimate of this effect could be obtained. Furthermore, important for BWR cases, quench models are necessary for correct prediction of rewetting phenomena, and may be in the form of a special purpose computer code for fuel temperatures.
  Also, it would be desirable if the selection of certain correlations (wall friction, drift flux), which are now user specified, could be made automatically from the flow pattern, especially in SMABRE and TRAC/PF1.

Finally, contininuity and smoothness of the constitutive models should be ensured.

- The SMABRE code, due to its simplicity and speed, is judged to be suitable as a supplementary tool to the large systems codes for small-break LOCA analysis, e.g. to perform extensive parametric studies, but not as an alternative to them. The code is well suited as a training simulator for plant operators.

- For analysis of small-break LOCA's with near-thermal equilibrium (near-saturation) conditions and without any considerable three-dimensional flow effects, the RELAP5 codes and TRAC/PF1 have equal physical modelling quality. However, the RELAP5/MOD1 or the RELAP5/MOD2 codes should be preferred over TRAC/PF1 because of their better facilities for representing hydraulic network, heat structures and the control/trip system of a nuclear reactor. Furthermore, RELAP5/MOD2 is clearly faster than TRAC/PF1.

- In cases, where small-break LOCA analysis includes considerable steam superheat in presence of liquid droplets, e.g. during emergency core cooling by liquid top spray in a BWR, TRAC/PF1 and RELAP5/MOD2 are equally accurate, whereas RELAP5/MOD1 cannot be used. However, due to its better facilities for representation of a reactor system and its speed, RELAP5/MOD2 should be preferred.

- Small-break LOCA analysis including three-dimensional flow effects in the reactor vessel requires the TRAC/PF1 code.

5. GENERAL CONCLUSION

Four thermohydraulic computer codes have been studied for small-break LOCA analysis: the fast running Finnish code SMABRE (especially for small-break LOCA's) and the American system codes TRAC-PF1, RELAP5/MOD1 and RELAP5/MOD2. The study has been performed theoretically and by a number of test case calculations, small- and intermediate-break LOCA experiments: LOFT L3-6 and L3-5, LOBI SD-SL-03, LOBI A2-81/ISP18 and FIX-II 3031.

The codes have performed reasonably well regarding the prediction of integral parameters (e.g. system pressure), but they fail in predicting certain details. Thus, in the three test cases (out of five) with low flow situations, the steam/water distribution in the reactor systems in connection with stratified flow in the horizontal pipes was not predicted correctly, and in a single test case fuel rod rewetting was erroneously predicted not to occur due to lacking quench models. No serious effects of this were observed, but under other conditions these deficiencies might lead to wrong predictions of core uncovery and/or fuel rod overheating.

The study has not revealed any significant difference in prediction accuracy between the codes. Consequently, a recommendation about which of the codes to use must rest on user-friendliness, speed and degree of detail:

- In cases with near-thermal equilibrium and mainly one-dimensional flow RELAP5/MOD1 or RELAP5/MOD2 should be preferred.

- In cases where steam superheat may occur (e.g. BWR spray cooling), but with mainly one-dimensional flow, RELAP5/MOD2 should be preferred.

- In cases where significant three-dimensional flow effects may occur TRAC/PF1 should be chosen.

- For extensive parametric studies of cases with near-thermal equilibrium and mainly one-dimensional flow SMABRE should be chosen. SMABRE is also well suited as a training simulator for plant operators.

The treated test cases suggest, according to the experimental data as well as the computer code predictions, that for small-break LOCA's of the investigated type there is no danger of core uncovery or core dryout that could lead to core melt-down if emergency core cooling water were supplied in time (HPIS and accumulator water in PWR's, spray cooling in BWR's). A single test case suggests that even a fraction of the designed emergency core cooling rate may be sufficient (accumulator water only in a PWR).

