NUREG/IA-0128
EGG-EAST-8719

International Agreement Report

International Code Assessment and Applications Program: Summary of Code Assessment Studies Concerning RELAP5/MOD2, RELAP5/MOD3, and TRAC-B

Prepared by
R. R. Schultz

Idaho National Engineering Laboratory
EG & G Idaho, Inc.
Idaho Falls, ID 83415

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

December 1993

Prepared as part of Arrangement on Research Participation and Technical Exchange between the Federal Minister for Research and Technology of the Federal Republic of Germany (BMFT) and the Japan Atomic Energy Research Institute (JAERI) and the United States Nuclear Regulatory Commission (USNRC) in a Coordinated Analytical and Experimental Study of the Thermo-hydraulic Behavior of Emergency Core Coolant during the Refill and Reflood Phase of a Loss-of-Coolant Accident in a Pressurized Water Reactor (the 2D/3D Program).

9401060241 931231
PDR NUREG
IA-0128 R PDR

Published by
U.S. Nuclear Regulatory Commission

Available from

Superintendent of Documents
U.S. Government Printing Office
Mail Stop SSOP
Washington, DC 20402-9328

and

National Technical Information Service
Springfield, VA 22161

NUREG/IA-0128
EGG-EAST-8719

International Agreement Report

International Code Assessment and Applications Program: Summary of Code Assessment Studies Concerning RELAP5/MOD2, RELAP5/MOD3, and TRAC-B

Prepared by
R. R. Schultz

Idaho National Engineering Laboratory
EG & G Idaho, Inc.
Idaho Falls, ID 83415

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

December 1993

Prepared as part of Arrangement on Research Participation and Technical Exchange between the Federal Minister for Research and Technology of the Federal Republic of Germany (BMFT) and the Japan Atomic Energy Research Institute (JAERI) and the United States Nuclear Regulatory Commission (USNRC) in a Coordinated Analytical and Experimental Study of the Thermo-hydraulic Behavior of Emergency Core Coolant during the Refill and Reflood Phase of a Loss-of-Coolant Accident in a Pressurized Water Reactor (the 2D/3D Program).

Published by
U.S. Nuclear Regulatory Commission

ABSTRACT

Members of the International Code Assessment Program (ICAP) have assessed the U.S. Nuclear Regulatory Commission (USNRC) advanced thermal-hydraulic codes over the past few years in a concerted effort to identify deficiencies, to define user guidelines, and to determine the state of each code. The results of sixty-two code assessment reviews, conducted at INEL, are summarized. Code deficiencies are discussed and user recommended nodalizations investigated during the course of conducting the assessment studies and reviews are listed. All the work that is summarized was done using the RELAP5/MOD2, RELAP5/MOD3, and TRAC-B codes.

CONTENTS

Figures

Tables

ACKNOWLEDGEMENT

The author of a summary report is more editor than anything else; particularly if the author has not written any of the reports, or done many of the report reviews, that are being summarized. So the summary report that follows is, for the most part, a compilation, that has been heavily edited, of the work done by the International Code Assessment Program (ICAP) assessment analysts and the INEL ICAP assessment reviewers. Consequently, the author thanks the assessors for providing the reports summarized herein and the people that were so helpful in reviewing the assessor's work.

The authors of the ICAP reports summarized in the following paragraphs can be found by reviewing the reference list. I should note that often I have taken quotes and conclusions directly from the subject author's work. In each instance, I've tried to give credit to the author such that it is clear that their innovative conclusions or work is not mine.

The INEL ICAP assessment reviewers, to which a great deal of credit should be given are: M. A. Bolander, J. D. Burtt, J. C. Cozzuol, J. C. Chapman, W. E. Driskell, C. P. Fineman, C. D. Fletcher, R. G. Hanson, C. M. Kullberg, C. S. Miller, M. G. Ortiz, R. A. Riemke, P. A. Roth, and R. A. Shaw. In particular I'd like to thank P. A. Roth for his work in first producing the summary review sheets and C. S. Miller and R. A. Riemke for giving Code Development's perspective.

In addition, thanks are due to Dr. C. P. Liou at the University of Idaho and Dr. C. J. Choi at the Korea Atomic Energy Research Institute for their assistance and advice in reviewing a number of the ICAP reports.

Finally, a great deal of thanks is due to G. S. Rhee (USNRC-RES) for his support and helpful comments.

SUMMARY

The International Code Assessment Program (ICAP) began in 1985 and is directed to:

- o Support the efforts of the U.S. Nuclear Regulatory Commission (USNRC) to determine the ability of advanced thermal-hydraulic codes to appropriately represent important physical phenomena and support the quantitative determination of the accuracy of these codes;
- o Share user experience on code assessment and to present a well documented assessment data base;
- o Share experience on code errors and inadequacies and cooperate in removing the deficiencies to maintain a single, internationally recognized version of each code; and
- o Establish and improve user guidelines for applying the code.

ICAP members include organizations in Belgium, the European Community (Joint Research Center ISPRA), the Federal Republic of Germany, Finland, France, Italy, Japan, the Netherlands, the Republic of Korea, Spain, Sweden, Switzerland, Taiwan, the United Kingdom, the Union of Soviet Socialist Republics, and the United States. The ICAP administrative activities are handled by the USNRC according to the policies set forth in the Guidelines and Procedures document.

The work summarized describes the ICAP RELAP5 code assessment reviews conducted through 1991 at INEL plus the TRAC-BWR assessment reviews completed during fiscal year 1992. Code assessments of RELAP5/MOD2, RELAP5/MOD3, TRAC-BF1 are summarized.

The sixty-two assessment studies that are summarized identified a number of code deficiencies in all three codes. The deficiencies particular to RELAP5/MOD2 were used as input to improve the MOD2 code and thus produce RELAP5/MOD3. Deficiencies specific to TRAC-BWR and RELAP5/MOD3 have been summarized to serve as input for upcoming code improvement efforts.

1.0 INTRODUCTION

The International Code Applications and Assessment Program (ICAP) began in 1985 and was directed to:

- o Support the efforts of the U.S. Nuclear Regulatory Commission (USNRC) to determine the ability of advanced thermal-hydraulic codes to appropriately represent important physical phenomena and support the quantitative determination of the accuracy of these codes;
- o Share user experience on code assessment and to present a well documented assessment data base;
- o Share experience on code errors and inadequacies and cooperate in removing the deficiencies to maintain a single, internationally recognized code version (for each code); and
- o Establish and improve user guidelines for applying the code.

ICAP members include organizations in Belgium, the European Community (Joint Research Center ISPRA), the Federal Republic of Germany, Finland, France, Italy, Japan, the Netherlands, the Republic of Korea, Russia, Slovenia, Spain, Sweden, Switzerland, Taiwan, the United Kingdom, and the United States. The participating organizations are listed in Table 1.1. The ICAP administrative activities are handled by the USNRC according to the policies set forth in the Guidelines and Procedures document (U.S. Nuclear Regulatory Commission, NUREG-1271, 1987).

As a matter of course, to meet the above objectives, the USNRC has asked that ICAP member code assessment reports be reviewed to (a) obtain a second opinion concerning the validity of each code assessment result, (b) assimilate the code assessment work, (c) create a comprehensive summary of the code deficiencies, and (d) provide input to any ongoing code development and correction effort at the code-source laboratory. As such, the code assessments and reviews have been conducted with care to produce a homogeneous output that will provide information to both code users and developers. The techniques and procedures are outlined in both the Guidelines and Procedures document (U.S. Nuclear Regulatory Commission, 1987) and an April, 1989 USNRC letter to all ICAP members (Rhee, 1989).

Historically the process for producing a reliable version of the code consists of three phases: (i) development, (ii) developmental assessment, and (iii) independent assessment. During the development phase the various models in the code are designed, mounted in the code body, and checked by the various model developers using simple checkout procedures. Following completion of each model and the integration of each model into the code, developmental assessment is undertaken to provide a partial check of the code-generated calculations. Developmental assessment is distinguished from independent assessment by three characteristics: (a) scope limited to select separate-effects and integral-effects analyses defined to reveal major inconsistencies and (b) a primary objective is to remove developmental errors. Finally, following distribution of the code to major code users, independent assessment is undertaken to fully define the code's operational envelope and capabilities. The ICAP assessment task is a portion of the "independent assessment" effort for the major thermal-hydraulic codes: RELAP5/MOD2, RELAP5/MOD3, TRAC-PF1/MOD1, TRAC-PF1/MOD2, and TRAC-BF1.

Table 1.1. ICAP Participating Countries and Codes

Country	Organization	RELAP5	TRAC-PWR	TRAC-BWR	COBRA-TF
Belgium	Tractebel	X			
Finland	Technical Research Center (VTT)	X			
France	Commissariat a l'Energie Atomique (CEA)	X	X		X
Federal Republic of Germany	1. Federal Ministry for Research and Technology 2. Siemens	X	X	X	
JRC-ISPRA	The Joint Research Center ISPRA, Establishment of the European Atomic Energy Community	X			
Italy	Italian Comitato Nazionale per la Ricerca e per lo Sviluppo Dell' Energia Nuclere e Delle Energie Alternative (ENEA)	X	X		
Japan	Japan Atomic Energy Research Institute (JAERI)	X	X	X	X
Korea	Korea Institute of Nuclear Safety (KINS)	X			X
Netherlands	Netherlands Energy Research Foundation	X	X		
Russia	(i) Ministry of Nuclear Industry and Power (ii) I.V. Kurchatov Institute for Atomic Energy	X	X		X
Slovenia	Josef Stefan Institute	X			
Spain	Consejo de Seguridad Nuclear	X	X	X	
Sweden	Swedish Nuclear Power Inspectorate and Studsvik Energiteknik AB	X		X	

Table 1.1. ICAP Participating Countries and Codes (continued)

Country	Organization	RELAP5	TRAC-PWR	TRAC-BWR	COBRA-TF
Switzerland	Paul Scherrer Institute	X		X	X
Taiwan (CCNAA)	Coordinating Council for North American Affairs (CCNAA)	X			
United Kingdom	(i) United Kingdom Atomic Energy Authority Central Electricity Generating Board Nuclear Installations Inspectorate National Nuclear Corporation British Nuclear Fuels Fuels Ltd.	X	X		
United States	U.S. Nuclear Regulatory Commission (USNRC)	X	X	X	X

The work recorded herein summarizes the detailed ICAP code assessment reviews completed by March, 1992 at INEL for RELAP5 and TRAC-BF1. (Note: Quick reviews of code assessment reports are often undertaken; the results of quick reviews are not necessarily reported.) Code assessment report reviews for the RELAP5/MOD2, RELAP5/MOD3, and TRAC-BF1 codes are included in this report. In general, the following guidelines were used to determine whether ICAP code assessment work should be summarized herein (Note: The assessment reports not included in the following guidelines were usually published as NUREG/IA reports.):

1. Once the major RELAP5/MOD2 code deficiencies had been identified and passed as input to the RELAP5/MOD3 developers, the remaining RELAP5/MOD2 reviews were limited to assessment studies that were probably still applicable to the RELAP5/MOD3 code.

2. Prior to early 1991 RELAP5/MOD2 reviews were performed in detail on the assessment work viewed as most likely to provide new insights to the code's use and to reveal code deficiencies.

3. Occasionally differences of opinion existed between the code assessment report authors and the reviewers concerning code deficiencies. If the reviewers concluded insufficient evidence was provided by the report author(s), then the reviewer's opinion was given concerning whether the label "code deficiency" should be assigned to a calculational behavior identified as a code deficiency by the assessment report author. However, usually the author(s)' opinion is given in the individual report summary.

4. The code's ability to calculate a particular phenomena or overall transient behavior has been ranked using the overall headings: excellent, reasonable, or minimal. These terms are defined in Table 1.2. The author has chosen to use these categories because the resulting approach is consistent (even if it is often qualitative), relatively quick to use, and easily understood.

The RELAP5/MOD2 (Ransom, et al., 1985), RELAP5/MOD3 (Carlson, et al., 1990; Fletcher and Schultz, 1992), and TRAC/BWR (Taylor, et al., 1984; Shumway, et al., 1984; Singer, et al., 1984; Shumway, et al., 1985; Weaver, et al., 1986; and Giles, et al., 1992) codes are advanced thermal-hydraulic systems analysis computer codes, developed at the Idaho National Engineering Laboratory (INEL). The principal distinguishing difference between RELAP5/MOD2 and its predecessor RELAP5/MOD1 (Ransom, et al., 1982) is the addition of a two-fluid nonequilibrium and nonhomogeneous hydrodynamic model for transient simulation of two-phase system behavior based on a six-equation two-fluid formulation. The principal distinguishing difference between RELAP5/MOD2 and RELAP5/MOD3 is a change in the interphase drag models and the capability of MOD3 to run on workstations as opposed to mainframe computers. Of course there are a number of other changes, for example the presence of a countercurrent flow limiting (CCFL) model and radiation heat transfer in MOD3 but not in MOD2. But the change in the interphase drag model is the most significant difference. Finally, the TRAC-BWR code series is distinguished from the RELAP5 series by (i) TRAC-BWR's inclusion of models peculiar to boiling water reactors and (ii) its three-dimensional thermal-hydraulic analysis capability.

Although all three codes had been assessed to a degree prior to the ICAP assessments (Ransom, et al., 1987b; Wheatley, et al., 1985; Carlson, et al., 1990; Shumway, et al., 1985), continued code assessment was mandatory to better define and isolate the codes' major deficiencies and capabilities.

The USNRC has made every attempt to produce a "balanced" assessment matrix for each code. Ideally the USNRC would like to have had a completed assessment matrix that matched the Organization of Economic Cooperation and Development's (OECD) Committee on the Safety of Nuclear Installations (CSNI) assessment matrix (see reference: Task Group on the Status and Assessment of Codes for Transients and ECCS, 1987). However, this was not possible. Since the ICAP members could only use data readily available to their organizations, quite often the data specified in the CSNI matrix were not used. The completed code assessment matrices for each of the three codes are described in the following sections.

The assessment studies performed by ICAP members on the three codes are divided into three parts. Part I contains a summary of work applicable to RELAP5/MOD2. Parts II and III contain summaries of work applicable to RELAP5/MOD3 and TRAC-BWR.

The assessments described in Part I were used extensively to produce the code updates used to create RELAP5/MOD3 (Carlson, et al., 1990). Also, the user's guidelines and nodalization studies were used to produce the RELAP5/MOD3 User's Guidelines (Fletcher and Schultz, 1992) even though RELAP5/MOD2 is less advanced and has different models in some cases than RELAP5/MOD3 since the information in Part I and various user guidelines produced by INEL and other users were all the information available.

The assessments described in Part II, mostly completed using the RELAP5/MOD3 Version 5M5 code, have been used in the effort that has produced Version 80. Code deficiencies that still remain in RELAP5/MOD3, that have been identified in Part II have been reported as code problems and hopefully will be corrected in the future.

The assessments described in Part III have identified several code deficiencies and thus are input for any future code maintenance efforts on TRAC-BWR.

Table 1.2. Code assessment comparison descriptors.

Descriptor	Definition
Excellent	An appropriate descriptor when the code exhibits no deficiencies in modeling a given behavior. Major and minor phenomena and trends are predicted correctly. The calculated results are judged by the analyst to be close to the data with which a comparison is being made. If the uncertainty of the data has been identified and made available to the analyst the calculation will, with few exceptions, lie within the uncertainty band of the data. The code may be used with confidence in similar applications. Neither code models nor the facility noding model requires examination or change.
Reasonable	An appropriate descriptor when the code exhibits deficiencies, but the deficiencies are minor; that is, the deficiencies are acceptable because the code provides an acceptable prediction of the test. All major trends and phenomena are predicted correctly. Differences between the test and calculated traces of parameters identified as important by the analyst are greater than those deemed necessary for excellent agreement. If uncertainty data are available, the calculation frequently will lie outside the uncertainty band. However, the analyst believes that the discrepancies are insufficiently large to require a warning to potential users of the code in similar applications. The assessment analyst believes that the correct conclusions about trends and phenomena would be reached if the code were used in similar applications. The code models and/or facility noding model should be reviewed to see whether improvements can be made.
Minimal	An appropriate descriptor when the code exhibits deficiencies and the deficiencies are significant; that is, the deficiencies are such that the code provides a prediction of the test that is only conditionally acceptable. Some major trends or phenomena are not predicted correctly whereas others are predicted correctly. Some RELAP5-calculated values lie far outside the uncertainty band of the data with which a comparison is being made. The assessment analyst believes that incorrect conclusions about trends and phenomena might be reached if the code were used in similar applications. The analyst believes that certain code models and/or the facility noding model must be reviewed, corrections made, and a limited assessment of the revised code or input models made before the code can be used with confidence for similar applications. A warning should be issued to the RELAP5 user community that the user applying the code in similar applications risks drawing incorrect conclusions. This warning should stay in force until the identified review, modification, and limited assessment activities are completed and the resultant characterization descriptor is "reasonable" or better.

PART I: SUMMARY OF RELAP5/MOD2 CODE ASSESSMENTS

2.0 SUMMARY OF ASSESSMENT RESULTS AND APPLICABILITY OF THE CODE

The RELAP5/MOD2 code was "frozen" in January, 1985 at Cycle 36.00 such that only error corrections could be inserted thereafter. The code was "frozen" to allow the world-wide user community to identify code deficiencies on a stable code version that did not change with time as each deficiency was found. Thus, the code was not a "moving target" for thermal-hydraulic modelers and code users.

2.1 HISTORY OF CODE

Following creation of Cycle 36.00, five updated versions were made, i.e., Cycles 36.01 through 36.05. Outstanding error corrections, that may have affected the code assessment process, were corrected prior to completion of each code assessment study. Thus it is believed that the differences between the six versions of RELAP5/MOD2 do not alter the conclusions concerning the code's capabilities, deficiencies, and preferred model nodalizations.

The state of the cycle 36.00 code configuration was evaluated (i) during the developmental assessment phase (Ransom, et al., 1987b) prior to release using thirteen phenomenological problems, twenty-one separate effects analyses, and seven integral experiment analyses, and (ii) soon after release by a code assessment study (Wheatley, et al., 1985) that included five small break loss-of-coolant accident (SBLOCA) analyses, two separate effects calculations, and an operational transient. The assessment studies indicated that the code is generally capable of calculating the phenomena of interest. In particular, all assessment studies predicted the occurrence of major events (with the exception of core heatup during SBLOCAs with core liquid level depression). Event timing was in good agreement with most simulations. However, these early analyses were already indicating shortcomings in the interfacial drag, entrainment, and reflood models of MOD2.

2.2 CODE DEFICIENCIES

The RELAP5/MOD2 code deficiencies, of most concern, are given in the documentation describing the changes required to create RELAP5/MOD3. In particular, RELAP5/MOD2 was found to have shortcomings in the following areas:

1. Counter-current flow limiting (CCFL): MOD2 depended on the code's interphase drag model to simulate CCFL and flooding. In general, the code calculated less liquid downflow than measured in vertical pipes and tubes and thus overcalculated CCFL and flooding. In addition, the code cannot calculate countercurrent flow through geometrically complex passages such as an upper tie plate (Weaver, et al, 1989).

2. Interfacial friction in bubbly/slug flow-regime: The interfacial friction correlation is inappropriate for rod bundles (Analytis and Richner, 1986; Croxford and Hall, 1989; Scriven, 1992a); the resulting high calculated interfacial shear results in underprediction of the collapsed liquid level histories both in low flooding rate reflood and boil-off. Also, researchers have noted that the code does not accurately calculate the axial void fraction profile, and thus level swell, in large vessels (Rosdahl and Caraher, 1986a; Stubbe, 1986).

3. Vapor pull through and liquid entrainment in horizontal pipe offtakes: The current vapor pull-through/liquid entrainment model in MOD2 does not adequately calculate the break mass flow rate when stratified fluid

conditions exist upstream of the eak (Scriven, 1992a; Ardron and Bryce; Hall, 1990).

4. Critical heat flux (CHF): The Biasi correlation (Collier, 1972) is used in the wall heat transfer package to initiate the transition from nucleate boiling to film boiling on a heated surface. The Biasi correlation overpredicts the maximum nucleate boiling heat flux in rod bundles by up to 60% (Weaver, et al., 1989; Sjoberg and Caraher, 1986).

5. Condensation in horizontal pipes: The current MOD2 models do not have the capability to correctly simulate condensation on a subcooled jet, in particular an emergency core cooling system (ECCS) injection jet (Weaver, et al., 1989).

6. Horizontal stratification inception criterion: The current transition criteria from stratified flow to nonstratified is based on the Taitel-Dukler correlation (Taitel and Dukler, 1976) using the vapor phase velocity. However, transition was shown to consistently occur when the stratified flow regime was still present under prototypical reactor conditions (Kukita, et al., 1987).

7. Reflood heat transfer: The MOD2 reflood heat transfer models were shown to be inadequate due to (a) an apparent overprediction of heat transfer rate between superheated steam and saturated water droplets in the dispersed flow regime, (b) a sharp discontinuity between the interphase drag formulations used in the inverted slug flow and slug flow regimes, and (c) an apparent overprediction of the interphase drag force in the high-void inverted slug flow regime. In addition, the code does not include a metal-water reaction model.

8. Critical flow modeling: Several deficiencies have been noted - (a) the saturated steam critical break mass flow rate and the subcooled critical break mass flow rate is overpredicted for nozzle geometries, as used in the Marviken critical flow experiments (Rosdahl and Caraher, 1986b), and require discharge coefficients of approximately 0.82 and 0.85 respectively, (b) nonphysical changes were noted in the computed discharge mass flow rates and were traced to an improperly calculated junction internal energy (Rosdahl and Caraher, 1986b), (c) changes in the discharge coefficient do not produce a linear change in the calculated break mass flow rate at low flow qualities (Rosdahl and Caraher, 1986b) due to a calculational feedback to the throat sonic velocity, (d) critical break flow oscillations were noted for superheated steam flows and were believed caused by sonic velocity oscillations (Stubbe and Vanhoenacker, 1990), (e) changes in the upstream conditions for superheated critical break mass flow affect the calculated mass flow rates more than indicated by the ideal gas law (Stubbe and Vanhoenacker, 1990), (f) modeling the upstream conditions using a time dependent volume gives incorrect initial temperatures of up to 1 K for superheated steam (Stubbe and Vanhoenacker, 1990), and (g) the break mass flow rate is systematically undercalculated, by approximately 30%, for break geometries similar to the LOFT configuration (Hall and Brown, 1990).

9. Inception of vertical stratification: The MOD2 vertical stratification model is triggered inappropriately and causes an incorrect, unphysical change in the fluid interphase drag (Moeyaert and Stubbe, 1988).

2.3 CODE ASSESSMENT MATRIX

The RELAP5/MOD2 code assessment matrices were defined by the ICAP members and are shown in Figs. 2.3.1, 2.3.2, and 2.3.3 for LBLOCAs, SBLOCAs, and operational transients respectively. All three matrices are based simply on the assessment studies performed by ICAP members and were defined using the phenomena of importance for each transient type as listed by the Committee on the Safety of Nuclear Installations (CSNI) of the Task Group on the Status and Assessment of Codes for Transients and Emergency Core Cooling Systems of the Principal Working Group No. 2 on Transients and Breaks for the Organization for Economic Cooperation and Development (see CSNI, 1987). Phenomena represented by data in a particular experimental data set are indicated by cross-referencing the test facility versus the phenomena (see Fig 2.3.1 - Matrix I). If the experimental data set contained good data for a phenomena of interest, then a filled-in circle is shown, for example the break flow data from the Marviken facility is good. If on the other hand, the data is of limited usefulness due to large uncertainty bands or other reasons, then an open circle is shown. Finally, if the data do not contain information on a particular phenomena, then a dash is shown.

Also indicated in the three matrices are correlations between the "test type" and the phenomena as well as the test type versus the test facility systems tests.

2.4 HOW TO USE AND INTERPRET PART I

The remainder of Part I is arranged so a short synopsis of each RELAP5/MOD2 assessment is given in Section 3. However, because forty-eight assessment studies are summarized Section 3 is quite lengthy and thus should be skipped by the reader only interested in the "highlights" of the RELAP5/MOD2 assessments. Section 4 contains a discussion of the various deficiencies that were found in the code from the perspective of doing steady-state calculations, SBLOCA transients, LBLOCA transients, and operational transients. Section 5 discusses the deficiencies from the perspective of doing full-scale plant calculations and Section 6 lists conclusions and observations of the assessment effort.

It is important for the reader to realize that even though code deficiencies were identified, there are a large number of transients that: (i) the code can be used to analyze and (ii) the analyst can expect reasonable calculational results. Of the forty-eight assessments twenty did not identify any deficiencies and thus produced reasonable representations of the transient being analyzed. Of the remaining twenty-eight assessments that contain various code deficiencies, some of the deficiencies are relatively minor. For example, one assessment identified the code's steady-state convergence algorithm (Hyvarinen and Kervinen, 1992) as being deficient. But such a deficiency did not prevent the code from being successfully used to produce a reasonable transient calculation. There are several other similar examples.

User guidelines and discussions of the nodalization studies that were completed by the RELAP5/MOD2 analysts are summarized for each assessment in Section 3. There is not a section that discusses all the user guidelines and nodalization studies because these topics have been addressed from an overall perspective in C. D. Fletcher and R. R. Schultz, *RELAP5/MOD3 Code Manual User's Guidelines*, NUREG/CR-5535, EGG-2596, Volume 5, January, 1992.

Matrix I

CROSS REFERENCE MATRIX FOR LARGE BREAKS IN PWRs

- Phenomena versus test type
  - ● simulated
  - ○ partially simulated
  - – not simulated
- Test facility versus phenomena
  - ● suitable for code assessment
  - ○ limited suitability
  - – not simulated
- Test type versus test facility
  - ● simulated
  - ○ partially simulated
  - – not simulated

		Test Type			System Tests		Sep. Effects				
					1:50	1:1600					
		Blowdown	Reflood	Refill	LOFT	SEMISCALE	MARVIKEN	RIT	NEPTUN	THETIS	CUMULUS
Phenomena	Break flow	●	●	●	○	○	●	–	–	–	●
	Phase separation	○	●	●	●	●	○	–	–	–	–
	Mixing and condensation during injection	○	●	●	○	○	–	–	–	–	–
	Core wise void + flow distribution	○	●	●	○	–	–	–	–	○	–
	ECC bypass and penetration	○	○	●	●	–	–	–	–	–	–
	CCFL (UCSP)	○	●	●	○	–	–	–	–	–	–
	Steam binding (liquid carry over, etc.)	–	●	○	○	○	–	–	–	–	–
	Pool formation in UP	–	●	●	○	○	–	–	–	–	–
	Core heat transfer incl. DNB, dryout, RNB	●	●	●	●	○	–	○	●	–	–
	Quench front propagation	○	●	○	●	○	–	–	○	–	–
	Entrainment (Core, UP)	○	●	○	○	○	–	–	–	–	–
	Deentrainment (Core, UP)	○	●	○	○	○	–	–	–	–	–
	1- and 2-phase pump behavior	●	○	○	○	●	–	–	–	–	–
Test Facility System Tests	LOFT	●	●	●							
	SEMISCALE	●	●	●							

important test parameter
- leak location/leak size
- pumps off/pumps on
- cold leg injection/ combined injection

* volumetric scaling

JEN-MATRIX1C

Figure 2.3.1 Code Assessment Matrix: LBLOCAs.

Matrix II

CROSS REFERENCE MATRIX FOR SMALL AND INTERMEDIATE LEAKS IN PWRs

- Phenomena versus test type
  - ● simulated
  - ○ partially simulated
  - – not simulated
- Test facility versus phenomena
  - ● suitable for code assessment
  - ○ limited suitability
  - – not simulated
- Test type versus test facility
  - ● simulated
  - ○ partially simulated
  - – not simulated

		Test Type							Test Facility: System Tests					Test Facility: Sep. Effects Tests									
		Stationary test addressing energy transportation on primary side	Stationary test addressing energy transp. on sec. side	Small leak overfeed by HPIS, secondary side necessary	Small leak w/o HPIS overfeeding, secondary side necessary	Intermediate leak secondary side not necessary	Pressurizer leak	U-tube rupture	PWR 1:1*	LOFT 1:50	LSTF 1:50	LOBI	SEMISCALE 1:1600	Northwestern	SMOOLIE	MACIASZEK/MENPONTEIL	SHROCK	Anderson/Menponteil	Pressurizer Test	CUMULUS	THETIS	ROYAL Inst. of Tech.	Christensen/EGIN
Phenomena	Natural circulation in 1-phase flow, primary side	●	●	●	○	–	●	●	●	○	●	●	●	–	–	–	–	–	–	–	–	–	–
	Natural circulation in 2-phase flow, primary side	●	–	○	●	●	○	–	–	●	●	●	●	–	–	–	–	–	–	–	–	–	–
	Reflux condenser mode and CCFL	●	–	–	●	●	–	–	–	○	●	○	○	–	–	–	–	–	–	–	–	–	–
	Asymetric loop behavior	–	–	●	●	–	○	●	–	–	○	○	○	–	–	–	–	–	–	–	–	–	–
	Leak flow	–	–	●	●	●	●	●	–	●	●	●	●	–	●	●	●	●	–	●	–	–	–
	Phase separation without mixture level formation	●	–	●	●	●	○	–	○	○	–	●	○	–	–	–	–	–	–	–	–	–	–
	Mixture level and entrainment in vertic. comp. s.g.**	–	●	●	●	●	●	●	–	–	○	○		–	–	–	–	–	–	–	–	–	–
	Mixture level and entrainment in the core	●	–	–	●	●	●	–	–	○	○	○	○	–	–	–	–	–	–	–	●	–	–
	Stratification in horizontal pipes	●	–	–	●	●	–	–	–	●	●	○	○	○	●	●	●	●	–	–	–	–	–
	ECC-mixing and condensation	–	–	○	●	●	●	●	–	○	○	○	○	○	–	–	–	–	–	–	–	–	–
	Loop seal clearance	–	–	–	●	●	○	–	–	●	●	●	●	–	–	–	–	–	–	–	–	–	–
	Pool formation in UP/CCFL(UCSP)	●	–	–	○	●	●	–	–	○	○	–	○	–	–	–	–	–	–	–	–	–	–
	Core wide void and flow distribution	●	–	–	○	●	●	–	–	○	○	–	–	–	–	–	–	–	–	–	●	–	–
	Heat transfer in covered core	●	●	●	●	●	●	●	○	●	○	●	●	–	–	–	–	–	–	–	–	–	○
	Heat transfer in partially uncovered core	●	–	–	○	●	–	–	–	●	○	○	○	–	–	–	–	–	–	–	●	○	○
	Heat transfer in SG primary side	●	○	○	●	●	○	○	–	○	●	●	○	–	–	–	–	–	–	–	–	–	–
	Heat transfer in SG secondary side	○	●	●	●	●	●	●	–	○	●	●	○	–	–	–	–	–	–	–	–	–	–
	Pressurizer thermohydraulics	○	–	○	○	●	●	●	○	○	–	○	–	–	–	–	–	–	●	–	–	–	–
	Surgeline hydraulics	○	–	–	○	●	●	○	–	○	–	○	○	–	–	–	–	–	●	–	–	–	–
	1- and 2- phase pump behavior	–	–	–	○	●	–	–	○	○	–	●	●	–	–	–	–	–	–	–	–	–	–
	Structural heat and heat losses***	●	–	○	●	●	○	○	–	○		○	○	–	–	–	–	–	–	–	–	–	–
	Noncondensible gas effects	●	–	–	–	–	–	–	–	–	–	●	○	–	–	–	–	–	–	–	–	–	–
	Phase separ. in T-junc. and effect on leak flow	–	–	–	●	●	–	–	–	○	–	○	–	–	–	–	–	–	–	–	–	–	–

		Stationary test addressing energy transportation on primary side	Stationary test addressing energy transp. on sec. side	Small leak overfeed by HPIS, secondary side necessary	Small leak w/o HPIS overfeeding, secondary side necessary	Intermediate leak secondary side not necessary	Pressurizer leak	U-tube rupture
Test Facility System Tests	PWR	–	–	○	–	–	●	●
	LOFT	–	–	●	●	●	●	–
	LSTF	●	●	–	●	–	–	–
	LOBI-II	●	●	●	●	●	●	●
	SEMISCALE	○	○	●	●	●	●	●

* volumetric scaling
** secondary side
*** problem for scaled test facilities
**** for intermediate leaks phenomene included in large break reference matrix may be also important

JEN-MATRIX2C

Figure 2.3.2 Code Assessment Matrix: SBLOCAs.

Matrix III

CROSS REFERENCE MATRIX FOR TRANSIENTS IN PWRs

- Phenomena versus test type
  - ● simulated
  - ○ partially simulated
  - \- not simulated
- Test facility versus phenomena
  - ● suitable for code assessment
  - ○ limited suitability
  - \- not simulated
- Test type versus test facility
  - ● simulated
  - ○ partiallly simulated
  - \- not limited

		Test Type											Test Facility: System Tests								Test Facility: Sep. Effects Tests			
		ATWS	Loss of feedwater, non ATWS	Loss of heat sink, non ATWS	Station blackout	Steam line break	Feed line break	Cooldown prim. feed and bleed	Reactivity disturbance	Over-cooling	Reactor trip	Loss of offsite power	Semiscale	LOFT 1:50	Doel 4 PP	KKPL PP	KNU 1 PP	YONG GWANG 2	LOBI-II 1:712	Tihange-2	Doel 4 PP	CUMULUS	CHRISTENSEN	EGEN et al.
Phenomena	Natural circulation in 1-phase flow	●	●	●	●	●	●	●	○	○	●	●	-	●	-	-	○	-	-	-	-	-	-	-
	Natural circulation in 2-phase flow	●	●	●	●	-	-	○	○	-	-	●	-	●	-	-	-	-	-	-	-	-	-	-
	Core thermohydraulics	●	●	●	●	○	○	○	●	○	○	●	○	●	○	○	○	○	-	○	-	-	○	○
	Thermohydraulics on primary side of SG	●	○	○	●	○	○	●	○	●	○	●	○	●	○	○	○	○	○	○	-	-	○	○
	Thermohydraulics on secondary side of SG	●	●	●	●	●	●	●	○	●	○	●	○	-	○	○	○	○	○	○	-	-	-	-
	Pressurizer Thermohydraulics**	●	●	●	●	○	○	○	○	●	○	●	-	-	-	-	-	-	-	-	●	-	-	-
	Surgeline hydraulics (CCFL, choking)**	●	●	●	●	○	○	○	○	○	-	●	-	-	-	-	-	-	-	-	○	-	-	-
	Valve leak flow***	●	●	●	●	●	●	●	●	●	○	●	-	●	-	○	-	-	○	-	-	●	-	-
	1- and 2- phase pump behavior	●	●	●	●	○	○	○	○	○	○	●	○	●	○	○	○	○	○	○	-	-	-	-
	Thermohydraulic-nuclear feedback	●	-	-	-	-	-	-	●	-	-	-	-	●	○	-	-	○	-	-	-	-	-	-
	Structural heat and heat losses****	○	○	○	○	○	○	○	○	○	○	○	○	○	-	-	-	-	-	-	-	-	-	-
	Boron mixing and transport	-	-	-	-	○	-	-	-	○	-	-	-	-	-	-	-	-	-	-	-	-	-	-
	Separator	○	-	-	-	-	-	-	-	-	-	-	○	-	-	-	-	-	-	-	-	-	-	-
Test Facility System Tests	PWR	-	-	-	-	-	-	-	-	○	●	●												
	LOFT	●	●	-	-	-	-	●	●	-	-	-												
	LOBI-II	-	○	-	-	-	-	-	-	-	-	-												
	SEMISCALE	-	-	-	-	○	-	-	-	-	-	-												

* Volumetric scaling

** for phenomena separate effects, pressurizer behavior, refer to small leak cross reference matrix

*** valve flow behavior will be strongly design-dependent, specific experimental data should be used if possible

**** problem for scaled test facilities

JEN-MATRIX4c

Figure 3.2.3 Code Assessement Matrix: Operational Transients.

3.0 SYNOPSES AND EXECUTIVE SUMMARIES OF THE ICAP ASSESSMENTS

The results of forty-eight code assessment studies are summarized. Ten of the studies are separate effects experiment assessments and the remaining thirty-eight of the studies are integral effects experiment assessments. The summaries of the separate effects experiments and the integral effects experiments are described in the following two sections.

3.1 SUMMARY OF ASSESSMENTS BASED ON SEPARATE EFFECTS EXPERIMENTS

Separate effects experiments are usually conducted to explore a particular phenomenon that may occur either independent of other phenomena or in conjuction with other phenomena. Either way, the purpose of separate effects experiments is to look at a piece of an overall picture to reduce the complexity of anticipated scenario behavior.

The ten separate effects assessments are summarized in Table 3.1 and in Sections 3.1.1 through 3.1.10.

Table 3.1 - Code Assessment Reviews: Separate Effects Experiments

Facility	Scale	Experiment	Reference	See Tables
Doel 4 PP	1/1	Pressurizer dynamics	Moeyaert, 1986	3.1.1
Several	NA	Horizontal stratification and critical flow	Ardron, 1988	3.1.2
Marviken	NA	Blowdown and critical flow: JIT 11/CFT 21	Rosdahl, 1986	3.1.3
CUMULUS	1/1	Critical flow	Stubbe, 1988	3.1.4
Marviken	NA	Level swell: JIT 11	Rosdahl, 1986	3.1.5
RIT[a]	NA	CHF and dryout	Sjoberg, 1986	3.1.6
THETIS	NA	Boiloff	Croxford, 1987	3.1.7
NEPTUN	NA	Reflooding	Richner, 1989	3.1.8
Several	NA	Subcooled boiling model	Brain, 1989	3.1.9
Northwestern	NA	Direct contact condensation on horizontal cocurrent stratified flow	Lee, 1991	3.1.10

[a] Royal Institute of Technology, Stockholm, Sweden

3.1.1 Pressurizer Dynamics Assessment Using Doel-4 Plant Data

Reference: P. Moeyaert and E. Stubbe, *Assessment Study of RELAP5/MOD2 Cycle 36.04 Based on Spray Startup Test for Doel-4*, NUREG/IA-0020, July, 1988.

Code version: RELAP5/MOD2 Cycle 36.04.

Facility: DOEL-4, nuclear power plant in Doel, East Flanders, Belgium.

Objectives: Evaluate the code's ability to model pressurizer thermal-hydrau response with pressurizer spray and heaters operational. The objectives were satisfied.

Major phenomena: Thermal nonequilibrium thermal-hydraulics in the pressurizer, the influence of form loss and countercurrent flow in the pressurizer surge line, heat transfer from the pressurizer vessel and heaters to a two-phase mixture, condensation of steam on cold spray liquid, asymmetric primary coolant loop behavior when one or two reactor coolant pumps are tripped.

Code deficiencies: None.

User guidelines:
(1) The transient pressurizer steam dome behavior was accurately represented by only two volumes.
(2) The initial pressurizer liquid level interface should be adjusted to lie near the center of the liquid/vapor interface cell.
(3) In direct contrast to a specific "RELAP5 User Guideline," the pressurizer surge line was connected to a larger than recommended hot leg volume to minimize mass error.
(4) For circumstances in which countercurrent flow may occur in the pressurizer surge line (particularly if the surge line geometry is likely to trigger counter-current flow limiting or flooding), e.g., condensation in the pressurizer steam dome from pressurizer spray coupled with pressurizer inventory vaporization from heater rod operation, the bottom of pressurizer vessel should be nodalized with multiple volumes to permit fluid state stratification.

Base calculation: The code was shown to be capable of accurately calculating the pressurizer thermal-hydraulic response to spray/heater operations by comparing the pressurizer depressurization rate and water level response data to the calculation.

Sensitivity studies: The influence of pressurizer initial water level relative to cell boundaries, the impact of initial spray temperature, and the impact of vessel structure temperatures on transient pressure response were studied. Also, the impact of reactor coolant pump status on pressurizer spray efficiency was examined.

Nodalization studies: Two studies were conducted to study effect of pressurizer dome modeling and pressurizer surge line hot leg connection.

Summary: The report gives the results of an assessment of RELAP5/MOD2 Cycle 36.04 based on a pressurizer spray startup test conducted in the Doel-4 power plant. Doel-4 is a three-loop Westinghouse pressurized water reactor plant with a nominal power rating of 1000 MWe, and equipped with Type E preheater steam generators.

The particular start-up test chosen for this assessment was pressurizer spray and heater test, SU-PR-101. This test investigated the effectiveness of the pressurizer spray to depressurize the plant with pressurizer heaters operational. The test can be considered a kind of separate effects test, principally involving thermal-hydraulic phenomena in the pressurizer and surge line. Although the pressurizer spray and heater systems do not strictly have a safety function, they have a large impact on the operational flexibility of the plant, ie., plant pressure control, especially during operational transients and some small break situations. Hence, a correct simulation of the pressurizer (including spray and heater control, as well as surge line representation) is essential in trying to calculate the plant behavior under off-normal conditions. Test SU-PR-101 provides a basis for evaluating the code's ability to accurately simulate pressurizer response in a full-scale reactor system. Furthermore, with the counteracting effects of having both spray and heaters operations, the test exercises a good number of code features in the simulation including: (i) non-equilibrium thermal-hydraulics in the pressurizer resulting from condensation, by the spray liquid in the steam dome, in conjunction with vaporization by heater operation, (ii) form loss effects and countercurrent flow in the surge line, (iii) heat transfer from the pressurizer shell and heaters to a two-phase liquid. The conduct of Test SU-PR-101 was as follows. Steady state, no load primary conditions were first established. The major heat input into the reactor coolant system was induced by the primary coolant pumps which maintained a rated mass flow and head for both the steady and transient portions of the test (no nuclear power was produced). The primary fluid temperature was held constant at the zero core power reference temperature by dumping steam generator steam to atmosphere. Initial pressurizer heater power was set at 29% of the maximum variable heater power to compensate for heat losses and the cooling effect of residual pressurizer spray flow. The pressurizer initial level was set at 26.3% of maximum. Once steady state was established, the depressurization portion of the transient was initiated by opening both of the available spray line valves, in parallel, from 0. to 100% in about 30 s, while simultaneously ramping pressurizer heater power to 60% of maximum variable heater power in about 4 s. The transient was continued until the pressurizer pressure had dropped 14.7 MPa, at which time both spray valves were closed manually over a period of 25 s. Pressurizer pressure and level data were recorded during the test.

The results of a comparison of Test SU-PR-101 data with the RELAP5/MOD2 Cycle 36.04 baseline calculation of the test indicate that the code is capable of accurately calculating the pressurizer thermal-hydraulic response to spray/heater operation. Figure 3.1.1.1 shows the measured and calculated steam dome pressure response for the transient and exhibits excellent agreement. The difference in depressurization rates between the test and calculation is smaller than 1% which is well within the data measurement uncertainty. Figure 3.1.1.2 shows the measured and calculated pressurizer water level response and again exhibits agreement well within the $\pm 5\%$ uncertainty band for the pressurizer level gauge. The excellent agreement between the measured and calculated depressurization rate for this test is an indication that the overall condensation model, which provides the dominant influence on the pressure response is acceptable.

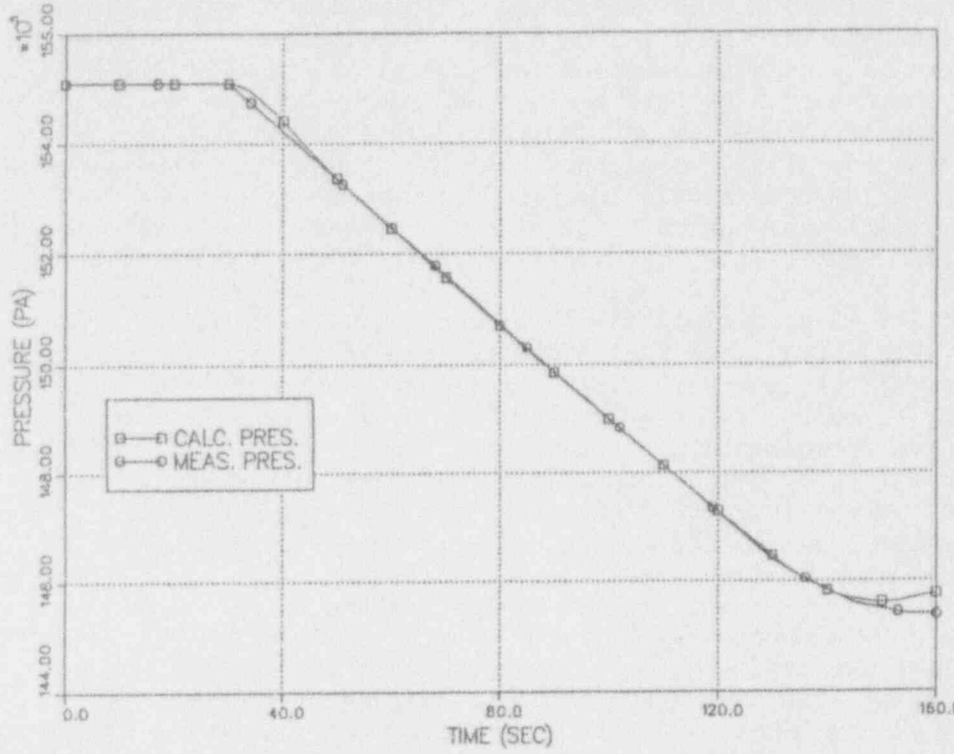


Figure 3.1.1.1 Doel-4 Spray Startup Test: Calculated and Measured Pressurizer Pressure.

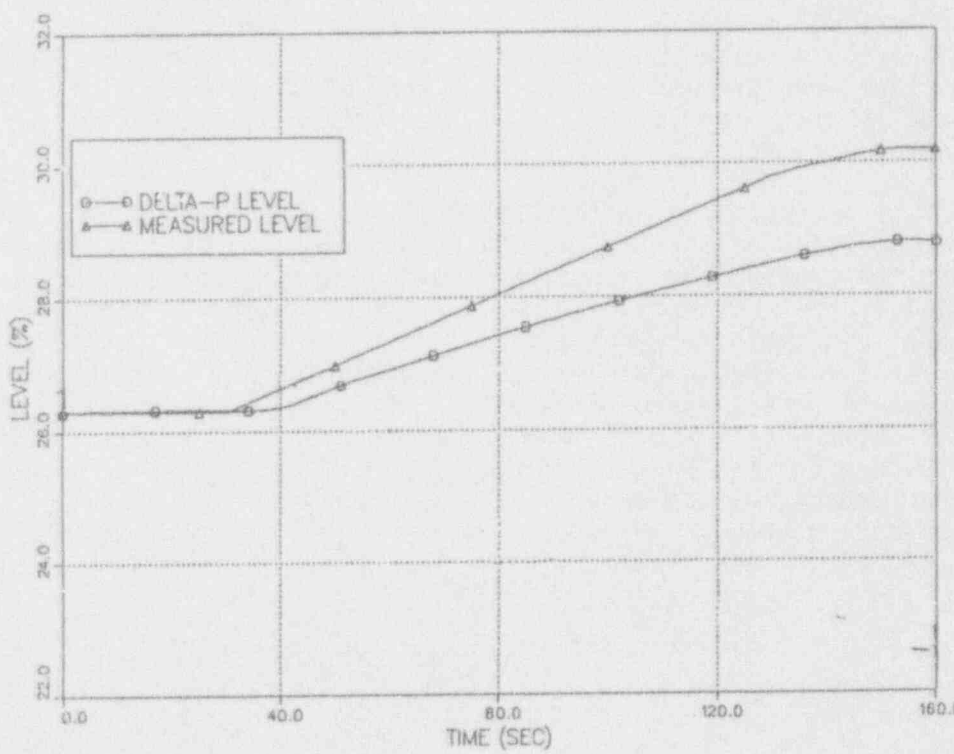


Figure 3.1.1.2 Doel-4 Spray Startup Test: Calculated and Measured Pressurizer Level

3.1.2 Horizontal Stratification Entrainment Model Assessment

Reference: K. H. Ardron and W. M. Bryce, *Assessment of Horizontal Stratification Entrainment Model in RELAP5/MOD2*, NUREG/IA-0039 (to be published).

Code version: RELAP5/MOD2, Cycle 36.04.

Facility:
1. Smoglie data; separate effects experiment at KfK.
2. Maciaszek/Menponteil data; separate effects experiments (CEA).
3. Shrock, et al. data from University of California, Berkeley; separate effects experiments.
4. Anderson/Benedetti data at Idaho National Engineering Laboratory (INEL); separate effects experiments.
5. Loss-of-Fluid Test (LOFT) at INEL.

Objectives: Assess the ability of the code to calculate the off-take branch quality from the horizontal stratification entrainment model.

Major phenomena: Vapor pull-through and liquid entrainment during horizontal stratified flow.

Code deficiencies: The code underpredicts the off-take branch quality when horizontal stratified flow is present.

Impact of deficiencies: Under prediction of the off-take branch quality results in over-predicting the off-take branch mass flow. Thus, this deficiency causes the code to overcalculate the break mass flow that exits the system.

User guidelines: None.

Base calculation: The base calculation was done using a simple model with a stratified flow regime simulated in a 206 mm pipe. The calculation showed a tendency, by the code, to underpredict the discharge flow quality.