Finally,it is one of the conclusions of the present study that the computer models for LOCA analysis are complicated and time comsuming (typical computer cost US $ 5000 - 10000 for a single small-break case with the RELAP5 codes or TRAC/PF1). Therefore, trained people to use and maintain them are necessary. In this connection the need to implement improved models for stratified flow should be stressed.

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APPENDIX. EXPERIMENTAL FACILITY DESCRIPTIONS

APPENDIX 1

The LOFT Test Facility.

The LOFT (Loss Of Fluid Test) experimental facility (see fig. A.1) is located at the Idaho National Engineering Laboratory in USA. It has a nuclear core with a nominal power of 55 MWt and it can be thought of as simulating a 3000 MWt PWR. The fluid system volumes of the LOFT facility are in general scaled according to the power scaling ratio. The flow areas, where possible and practical, are scaled with the same ratio. The main components of a commercial PWR such as the pressurizer, steam generator, emergency core cooling system and pumps are includud in the test facility. The aim of the tests performed in the facility is to provide data on the thermal, hydraulic, nuclear, and structural processes expected to occur during a LOCA in a PWR. The LOFT facility is relatively large, thus the ambient heat losses are not so big a problem as in smaller-scale facilities.

The test facility has only one active loop with two parallel pumps and one steam generator. The other, which is called the broken loop, consists of a hot leg and a cold leg both with quick opening valves and connected to a suppression tank. These valves are closed in the small break LOCA experiments. For small-break LOCA's an additional break device has been installed. This consists of a horizontal tube connected to the intact loop cold leg at the location shown in Fig. A.1.

Originally the LOFT facility was intended for large break LOCA experiments and therefore the primary side is more extensively instrumented than the secondary side, because the secondary side has not a very significant role during large break LOCA's.

side depends strongly on the conditions of the secondary side. The lack of the adequate instrumentation may cause problems when comparing the experimental and calculated results. In the experiments the core could be bypassed through three different paths. The first bypass was an annulus space around the core, the second one was the reflood assisst bypass system between the broken loop hot and cold legs, and the third one was the channel between the inlet annulus and the upper plenum.

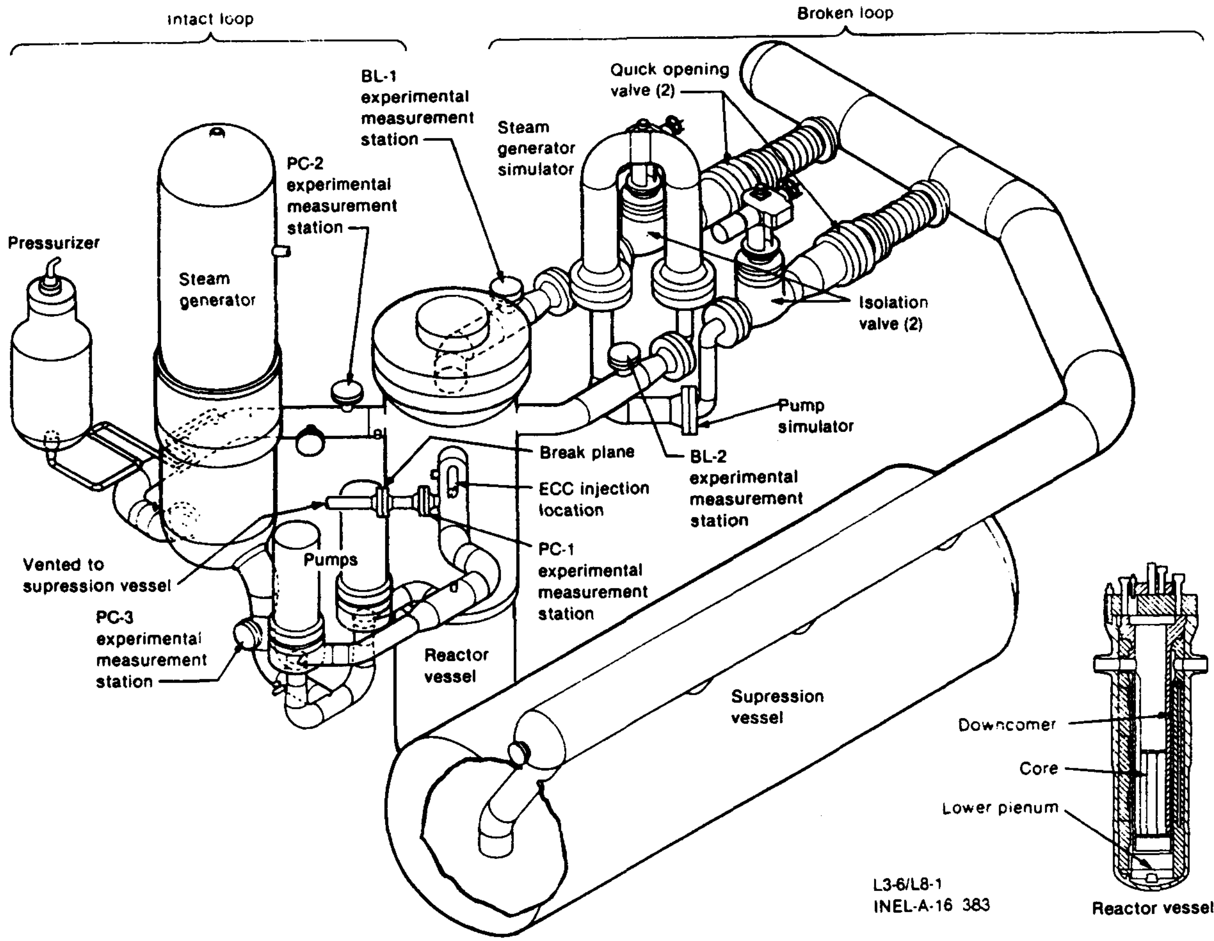

Figure A.1. Axonometric projection of LOFT system.

APPENDIX 2

The LOBI Mod.1 Test Facility

The LOBI test facility, see fig. A.2, is a scaled model of a 1300 MWe pressurized water reactor of KWU design[33]. It is sited at the EEC Joint Research Center Establishment, Ispra, Italy. The volume, the power, the flows etc. are scaled 1:712 but the heights are scaled 1:1 for better modelling of gravitational heads. Also lengths of heat transfer surfaces, differential pressure and temperature distributions are scaled 1:1 and so are the power/volume and the break/coolant volume ratios.

However, compromises have led to distorsions in metal-to-coolant volume ratio, in reduced pipe lengths and in a too large downcomer gap and in scaling of pump characteristics. Filler material adjusting the square core geometry to the circular downcomer also adds to the distortion of the total thermal capacity/ coolant volume ratio.

LOBI includes two active loops with pumps and steam generators: the intact loop, representing three of the four loops of the reference reactor, and the broken loop, representing one reactor loop. The break can be positioned in the hot leg, in the pump surge loop or in the cold leg between pump and reactor vessel. The accumulators, one for each loop, can be connected to the hot and/or the cold legs and the pressurizer can be connected to the hot leg of either loop. The break can have any size up to 2x100%.

To simulate the turbines and the condenser of a real plant, LOBI has two condensers in parallel followed by a cooler and thus a tertiary side where the water is cooled via a cooling tower. The LOBI core consist of 64 full length directly heated nitrogen filled tubes, but the power connections below and above the core accounts for about 13.6% of the total power transferred to the fluid.

The mod.1 configuration used in the SD-SL-03 test was designed and instrumented for large break tests, and consequently care has not been taken to simulate effects which are important only at small breaks. For instance the steam generators cannot be isolated from the condensers (as is the normal power plant procedure in case of a break), there is no thermal insulation of measurement inserts and of the pipework around the possible break positions, and there is no high pressure injections system except for the pump bearing coolant injection.