Sensitivity studies: An updated version of Cycle 36.04 was created with the horizontal stratified entrainment (HSE) model modified by using correlations based on the facility data listed above. The update included changes to model the (a) critical entrainment depth and the (b) discharge flow quality.

Nodalization studies: None.

Summary: The development of an entrainment model for horizontal stratified flow is documented. The report also compares results of RELAP5/MOD2 Cycle 36.04 calculations implementing that model and RELAP5/MOD2 Cycle 36.04 frozen version to the separate effects data used in the model development and to integral test data. The model determines the off-take branch flow quality from a horizontal main branch when stratified flow is present in the main branch. This model accommodates the orientations of the off-take branch with respect to the horizontal: upward, horizontal, or downward. The authors correlate the critical depth for entrainment to the separate effects data. The critical depth is the liquid level in the main branch at the onset of entrainment. They then used that critical depth in off-take branch quality correlations previously suggested by C. Smoglie and V. E. Shrock, S. T. Revankar, R. Mannheiner, and C. H. Wang.

Accurately modeling entrainment through an off-take branch from a horizontal stratified flow is essential for a best-estimate transient analysis code such as RELAP5. Accurately calculating the break flow will depend, in part, on accurately calculating the entrainment through the break. Similarly, a best-estimate code must accurately calculate entrainment into the surge line for transients that discharge through the pressurizer relief valve. The entrainment model in the RELAP5/MOD2 Cycle 36.04 frozen version is deficient for some applications.

The authors developed the horizontal stratified entrainment (HSE) model using four separate effects data sets (KfK, CEA, UCB, and INEL). These data are from studies of air-water and steam-water, two-phase flows in an off-take branch connected to a larger diameter horizontal pipe. The pressures in the experiments ranged from 0.2 to 6.2 MPa. Steady state stratified flows were established in the main branch with known gas and liquid flow rates and known liquid depth. The quality and the mass flow rates were measured in the off-take branch. The integral test data was from a LOFT small hot leg break LOCA experiment.

The flow qualities determined using the modified RELAP5 version compared well to the separate effects data while the frozen version qualities were generally lower than the data. Fig 3.1.2.1 taken from the report, is an example of the results from the frozen version calculations compared to the separate effects data for a horizontal, centered, off-take branch. The frozen version qualities compared especially poorly to the upward oriented off-take data.

Flow quality is not solved for directly in the frozen RELAP5 code. Instead, the code solves for the phasic junction velocities and the volume void fractions. The off-take junction void fraction is then determined in the HSE model from the volume fractions. The flow quality can then be calculated using the junction quantities by accounting for the ratio of the phasic velocities. In the authors' model, the mass flow rate of the continuous phase in the off-take branch is used to determine the critical depth which in turn is used to calculate the flow quality. The off-take continuous phase mass flow rate is calculated using the junction velocity and junction void fraction. The complex role of the off-take junction phasic velocities in calculating the flow quality must temper the direct comparison of the modified and frozen RELAP5 version flow qualities.

When the suggested entrainment model was implemented, the authors defined a transition region between dispersed and horizontal stratified flows. Within the transition region the off-take branch quality was interpolated between the HSE model flow quality and the normal donored quality for dispersed flow. The authors suggested that further experimental work is needed to help develop models for separation and entrainment in this transition region.

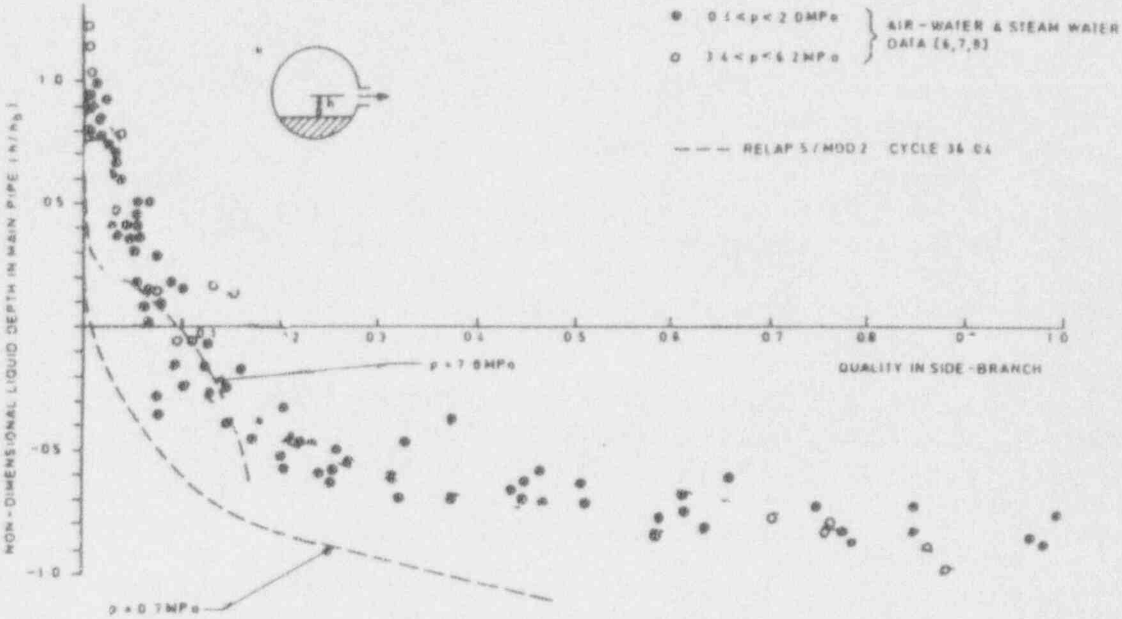


Figure 3.1.2.1 Ardron and Bryce: Calculated and Measured Discharge Flow Quality and Liquid Depth for Horizontal Side Branch.

3.1.3 Critical Flow Assessment Using Marviken Data

Reference: O. Rosdahl and D. Caraher, *Assessment of RELAP5/MOD2 Against Critical Flow Data From Marviken Tests JIT 11 and CFT 21*, NUREG/IA-0007, September, 1986.

Code version: RELAP5/MOD2, Cycle 36.02.

Facility: Marviken at Vikbolandet, Sweden.

Objectives: Evaluate the code's ability to calculate critical flow for saturated steam (JIT 11), subcooled and saturated liquid (CFT 21).

Major phenomena: Critical flow at pipe breaks.

Code deficiencies:
(1) Step increases in the critical flow for saturated steam.
(2) Magnitude of critical flow for saturated steam.
(3) Atypical response in the critical flow for changes in the discharge coefficient.
(4) Critical mass flow for a subcooled liquid.

Impact of deficiencies: Timing and chronology of transient events during a loss-of-coolant accident (LOCA) or LOCA simulation.

User guidelines:
(1) Little benefit is gained in modeling discharge piping having a length-to-diameter ratio (L/D) greater than 4.0 when steam is being discharged.
(2) Short discharge nozzles with L/D < 2.0 should not be explicitly modeled.
(3) Discharge coefficients less than 1.0 may be necessary to attain accurate critical mass flow rates for saturated and subcooled liquid and saturated steam.

Base calculations: The study was done by conducting four calculations to simulate the JIT 11 experiment and eight calculations to simulate the CFT 21 experiment. The calculations were parametric studies varying the time step, the number of model nodes, and the break discharge coefficient.

Sensitivity studies: Calculations were performed to evaluate the effect of varying the break discharge coefficient.

Nodalization studies: The JIT 11 nozzle (L/D = 3.95) was simulated in four cases ranging from the simulation of only the break area to simulating the nozzle using a five cell pipe. The CFT 21 nozzle (L/D = 3.0) was modelled in two cases ranging from the simulation of only the break area to simulating the nozzle with a one cell pipe.

Summary: The Marviken Jet Impingement Test (JIT) 11, yielding saturated steam critical mass flow data, and the Marviken Critical Flow Test (CFT) 21, yielding subcooled and two-phase critical mass flow data, were used to assess the RELAP5/MOD2 Cycle 36.04 code.

The experimental facility consisted of a large vessel 5.2 m in diameter and 22 m high having a total volume of 420 m^3. A discharge pipe containing a valve, a nozzle, rupture discs, and assorted transducers was attached to the bottom of the vessel. For JIT 11 a standpipe, 1 m in diameter and 18 m high, was mounted within the vessel to prevent any liquid from entering the discharge pipe. The nozzle used for the saturated steam flow test (JIT 11) had a diameter of 0.3 m and a length of 1.18 m. The nozzle used for the subcooled critical flow test (CFT 21) had a 0.5 m diameter and was 0.96 m in length.

For all the RELAP5 simulations the experimentally measured fluid conditions in the vessel were used as boundary conditions. This technique allowed the simulations to focus on the flow in the discharge pipe.

The simulations of saturated steam flow overpredicted the experimental discharge flow rate by 20 to 25 percent. Explicitly representing the nozzle region by up to five computational cells had little effect on computed results. It was concluded that, when simulating saturated steam critical flow with RELAP5, a discharge coefficient of about 0.8 needs to be applied. Furthermore, short lengths of pipe (L/D < 4) at the discharge should not be explicitly modeled.

Numerical discontinuities in calculated critical flow rate were found to occur in some of the saturated steam flow simulations. The cause of the discontinuities was traced to an approximation made in the equation used for determining the internal energy at a junction in subroutine JCHOKE.

When simulating CFT 21 RELAP5 was found to overpredict critical flow rates of subcooled liquid by 18 to 20 percent when the nozzle not explicitly included in the RELAP5 model (only its flow area was included). Good agreement with experimental results was attained by using a discharge coefficient of 0.85.

When the nozzle was included in the RELAP5 model RELAP5 underpredicted the measured flow rates. Applying discharge coefficients greater than unity did little to improve computed results but greatly increased computational times. It was concluded that when modeling discharge regions using RELAP5 explicit representation of short lengths of piping near the discharge location should be avoided.

For low quality two phase flow RELAP5 was in good agreement with experimental data when the vessel fluid state (RELAP5 boundary condition) was based upon gamma densitometer measurements. When the fluid state was based upon differential pressure measurements RELAP5 overpredicted the measured flow rate by up to 30 percent. Since the actual fluid state in the vessel probably lies between those used as boundary conditions it was concluded that RELAP5 would generally need a discharge coefficient between 0.8 and 0.95 when used to simulate low quality critical flow.

Application of a discharge coefficient to the RELAP5 simulation of low quality two-phase flow did not achieve an expected result. Using a discharge coefficient of 0.85 instead of 1.0 resulted in only a 8 percent reduction in flow rate rather than the 15 percent expected.

It was discovered that, because of the logic used in subroutine JCHOKE to select between the subcooled and saturated flow calculations and because of an apparent dependency of local equilibrium quality on discharge coefficient, the sonic velocities used in the RELAP5 choking criterion could increase when a discharge coefficient was applied, thus partially offsetting the velocity reduction represented by the discharge coefficient.

3.1.4 Critical Flow Assessment Using CUMULUS SRV Data

Reference: E. J. Stubbe and L. Vanhoenacker, *Assessment Study of RELAP5/MOD2 Cycle 36.04 Based on Pressurizer Safety and Relief Valve Tests*, NUREG/IA-0034, July, 1990.

Code version: RELAP5/MOD2, Cycle 36.04.

Facility: CUMULUS, Electricite de France.

Objectives: Evaluate the code's capability to simulate critical break mass flow thorough safety/relief valves. The objectives were satisfied.

Major phenomena: Critical break mass flow and flow pressure drop.

Code deficiencies:

1. Mismatch between imposed init al temperature (input) and initial input processing for time dependent volume of up to 1 K for superheated steam conditions.
2. Flow rate oscillations for superheated steam conditions (believed to be sonic velocity oscillations by authors).
3. Critical mass flow rate of steam at various degrees of superheat is more sensitive in RELAP5 than indicated by the perfect gas law.

Impact of deficiencies:

1. The mismatch in the imposed-calculated temperature is due to inputting pressure and temperature, whereas the code uses pressure and internal energy. This deficiency can be corrected by adjusting the input appropriately.
2. The flow oscillations are probably due to density changes from volume center to junction location. This deficiency will result in oscillatory system conditions.
3. The sensitivity of the critical flow model to various degrees of superheat probably will not affect most user problems.

User guidelines: The flow area and discharge coefficient should be carefully set to match measurement locations when possible for assessment calculations.

Base calculations: Base calculations were done for six different experiments. The calculations were for steady-state flow conditions through open safety-relief valves.

Sensitivity studies: Sensitivity calculations were run to evaluate the effect and cause of the oscillating choked flow and the model's sensitivity to the degree of superheat.

Nodalization studies: None.

Summary: To qualify pressurizer safety and relief valves planned for use in the Doel plants, the candidate valves (SEBIM - a French assisted safety valve) were tested in the CUMULUS facility. The resulting data were used to perform a RELAP5/MOD2 simulation to establish the feasibility of a lumped valve simulation approach and to evaluate the code's calculation of the main parameters (pressures, flow rates) over a wide range of superheated vapor and subcooled water conditions.

To determine the proper RELAP5 discharge coefficients to model the SEBIM valve flow path, the data from one steam test and one subcooled flow test were used. Subsequent calculations, using the above discharge coefficients with the code, resulted in excellent comparisons between the calculations and the data for one other experiment with choked steam flow and two other experiments with subcooled flow. The data from a fourth experiment, conducted with a 43 K subcooling margin, was matched to within 4%. Although the match was outside the data uncertainty band, the comparison was considered to be reasonable.

3.1.5 Level Swell Assessment Using Marviken Data

Reference: O. Rosdahl and D. Caraher, *Assessment of RELAP5/MOD2 Against Marviken Jet Impingement Test 11 Level Swell*, NUREG/IA-0006, September, 1986.

Code version: RELAP5/MOD2 Cycle 36.02.

Facility: Marviken, Vikbolandet, Sweden.

Objectives: Assess the code's ability to simulate level swell in a large vessel.

Major phenomena: Two-phase level swell, including interfacial drag in the bubbly and slug flow regimes, pool boiling, flow regimes and flow regime transition, and void fraction distribution. Such phenomena are representative of the depressurization behavior of pressurizers, surge tanks, and SGs.

Code deficiencies: Void fraction axial distributions were poorly predicted. Small void fractions were under-predicted, mid-range void fractions were under-predicted, and large void fractions were over-predicted. Other poorly simulated parameters, not listed as deficiencies, were: (a) the flow regime map doesn't model counter-current flow for annular flows, (b) the interfacial friction changes too rapidly near the bubbly-slug flow transition, (c) the mass error algorithm is unreliable, and (d) the relaxation algorithm used in the interphase drag coefficient calculation is sensitive to the time step size during periods of rapidly changing interphase drag.

Impact of deficiencies: Over-predicted void fractions could result in early dryouts during blowdown and the critical flow rates based on conditions near the two-phase interphase wouid be under predicted.

User guidelines: The practices of reducing the time steps and increasing the number of model nodes may not increase the quality of the calculated level swell behavior.

Base calculation: The code predicted the gross behavior of the level swell phenomena well, but did not predict the void fraction axial distribution well. The calculation was done with 20 nodes to model the vessel below the standpipe.

Sensitivity studies: Time step sensitivity calculations were done using the 100 node and 20 node (base) models. Varying the maximum time step from 0.1 to 0.05 s (the material Courant limit was 0.12 s), showed the code to be sensitive to time step size during periods of rapidly changing interphase drag (bubble-to-slug flow).

Nodalization studies: Studies were done using 40 and 100 node models. Results from the 100 node model calculation showed erratic behavior caused by fluctuating calculated axial void fraction profiles.

Summary: The purpose of the simulations was to assess the ability of the RELAP5/MOD2 code to simulate level swell in a large vessel.

The experimental facility consisted of a large vessel 5.2 m in diameter and 22 m high having a total volume of 420 m^3. A standpipe 1 m in diameter and 18 m high was inserted in the vessel. A discharge pipe containing a valve, nozzle, and rupture disks was attached to the lower end of the standpipe at the bottom of the vessel.

The vessel was filled to the 10.2 m elevation with nearly saturated liquid; the remaining part of the vessel and the standpipe were filled with saturated steam. The initial pressure in the vessel was 5.0 MPa.

The test was initiated by breaking the rupture disks. Because of the standpipe configuration, only steam flowed from the vessel. Differential pressures were recorded at various elevations in the vessel, thus allowing a history of fluid density versus elevation to be obtained. Discharge mass flow rate was also measured. The experiment was terminated when the pressure in the vessel reached 1.9 MPa.

After the rupture disks were punctured bulk flashing occurred in the liquid. The level of the resulting two-phase mixture rose rapidly and reached a maximum height of about 18 m - the top of the standpipe - within 15 s. The mixture level declined slowly thereafter receding to near the 14 m elevation by the time the test ended. For elevations below the 13 m height the differential pressure measurements remained fairly constant over the 15 to 80 s time period; indicating that the void fraction was fairly constant.

RELAP5/MOD2 simulations were performed using 20, 40, and 100 nodes to model the annular region in the vessel below the top of the standpipe. The experimental mass flow rate was used as a boundary condition. Differential pressures calculated by RELAP5 were compared to measured data.

The 20 node and the 40 node simulations showed similar results. Both calculations indicated that RELAP5 underpredicted the void fraction of the swelled two-phase mixture for elevations below 9.?8 m and overpredicted the void fraction for higher elevations. The results imply that the interfacial drag forces in RELAP5 fell off too rapidly with increasing void fraction. Consequently RELAP5 carried less liquid to the upper elevations than indicated in the data and RELAP5 allowed the liquid to drain from the upper elevations more quickly than indicated in the data.

The 100 node simulation was characterized by very erratic differential pressure histories which were, for some elevations, much different from the differential pressure histories of the 20 and 40 node cases (see Fig. 3.1.5.1). Moreover, the 100 node simulation was found to be sensitive to time step size - changing the step size from 0.1 s to 0.05 s (material Courant limit = 0.12 s) produced large changes in void fraction profiles. The behavior of the 100 node simulation is believed to be related to the interphase drag model because of its strong dependence on void fraction in the bubble-to-slug transition region; its explicit connection to the numerical solution; and its algorithm for damping large changes in computed values.

Time step studies on the 20 node model revealed that the RELAP5 calculation was sensitive to time step size during the time period from 0. to 30.s when the level swelled to its maximum height. The original calculation was allowed to proceed with a specified maximum time step of 0.5 s. To observe whether the code exhibited time step dependencies, the same problem was performed with maximum time step specifications of 0.1 s and 0.05 s. These cases are shown in Fig. 3.1.5.2. The case with the largest specified maximum time step control corresponded the closest to the data. The cases with smaller specified maximum time steps gave similar results to one another but results quite different from the original calculation for the first 30 s. Hence, the user must exercise caution in using the code's automatic time step algorithm.

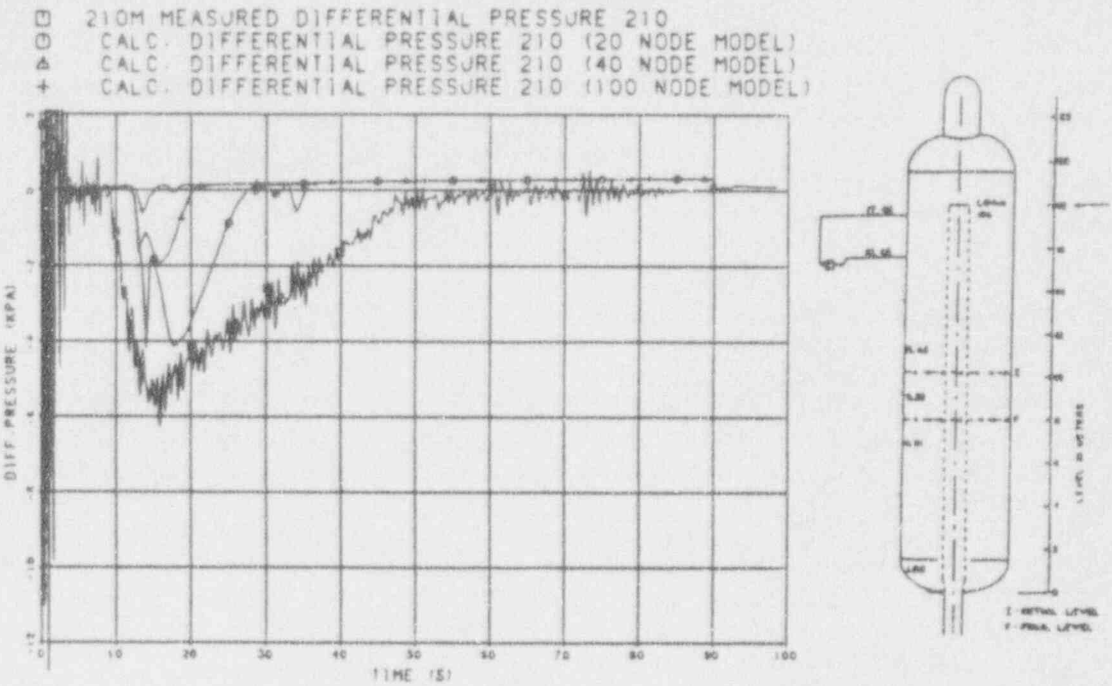


Figure 3.1.5.1 Marviken JIT-11 Experiment: Calculated and Measured Differential Pressure Over 15.45 m to 17.45 m Span.

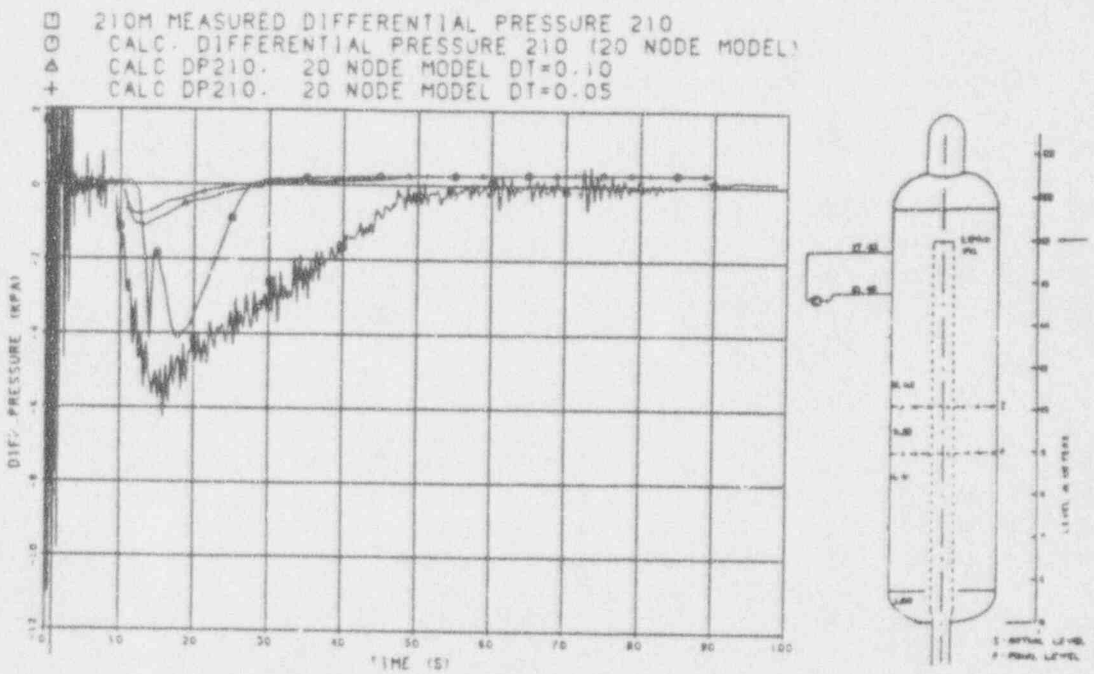


Figure 3.1.5.2 Sensitivity of dp210 to time step size.

3.1.6 CHF Correlations and Dryout Assessment: Royal Institute of Technology

Reference: A. Sjoberg and D. Caraher, *Assessment of RELAP5/MOD2 Against 25 Dryout Experiments Conducted at the Royal Institute of Technology*, NUREG/IA-0009, October, 1986.

Code version: RELAP5/MOD2, Cycle 36.02.

Facility: Separate effects facility at the Royal Institute of Technology, Stockholm, Sweden

Objectives: To assess the code's ability to calculate heater-rod thermal response during core dryout. The objectives were achieved.

Major phenomena: The time and location of CHF are presented in conjunction with axial heater rod temperature distributions.

Code deficiencies:
1. Biasi CHF correlation doesn't adequately predict location of CHF.
2. CHF models in general.

Impact of deficiencies: The location of CHF is not properly calculated. The code under predicted heater rod temperatures at high system pressures (i.e., P> or = to 10 MPa) and over predicted temperatures at lower pressures.

User guidelines: More than two radial nodes in thin-walled heater-rod cladding unnecessary.

Base calculations: Calculations were done for each of the 25 experiments. Calculational deficiencies are as noted above. The deficiencies resulted in large differences between the calculated and measured heater rod temperatures between the point of CHF and 30 cm downstream. However, for locations more than 30 cm downstream the temperatures were accurately calculated.

Sensitivity studies: None. Base code calculations were repeated with modifications to force CHF at the desired location.

Nodalization studies: Two nodalization studies were done. First, the radial heat transfer nodes were increased from two to ten; the computed temperature difference was less than 0.5 K. Second, the axial nodalization was reduced from 47 to 14; unfortunately, conclusions of the axial nodalization study were clouded by the inability of the code to calculate CHF properly.

Summary: Calculations were performed for twenty-five of the post-dryout heat transfer experiments conducted at the Royal Institute of Technology in Stockholm, Sweden. The experimental test section was a 7 m long, 1.5 cm diameter heated tube. Experimental pressures ranged from 3 to 20 MPa; mass fluxes ranged from 500 to 2000 kg/m^2-s; heat fluxes from 10 to 125 W/cm^2; and inlet subcooling from 7 to 13 K.

The RELAP5 model used for the simulations consisted of 47 fluid volumes - fairly coarse noding was used in the lower 3 m of the test section while cell lengths of 10 cm were used above the 3 m elevation. For nearly all experiments being simulated the tube region below 3 m remained in nucleate boiling. Time dependent pressure, temperature, and flow boundary conditions were imposed to simulate the fluid entering the test section. The region downstream of the test section was modelled by a time dependent pressure boundary condition. A constant, axially uniform heat flux was imposed on the tube, replicating the experimentally measured heat flux. An insulated boundary condition was imposed upon the outside edge of the tube wall while the inner edge received its boundary condition - a heat transfer coefficient and temperature sink - from the RELAP5 heat transfer package. Once the RELAP5 simulations reached steady-state the calculated axial temperature distribution along the tube was compared to experimental measurements.

The first series of RELAP5 simulations showed very poor agreement with the experimental data because RELAP5 predicted critical heat flux (CHF) to occur much farther downstream than shown in the data. A subsequent comparison of the Biasi CHF correlation (used in RELAP5) to 177 of the post-dryout experiments showed that the mean difference between the measured and Biasi CHF was -60.8%, the negative sign indicating that the Biasi heat flux was greater than the actual heat flux.

A second series of simulations was conducted using a version of RELAP5 which was updated so that the calculate CHF location corresponded to the measured location. This technique allowed the objectives of the simulations - assessing post-CHF heat transfer - to be achieved.

The forced-CHF simulations showed that RELAP5 accurately simulated the temperature distribution in the region more than 30 cm downstream of the CHF point. In all but one simulation the difference between measured and calculated temperatures in this region was less than 10%. In general, RELAP5 underpredicted the temperature for the higher pressure (P>10 MPa) experiments and overpredicted the temperatures for the lower pressure experiments.

In the region immediately (0 to 30 cm) downstream of the CHF point large differences between measured and calculated temperatures were evident. In this region the axial temperature gradient is very large (200 to 300 K over 10 to 20 cm). The differences between calculated and measured temperatures could, in some cases, be attributed to the discreetness of both the RELAP5 model and the temperature measurements. In other cases the differences were due to transition boiling occurring in the experiment but not being calculated by RELAP5.

Nodalization studies showed that, if CHF were forced to agree with experimental measurements, adequate simulation of the measured temperature distribution could be achieved with a RELAP5 model having only 14 equally sized nodes representing the test section. For this case the node length (0.5 m) corresponded to that typically used in power plant simulations.

Examination of time history plots from several of the RELAP5 simulations revealed that the steady-state convergence algorithm in RELAP5 could probably be relaxed and still yield an acceptable steady-state while reducing running time by up to 40%.

A potential numerical problem with the CHF calculation was discovered during the RELAP5 simulations in which CHF was not forced. In subroutine PREDNB the calculated CHF is altered based upon results from the iteration scheme used to obtain a wall temperature corresponding to CHF. This alteration sometimes induces discontinuities into the CHF calculated at adjacent time steps in the RELAP5 solution.

3.1.7 Boiloff Experiment Assessment: THETIS Experiments

Reference: M. G. Croxford and P. C. Hall, *Analysis of the THETIS Boildown Experiments Using RELAP5/MOD2*, NUREG/IA-0014, July, 1989.

Code version: RELAP5/MOD2, Cycle 36.04.

Facility: THETIS out-of-pile facility at the United Kingdom Atomic Energy Authority (UKAEA) at Winfrith.

Objectives: Evaluate the code's capability to calculate core boiloff rates and fuel thermal response. The objectives were achieved.

Major phenomena: Core boiloff rates, axial vapor fraction profiles, and heater-rod temperatures histories.

Code deficiencies: Interphase drag models.

Impact of deficiencies: Core boiloff rates, two-phase mixture levels, and the calculated thermal response of the core may all be affected.

User guidelines: The same core uncovery and boiloff rates were obtained with cores modeled with 6 and 24 volumes at pressures equal to or above 4 MPa. Thus, only a simple core nodalization is necessary.

Base calculations: The base calculations were made with a 24-volume core for experiments conducted at system pressures of 2, 5, 10, 20, and 40 bars. The calculated and measured two-phase mixture level was in good agreement for pressure equal 40 bars. However, the void fraction below the mixture level was over calculated. Differences between the calculation and the data became progressively greater at lower pressures.

Sensitivity studies: Oscillations observed in the calculations for the 5 bar pressure level experiment was traced to the periodic triggering of the vertical stratification model by forcing the code to bypass the vertical stratification model.

Nodalization studies: The base calculations, conducted with 24-volumes in the core, were repeated with a 6-volume core. Essentially the same core uncovery and boiloff rates were calculated with the coarse nodalization as with the fine nodalization.

Summary: Comparisons of the RELAP5/MOD2 calculations and data giving the mixture level, void fraction distribution, and exposed rod heat-up rates obtained in the THETIS boildown experiments are given.

The THETIS facility consisted of a vertical bundle of electrically-heated rods enclosed in a 130.6 mm inner diameter circular shroud tube placed inside a vertical cylindrical pressure vessel. The shroud tube was closed at the bottom but open at the top. Systems were provided to supply a constant measured flow rate of make-up water to the bottom of the test section and to maintain the rig pressure at a pre-selected value. The pin bundle consisted of 61 pins of 12.2 mm outer diameter. Fifty-seven of the pins were electrically heated fuel pin simulators with a heated length of 3.6 m. A chopped cosine axial power profile was used in the fuel pin simulators.

The experiments were conducted by setting the test section power to 100 kW and by achieving an equilibrium condition where the system pressure was constant and the test section material reached an unchanging temperature profile. In the equilibrium state, sufficient make-up flow was provided to compensate for the liquid boil-off rate so that the entire heated length of the rod bundle was wetted. Thereafter the data acquisition equipment was started and the transient was initiated by reducing the makeup flow to zero. The bundle liquid level was allowed to boil off. Tests were conducted at pressures of 2, 5, 10, 20, and 40 bars.

The RELAP5/MOD2 code was shown to give reasonable agreement with the high pressure core boildown data. Specifically:

1. When a fine node (24 axial volumes) nodalization was used to model the THETIS rod bundle, excellent agreement was obtained with mixture level boildown rates at test section pressures of 20 and 40 bars. However, at pressures less than 10 bars the boildown rates were considerably overpredicted.

2. RELAP5/MOD2 has a tendency to overpredict the void fraction below the two-phase mixture level with errors increasing with decreasing pressure.

3. Calculations performed with a coarse nodalization (six axial cells) of the rod bundle, typical of that used in simulating plant analyses, showed reasonable agreement with the data at pressures above 20 bars. However, oscillations were encountered in the simulation of the steady-state condition prior to boildown. These oscillations were found to be due to the periodic triggering of the RELAP5/MOD2 vertical stratification model.

4. The code gave a reasonable calculation of the heat-up of exposed rods above the two-phase mixture level.

5. RELAP5/MOD2 showed a decided improvement over RELAP5/MOD1 in simulating the THETIS experiments. In particular, accuracy, stability, running time, and mass error were much improved.

3.1.8 Reflood Simulation Assessment: NEPTUN Experiments

Reference: M. Richner, G. Th. Analytis, S. N. Aksan, *Assessment of RELAP5/MOD2, Cycle 36.02, Using NEPTUN Reflooding Experimental Data*, NUREG/IA-0054, (to be published).

Code version: RELAP5/MOD2 Cycle 36.02.

Facility: NEPTUN and FLECHT-SEASET experimental facilities.

Objectives: The perceived objective for the assessment the evaluation of the overall code capability to perform reflood type calculations.

Major phenomena: Reflood core heat transfer, the influence of interphase friction correlations on the calculation of the reflood rate and heat transfer.

Code deficiencies: Interphase friction and wall heat transfer were the two code areas primarily responsible for large deviations between predictions and measurements for the low flooding rate experiments.

User guidelines: For a reactor core 15-20 volumes are recommended to represent the core region.

Base calculation: The base nodalization was used for a range flooding rates constituting a set of base calculations. The code was shown to predict the reflood response best for the moderate and high flooding rates and not in agreement with experimental data for the case of low reflood rates.

Sensitivity studies: Sensitivity studies were performed to investigate the inclusion of a modified Bromley correlation for film boiling heat transfer, inclusion of the CATHARE correlation for interphase friction in bubbly and slug flow, and for the use of the Forslund-Rohsenow film boiling heat transfer correlation for void fractions greater than 0.8.

Nodalization studies: Nodalization studies were performed to determine the appropriate number of core nodes. Calculations were performed using 10, 18, and 32 volumes to represent the core.

Summary: The assessment calculations described in the above report address the code's capability to simulate reflood phenomena using data generated in the NEPTUN facility and the FLECHT-SEASET facility.

The NEPTUN facility was constructed to study reflooding in bundle geometries. The test section heater rod bundle consisted of 33 electrically-heated rods and four guide-tubes. The outer dimensions of the NEPTUN heater rods are similar to those of typical pressurized water reactor fuel rods, but the NEPTUN rods are only half the length. The NEPTUN rods are 1.68 m in length and are 10.7 mm in diameter with a pitch to diameter ratio of 1.33. The heater rod axial power profile is a chopped cosine shape with a maximum peaking factor of 1.58.

Forty reflood experiments were conducted in the NEPTUN facility, seven of which have been chosen for the assessment study. The test procedure for each experiment consisted of three steps: (i) The experiment liquid inventory is circulated until the desired condition is achieved and the test section is filled with saturated steam at the desired pressure. (ii) The heater rod power is switched on and the heater rod temperatures are allowed to increase. (iii) Shortly before the heater rod temperatures reach at predetermined level, the reflood water valve is opened and water is allowed to enter the test section.

Analysis was begun using a model with 18 cells representing the core. Other nodalizations were studied to define a base case. Specifically, core nodalizations with 10, 18, and 32 cells were studied. Heat structures were defined to have 18 fine mesh points. On the basis of these analyses it was seen that the 18 cell nodalization provided convergent results. In particular, the 18 cell nodalization model results differed little from the results obtained using the 32 cell nodalization for the high reflooding rate experiment simulations.

3.1.9 Subcooled Boiling Model Assessment

Reference: C. R. Brain, *Assessment of the Subcooled Boiling Model Used in RELAP5/MOD2 (Cycle 36.05, Version E03) Against Experimental Data*, NUREG/IA-0056, March, 1992.

Code version: RELAP5/MOD2 Cycle 36.02, Winfrith Version E03.

Facility: Two vertical tube experiments conducted by (i) Christensen and (ii) Egen, Dingee, and Chastain

Objectives: The assessment was performed to determine the code's capabilities for predicting subcooled nucleate boiling under high-pressure, high-heat-flux conditions similar to those expected during a PWR ATWS.

Major phenomena: Fluid void fraction distributions in separate-effects heated-channel experiments.

Code deficiencies: The code tended to predict onset of fluid voiding lower in the channel than measured for experiments with high inlet subcooling.

Users guidelines: None.

Base calculation: The base-case calculations were performed using RELAP5/MOD2/36.05, Winfrith Version E03. The code error corrections and improvements beyond the RELAP5/MOD2, Version 36.05 INEL released code have not been documented.

Sensitivity studies: None.

Nodalization studies: None.

Summary: During an anticipated transient without scram (ATWS) accident in a pressurized water reactor, the core power is not tripped and the resulting heat-load mismatch causes the primary coolant system to be significantly overpressurized. The peak pressure attained will be sensitive to the volume of vapor produced within the primary coolant system as a result of subcooled nucleate boiling phenomena.

For the purpose of determining the capabilities of RELAP5/MOD2 to simulate subcooled nucleate boiling phenomena at high pressures, the code was assessed against data from two series of high-pressure, steady-state, subcooled boiling experiments. Both experiments generated single rectangular heated-tube data for vertically upward flowing water. The Christensen data involved pressures up to 6.9 MPa in a 127 cm long, 1.11 cm by 4.44 cm channel. The Egen, Dingee, and Chastain data are at a pressure of 13.79 MPa in a 68.6 cm long, 2.5 cm by 0.26 cm channel. The channels were electrically-heated. The inlet fluid temperatures and flow rates, and the test section pressures, were measured and controlled. The primary test output data are the tube fluid void fractions at various elevations, measured using gamma densitometers.

The experiments were modeled with the RELAP5 code using pipe components to represent the test section; 20 cells were used for the Christensen tests and 27 cells were used for the Egen, Dingee, and Chastain tests. Fluid conditions at the inlet were specified using a time dependent volume and the inlet flow boundary condition was set using a time dependent junction component. The pressure boundary condition was set at the test section exit by a second time dependent volume connected to the test section via a single junction component.

The RELAP5-calculated volume void fractions in each of the cells were plotted against test section height for seven of the Christensen tests and five of the Egen, Dingee, and Chastain tests for a variety of inlet mass flow rates and fluid subcoolings. Comparison is made with experimental data giving the measured void fraction at selected points along the tube. An example comparison (for Christensen Run 16) is shown in Fig. 3.1.9.1. This test was performed at a pressure of 6.89 MPa, a power of 70 kW, an inlet mass flux of 808 kg/m_2-s, and an inlet subcooling of 12.1 K. The anomaly in the calculated data apparent at the channel exit is caused by the averaging technique at the model's outlet junction and does not represent an actual deviation from the test data.

For cases of low subcooling (all the Christensen runs and Egen Runs 13 and 19), the agreement between experiment and the calculation is reasonable. The code adequately predicts both the slope and magnitude of the experimental data. Where differences were found, they were generally found to be due to inadequacies of the experimental data, not due to any code deficiency. However, for cases of high subcooling (Egen Runs 7, 15, and 16), the code was shown to overpredict the void fraction near the test section inlet. Because the heat addition process was continuous along the tube length, the overprediction of void at the test section inlet resulted in a general overprediction of void at all test section locations. This disagreement is shown in Fig. 3.1.9.2. for Egin Run 16. This test was performed at a pressure of 13.79 MPa, a power of 60 kW, an inlet mass flux of 1153 kg/m^2-s, and an inlet subcooling of 74.7 K.

To evaluate the effect that the code deficiency in void prediction at high-subcooling would have for an ATWS event, the code-calculated channel-average void fraction was compared with the upper and lower bounds of the experimental data for each of the tests. This comparison indicated that RELAP5 systematically overpredicted the channel-average void fraction. However, in five of the twelve cases the calculation fell within the experimental uncertainty band, and in three other cases the calculation fell only slightly outside it. In the worst case a void fraction error of 6% was indicated. This agreement was considered reasonable, and the study concluded that the code may be used with reasonable confidence to calculate the subcooled nucleate boiling void fraction during PWR ATWS sequences.

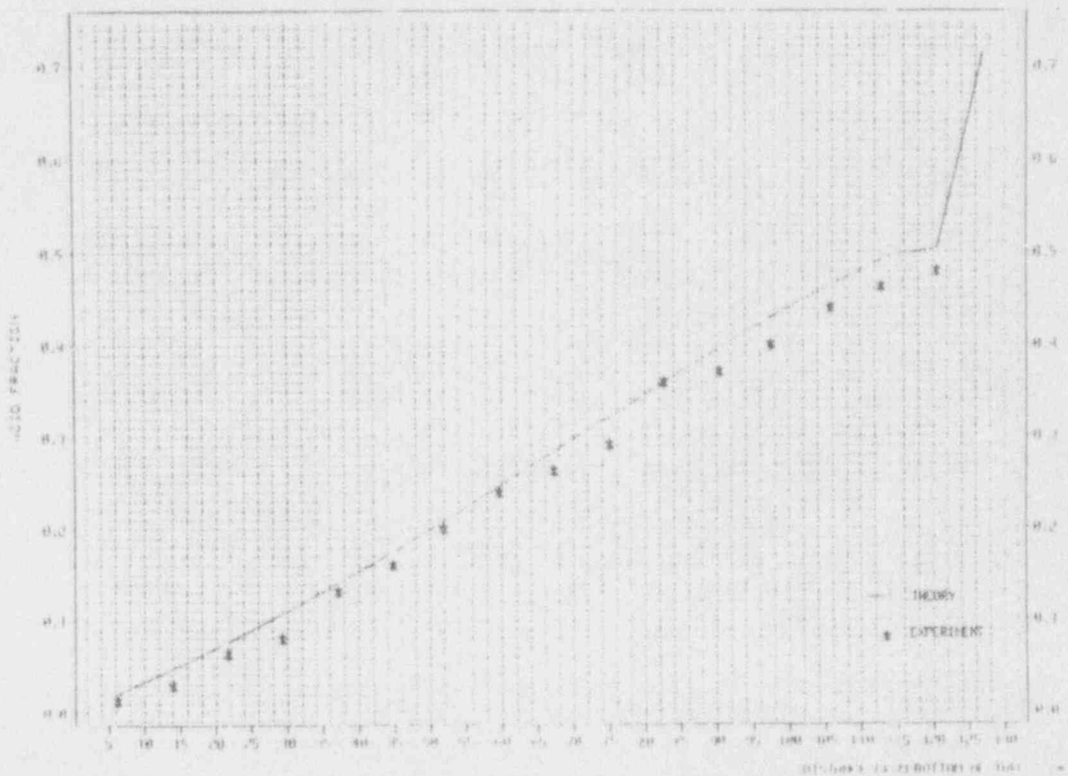


Figure 3.1.9.1 Comparison of RELAP5/MOD2 (36.05, E03) with Christensen RUN16.

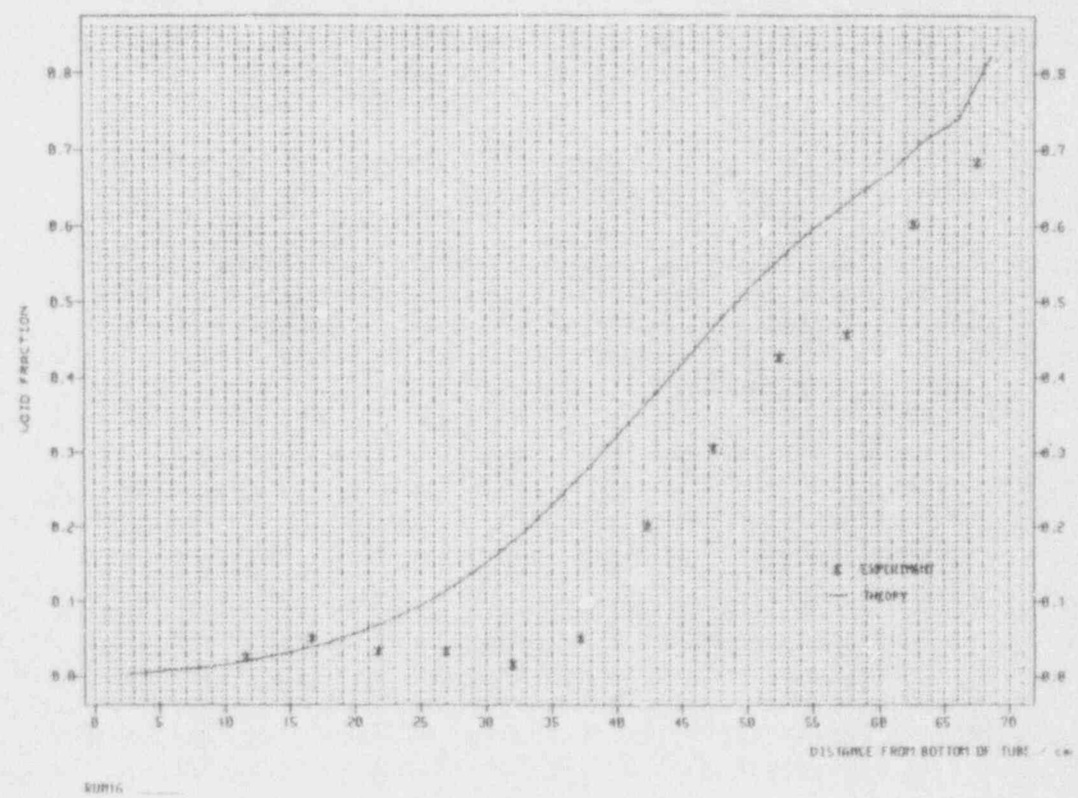


Figure 3.1.9.2 Comparison of RELAP5/MOD2 (36.02, E03) with EGEN RUN16.

3.1.10 Direct Contact Condensation on Horizontal Cocurrent Stratified Flow

Reference: S. Lee and H. J. Kim, *RELAP5 Assessment on Direct-Contact Condensation in Horizontal Cocurrent Stratified Flow*, NUREG/IA-0077, April, 1992.

Code version: RELAP5/MOD2 Cycle 36.04 and RELAP5/MOD3 version 5M5.

Facility: Horizontal rectangular test channel located at Northwestern University, Chicago, IL.

Objectives: Assess the code's capability to calculate the proper condensation rate on a liquid stratified flow interface.

Major phenomena: Direct-contact condensation on the liquid-steam interface between horizontal cocurrent steam-water flow.

Code deficiencies: The code usually undercalculated the liquid fluid depth.

User guidelines: A coarse nodalization was sufficient to calculate the experimental condensation rate. The original nodalization with 10 nodes, each representing a length of 16 cm, gave the same results as a more detailed nodalization with 20 nodes.

Base calculation: Four base-case calculations were performed, based on runs 253, 259, 279, and 293, using RELAP5/MOD2/36.04 and RELAP5/MOD3 version 5M5. The parametric studies focused on the effect of various water flow/steam flow combinations with a constant channel water level.

Sensitivity studies: None.

Nodalization studies: The original nodalization with 10 nodes, each representing a length of 16 cm, gave the same results as a more detailed nodalization with 20 nodes.

Summary: Both RELAP5/MOD2 Cycle 36.04 and RELAP5/MOD3 version 5M5 were assessed using steam condensation rate data generated at Northwestern University.

The experimental facility was composed of a rectangular channel that represented the test section, steam and water inlet plena, and a water tank. The water line was a closed loop while the steam line was built to provide steam to the test section. The channel was 1.6 m long, 0.3 m wide, and was 0.06 m deep. Uniform flow was assured by constructing large plena that assured low plenum velocities. The tests were performed at atmospheric pressure with steam flow rates ranging from 0.04 kg/s to 0.16 kg/s, water flow rates ranging from 0.2 kg/s to 1.45 kg/s, and water inlet temperatures ranging from 25 C to 50 C. The injected steam was slightly superheated. The condensation data was obtained by measuring the water flow rate at incremental positions along the channel length.

The test section was nodalized by using a PIPE with 10 cells (each 16 cm long). The code calculations of the condensation rates was in reasonable agreement with the data. However, differences were observed between the calculated channel water depth and the local heat transfer coefficient particularly for cases with a wavy interface. A nodalization study was conducted by increasing the test section cell from 10 to 20. No difference in the calculated condensation rates were observed.

3.2 SUMMARY OF ASSESSMENTS BASED ON INTEGRAL EFFECTS EXPERIMENTS

Integral effects experiments are usually conducted to: (i) produce experimental behavior that can be linked to a full-scale facility and (ii) provide a representative picture of the interactions between various interconnected phenomena.

The thirty-eight integral effects experiments are listed in Table 3.2 and each study is described in sections 3.2.1 through 3.2.38.