Furthermore, to maintain the cooling of the electric power connection structures eight holes of five millimeter diameter have been drilled through the upper part of the core barrel tube thus providing a downcomer to upper plenum bypass of considerable size (about 5% of nominal flow at nominal full power steady state conditions). This bypass has the effect that the break size necessary to cause loop seal clearance is larger than for a real plant.

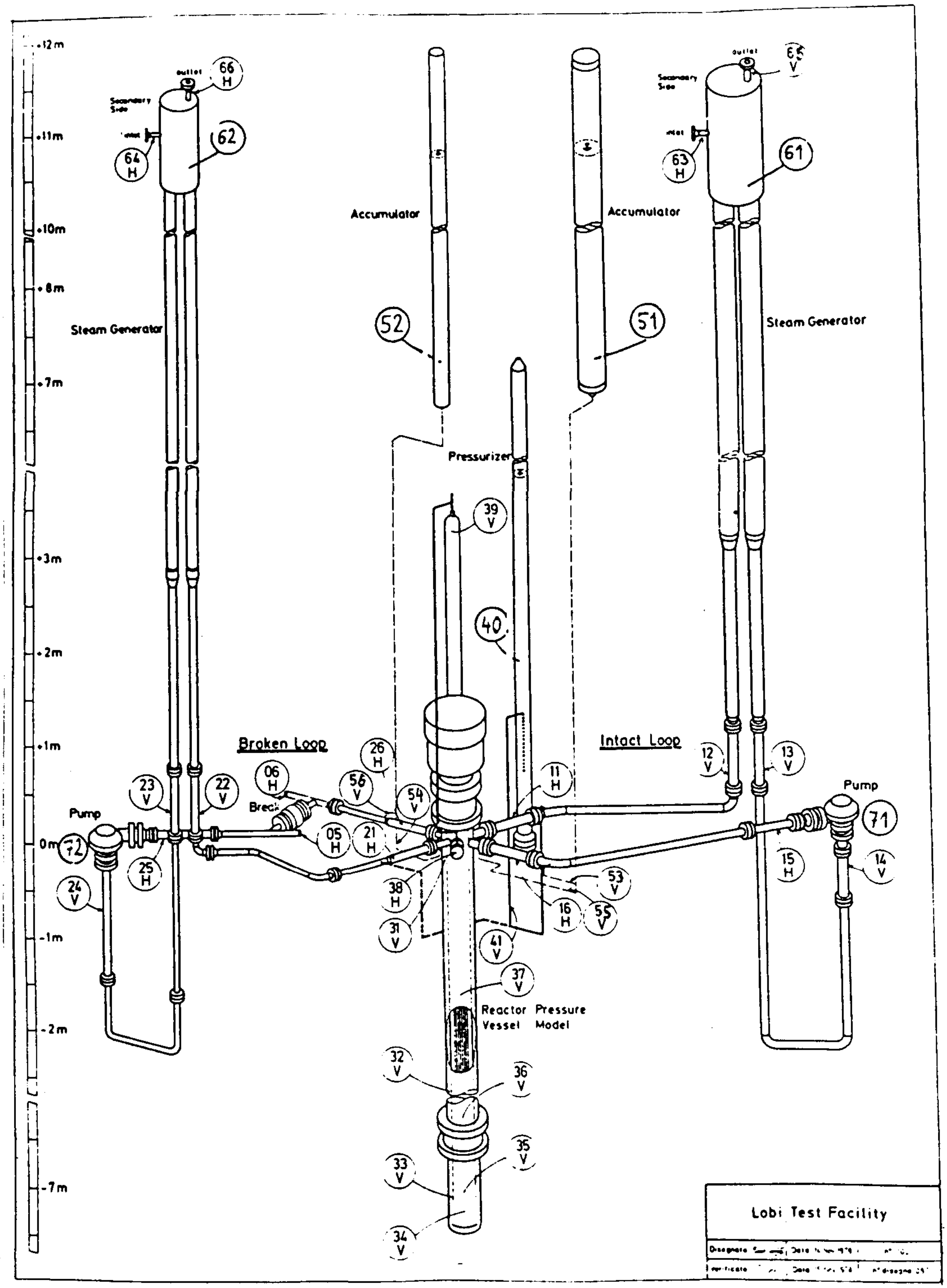

Fig. A.2. : LOBI Test Facility
for Cold Leg Break Configuration,
with Measurement Locations

APPENDIX 3

The LOBI Mod.2 facility

A sketch of the facility is given in fig. A.3. As in mod.1, the intact loop has a size corresponding to three loops of the reference plant while a broken loop represents one loop of the reference plant.

Contrary to mod.1 this configuration of the LOBI facility is especially designed for small break LOCA's and other special transients[14]. As compared to the mod.1 configuration the most important modifications are

- new steam generator with "true" inverted U-tubes, annular downcomer, coarse and fine separators designed especially to meet the requirements of small-break experiments
- addition of auxilliary secondary feedwater system (this was, however, not used in the A2-81/ISP18 experiment)
- installation of high pressure injection system (HPIS)
- modified reactor pressure vessel with a more realistic (smaller) downcomer width of 12 mm
- adaption of instrumentation to the lower measuring ranges in pressure and flow rates relevant for small break LOCA's
- improved thermal insulation and reduction of heat losses associated with pumps and with instrumentation cooling.

Furthermore, the number of the 5 mm Ø bypass-holes between upper plenum and the downcomer top has been reduced to 2. Together with some additional leakages between the hot legs and the downcomer this causes bypass to about 3% of the total recirculation flow in the steady state. In addition, a "locked rotor resistance" valve has been installed after the pump in the broken loop in order to simulate properly the effect of a locked pumpwheel in a main circulation pump of the reference plant.

As indicated in fig. A.3 the accumulators were inactive in the A2-81/ISP18 experiment. The pressurizer was connected to the intact loop hot leg while the high pressure injection (HPIS) was given into the intact loop cold leg.

In the A2-81/ISP18 experiment the break (1%) was a 3 mm hole in the broken loop cold leg. The break line is equipped with a turbine flow meter. However, this was unreliable in the A2-81/ISP18 experiment because two-phase flow developed in the break line.

The horizontal main pipes, the hot and cold legs, are equipped with double beam gamma-densitometers. The difference in the diametrically and the peripherally measured density is an indication of uneven water distribution, i.e. (during small-break LOCA's) a stratified flow. Observations of stratified flow in these pipes may also be obtained from the top- and bottom-mounted thermocouples for fluid temperatures: high top (steam) temperature and low bottom (water) temperature.