Table 3.2 - Code Assessment Reviews: Integral Effects Experiments[a]

Facility[b]	Scale	Experiment	Reference	See Tables
Doel 2 PP	1/1	SGTR	Stubbe, 1986	3.2.1
Doel 4 PP	1/1	Manual Loss of Load	Stubbe, 1988	3.2.2
		Reactor trip	De Vlaminck, 1990	3.2.3
KKPL PP	1/1	Reactor trip at full load: D-100-301	Gerth, 1986	3.2.4
KNU 1 PP	1/1	Loss of offsite power	Chung, 1990	3.2.5
Loviisa-2[c]	1/1	Stuck-open turbine bypass valve transient.	Yrjola, 1989	3.2.6
Tihange-2	1/1	Reactor trip at 100% power.	Rouel, 1989	3.2.7
Yong Gwang 2	1/1	Net load trip test	Arne, 1990	3.2.8
LSTF	1/48	SBLOCA: 5% CL break with no HPI - SB-CL-18.	Lee, 1991	3.2.9
LOFT	1/60	LBLOCA: 200% CL break with operational reactor coolant pumps: L2-3.	Bang, 1992	3.2.10
		LBLOCA: 200% CL break with delayed ECC, rapid pump coastdown: L2-5.	Kao, 1988 Bang, 1990	3.2.11 3.2.12
		LBLOCA: 200% CL break with no HPI, normal pump coastdown and LOOP LP-02-6.	Lubbesmeyer, 1991	3.2.13
		LBLOCA: 200% CL break with no HPI, rapid pump coastdown and LOOP: LP-LB-1.	Lubbesmeyer, 1991	3.2.14
		IBLOCA: 14% CL break in accumulator line: L5-1.	Lee, 1990	3.2.15
		SBLOCA: 2.5% CL break with HPI and reactor coolant pumps off: L3-5.	Eriksson, 1987 Scriven, 1988	3.2.16 3.2.17

Table 3.2 (continued) - Code Assessment Reviews: Integral Effects Experiments[a]

Facility[b]	Scale	Experiment	Reference	See Tables
		SBLOCA: 2.5% CL break with HPI and reactor coolant pumps on: L3-6.	Eriksson, 1987 Scriven, 1988	3.2.18 3.2.17
		SBLOCA: 1.0% HL break with HPI and reactor coolant pumps off: LP-SB-01.	Hall, 1986	3.2.19
		SBLOCA: 1.0% HL break with HPI and reactor coolant pumps on: LP-SB-02.	Hall, 1987	3.2.20
		SBLOCA: 0.4% CL break with no HPI and reactor coolant pumps on: LP-SB-03.	Harwood, 1986 Guntay,	3.2.21 3.2.22
		SBLOCA: 0.1% CL break: L3-7.	Lee, 1988	3.2.23
		Loss of feedwater induced ATWS: L9-3.	Birchley, 1988	3.2.24
		Loss of offsite power anticipated transient without trip: L9-4.	Keevil', 1988	3.2.25
		Loss of feedwater transient and feed and bleed sequence: LP-FW-01.	Croxford, 1988	3.2.26
		V sequence: LP-FP-2.	Pena, 1989	3.2.27
LOBI	1/712	SBLOCA: 3.0% CL break with HPI and reactor coolant pumps off: BL02.	Scriven, 1987	3.2.28
		Loss of feedwater: ST-02 (BT-00)	Scriven, 1988	3.2.29
FIX-II[c]	1/777	LBLOCA: 200% recirculation line break: Test 5061.	Eriksson, 1987	3.2.30
		IBLOCA: 31% recirculation line break: Test 3027.	Eriksson, 1986	3.2.31

Table 3.2 (continued) - Code Assessment Reviews: Integral Effects Experiments[a]

Facility[b]	Scale	Experiment	Reference	See Tables
		IBLOCA: 10% recirculation line break: Test 3051.	Eriksson, 1986	3.2.32
Semiscale	1/1700	LBLOCA: 200% CL break with ECC: S-06-3.	Liang, 1988.	3.2.33
		SBLOCA: 5% CL break, 0.9% bypass - S-LH-1.	Hall, 1989	3.2.34
		SBLOCA: 5% CL break, 3% bypass - S-LH-2.	Brodie, 1992	3.2.35
		SBLOCA: 0.5% CL break, with no HPI - S-NH-1.	Lee, 1991	3.2.36
		Steam line break: S-FS-1.	Rogers, 1989	3.2.37
REWET-III[c]	1/2333	Natural circulation	Hyvarinen,	3.2.38

[a] Nomenclature
- HL = hot leg
- CL = cold leg
- HPI = high pressure injection
- IL = intermediate break
- LB = large break
- LOCA = loss-of-coolant accident
- LOOP = loss of off site power
- PP = power plant
- SB = small break
- SGTR = steam generator tube rupture

[b] Note: All facilities are either full sized Westinghouse pressurized water reactors (W-type PWRs) or W-type simulators unless indicated otherwise.

[c] The Loviisa PP is a VVER-440 PWR. The REWET-III facility is a VVER-440 PWR simulator. The FIX-II facility is a simulation of an ASEA-ATOM boiling water reactor.

3.2.1 Doel-2 Steam Generator Tube Rupture Incident Assessment

Reference: E. J. Stubbe, *Assessment Study of RELAP5/MOD2 Cycle 36.01: Based on the Doel-2 Steam Generator Tube Rupture Incident of June 1979*, NUREG/IA-0008, October, 1986.

Code version: RELAP5/MOD2, Cycle 36.01

Facility: DOEL-2 nuclear power plant in Doel, East Flanders, Belgium.

Objectives: Assess the code's ability to calculate the integral behavior of the plant during a steam generator tube rupture (SGTR) incident. The objectives were met.

Major phenomena: Thermal-hydraulic phenomena associated with a SGTR. Specifically the secondary system behavior, primary coolant conditions, the pressurizer response, and the behavior of the primary inventory level.

Code deficiencies: Two deficiencies were noted: (a) Calculated excessive water level swell due to excessive interphase momentum transfer, and (b) excessive condensation due to excessive interphase mass/heat transfer.

Impact of deficiencies Inaccurate calculated secondary system behavior.

User guidelines: None.

Base calculations Above deficiencies resulted in inaccurate calculated secondary system behavior.

Sensitivity studies The impact of opening the atmospheric dump valve instantaneously, instead of over a 300 s period (base calculation) was studied; the results showed the code to give an unrepresentative level swell behavior.

Nodalization studies: None.

Summary: Doel-2, a Westinghouse two-loop pressurized water reactor rated at 392 MWe suffered a steam generator tube rupture (SGTR) event when one tube failed on June 25, 1979. When the SGTR event occurred, the plant was being heated-up in preparation for going on line. The total plant power level was 11 MWt, i.e., 2.5 MWt from each reactor coolant pump and 6 MWt core decay heat. The system pressure was 15.5 MPa and the primary inventory coolant temperature was 528 K. The plant operators followed their normal recovery procedures.

The resulting code assessment was based on the plant data recorded during the above event. Although the initial plant conditions were well known, the quantity and the quality of the available data from real plants are in general inferior to well-instrumented test facility data. Furthermore, the timing and intensity of the operator involvement during the transient was not readily available and had to be inferred from the available data.

The objective the assessment study was to evaluate the code's capability to simulate the Doel-2's SGTR event. The assessment was performed by comparing trends rather than comparing absolute values.

Several important observations concerning the code were obtained in the study:

1. The code is capable of simulating the phenomena that occurred during the SGTR event in a reasonable fashion. However, the lack of precise boundary conditions limited the extent of the assessment.

2. Impressive improvements over the capability present in RELAP5/MOD1, Cycle 19 were observed, particularly in the calculation of the break flow rate and the pressurizer inventory due to counter-current flow.

3. Excessive water level swell observed in the intact SG during cooldown may be due to excessive interphase momentum transfer in the SG riser when bulk boiling was initiated.

4. Excessive interphase mass and heat transfer for condensation and evaporation in quasi-stagnant flow conditions were calculated in the isolated affected steam generator.

3.2.2 Doel-4 Manual Loss of Load Test Assessment

Reference: E. J. Stubbe and P. Deschutter, *Assessment Study of RELAP5/MOD2 Cycle 36.04 Based on the Doel-4 Manual Loss of Load Test of November 23rd 1985*, NUREG/IA-0043, March, 1992.

Code version: RELAP5/MOD2, Cycle 36.04

Facility: DOEL-4 nuclear power plant in Doel, East Flanders, Belgium.

Objectives: Assess the code's capability to simulate the Doel-4 nuclear power plant "islanding" transient. An "islanding" transient is a transient scenario that occurs when the plant is isolated from all external systems and becomes an "island." During such a transient, the plant can only rely on its own on-site equipment and capabilities.

Major phenomena: The plant pressures, fluid temperatures, and levels in both the steam generator and pressurizer are the parameters of most concern.

Code deficiencies: Excessive interphase drag at low void fractions in the steam generator was noted. This deficiency was apparent because the code was unable to calculate the correct steam generator level behavior. The deficiency was especially apparent during a period of rapidly changing level.

Impact of deficiencies: This deficiency results in an incorrectly calculated mixture level and also incorrectly calculated level instrumentation readings.

User guidelines: None.

Base calculations: The base calculation was performed with the steam generator structural heat slabs removed because the model was too large to use with their computer system memory configuration.

Sensitivity studies: The impact of running the calculation with the steam generator structural heat slabs removed was evaluated using only a single steam generator. The results showed the structural heat slabs had only a small effect on the results, except when large changes in temperature occurred. Since the calculation simulated a relatively slow transient, the authors justified their removal of the steam generator structural heat slabs.

Nodalization studies: None.

Summary: Doel-4 is a full scale PWR, featuring a 3-loop Westinghouse design with one circulating pump and one steam generator in each loop. The core contains 157 fuel assemblies with 264 fuel rods per assembly and generates 2988 MWt under nominal conditions. The reactor coolant pumps, rated 4.5 MW each circulate 6.4 m^3/s of coolant per loop. A 45.3 m^3 pressurizer is connected to the hot leg of loop B. Doel-4 also features a preheater steam generator design, in which the feedwater comes in the bottom of the SG, on the cold side of the inverted U-tubes. Like all Belgian nuclear power plants, Doel-4 is designed and tested to ride through a "loss of external load" transient. In this transient the turbine-generator set is isolated, and the reactor is returned from full power to house load power level (typically 5% of nominal power), quickly and without scram. This is done to test the plant built-in flexibility to cope with grid-transients. This report discusses such a test, successfully performed on Doel-4 on November 23, 1985, and its comparison with a RELAP5/MOD3 simulation of the same test.

The experimental data from the test was recorded with a dedicated data acquisition system, capable of recording 240 plant parameters. The test was initiated by manually opening the main high voltage breaker when the plant was at full power. The behavior of the plant is described in terms of the primary system parameters, the steam generator parameters, and the pressurizer parameters. Plots of short term (0 to 100 s) as well as long term (100 to 600 s) behavior are presented. It was determined during the experiment that the period of 600 s covers the most important phenomena which govern a successful transition from full power to house load.

The RELAP5 input model was constructed using the recommended methods and procedures given in the RELAP5/MOD2 manual. The primary and secondary systems (feedwater, steam generator, main steam) were both modeled explicitly by control volumes and junctions respecting the true geometry and hydraulic features of the components. The piping and component walls and internals in contact with the coolant were represented by heat structures, with the exception of the steam generators: a need to reduce the size of the model motivated a parametric study which was performed on one steam generator, with and without structural heat slabs; the results of tnis study indicated little impact on the results, thus the structural heat slabs were eliminated from all steam generator models. The auxiliary components and systems (pressurizer relief and safety valves controls, the main feedwater system, the auxiliary feedwater system, the steam generator relief and safety valve controls, the steam dump to the condenser) were simulated functionally by using control system packages that reproduced their effect. Finally, boundary conditions were imposed to the explicitly modeled components.

Comparison of numerical results with recorded data indicate an overall acceptable agreement; thus satisfying the main objectives of the assessment: (1) to prove RELAP5's capability and (2) to establish the quality of the Doel-4 model. Small discrepancies were observed, which can be traced to : slight differences between the steady-state conditions of the simulation and initial conditions of the plant; acoustic effects on the sensors, which may require a much finer nodalization of the model in order to be predicted ; and the removal of the heat slabs from the steam generator model which account for an excessive rise of the cold leg temperature. Excessive level swell in the steam generator, in the RELAP5 prediction, is attributed to deficiencies of the interfacial model.

The study revealed some important feedback mechanisms which could lead to plant divergence, and hence reactor trip. It show that by optimizing the feedwater flow controller gain, stability can be improved considerably.

3.2.3 Doel-4 Reactor Trip Assessment

Reference: M. De Vlaminck, P. Deschutter, and L. Vanhoenacker, *Assessment Study of RELAP5/MOD2 Cycle 36.05 Based on the Doel-4 Reactor Trip of November 22nd, 1985*, NUREG/IA-0051, March, 1992.

Code version: RELAP5/MOD2, Cycle 36.05

Facility: DOEL-4 nuclear power plant in Doel, East Flanders, Belgium.

Objectives: Assess the code's capability to simulate the Doel-4 nuclear power plant reactor trip transient. Events of interest were the timing of various equipment trips and also the simulation of the steam generator secondary liquid level.

Major phenomena: The analyzed transient consisted of a turbine trip on high steam generator level, followed by a reactor trip. The thermal-hydraulic phenomena and parameters addressed are the neutron and thermal power, the evolution of the primary pressure, the pressurizer water level, the temperature distribution in hot and cold legs, the secondary pressure, the steam flow, and the instrumentation delays.

Code deficiencies: Excessive interphase drag at low void fractions in the steam generator was noted. This deficiency was apparent because the code was unable to calculate the correct steam generator level behavior.

Impact of deficiencies: This deficiency results in an incorrectly calculated mixture level and also incorrectly calculated level instrumentation readings.

User guidelines: None.

Base calculations: The base calculation was performed using a baseline model modified to contain time lags to better follow the hot and cold leg temperature transient behavior together with modified steam dump flow curves and an adjusted opening sequence for the steam generator relief valves.

Sensitivity studies: Five sensitivity calculations were performed to examine the effect of varying the opening times for the steam generator relief valves, the volume modeling adjacent to the narrow range secondary level measurements, the auxiliary feedwater temperature, increasing secondary inventory levels, and varying on the fuel properties.

Nodalization studies: None.

Summary: The Doel-4 nuclear power plant is a 3000 MWt, 3-loop, Westinghouse designed pressurized water reactor. As part of its first cycle testing program, a turbine trip on high steam generator level followed by a reactor trip was performed on November 22, 1985. This test was specifically performed to test the operation of the steam dump control systems. These data were used to assess the capabilities of RELAP5/MOD2 to simulate the transient.

The scope of the simulation includes the primary coolant system, the three loops, and the secondary system. The assessment consists of nine simulations of which one (Run 12) was taken as the base calculation.

The short term simulation highlighted the rapid changes occurring in both the primary and secondary systems due to the sudden drop in reactor power. It also indicated the strong sensitivity of the calculated pressures and temperatures on the timing of sensors and effective control actions. Sensitivity studies were conducted to adjust the instrumentation response times during this initial period of the transient. Boundary conditions required some adjustment as well. Sensitivity studies were conducted to determine the steam dump valves capacity at partial opening positions.

Sensitivity calculations and comparison with the recorded data identified a deficiency in the interphase drag calculation in the code. Two phase flow appears to carry excess water with it, causing an overestimation of the void fraction in the steam generator's riser. It was necessary to artificially increase the initial water content in the steam generator, by several metric tons, in order to reproduce the steam generator level after the trip.

Except for the narrow range level, it was possible to closely approximate the parameters related to the primary system and the steam generator to the recorded data, by adjusting the boundary conditions and instrumentation timing characteristics.

3.2.4 Phillipsburg-2 Reactor Trip at Full Load Assessment

Reference: G. Gerth, *Assessment Study of RELAP5/MOD2 Cycle 36.04 Based on the Commissioning Test Reactor Trip at Full Load at the Phillipsburg 2 Nuclear Power Plant*, NUREG/IA-0057, April, 1992.

Code version: RELAP5/MOD2, Cycle 36.02.

Facility: Phillipsburg-2 (KKPL), Federal Republic of Germany.

Objectives: Evaluate the capability of the code to simulate the thermal-hydraulic and control system behavior of a full-scale plant during an operational transient. The objectives were met.

Major phenomena: Reactor power, core differential temperature, core inlet and outlet temperatures, primary system pressures, pressurizer level, secondary system pressure, steam bypass valve flow, steam generator levels, and feedwater flow rate.

Code deficiencies: Secondary side heat transfer not calculated correctly during steady-state calculations.

Impact of deficiencies: This deficiency is routinely accounted for and the input adjusted appropriately for steady-state calculations.

User guidelines:

1. If a turbine is modeled using a time dependent volume, the operational pressure difference across the valve should be restricted to prevent choking.
2. Modeling multiple main steam bypass valves (and presumably any bank of valves) with only one valve will cause a deviation between the calculated and real valve flow rates. However, one possible solution to this problem, may be to model the valve bank using a single servo valve together with a variable valve area dependent on an appropriate control logic to simulate valve openings and valve hysteresis. Unfortunately, such a modeling scheme requires that a single downstream pressure be assumed.

Base calculations: Only a base calculation was conducted. Generally good agreement between the calculation and the data were obtained.

Sensitivity studies: None

Nodalization studies: None.

Summary: The assessment was performed using plant startup data from a prototype reactor. The test simulated a reactor trip from full power-conditions. The plant from which the data were obtained is the Phillipsburg-2 pressurized water reactor constructed by KWU in the Federal Republic of Germany.

The Phillipsburg-2 reactor has a rated thermal power of 3765 MWt. The reactor is a four-loop design; each loop contains a steam generator, primary coolant pump, and interconnecting piping.

The plant startup test was initiated with the plant in full-power operation by a manual reactor trip. The reactor trip resulted in a turbine trip that effected isolation of the steam generator main feedwater and main steam flows. The primary coolant pumps continued operating. Following the reactor trip, core stored energy was removed to the steam generators, resulting in the momentary lifting of the turbine bypass valves. As the core heat input to the fluid decreased toward the decay heat rate and as auxiliary feedwater flow was established, the turbine bypass valves closed.

In the final plant condition the reactor coolant pumps were operating and the core decay heat was removed to the steam generators. The primary coolant system pressure was controlled by the pressurizer heater and spray systems and the pressurizer level was controlled by the makeup system. On the secondary side, auxiliary feedwater was controlled to maintain setpoint levels and the turbine bypass valves opened as needed to control the setpoint pressure.

This assessment was performed to determine code capabilities for simulating the thermal-hydraulic and control system behavior of a full-scale plant during an operational transient. An assessment of this type differs from those involving experimental facilities in three respects: questions of scaling and non-representative behavior of the test facility are removed, the data are available only for transients that are much less challenging for the code to simulate than design-basis accidents, and the quality and quantity of the data are not up to the standards of experimental facilities. This assessment therefore provides a somewhat qualitative benchmark of overall code capabilities for simulating transients in a prototype reactor.

A qualification of the data was performed to eliminate suspect data prior to comparison with the calculation. Only limited plant data uncertainty information was available. The uncertainty of the measured data is generally taken as 1 to 2% of the instrument full range.

A full power RELAP5 steady state calculation was first performed to obtain satisfactory conditions from which to start the transient calculation. Difficulty was encountered in obtaining concurrent agreement between calculated and measured primary coolant system temperatures and steam generator secondary pressures. It was elected to conserve the simulation of secondary pressure, thus requiring the calculated primary coolant temperatures to rise approximately 2.5 K above the measured values. This compromise is due to a code limitation resulting from the incorrect calculation of the heat transfer coefficient on the secondary side of the steam generator tubes. The deficiency arises because the calculated heat transfer coefficient does not account for the unique flow geometry in the steam generator boiler region. At INEL, a standard practice has developed to use the steam generator tube-to-tube spacing (i.e. the gap between the outside of the tubes) as the hydraulic diameter. With this method, the heat transfer is enhanced artificially to account for the deficiencies involved in applying simple tube bundle correlations in the complex geometry of the baffled steam generator boiler. The prototype primary system temperatures and secondary pressures may thereby be matched in the model.

The RELAP5 simulation of the full-power manual reactor trip transient at the Phillipsburg-2 plant spanned the first 250 s of the transient. The calculated and measured data for the following parameters are compared: reactor power, core differential temperature, core inlet and outlet temperatures, primary system pressure, pressurizer level, secondary system pressures, steam bypass valve flow, steam generator levels, and feedwater flow rate.

Reasonable agreement between most of the calculated and measured parameters is noted. However agreement between the pressure vessel inlet and outlet temperatures (see Fig. 3.2.4.1) could have been markedly improved by just adjusting the U-tube heated diameters as recommended in the User's Guidelines (Fletcher and Schultz, 1982). Also, agreement between the calculated and measured turbine bypass valve position (see Fig. 3.2.4.2), and consequently the mass flow rates, could probably have been improved by a more detailed valve simulation. It is believed by the reviewers that the difference between the calculated and measured bypass valve position was caused by the modeling of the bank of six bypass valves using a single servo valve. It is not clear whether the anomaly is due to an incorrect lumping of the six valves into a single valve or due to calibration uncertainty in the actual valves. A method of satisfactorily modeling the operation of a valve bank using a single valve component may be accomplished using a single servo valve and varying the flow area based on the number of valves open at any given time. Control logic may be used to determine if each of the valves in the bank is open including the effects of valve hysteresis. A limitation with this approach is that a single downstream pressure must be assumed. This is satisfactory as long as the valve flow is choked, but may prove unsatisfactory if the valve flow rate is controlled by friction. It should be noted that the agreement between calculated and measured valve flow response is particularly sensitive to uncertainty in the valve lift and reseat pressures. This anomaly also may have resulted in a momentary disagreement between calculated and measured steam generator secondary levels during the time frame when the turbine bypass valves were active.

In summary, the comparison of calculated and measured data from a plant commissioning test indicates RELAP5/MOD2 Version 36.02 is capable of simulating operational transients in pressurized water reactors. No serious code deficiencies were identified. Minor deficiencies in the modeling of a transient such as this may be circumvented using various modeling options.

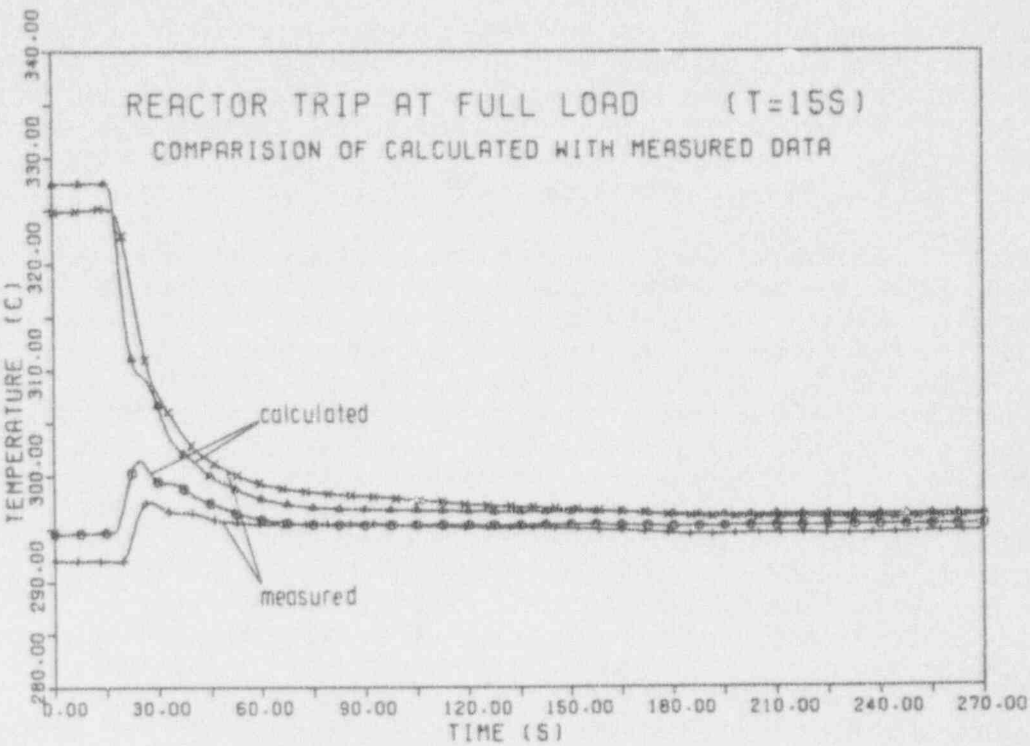


Figure 3.2.4.1 Philippsburg-2 Plant: Calculated and Measured Pressure Vessel Inlet and Outlet Temperatures.