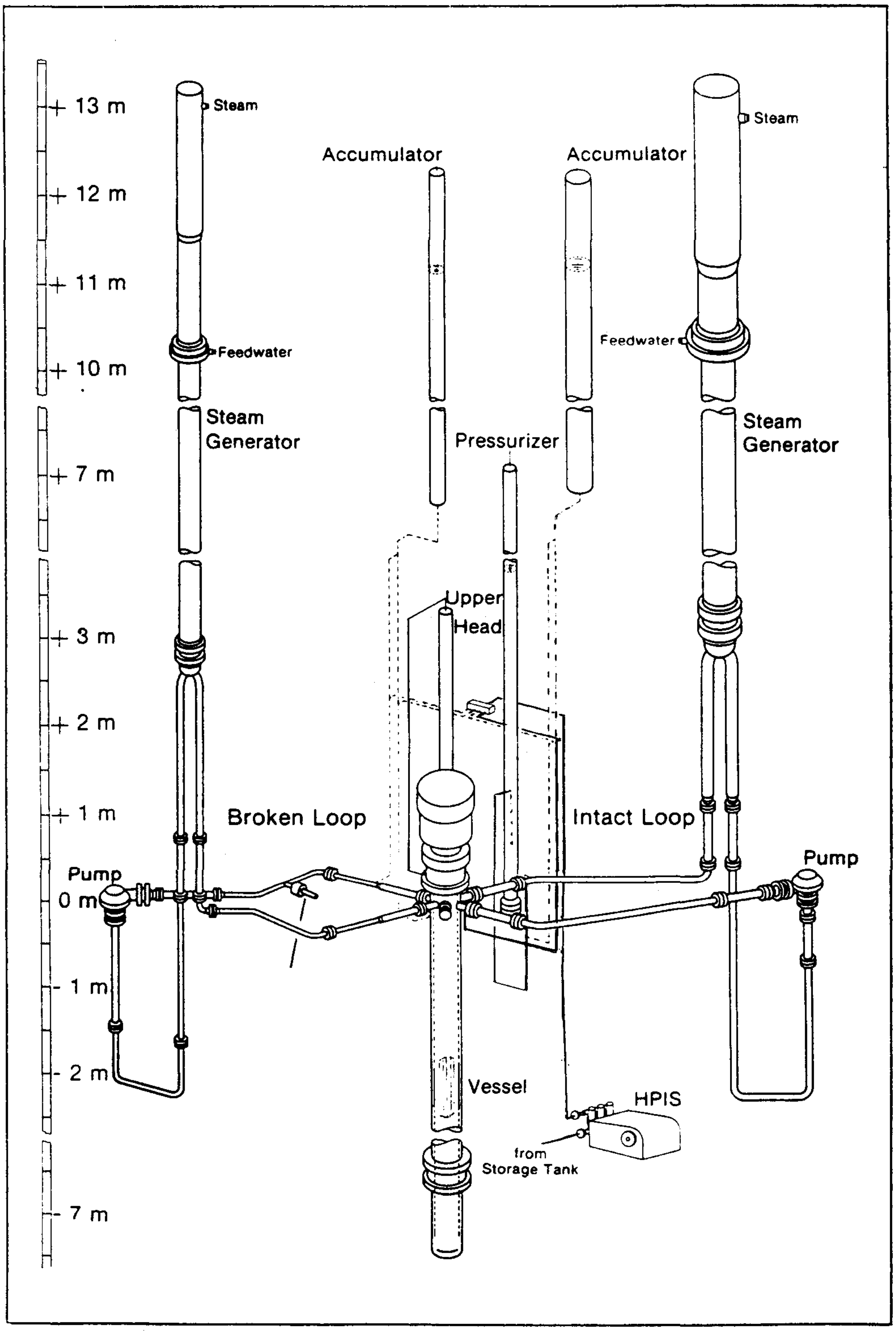

FIG. A.3 LOBI mod.2 Test Facility, primary circuit. Inactive flow lines are shown by dashed lines.

APPENDIX 4

The FIX-II Test Facility.

The purpose of the FIX-II experiments was to provide measurements from simulations of transients following pipe ruptures in BWRs of the ASEA-ATOM external pump design. The FIX-II facility, see fig. A.4, has been built to a 1:777 volume scaling of the Swedish Oskarshamn 2 reactor. However, instead of modelling the external power removal in the full-scale turbines an extra steam condenser volume has been added as an uppermost part of the FIX-II pressure vessel. The condenser water sprayed into that volume is taken from the downcomer and passed through a cooler outside the vessel.

Besides the added steam condenser space also some additional differences in the FIX-II construction compared to that of the reference plant are of significance

- The single test fuel element is equipped with 6x6 directly and electrically heated fuel rod simulators instead of the 8x8 nuclear heated rods in a fuel element of the reference plant.

- The bypass and guide tube volumes are simulated outside the vessel. A separate electrical heating is arranged in that bypass, and the desired part of recirculation flow into the bypass is obtained by an adjustable throttle valve.

- To adapt the single fuel element simulator of square cross section to the circular cross section of the pressure vessel, teflon filler bodies were installed. The surrounding space is cooled by separate water.

- FIX-II has two recirculation lines. One of them is scaled for the line on which the break occurs, the other line represents the remaining three intact recirculation lines of the reference plant and is scaled accordingly.

A description of the FIX-II facility including detailed geometric data is found in Ref. 12. Fig. A.4 shows a general view of the facility with the main constructtion elements involved in the experiment no.3031.

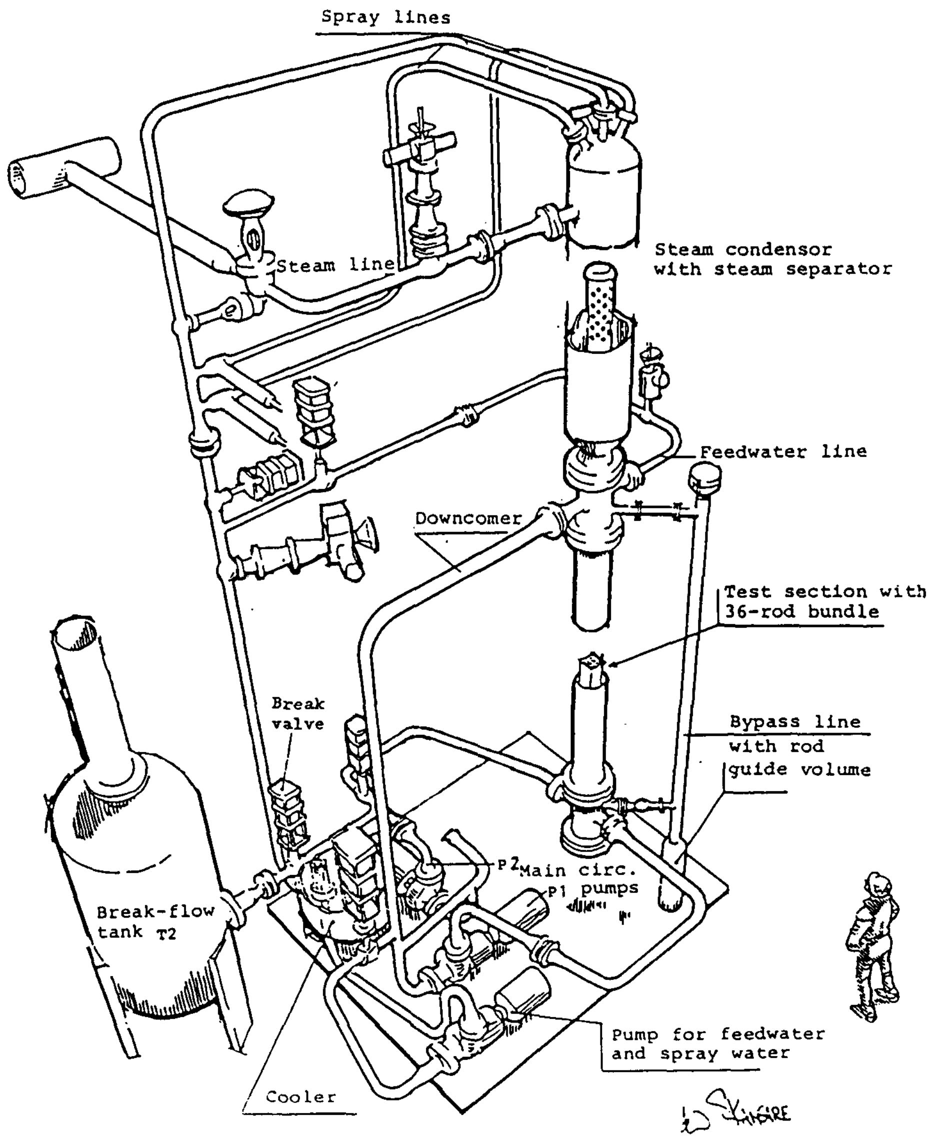

Fig. A.4 FIX-II, general view as equipped for experiment No 3031