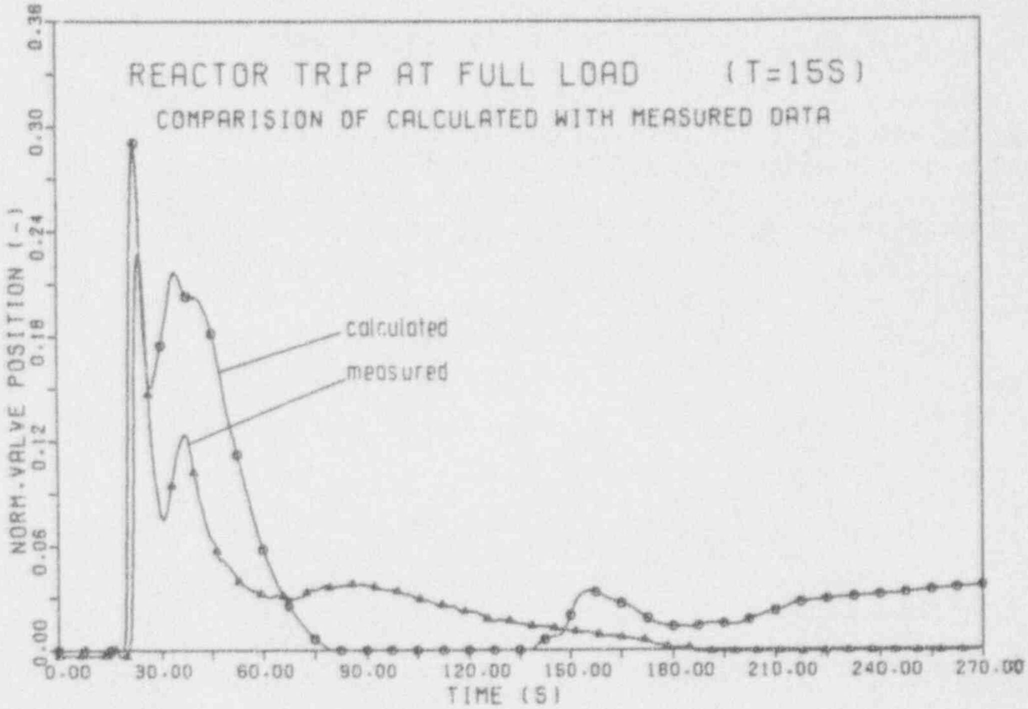


Figure 3.2.4.2 Philippsburg-2 Plant: Calculated and Measured Bypass Valve Position.

3.2.5 Kori Nuclear Unit No. 1 Loss of Offsite Power Assessment

Reference: B. D. Chung, H. J. Kim, and Y. J. Lee, *Assessment of RELAP5/MOD2 Code Using Loss of Offsite Power Transient Data of KNU No. 1 Plant*, NUREG/IA-0030, April, 1990.

Code version: RELAP5/MOD2, Cycle 36.05.

Facility: Kori Nuclear Unit No. 1, Kori, Kyongnam, South Korea.

Objectives: Evaluate the capability of the code to simulate the thermal-hydraulic behavior of a full-scale plant during a loss of offsite power event.

Major phenomena: Global parameters of interest included the primary pressure, primary fluid temperatures upstream and downstream of the steam generator, the secondary water level and the secondary pressure behavior.

Code deficiencies: None.

Impact of deficiencies: None.

User guidelines: None.

Base calculations: The base calculation was conducted using the author's first-cut nodalization. The subsequent calculations showed good comparisons between the data and the calculation for the secondary pressure, primary loop flow rates and temperatures. The authors found the primary pressure calculation was too high and the steam generator collapsed liquid level calculation did not match the data.

Sensitivity studies: None

Nodalization studies: Nodalization calculations were performed to examine the effect of several different secondary nodalizations on the steam generator liquid level calculation. By altering the baseline nodalization to allow communication between the downcomer plenum and the steam dome, an acceptable match between the calculated steam generator liquid level and the data was obtained.

Summary: The assessment study was performed using the data from a loss-of-offsite power transient that occurred at the Kori Nuclear Unit No. 1 (KNU No. 1) on June 9, 1981. The transient was initiated by a spurious signal generated by the steam generator (SG) A level control system. The plant was operating at 77.5% rated power.

The Kori Nuclear Unit No. 1 is a two-loop Westinghouse pressurized water reactor with a full-power rating of 587 MWe. The plant SGs are Model 51s with inverted U-tubes.

Following the initiating event, a secondary water level/feedwater flow mismatch was recorded in SG A was recorded 100 s later. The loss-of-offsite power occurred at 131 s. Both reactor coolant pumps had tripped by 163 s. By 392 s the off-site power was restored.

The transient was simulated using a model developed by both the Korea Advanced Energy Research Institute and the utility operating KNU No. 1. The code calculation showed a stable steady-state condition and gave a reasonable prediction of the plant transient behavior. Although the calculation did not match the measured primary plant behavior exactly, the correct trends and plant behavior were calculated.

A nodalization study was undertaken because the base calculation did not accurately predict the behavior of the SG secondary collapsed liquid level in SG B. The effect of several different secondary nodalizations on the steam generator liquid level calculation were studied. By altering the baseline nodalization to allow communication between the downcomer plenum and the steam dome, an acceptable match between the calculated steam generator liquid level and the data was obtained.

3.2.6 Loviisa-2 Stuck-Open Turbine Bypass Valve Transient Assessment

Reference: V. Yrjola, *Assessment of RELAP5/MOD2 Cycle 36.04 Against the Loviisa-2 Stuck-Open Turbine Bypass Valve Transient on September 1, 1981*, NUREG/IA-0047, March, 1992.

Code version: RELAP5/MOD2, Cycle 36.04.

Facility: Loviisa Unit No. 2, Loviisa, Uusimaa, Finland. Loviisa-2 is a Soviet VVER-440 type-design with horizontal steam generators and a loop seal in both the hot leg and the cold leg in each loop. The plant has six loops.

Objectives: Examine whether RELAP5/MOD2 has the capability to simulate transients in a plant with horizontally-oriented steam generators.

Major phenomena: Energy transfer between the primary and secondary systems, spray and condensation in the pressurizer, mass flows and temperature changes in the primary, depressurization of the secondary, and circulation in the horizontal steam generators are the major phenomena addressed in the report.

Code deficiencies: None.

Impact of deficiencies: None.

User guidelines: None.

Base calculations: The base calculation was performed using a model with three loops. One model loop simulated the broken loop, one model loop simulated two loops that are connected to the pressurizer, and one model loop simulated the remaining three plant loops. The steam generator was modelled by using a horizontal PIPE component for the primary. The steam generator was modelled by using four volumes: three PIPE volumes and one separator volume.

Sensitivity studies: Calculations were performed to study the plant transient sensitivity to the initial conditions, the primary loop flow rates, and the turbine bypass valve behavior.

Nodalization studies: None.

Summary: The assessment is based on data recorded during an overcooling transient that occurred at the Loviisa Unit 2 on September 1, 1981. The objective of the study was to assess the applicability of the code for a real plant transient analysis and in particular to examine the capability of the code as it is extended to model a plant of Soviet design which has horizontal steam generators.

The Loviisa power plant consists of two Soviet VVER-440 type pressurized water reactors having a net power output of 445 MWe each.. In VVER-440 reactors the primary system consists of six parallel loops each with a horizontal steam generator, a main circulation pump, and a main loop isolation gate valves. There are loop seals in both the hot and cold legs of each loop of this reactor system.

The transient that occurred on September 1, 1981 was initiated from full power by a reactor trip. Faulty operation of the level gauges in four steam generators caused the trip signal. An associated stuck-open failure of one turbine by-pass valve caused a fast cooldown. The high pressure safety injection started to operate, but it was quickly turned off by the operator. The downcomer temperature dropped from 538 K to 488 K in fifteen minutes. The temperature decrease ceased when the operator closed the shut-off valve in the open by-pass line.

The code assessment was based on the available plant data saved using the normal plant instrumentation for the early portion of the transient. Because of plant computer memory limitations, the remainder of the transient data was taken from plant recorder plotted histories. Although the recorded data are comprehensive, there is sometimes not enough data to explain all phenomenon and thus engineering judgement was used when needed.

The RELAP5 model was built by personnel at the Technical Research Centre of Finland. The loop with the stuck-open valve was modelled as one loop, the two loops connected to the pressurizer were lumped together as another loop, and the remaining three loops were lumped together as one. The horizontal steam generator design and its gravitational water separation was modeled using the mechanistic separator model of the RELAP5 code. The natural circulation in the steam generator secondary was artificially modelled.

The RELAP5 simulation compared qualitatively with the data. The main quantitative disagreement between the data and the calculation was in the primary pressure and the pressurizer level; this however was traced to an imprecise nodalization (too coarse) of the pressure vessel upper head. Other differences between the data and calculation were observed, but the limited instrumentation did not allow a clear identification of the source.

The calculation gave the researchers insight to the behavior of the pressurizer and the pressurizer spray, which could not be determined from the data. The simulation showed that condensation on the pressurizer walls was enough to stop repressurization. The wall heat transfer in the pressurizer volume, where both liquid and vapor were present, experienced anomalous behavior during the fast insurge period. The vapor in this volume was superheated faster than the vapor in the volumes above it.

Oscillatory behavior of the servo valve model was found. Time step reductions removed the oscillation, but the root causes of the oscillations could not be identified.

A sensitivity study of the initial conditions showed that the results were sensitive to the primary mean temperature; other parameters changed within their range of measurement uncertainty did not cause significant changes in the result.

3.2.7 Tihange-2 Reactor Trip Transient Assessment

Reference: G. P. Rouel and E. J. Stubbe, *Assessment Study of RELAP5/MOD2 Cycle 36.05 Based on the Tihange-2 Reactor Trip of January 11, 1983*, NUREG/IA-0044, March, 1992.

Code version: RELAP5/MOD2 Cycle 36.05

Facility: Tihange-2 nuclear power plant in Belgium

Objectives: Evaluate the code's ability to model the transient and evaluate the sensitivity of the model to uncertainties in the boundary conditions and to parametric variations of the steam generator model.

Major phenomena: Steam generator response including level swell and condensation on the secondary side, level and pressure response in the pressurizer.

Code deficiencies: The two-phase code models do not tolerate high thermal disequilibrium conditions for the bubbly flow regime under fast pressurization.

User guidelines: None.

Base calculation: The code was shown to be capable of simulation of the basic thermal hydraulic phenomena that occur in a full scale power plant following a reactor trip.

Sensitivity studies: Parametric studies to investigate the effect of boundary conditions uncertainty showed that minor changes in timing of events or dynamics of the steam dump result in large variations in the calculated steam generator response.

Nodalization studies: None.

Summary: The assessment was performed using data obtained during the Tihange-2 reactor commissioning tests. Tihange-2 is a 2785 MWt, three-loop Framatome designed pressurized water reactor. The purpose of the test was to evaluate the dynamic behavior of the plant including the steam dump control systems and the feedwater regulating valve response.

The objectives of the assessment were to determine the code's ability to model the aforementioned transient; to evaluate the quality of the model itself; and to evaluate the sensitivity of the plant transient to uncertainties in the boundary conditions and to parametric variations of the steam generator model.

The model included the primary system (with each loop modelled individually), the secondary system and the necessary components of the plant control system. The code assessment was based on eight calculations including the base calculation. The various runs investigated the effect of uncertainties concerning the boundary conditions and also investigated the effect of the steam generator nodalization.

Conclusions resulting from this study were:

1. RELAP5/MOD2 is able to simulate the thermal-hydraulic phenomena that occur in a full-scale nuclear power plant following a reactor trip.

2. Despite the high quality of the plant data acquisition system, the data are affected by a rather large uncertainty due to the imprecision or offset of the many sensors. The plant sensor offsets were large compared to those usually found in separate effects tests and scaled integral tests.

3. The basic merit of this type of assessment is to allow a gauge of the scaling effects on the code models and correlations and guidance concerning whether additional separate effects tests should be conducted.

4. Agreement between the data and calculated parameters was much better for the primary system than for the secondary system.

5. The parametric study clearly shows the importance of using accurate boundary conditions for the plant models. Relatively small changes of the timing and dynamics of the steam dump induce large variations in the steam generator parameters, for example the water level indication. This study underscores the need for the code users to have (a) a good understanding of the code and its limitations and (b) a detailed understanding of the plant and its instrumentation.

6. The two-phase code constitutive models do not tolerate high thermal disequilibrium conditions for the bubbly flow regime under fast depressurization. Due to premature condensation (in the simulation) the vapor phase returned to the quasi-saturation condition too quickly as apparent from the temporary stagnation of the pressure and an abnormal water level response in the steam generators. It should be noted that these two parameters are basically the only two parameters that the operator observes in the control room and upon which plant protection and control systems actions are based.

3.2.8 Yong Gwang 2 Net Load Trip Test Transient Assessment

Reference: N. Arne, S. Cho, and S. H. Lee, *Assessment Study of RELAP5/MOD2 Computer Code Against the Net Load Trip Test Data from Yong-Gwang Unit 2*, Korea Electric Power Co, January, 1990.

Code version: RELAP5/MOD2 Cycle 36.04

Facility: Yong-Gwang Unit 2, Yonggwang, Chonnam, Korea.

Objectives: Evaluate the code's ability to model the Yong-Gwang Unit net load trip test transient.

Major phenomena: Rates of change in the primary and secondary state variables such as pressure, temperature, pressurizer level, and secondary water level as driven by the plant control system response to plant power level changes.

Code deficiencies: None.

User guidelines: None.

Base calculation: The calculation, although using relatively crude reactivity coefficients, showed reasonable agreement with the plant operating data.

Sensitivity studies: A sensitivity calculation was performed to evaluate the effect of doubling the control rod reactivity worth. Better agreement with the core power data was shown.

Nodalization studies: A study was done to determine whether the transient could be better simulated using boundary conditions imposed on the model to simulate the balance-of-plant versus a more detailed balance-of-plant model. The more detailed balance-of-plant model gave a better simulation.

Summary: This report documents the assessment of RELAP5/MOD2 Cycle 36.04 using the plant Net Load Trip Test (NLTT) data from Yong-Gwang Unit 2 of the Republic of Korea.

Yong-Gwang Unit 2 is a 996.8 MWt Westinghouse three loop PWR. Each loop has a reactor coolant pump and a steam generator. A pressurizer is connected to the hot leg of one coolant loop. Its control system maintains the vessel pressure at a set value and prevent reactor trips due to plant transients. A control rod control system maintains a programmed average vessel temperature by regulating the core activity.

The steam is conveyed to the turbine generator system through a main steam pipe. This pipe is equipped with power operated relief valves, safety valves, main steam isolation valves, and atmosphere and condenser dump valves. These valves are parts of steam generator level and steam dump control systems, and play an important role in NLTT transients.

The NLTT took place when the plant was at 100 percent power and all control systems were in automatic mode. NLTT was initiated by a large and sudden load rejection. Sensing the difference between the reactor power and turbine load, the control rods were lowered to keep the vessel temperature at a programmed level, but the immediate rate of core power reduction was very small (0.33 % full power per seconds). Meanwhile, the rate of energy removal from the reactor vessel became limited by the steam flow reduction. As a result, the primary system pressure tended to rise but was prevented from doing so by the pressurizer spray flow.

For the secondary system, the steam flow was initially reduced by turbine valve closing. The resulting temperature and pressure rises tripped open the steam dump valves immediately. Steam blowdown began and caused gradual depressurization in the steam generator. The steam dump valves were programmed to remain open for about 35 s, then begin to close gradually and eventually ended the blowdown. Meanwhile, the feedwater control system strived to maintain the steam generator water level.

At 180 seconds the reactor power dropped to 50 percent of the rated full power and the plant control was switched to manual. The test ended at that time. During the transients, no safety injection of coolant was initiated, no steam generator and pressurizer safety valves were actuated, and the turbine did not reach the overspeed trip setpoint.

A base case simulation, a sensitivity study on control rod reactivity worth, and a nodalization study on steam dump system modeling were conducted. The conclusions are:

1. RELAP5/MOD2 yielded reasonable predictions of the primary system thermal-hydraulic parameters such as reactor power, vessel average temperature, and pressurizer level and pressure during NLTT.

2. RELAP5/MOD2 yielded reasonable predictions of the secondary system thermal-hydraulic parameters such as steam generator water level, steam pressure, steam flow, and feedwater flow during NLTT.

3. Because the control rod reactivity worth had a large uncertainty, the core power had a complementary uncertainty. Using the control rod reactivity worth uncertainty band as a limit for conducting the sensitivity study, the code was shown to give reasonable agreement with the core power data.

4. For NLTT transients, a valve junction model (one turbine volume, four condenser volumes, and four servo valve junctions) yields better results than a boundary condition model (one turbine volume, one condenser volume, and one time dependent junction).

3.2.9 Large Scale Test Facility (LSTF) 5% Cold Leg SBLOCA Experiment Assessment

Reference: S. Lee, B. D. Chung, and H. J. Kim, *RELAP5 Assessment Using LSTF Test Data SB-CL-18*, Korea Institute of Nuclear Safety, February, 1991.

Code version: RELAP5/MOD2 Cycle 36.04

Facility: The ROSA-IV Program's Large Scale Test Facility (LSTF), Tokai, Japan.

Objectives: Evaluate the code's ability to simulate the important phenomena occurring during a SBLOCA such as break critical flow, loop seal clearing and core uncovery, and core heatup.

Major phenomena: Critical flow, countercurrent flow limiting (CCFL), loop seal clearing and core uncovery, core heatup, stratified two-phase in the horizontal legs, vessel inventory boiloff, and vessel refill due to accumulator injection.

Code deficiencies:

1. Two-phase break flow rate underpredicted and steam critical flow rate overpredicted.

2. Overcalculated liquid holdup in the upflow side of the steam generator U-tubes.

User guidelines: None.

Base calculation: The base calculation was performed using the nodalization recommended by the INEL. Deficiencies were detected in the code's critical flow model and the interphase drag model (overcalculation of liquid holdup in the upflow side of the steam generators).

Sensitivity studies: Sensitivity calculations were performed to evaluate the effect of using the abrupt are change option and the smooth area change options at the break junction. Better agreement was found between the data and the calculation when the smooth area change option was used.

Nodalization studies: None.

Summary: The code was assessed using the experimental data obtained during the SB-CL-18 experiment conducted in the LSTF. The experiment was conducted to investigate the thermal-hydraulic mechanisms responsible for the early core uncovery, including the manometric effect due to an asymmetric coolant holdup in the steam generator upflow and downflow side during the 5% cold leg small break loss-of-coolant accident (SBLOCA). The simulation capability of the code of the phenomena occurring during the SBLOCA is the subject of the report.

The LSTF is a 1/48 volumetrically scaled nonnuclear model of a Westinghouse type 3423 MWt four loop PWR. The facility is designed to simulate SBLOCAs (up to 10%) and operational transients at the same high pressures and temperatures as the reference PWR. The LSTF has two equally sized loops that differ only in the possible break geometries and in the presence of a pressurizer in one of the loops. The 1064 electrically-heated rods and the 104 unheated rods are used to simulate the 17x17 fuel assembly of the PWR core. The design scaling compromise is the 10 MW maximum core power limitation, 14% of the scaled reference PWR rated power. Each steam generator (SG) with 141 full-sized U-tubes in a scaled secondary volume is designed in considering the steady-state flow conditions at 14% of the scaled reference PWR SG flow.

The baseline calculations show good agreement with the experimental data in predicting thermal-hydraulic phenomena. The authors, however, point out several differences regarding the evolution of phenomena and affecting the timing order. Specific deficiencies noted by the authors are as follows:

1. The calculated break flow rates show some discrepancy with experimental data in RELAP5/MOD2 version 36.05. Underestimation of the two-phase break flow resulted in an insufficient mass discharge from the primary system prior to loop seal clearing. Overpredicted vapor phase break flow caused a fast primary mass loss and an earlier accumulator injection after loop seal clearing.

2. The liquid holdup in the upflow side of the SG U-tubes was overcalculated. This caused a plug effect hindering the loop seal downflow side level decreasing and delaying the loop seal clearance.

The sensitivity studies were performed to improve the calculational agreement with the data. The calculations were performed for several different values of the break junction options such as abrupt area changes, smooth area changes, and two-phase flow discharge coefficient. The results showed no remarkable improvement in predicting the break flow. However, in the case where a smooth area change option was used, an improved prediction of the break flow for single-phase flow was observed.

In conclusion, the code can predict the major phenomena occurring during a 5% cold leg break LOCA although some deficiencies in predicting the break flow and liquid holdup in the steam generator U-tubes were noted.

3.2.10 LOFT Large Break LOCE L2-3 Assessment

Reference: Y. S. Bang, H. J. Kim, and S. H. Kim, *Assessment of RELAP5/MOD2 Cycle 36.04 with LOFT Large Break LOCE L2-3*, NUREG/IA-0070, April, 1992.

Code version: RELAP5/MOD2 Cycle 36.04

Facility: Loss of Fluid Test (LOFT) Facility, Idaho Falls, ID, USA.

Objectives: To assess the capability of RELAP5/MOD2, Cycle 36.04 to predict the thermal-hydraulic phenomena during a large break LOCA with the primary coolant pump operational, to reexamine previously identified code deficiencies, and to examine the effect of using a modified heated surface rewet criteria.

Major phenomena: Early dryout and rewet of the core; later dryout, refill, and reflood; break critical flow; and emergency core cooling system injection.

Code deficiencies:

1. Although the break critical flow rate during the transition phase between subcooled and two-phase break flow was cited as a deficiency, the match between the data and the calculation was reasonable. No evidence, other than the data/calculation mismatch, was given in the assessment to support the conclusion that the RELAP5/MOD2 transition model is deficient.

2. Code was unable to properly calculate the early dryout and rewet behavior of the core.

3. The core heatup during the blowdown period was undercalculated and the calculated rewet occurred earlier. This deficiency was caused by the inadequacy of the Biasi critical heat flux correlation.

User guidelines: None.

Base calculation: The base calculation was performed using the nodalization originally specified by the INEL and thereafter modified by Bang et al. Bang et al's nodalization differs from the INEL original work principally in the use of only one channel with 12 cells. In general, a reasonable calculation of the loop hydraulic behavior was obtained. However, the core heatup behavior was not well calculated.

Sensitivity studies: Sensitivity calculations were performed to evaluate the effect of the Paul Scherrer Institute update designed to better-calculate the core heatup behavior following departure from nucleate boiling. The sensitivity calculations, including the PSI updates, produced a reasonable match with the data during the post-CHF core heatup and reflood period of the transient.

Nodalization studies: None.

Summary: This report documents the assessment of RELAP5/MOD2 Cycle 36.04 using the data from Test L2-3 of Loss of Fluid Test (LOFT) facility.

The LOFT facility was designed to simulate the major components and system response of a commercial PWR during LOCA with various sizes. The facility consists of five components: the reactor vessel, the intact loop, the broken loop, the blowdown suppression system, and the ECCS. All components are instrumented such that variables can be measured and recorded during loss of coolant experiments.

Test L2-3 simulated a postulated LOCA resulting from a 200% double-ended offset shear break in the cold leg of the primary coolant system. At the time of experiment initiation, the LOFT reactor was operating at a 39.4 kW/m maximum linear heat generating rate, corresponding to 100% power in a typical large PWR.

To simulate the LOFT system specific to L2-3 experiment, the reactor core was modeled by two separate flow channels and the downcomer by two equally split flow channels. Three heat structures were used to describe the LOFT fuel assemblies.

The result of the base case calculation using the frozen code of RELAP5/MOD2 were compared with the experiment data in terms of loop flows, secondary side pressure, ECCS performance, reactor vessel behavior, and fuel rod thermal response. The overall hydraulic behavior was reasonably predicted, while the fuel rod thermal response was minimally predicted.

The main reasons for the discrepancies between the calculation and the experimental data are: (1) a mismatch between the calculated and measured critical break flow when the flow changes from subcooled to two phase, resulting in an under prediction of cold leg break flow and thus over prediction of coolant inventory, (2) a deficiency in the CHF correlation at high flow rate, resulting in minimal prediction of the early core heatup during blowdown, and (3) a lack of rewet criteria specific to the phenomena present during early rewet.

To re-identify the deficiencies found in the base case calculation and to determine the effectiveness of improving the rewet criteria, a sensitivity calculation was performed using an updated code with the PSI modification. The results showed that the rewet phenomena was better predicted with the PSI modified rewet criteria.

3.2.11 LOFT Large Break LOCE L2-5 Assessment

Reference: Lainsu Kao, K. S. Liang, J. L. Chiou, L. Y. Liao, S. F. Wang, and Y. B. Chen, *Assessment of RELAP5/MOD2 Using LOCE Large Break Loss-of-Coolant Experiment L2-5*, NUREG/IA-0045, April, 1992.

Code version: RELAP5/MOD2 Cycle 36.04

Facility: Loss-Of-Fluid Test Facility (LOFT) at the Idaho National Engineering Laboratory.

Objectives: Determination of the sensitivity of code input uncertainties on the calculated response to the LBLOCA response. The input parameters varied were:

1. cold leg initial temperature
2. cross-flow junction uncertainty
3. discharge coefficient
4. reflood fine mesh number
5. form loss coefficients
6. fuel gap dimensions
7. accumulator conditions
8. reflood heat transfer options
9. coolant pump operation

Major phenomena: System depressurization rate, subcooled and saturated break flow, core heat transfer as indicated by the measured cladding temperature.

Code deficiencies: Two suspected deficiencies and one deficiency were identified.

1. The suspected deficiencies are:
   a. The condensation model appeared to be inadequate during the ECCS injection period of the transient as evidenced by low cold leg and downcomer fluid temperatures.
   b. The code seemed to calculate a discontinuous heat transfer coefficient before and after the activation of the reflood model.
2. By using sensitivity studies, the code was found to be unable to calculate blowdown quench.

Base calculation: The baseline calculation showed reasonable agreement with the experimental data with the exception of the cladding temperatures at the top and bottom of the core. Because only the middle of the core was only calculated to dryout, the temperatures near the core inlet and core outlet deviated markedly from the measured data.

Sensitivity studies: The sensitivity studies resulted in only minor differences relative to the baseline calculation of the LBLOCA transient.

Nodalization studies: Two nodalization studies were performed. One addressed changing the normal junction between the broken loop hot leg and the vessel from a normal junction to a cross-flow junction. The second addressed changing the number of fine mesh nodes used in the reflood model from 8 to 32.

Summary: The assessment study was based on the data recorded in the LOFT L2-5 experiment. A baseline calculation was performed followed by several sensitivity studies.

The LOFT facility was a 50 MWt pressurized water reactor system. The facility was a 1/50-scaled representation of a prototypical PWR. The experimental facility was designed to provide capability to investigate the thermal-hydraulic and nuclear core behavior during postulated LOCA events as well as anomalous transients. The facility consisted of five major systems: reactor system, primary coolant system, blowdown suppression system, emergency core cooling system, and the secondary coolant system.

The L2-5 experiment was conducted to study the behavior of a 200% double-ended offset shear large break loss-of-coolant accident (LOCA) with the primary coolant pumps unpowered and decoupled from their flywheels. Thus, the pumps have a minimum of influence on the loop flow once the experiment is initiated and no flow surges were expected to be seen through the core unlike the core thermal-hydraulic behavior observed in LOFT experiments L2-2 and L2-3. During experiments L2-2 and L2-3 a flow surge through the core caused the core rewet. However, during experiment L2-5 the core did not rewet during the early portion of the transient. It should be noted that such a primary coolant pump coastdown is nontypical.

A baseline calculation was performed using the sequence of events reported for the experiment and using the recommended modelling options and guidance provided in the code documentation. The baseline calculation was performed beginning with break initiation and was continued to include the subsequent blowdown, lower plenum refill, core reflood, and core quench.

Major events and the timing of the events were reasonably predicted by the code. The code did not predict early rewet in agreement with the data. Important parameters such as pressure, break flow, and cladding temperature were calculated with reasonable agreement in comparison to the data. The calculated peak cladding temperature was 1112 K compared to the measured value of 1077 K.

Sensitivity analyses of the test simulation with respect to various code input options including the use of cross-flow junctions, different discharge coefficients, reflood options, and the density of the reflood mesh were studied. Also, the effect of the broken loop initial temperature, the accumulator conditions, some form loss coefficients, and the fuel gap dimensions were studied. (Note: The researchers also studied the effect of multiplying the output from the Biasi CHF model by a factor of 0.6; this calculation is not discussed in this summary report.)

It was observed that the calculated PCTs were insensitive to the selected parameters except for variations in the fuel gap dimensions. When the fuel gap dimensions were doubled the PCT was increased by 130 K.

3.2.12 LOFT Large Break LOCE L2-5 Assessment

Reference: Y. S. Bang, S. Y. Lee, and H. J. Kim, *Assessment of RELAP5/MOD2 Cycle 36.04 Using LOFT Large Break Experiment L2-5*, NUREG/IA-0032, April, 1990.

Code version: RELAP5/MOD2 Cycle 36.04

Facility: Loss-Of-Fluid Test Facility (LOFT) at the Idaho National Engineering Laboratory.

Objectives: Assess the capability of the code to calculate the important phenomena that occur during a LBLOCA.

Major phenomena: Cladding temperature, break critical flow, core CHF and rewet.

Code deficiencies: Interfacial friction suspected to be significantly overcalculated.

Base calculation: The baseline calculation of the hot channel cladding temperatures and quench temperatures were significantly too low. The mass flow rates at the downcomer inlet, the core inlet, and the cross flow junctions were unstable and exhibited oscillations with high frequencies together with large amplitudes that are believed to be caused by excessive interfacial friction.

Sensitivity studies: A sensitivity calculation was performed using a set of six updates provided by the Paul Scherrer Institute. The sensitivity calculation showed better agreement to the measured quenching temperatures and the calculated mass flow rates in the hot legs were less oscillatory than the baseline calculation.

Nodalization studies: Two nodalization studies were performed. One addressed (i) the effect of eliminating crossflow junctions between downcomer cells in the broken and intact loops at a given elevation and (ii) a finer core nodalization, i.e., the core cells were increased from 12 to 14 and the new nodalization was sized to place the cell midpoint at the elevation of the existing instrumentation. The second nodalization study was the same as the first, but with two channels in the core: one simulated the "hot" bundle and the other simulated the remaining core bundles. The "hot" bundle contained 14 cells and the average bundle contained only 6 cells.

Summary: The assessment study was based on the data recorded in the LOFT L2-5 experiment. A baseline calculation and three nodalization studies were performed followed by one sensitivity study.

The LOFT facility was a 50 MWt pressurized water reactor system. The facility was a 1/50-scaled representation of a prototypical PWR. The experimental facility was designed to provide capability to investigate the thermal-hydraulic and nuclear core behavior during postulated LOCA events as well an anomalous transients. The facility consisted of five major systems: reactor system, primary coolant system, blowdown suppression system, emergency core cooling system, and the secondary coolant system.

The L2-5 experiment was conducted to study the behavior of a 200% double-ended offset shear large break loss-of-coolant accident (LOCA) with the primary coolant pumps unpowered and decoupled from their flywheels. Thus, the pumps have a minimum of influence on the loop flow once the experiment is initiated and no flow surges were expected to be seen through the core unlike the core thermal-hydraulic behavior observed in LOFT experiments L2-2 and L2-3. During experiments L2-2 and L2-3 a flow surge through the core caused the core rewet. However, during experiment L2-5 the core did not rewet during the early portion of the transient. It should be noted that such a primary coolant pump coastdown is nontypical.

A baseline calculation was performed using the sequence of events reported for the experiment and using the recommended modelling options and guidance provided in the code documentation. The baseline calculation was performed beginning with break initiation and was continued to include the subsequent blowdown, lower plenum refill, core reflood, and core quench. For the base calculation, the reactor vessel was modelled using a split downcomer with crossflow junctions and a single core channel. Four cross flow junctions connected four downcomer cells, and one junction connected a split upper annulus. Results of the base case calculation indicated unrealistically high ECC bypass flow through the cross flow junctions between the intact side downcomer and the broken side downcomer, flow oscillations due to overpredicted interfacial shear in the rod bundle geometry, and undercalculated core heatup, poor correspondence with the measured quench temperature and no top-down quenching.

To determine the effectiveness of nodalization changes and to quantify their effects on the thermal-hydraulic responses, studies were performed for three different cases of reactor vessel modelling. They were as follows: split downcomer modelling with a crossflow junction only at the upper annulus (Case A), finer axial modelling of the core (Case B), and two core channel modeling (Case C). The elimination of the downcomer crossflow junctions resulted in a correct ECC bypass flow. This change was kept for the following calculations. Adding two extra cells in the single core channel to better represent measurement locations did not result in better calculated/measured cladding temperature correspondence. Modelling the core with two channels resulted in increased calculated cladding temperatures but still did not provide good correspondence with the data.

The last calculation incorporated model changes attributed to the Paul Scherrer Institute. It was found that the changed interfacial shear in the rod bundle and the heat transfer correlations during the reflood phase resulted in reduced nonphysical flow oscillations and accurate calculation of the quenching temperature. However, the cladding temperature was overcalculated after departure from nucleate boiling until just prior to quench. The intact loop pump speed was not calculated correctly during any of the studies.

3.2.13 LOFT Large Break LOCE LP-02-6 Assessment

Reference: D. Lubbesmeyer, *Post-Test Analysis and Nodalization Studies of OECD LOFT Experiment LP-02-6 with RELAP5/MOD2 Cycle 36-02*, NUREG/IA-0088, (to be published).

Code version: RELAP5/Mod2 Cycle 36.02

Facility: Loss-Of-Fluid Test Facility

Objectives: Evaluation of code capability to capture the bottom-up quenching behavior that occurred early in the transient (5 to 10 seconds into blowdown)

Major phenomena: Bottom-up rewetting during the blowdown phase of the LBLOCA transient.

Code deficiencies: A potential code deficiency was identified in that the code identified two different flow regimes in a volume dependent on whether the code was determining the interfacial shear stresses and interfacial heat transfer or the wall heat transfer.

User guidelines: Detailed nodalization provided a more accurate calculation of the fuel cladding temperature during the large break LOCA.

Base calculation: A base calculation was performed to provide a basis for nodalization studies that followed.

Sensitivity studies: None.

Nodalization studies: Nodalization studies were conducted specifically to investigate the effect of coarser noding in the pressurizer, the steam generator secondary and the intact loop piping. The study also evaluated the effect of coarser radial noding in the fuel rods.

Summary: The assessment study gives the results and analyses of nine post-test calculations of the LOFT LP-02-6 experiment using several nodalizations. The author began with a "standard" nodalization comparable to that used by the code developers at INEL, the number of volumes and junctions were reduced (especially in the pressurizer, the steam generator secondary side, and the intact loop. Additionally, the number of radial zones in the fuel rods were reduced for other nodalization studies.

The LOFT facility was a 50 MWt pressurized water reactor system. The facility was a 1/50-scaled representation of a prototypical PWR. The experimental facility was designed to provide capability to investigate the thermal-hydraulic and nuclear core behavior during postulated LOCA events as well as anomalous transients. The facility consisted of five major systems: reactor system, primary coolant system, blowdown suppression system, emergency core cooling system, and the secondary coolant system.

Experiment LP-02-6 was conducted October 3, 1983, in the LOFT facility. It was the first large break loss-of-coolant accident simulation and the fourth experiment overall conducted under the auspices of the OECD. It simulated a double-ended offset shear of one inlet pipe of a four loop PWR coincident with loss of offsite power. The experiment addressed the response of a PWR to conditions closely resembling a USNRC "Design Basis Accident" in that prepressurized fuel rods were installed and minimum US emergency coolant injections were used.

During this experiment, the cladding temperatures remained lower than 1060 K. This resulted from high mass flow, early bottom-up rewetting during the blowdown phase of the experiment between 4.5 and 8 s after opening the break valves. Additionally, in the upper part of the core, heat-up may have been delayed due to partial bottom-up quenching between 15 and 18 s of the experiment.

For the plant analyzed, the "adequate nodalization" is usually unknown and only some very rough criteria are given to the code user. This makes the accuracy of a prediction strongly related to the "experience" of the code user, a quite unsatisfactory conclusion. Therefore, the LP-02-6 experiment was analyzed with different nodalizations of the LOFT system. Starting with a nodalization similar to that used by the INEL code developers (developed for small break LOCAs), the number of volumes, junctions, and heat structures in the primary loop of the LOFT system was reduced by nearly one-half. The entire vessel was unchanged to meet the requirements of the given experimental axial positions, especially for the cladding temperature measurements. Also, the (i) influence of fine meshing in the core zone during reflooding on quench time and temperature and (ii) influence of the time for reflood initialization with respect to the code's predicting capabilities of the quench phenomena were investigated.

The code gave a reasonable calculation of the overall thermo-hydraulic behavior of experiment LP-02-6 although it failed to predict the early bottom-up rewetting which happened between 4 and 8 s of the transient (blowdown phase) quenching the whole core. Independently of the chosen nodalization, most of the investigated parameters like pressures, mass flows in the broken and intact loops, pump speed, and ECC systems have error bounds less than $\pm$ 20 %. However, the cladding temperatures usually have been both over- and under-predicted (depending on the investigated core level) up to 150 K. For all nodalizations, the hot spot has been calculated at a position downstream of experimentally inferred position 0.686 m from the bottom of the core. The computer code always calculated the hot spot at axial level 31. The good agreement of most of the code results with the measured LOFT data is not really surprising because the code has been extensively used to eliminate insufficiencies both in the codes and in the plant specific nodalization for the input model. One has to be aware that both the code and LOFT specific nodalization (also used here as the basic nodalization scheme), are somewhat "LOFT tuned" which resulted in quite acceptable results.

With respect to the computation time, the degree of specification of the nodalization (the number of volumes and junctions) is, of course, an important factor. However, a faster calculation does not always result from a lower number of volumes and junctions. Sometimes the reduction in computer time resulting from reduction of nodalization is small.

For large break LOCA's, the nodalization seems important only for the cladding temperatures, where significant differences can be observed for the different nodalizations under investigation. But, opposite to our findings when analyzing the LP-LB-1 experiment (see Lubbesmeyer, NUREG/IA-0089) the differences in the times of final quenching are usually rather small and within a band of less than ten seconds. Also for the other parameters, the deviations between the results of the calculations with the different nodalizations under investigation remain relatively small.

The time of initiating the reflood option determines the "quench behavior" of the code because it starts the fine-meshing in the core-zone. It thus enables a more correct tracing of the axial cladding temperature distribution and consequently a better reflood modeling. Therefore, the comparison of two of three possible methods of initiating the reflood option have manifested a strong dependence of the results on the reflood option setting. An external trip based on only the fluid level in the core leads to much lower values of the cladding temperatures at nearly all axial levels of the LOFT core. But still, the early bottom-up rewetting was not correctly calculated. It is the bottom-up rewetting that quenches the whole core within the first 4.5 to 8 s of the experiment and therefore has a very important influence on the behavior of the whole system

Early bottom-up rewetting is probably a consequence of the coast-down behavior of the primary pumps. RELAP5/MOD2 has given an indication of this dependence. Looking at the different mass fluxes as calculated by the code using different assumptions about the pump coast-down behavior, one easily observes the strong relationship between coast-down behavior and mass flux. Rapid pump coast down leads to much lower core in and out mass fluxes than normal coast-down

Finally, Lubbesmeyer noted that during the refill phase of the LP-02-6 LOCE the code selected different flow regimes on the one hand for its calculation of the interfacial shear stresses and interfacial heat transfer and on the other hand for the determination of the wall heat transfer. At the same axial position and at the same time the code indicated both wet and dry surfaces by defining mist flow and slug flow for the same volume. In this instance, RELAP5/MOD2 logic assumed both wet and dry surface by defining mist flow and slug flow for the same volume at the same time. It should be noted that this apparent inconsistency is due to the manner in which RELAP5 uses independent logic for the flow regimes and heat regimes.

3.2.14 LOFT Large Break LOCE LP-LB-1 Assessment

Reference: D. Lubbesmeyer, *Post-Test Analysis and Nodalization Studies of OECD LOFT Experiment LP-LB-1 with RELAP5/MOD2 Cy 36.02*, NUREG/IA-0089, (to be published).

Code version: RELAP5/Mod2/36.02

Facility: Loss-Of-Fluid Test Facility

Objectives: Evaluation of the code capability to capture the top-down quenching phenomena that was observed in the LOFT LP-LB-1 experiment was the main objective.

Major phenomena: Top-down rewetting during the blowdown phase.

Code deficiencies: A potential code deficiency was identified in that the code identified two different flow regimes in a volume dependent on whether the code was determining the interfacial shear stresses and interfacial heat transfer or the wall heat transfer.

User guidelines: When modeling the fuel rod the number of radial meshes should be: 10 for the high power fuel rods, and 5 for the lower power fuel rods. This is the scheme used at the INEL as a modeling guideline.

Base calculation: A base calculation was performed using the recommended nodalization scheme and the frozen code version.

Sensitivity studies: None.

Nodalization studies: Several nodalization changes were made to the initial input model to determine impact on the simulation of the LOFT LP-LB-1 transient and specifically the calculation of the top-down rewetting of the fuel during the late blowdown/early refill period of the transient.

Summary: Experiment LP-LB-1, conducted February 3, 1984, in the Loss-Of-Fluid-Test (LOFT) facility was the data base for the nodalization study reported here. This test was run under the auspices of the OECD. It simulated a double-ended offset shear of one inlet pipe of a four loop PWR and was initiated from conditions representative of licensing limits in a PWR. Additional boundary conditions for the simulation were loss of offsite power, rapid primary coolant pump coastdown, and UK minimum safeguard emergency core coolant injection rates.

The LOFT facility was a 50 MWt pressurized water reactor system. The facility was a 1/50-scaled representation of a prototypical PWR. The experimental facility was designed to provide capability to investigate the thermal-hydraulic and nuclear core behavior during postulated LOCA events as well as anomalous transients. The facility consisted of five major systems: reactor system, primary coolant system, blowdown suppression system, emergency core cooling system, and the secondary coolant system.

During this experiment, all fuel rods in the central fuel assembly (box 5) experienced temperatures in excess of 1100 K in their high power regions (about 24 inches from the bottom of the core). The maximum cladding temperatures reached peak values of 1261 K during blowdown and 1257 K during refill/reflood which were the highest temperatures ever measured in LOFT. The core-wide temperature increase continued until a partial core top-down quench occurred, starting at 13 s, which affected the top third of the core. This top-down rewetting was one of the key-phenomena of the LOFT experiment LP-LB-1.

When a plant is originally analyzed the "adequate nodalization" is usually unknown and only some very rough criteria can be given to the code user. This may make the accuracy of a prediction strongly related to the "experience" of the code user, a quite unsatisfactory situation. Therefore, the author analyzed the LP-LB-1 experiment using different nodalizations of the LOFT system. Starting with a nodalization similar to the one use by the code developers at INEL (especially developed for small break LOCAs), the author reduced the numbers of volumes, junctions and heat structures in the primary loop of the LOFT system to nearly half whereas the entire vessel stayed unchanged to meet the requirements of the given experimental axial positions, especially for the cladding temperature measurements. Also, 1) the influence of fine meshing in the core zone during reflooding on quench time and temperature, and 2) the influence of the time for reflood initialization with respect to the code's predicting capabilities of the rewetting phenomena were investigated.

The code calculated the general thermo-hydraulic behavior of experiment LP-LB-1 reasonably although it failed to describe the top-down rewetting which happened in the upper third of the core between 15 and 20 s of the transient (blowdown phase). Independently of the chosen nodalization, most of the investigated parameters like pressures, mass flows in the broken and intact loops, pump speed, and ECC systems have error bounds less than $\pm$ 20 %. However the cladding temperatures usually have been under-predicted between 10 and up to 150 K (hot spot). The good agreement of most of the RELAP5/MOD2 results with the measured LOFT data is not really surprising because the code has been extensively used to eliminate insufficiencies both in the codes and in the plant specific nodalization for the input model. One has to be aware that both the code and LOFT specific nodalization (also used here as the basic nodalization scheme), are somewhat "LOFT tuned" which resulted in quite acceptable results.

With respect to the computation time, the degree of specification of the nodalization (the number of volumes and junctions) is of course an important number. However, a faster calculation does not always result from a lower number of volumes and junctions. Sometimes the reduction in computer time resulting from reduction of nodalization is small because of numerical instabilities.

The cladding temperatures are usually under-predicted as stated above. In addition, for all nodalizations, the hot spot has been calculated downstream from the experimentally inferred position, 0.61 m from the bottom of the core. The code always calculated the hot spot at axial level 31.

For large break LOCA's, the nodalization seems to be important only for the cladding temperatures, where significant differences are observed for the different nodalizations under investigation. Especially, the times of final quench differ from nodalization to nodalization by 20 to 30 s.

For the other parameters, the deviations between the results of the calculations with the different nodalizations under investigation have error bounds of less than ± 20 %. Surprisingly, the calculated results using less detail than the base nodalization usually seem to be closer to the experimental data. The base nodalization is similar to the INEL nodalization of LOFT.

A possible problem with the computer code seems to be in modeling the stored energy of the vessel material, especially in relation to calculating the time of final quenching. When accounting for the heat capacity of the downcomer walls as well as other core material, the predictions have been found to compare worse than when these effects are neglected.

The modeling of the fuel rod (number of radial meshes) has shown an important influence on the cladding temperatures as well as on the center fuel temperatures. Compared to the equivalent results obtained using other nodalizations, the temperature traces when the radial meshes are reduced from 10 to 5 (hot) and 5 to 4 (average) differ significantly at very low and very high core elevations. But the influence of the different nodalizations had a small influence on the other thermo-hydraulic parameters.

The time point of initiating the reflood option determines the "quench behavior" of the code because it starts the fine-meshing in the core zone. A better initiation time enables a more correct tracing of the axial cladding temperature distribution and consequently better modeling of the reflood phase. A comparison of three methods of initiating the reflood option show the results have a strong dependence on the initiation method.

The results of RELAP5/MOD2 calculations using either of the two code-internal trips for the initiation of the reflood option are identical.

An external trip based solely on the fluid level in the core lead to much lower values of the cladding temperatures at nearly all axial levels of the LOFT core. Still, the top-down rewetting in the upper third of the core was not calculated correctly. (The "good" results at level 43.8 seem to be coincidental.)

3.2.15 LOFT Intermediate Break LOCE L5-1 Assessment

Reference: E. J. Lee, B. D. Chung, and H. J. Kim, *ICAP Assessment of RELAP5/MOD2 Cycle 36.04 Using LOFT Intermediate Break Experiment L5-1*, NUREG/IA-0069, April, 1992.

Code version: RELAP5/Mod2/36.04

Facility: Loss-Of-Fluid Test Facility

Objectives: Assess the code's capability to simulate an intermediate break LOCE.

Major phenomena: A blowdown/refill process due to IBLOCA was studied. Depressurization and voiding throughout the system followed the break. Dryout occurred and was followed by core thermal excursion. The overall phenomena resembled those of LBLOCAs except that the rates of depressurization and the liquid drain-off in the core were slower, and the core was uncovered for a shorter time period.

Code deficiencies: The code could not be run after the accumulator component emptied without removing the accumulator component altogether from the model. Although the users suspected the model based on modified Zuber CHF correlation deficient for low mass fluxes, evidence supporting this view was sketchy.

User guidelines: Three user guidelines were listed by the authors. However, their guidelines were judged to be indistinct, may be specific to their particular interest, and are awaiting corroboration by other users.

Base calculation: The base calculation was performed using a model nodalization based on the developmental assessment studies described in Ransom, 1985. Neither the reflood nor gap conductance model options were used in the base case.

Sensitivity studies: Two sensitivity studies were completed to evaluate the performance of the code when the reflood and gap conductance options were activated. The authors' conclusions were that the reflood option alone should be used for such a calculation. Reasonable results were obtained.

Nodalization studies: The nodalization studies were designed to evaluate the effect of simulating the core behavior using (i) the base case nodalization of a single flow channel with two heat structures, (ii) a single channel with a single heat structure and (iii) two flow channels with two heat structures. The same number of axial nodes was used in all cases. The authors found the base case nodalization to give the best agreement with the data.

Summary: RELAP5/MOD2 Cycle 36.04 was assessed against the experimental data of the Loss of Fluid Test facility (LOFT) L5-1 Intermediate Break Loss of Coolant Accident (IBLOCA). The objectives of the assessment were to show the applicability of using the code for LOFT IBLOCA Test L5-1 and similar transients of a typical PWR IBLOCA and to optimize the modelling software.

The LOFT is a 50 MWt integral effect test facility designed to simulate the responses of the major components and system responses of a commercial four-loop

PWR during loss of coolant accidents. The intermediate break experiment series L5 was designed to identify and evaluate the LOFT system thermal-hydraulic response during an IBLOCA caused by a simulated 11.2-inch I.D. accumulator line rupture.

A base case model and four sensitivity calculations were performed. In the base case, the core was modeled by a single flow channel and two heat structures and without the reflood and gap conductance options activated. Two nodalization sensitivity studies and two model option sensitivity studies were carried out to investigate the following effects on the PCT predictions: single flow channel and single heat structure, two flow channels and two heat structures, reflood option added, and both reflood and gap conductance options added.

The nodalization for the base case consists of 130 volumes, 136 junctions and 143 heat structures. The input deck was basically equivalent to that used for the LOFT L3-7 simulation by E. J. Lee et al. 1981.

Prior to the break, the LOFT was operated at a thermal power of 45.9 MWt, vessel temperature difference of 27.0 K, a core mass flow rate of 308.2 kg/s, and a system pressure of 14.93 MPa. The duration of the transients simulated was 300 seconds.

Depressurization occurred immediately following the break. The events observed in the experiment are as follows. At 0.17 seconds, the reactor was scrammed and the secondary side inlet/outlet valves started to close. At 0.2 seconds, the water at the upper plenum reached saturation. At 0.4 seconds, the high pressure injection system (HPIS) set point of 10.6 MPa was reached. HPIS started 2.88 seconds later. The primary coolant pump tripped at 4.0 seconds and the water in the broken loop cold leg reached saturation at 10.5 seconds. At 12.1 seconds the steam control valve closed. The pressurizer became empty at 15.5 seconds. The fuel cladding thermal excursion started at 108.4 seconds. The accumulator A injection started at 185.8 seconds while the fuel cladding temperature continued to climb until a peak of 715 K was reached at 198 seconds. the low pressure injection system (LPIS) started at 201 seconds and eventually repressurized the system.

The base case results reasonably matched the experimental data. For areas where the experimental data was lacking, the calculations indicated: (1) the water supplied from the accumulator and the LPIS more than compensated for the loss of reactor coolant inventory and repressurized the system, (2) the accumulator and the LPIS influenced safety more than HPIS for IBLOCA, and (3) significant core uncovery occurred but this was later reversed by the accumulator flow.

There are two noticeable discrepancies. First, the calculated PCT was too low and occurred too late. Secondly, the fuel cladding temperature measurements in a peripheral assembly indicated an early heatup, quenching, re-heatup and final quenching. The calculations did not catch the early heatup.

The authors concluded that:

1. The base case results reasonably predicted the LOFT IBLOCA L5-1 experimental data.

2. The single flow channel/two heat structures core model yields better PCT prediction than the single flow channel/single heat structure and the two flow channels/two heat structures core models.

3. It is preferable to use the reflood option alone and not together with the gas conductance option for IBLOCA applications.

4. Either the over estimation of the system water inventory or the inadequacy of the prediction of CHF occurrence caused the delay in the calculated dryout time.

5. One dimensional modeling is the cause for the code not being able to predict the experimentally observed early peak peripheral cladding temperature.

3.2.16 LOFT Small Break LOCE L3-5 Assessment

Reference: J. Eriksson, *Assessment of RELAP5/MOD2, Cycle 36.04 Against LOFT Small Break Experiment L3-5*, NUREG/IA-0037, March, 1992.

Code version: RELAP5/MOD2 Cycle 36.04.

Facility: Loss-of-Fluid-Test Facility (LOFT), Idaho Nation-l Engineering Laboratory.

Objectives: Study the sensitivity of an integral effects exper t simulation to changes of steam generator modeling and of core bypass flow. rovide a basis for comparison with the L3-6 experiment; L3-5 was conducted with the reactor pumps off and the L3-6 experiment was conducted with the reactor pumps on. The objectives were met.

Major phenomena: Primary pressure response, fluid temperatures, break mass flow rate, and primary to secondary interactions.

Code deficiencies: None.

User guidelines: None.

Base calculation: The author concluded that the transient predictions compared reasonably well with the experimental data.

Sensitivity studies: Two sensitivity calculations were conducted:

1. The steam generator shell region model was modified to increase the void fraction with increasing elevation. However, only a limited improvement over the base calculation was observed.
2. The modelled downcomer to upper plenum leakage was split into two junctions; improvement was noted in the clad temperature and break fluid density comparisons with data.

Nodalization studies: None.

Summary: The basis of the assessment study was the LOFT L3-5 experiment. The L3-5 experiment was conducted in conjunction with the L3-6 experiment to study the effect having the reactor coolant pumps off and on respectively during a small break loss-of-coolant accident (SBLOCA).

LOFT was a 50 MWt integral effect test facility designed to simulate the responses of the major components and system responses of a commercial four-loop PWR during either loss-of-coolant-accidents or operational transients. LOFT contained a half-height nuclear-powered core and was a 1/50-volumetrically scaled system.

The LOFT L3-5 experiment simulated the behavior of a 4-inch diameter SBLOCA in a four-loop Westinghouse reactor with a rated core power of 1000 MWe. Shortly after the break occurred the reactor coolant pump was stopped. The experiment simulated injection from the high pressure injection system (HPIS), but the experiment was terminated prior to reaching the low pressure injection system (LPIS) setpoint.

A base case and two modelling sensitivity studies were performed to study examine the importance of the model nodalization in the steam generator and the core bypass.

The base case calculation underpredicted the primary system pressure, following subcooled depressurization, until about 900 s after the break. Discrepancies between the primary and secondary side temperatures affected the depressurization rate adversely when compared to data.

A sensitivity study on the steam generator modelling was performed. Main steam valve leakage was used that more closely matched the experimental conditions. This change improved pressure drop comparisons. A change in the steam generator downcomer level produced only slightly better agreements with data.

Modelling changes to the core bypass leakage paths were made in an attempt to terminate loop flow, as seen in the experiment, and to lower the pressure difference between the vessel inlet and outlet. The changes had a positive impact on core fluid distribution, leading to a better prediction of core clad temperature.

3.2.17 LOFT Small Break LOCE L3-5 and L3-6 Assessments

Reference: A. H. Scriven, *Application of the RELAP5/MOD2 Code to the LOFT Tests L3-5 and L3-6*, NUREG/IA-0060, April, 1992.

Code version: RELAP5/MOD2 cycle 35.05 Winfrith Version E03.

Facility: Loss-of-Fluid Test facility

Objectives: The study was performed to evaluate the capability of the code to simulate a small break loss-of-coolant accident in a pressurized water reactor.

Major phenomena: The study focuses on the primary-to-secondary heat transfer during natural circulation conditions in the primary system including reflux heat transfer in the steam generator, and countercurrent flow in horizontally-stratified hot legs.

Code deficiencies: The code was observed tc over predict interphase drag.

User guidelines: (i) Small break modeling was recommended as follows: The break was modeled using a cross-flow junction connected to a horizontal pipe that terminated in a valve connecting to a time-dependent containment volume. A break discharge coefficient of 0.84 was used and the Ardron-Bryce offtake entrainment model was used at the break. (ii) Expected valve behavior should be represented as closely as possible as the calculation was very sensitive to small changes in valve conditions.

Base calculation: The code performed well for both of the test cases in this assessment study.

Sensitivity studies: Sensitivity studies were performed to investigate effects of modeling considerations for the steam outlet valve, of injecting HPIS coolant directly into the downcomer, of break modeling changes, and of pump speed control and inertia modifications.

Nodalization studies: A nodalization study was performed relative to the modeling of the break region.

Summary: This report assesses RELAP5/MOD2 Cycle 36.05 (Harwell Version E03) through a comparison of calculated results with LOFT Experiments L3-5 and L3-6. These experiments simulated the response of a pressurized water reactor to a small break loss-of-coolant accident. For both tests a cold leg break with a flow area equal to 2.5% of the cold leg cross section was simulated. The experiments were performed with comparable conditions and differed with respect to operation of the main coolant pumps. In Test L3-5 the pumps were tripped shortly after the break opened. In test L3-6 the pumps were allowed to continue operating until approximately 40 minutes after the break opened, at which time they were tripped. The purpose of the tests was to evaluate the effect of tripping, as compared with not tripping pumps, following a small break LOCA. The tests showed a potential for continued pump operation to result in significant core damage if the pumps are later tripped or fail due to cavitation effects.

LOFT was a 50 MWt integral effect test facility designed to simulate the responses of the major components and system responses of a commercial four-loop PWR during either loss-of-coolant-accidents or operational transients. LOFT, with a half-height nuclear-powered core, was a 1/50-volumetrically scaled system.

The model was derived from one originally created by INEL for RELAP5/MOD1 to model the LOFT LP-SB-03 experiment. For the calculations reported here the input for the steam outlet valve was completely rewritten (a sensitivity to this modeling was identified), the high pressure injection was modified to flow into the reactor vessel downcomer (in the same manner as in the experiment), the break modeling was changed, and the pump speed control and inertia were modified (to be compatible with the test procedure and in response to an identified sensitivity to the pump inertia).

An extensive study was performed to determine the best manner to model the main coolant pump inertia. For overall experiment prediction, the inertia is important for those cases where the pumps are tripped early, near the time of scram, such as in Test L3-5. This is so because the pump coastdown occurs during a period when the core stored energy is being removed, and any misrepresentation of the coastdown is therefore reflected in the overall system parameters. Normally, pump inertial effects are simple to model. However, in the LOFT experiments the pump inertial effects were complicated because initially the pump is connected to a flywheel then decoupled as the pump speed drops to about 70 radians/s. Moreover, there is an uncertainty regarding the true combined inertia of the pump and flywheel. LOFT documentation specifies a combined inertia of 316.04 kg-m^2, a value that this assessment indicates is much too high. This reviewer suspects that value may actually be the combined inertia in lbm-ft^2 rather than in kg-m^2. If this is the case, then the actual combined inertia is 13.32 kg-m^2, a value much more in line with the remainder of the analysis. "In summary there exists an uncertainty in how to model the pump inertia. This has not been fully resolved. The value of 4 kg-m^2 was used for these calculations, but a better model may be to use the variable inertia model with the speed dependent terms set to zero giving two values for the inertia, 10 kg-m^2 before decoupling, and 1.43 kg-m^2 after decoupling."

A variety of break modeling options were considered. The scheme decided upon used a crossflow junction, with the modified Ardron-Bryce entrainment break offtake model, that removed flow from the main coolant pipe and emptied into a horizontal pipe. A valve at the other end of the pipe discharged to a time dependent volume representing the containment. A two-phase discharge coefficient of 0.84 was used at the valve junction.

Suspected anomalies regarding the LOFT upper plenum bypass path were uncovered. Specifically the experimental data was found to indicate the bypass is both smaller, and located lower in the reactor vessel, than previously thought. The overall behavior during small break LOCAs is particularly sensitive to the size and locations of this bypass. The comparison of experimental and calculated data therefore suffered from bypass uncertainties. Two calculations, one with maximum, and one with no bypass were therefore performed for test L3-5 (pumps off) to bracket the possible bypass effects.

A sensitivity of the experimental results was noted to relatively minor aspects of main steam line valve operation. Small errors in the prediction of the steam flows during the period shortly after scram were found to greatly affect the primary-to-secondary heat transfer and therefore the observed system behavior. The model was modified to match the 5%/s maximum valve change rate, starting from a 60% open position. A linear relation between valve flow area and stem position was assumed in the model, ignoring some non-linearity in this relation. In addition, a valve-closed leak of 0.5% was used for L3-6; 0.25% was used for L3-5. This valve modeling scheme was considered the best available, given the uncertainties in actual valve performance. It was noted that it is not generally understood by modelers that it is important to match the initial valve position and steam mass flow rate, and then model accurately the exact manner in which the valve is closed from the initial position.

Reasonable matches between calculated and measured primary and secondary pressures, flows, temperatures, and levels were attained. The author reported that the experimental data and the RELAP5/MOD2 calculation indicate that reflux natural circulation was experienced during the test. The RELAP5 time step control logic, that continually divides the time step size by 2 until a suitable solution is attained, was faulted for leading to excessive computational costs. A suggestion was made to alter the logic to allow the solution to proceed at time steps nearer the Courant limit than currently.

For Test L3-6, the pumps-on experiment, a generally good comparison between calculated and measured data was obtained. The comparisons of hot and cold leg densities, break flow, total system inventory, and pressurizer level were particularly favorable. On the negative side, the test data shows stratified hot leg conditions by 600 s; RELAP5 did not indicate stratification until about 1000 s. This is consistent with a known need to improve RELAP5 criteria for transition to stratified conditions. Other disagreements between the code calculation and experiment concerned the steam generator secondary-side level indications and the draining of the steam generator U-tubes primary-sides. It is believed the discrepancy resulted from RELAP5 overprediction of interphase drag in bubbly flow. This effect is suspected of causing an underprediction of the initial steam generator secondary side mass and an overprediction of the time needed to void the U-tube primaries and thermally-decouple the primaries and secondaries. However, neither of these discrepancies was particularly significant for prediction of the L3-6 experiment.

Two calculations were performed for the L3-5 experiment, the pumps-off test. One of these calculations was without the upper plenum bypass in the model and one was with the bypass. The case without the upper plenum bypass proved a much better comparison with the test data. It was therefore concluded that the no-bypass situation more closely represented that in the test facility. However, because of the uncertainties regarding the bypass, the usefulness of the L3-5 code/data experiment comparison is reduced. Generally-favorable comparisons between code and experimental data are indicated for the primary system pressure, and in the break mass flow rate.

One of the purposes of calculating the L3-5 test was to assess the RELAP5 modified break offtake model. However the level did not remain near the break location for an appreciable period of time, and therefore this experiment was not a challenging test of the offtake model.

3.2.18 LOFT Small Break LOCE L3-6 Assessment

Reference: J. Eriksson, *Assessment of RELAP5/MOD2, Cycle 36.04 Against LOFT Small Break Experiment L3-6*, NUREG/IA-0033, July, 1990.

Code version: RELAP5/MOD2, Cycle 36.04.

Facility: Loss-of-Fluid-Test Facility (LOFT), Idaho National Engineering Laboratory.

Objectives: Evaluate steam generator secondary side boiler region nodalization and feedwater valve timing with respect to secondary side liquid mass. Also, two-phase pump characteristics were studied. Provide a basis for comparison with the L3-5 experiment; L3-5 was conducted with the reactor pumps off and the L3-6 experiment was conducted with the reactor pumps on. The objectives were met.

Major phenomena: Primary pressure response, fluid temperatures, break mass flow rate, and primary to secondary interactions.

Code deficiencies: None.

User guidelines: None.

Base calculation: The base calculation compared well with the experimental data. Shortcomings of the calculation included a secondary water level that was too low and some deficiencies in the two-phase pump head calculation.

Sensitivity studies: Two sensitivity calculations were undertaken to improve the secondary water level and the pump two-phase pump head characteristics. Although an improvement was noted locally for both changes, no substantial improvements were observed elsewhere in the system calculation.

Nodalization studies: None.

Summary: The assessment was based on the LOFT L3-6 experimental data. LOFT was a nuclear powered, scaled-down PWR experimental facility. The facility represented all of the major components in a commercial PWR including ECCS. The LOFT L3-6 experiment was a small break transient with the break occurring in the cold leg. The break size was equivalent to a 4-inch break in a commercial PWR. The transient scenario was the same as the LOFT L3-5 small break experiment, except the reactor coolant pumps were not tripped, but allowed to run through the duration of the test.

Three calculations were performed; a baseline calculation and two sensitivity calculations. In the baseline calculation neither the code nor the input model was modified. One of the sensitivity calculations addressed the calculated liquid level in the boiler region of the steam generator. The other sensitivity calculation addressed the undercalculation of the reactor coolant pump two-phase head.

The baseline calculation showed, in general, good agreement with the measured data. Figure 3.2.18.1 shows the comparison of break flow, while Fig. 3.2.18.2

shows the comparison of upper plenum pressure (data versus Case A). The calculated values are within the uncertainty of the measured data. However, the baseline calculation showed differences in the steam generator secondary side downcomer liquid level (Fig. 3.2.18.3, data versus Case A) and reactor coolant pump differential pressure (Fig. 3.2.18.4, data versus Case A).

The assessment shows nodalization changes made to the input model to improve the steam generator secondary side downcomer liquid level calculation. The flow in the steam generator boiler region was directed vertically as opposed to zig-zagging across the boiler region. This orientation allowed a different void rise in the boiler and increased the liquid mass in the boiler region. In addition, the data suggested the feedwater valve closed later than was input in the input model. Thus the time to feedwater closure in the input model was increased to be more representative of the data. As a result of these changes, the calculated downcomer liquid level was more representative of the data (Fig. 3.2.18.3, data versus Case B), but still not as good as desired. The author concluded that the reason for the calculated low downcomer level was not fully understood. Analysis should be performed to further investigate this discrepancy.

The assessment also documents changes made to the input model to address the difference in the reactor coolant pump differential pressure. The two-phase pump head was undercalculated. References suggested less degradation for the void fraction range when compared with the pump characteristics of the input for the LOFT baseline input. As a result, pump characteristics used by Grush, et al., 1984, were applied to the input model and the calculation rerun. The new pump data resulted in a better calculation of the pump head as shown in Fig. 3.2.18.4 (data versus Case C). However, those phenomena associated with the effects of two-phase pump head phenomena, such as loop seal level and vessel downcomer liquid level, were unchanged.

In conclusion, the results of the baseline calculation of the LOFT L3-6 experiment compared well with the measured data. In those areas where the calculation was not as good, such as steam generator secondary side downcomer liquid level and reactor coolant pump differential pressure, improvements to the nodalization, event timing or component characteristics showed better agreement with the data relative to that phenomena.

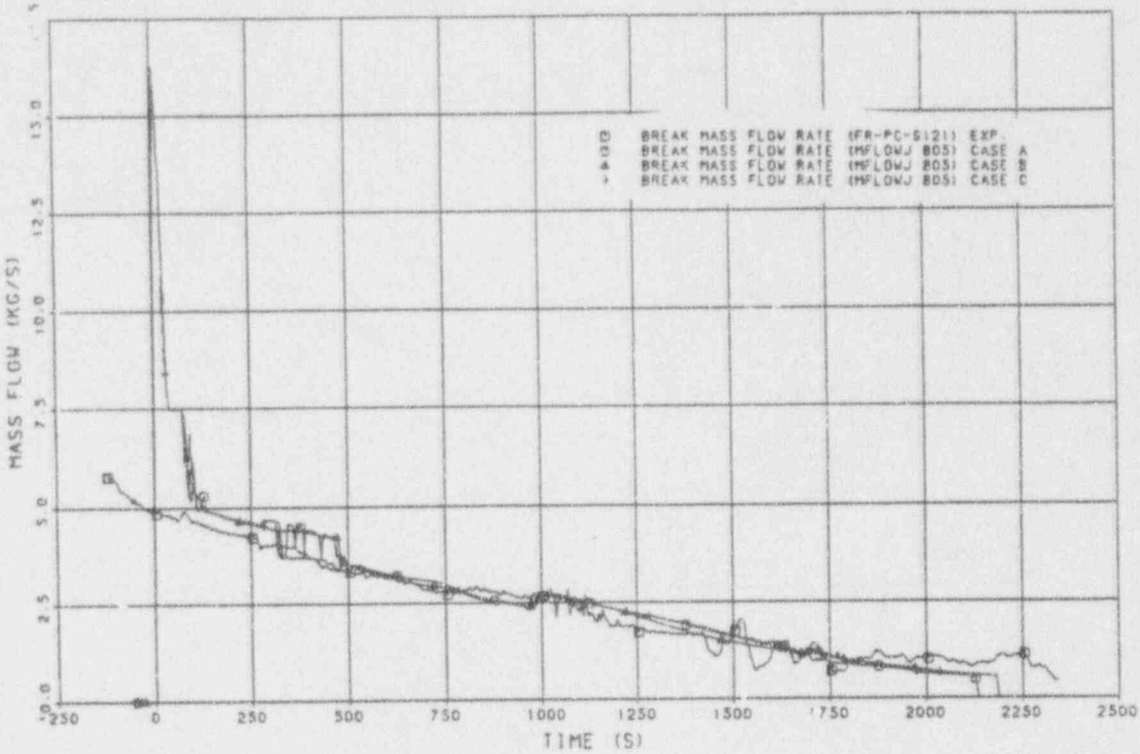


Figure 3.2.18.1 LOFT L3-6 Experiment: Calculated and Measured Break Mass Flow Rate.

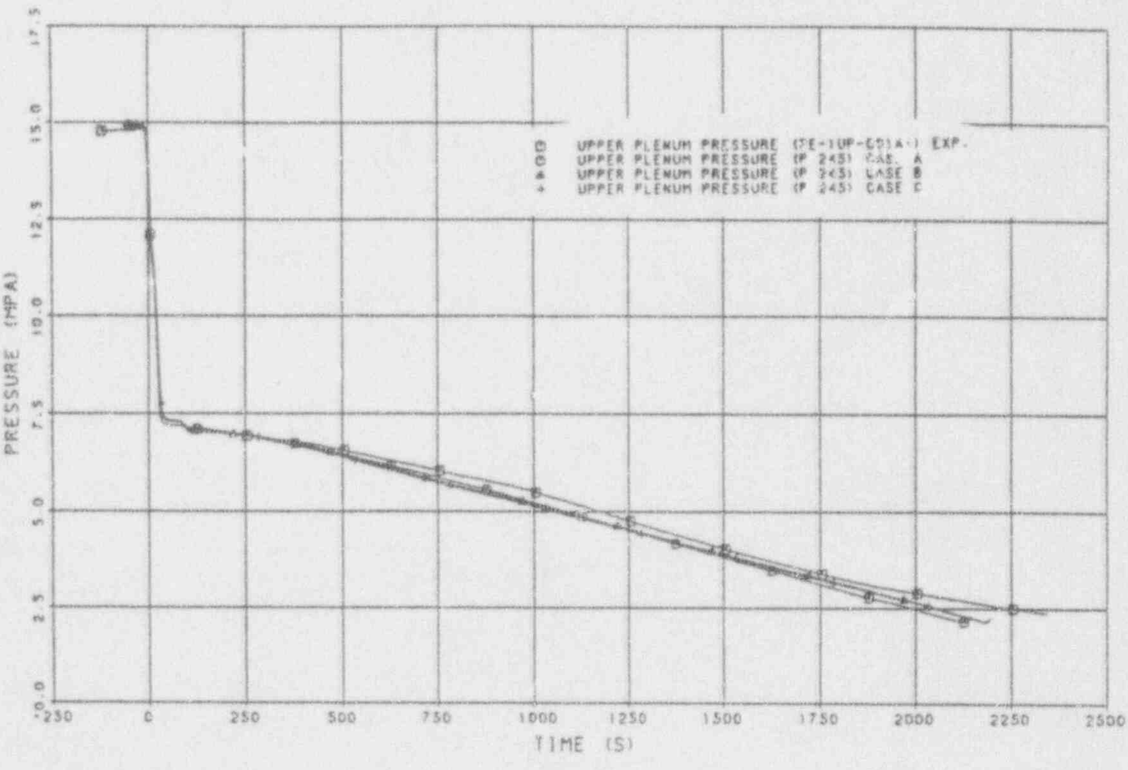


Figure 3.2.18.2 LOFT L3-6 Experiment: Calculated and Measured Upper Plenum Pressure.

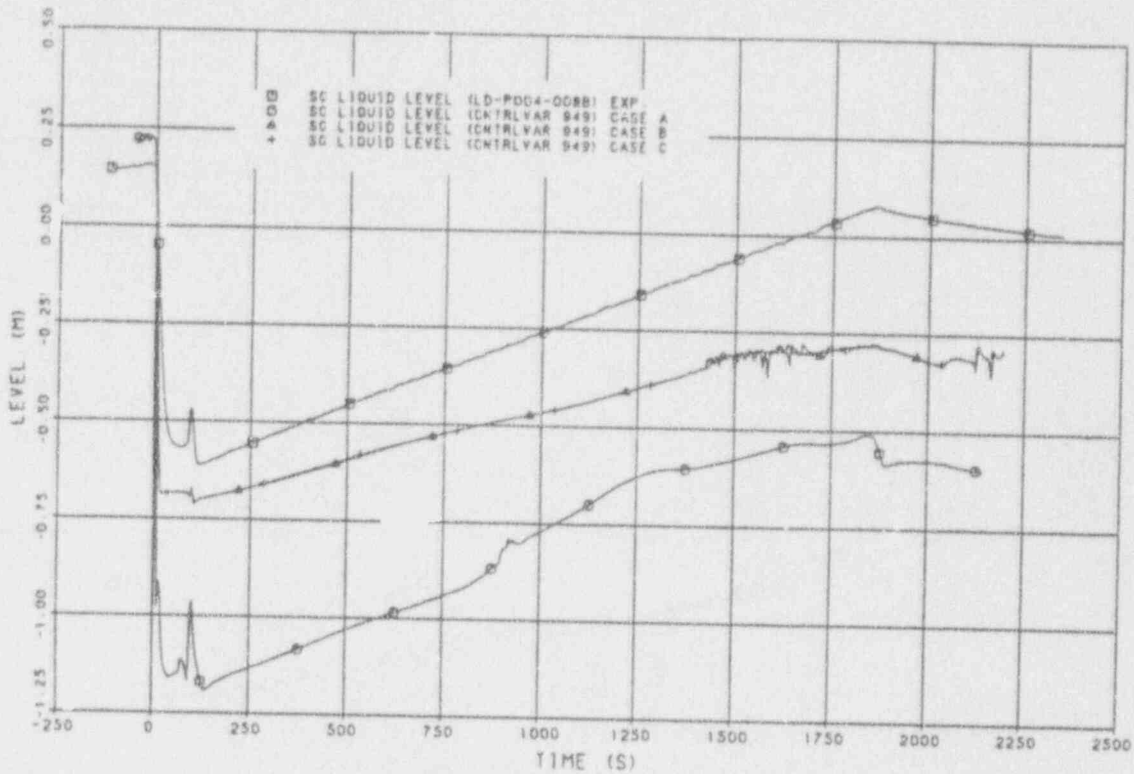


Figure 3.2.18.3 LOFT L3-6 Experiment: Calculated and Measured Secondary Liquid Water Level.

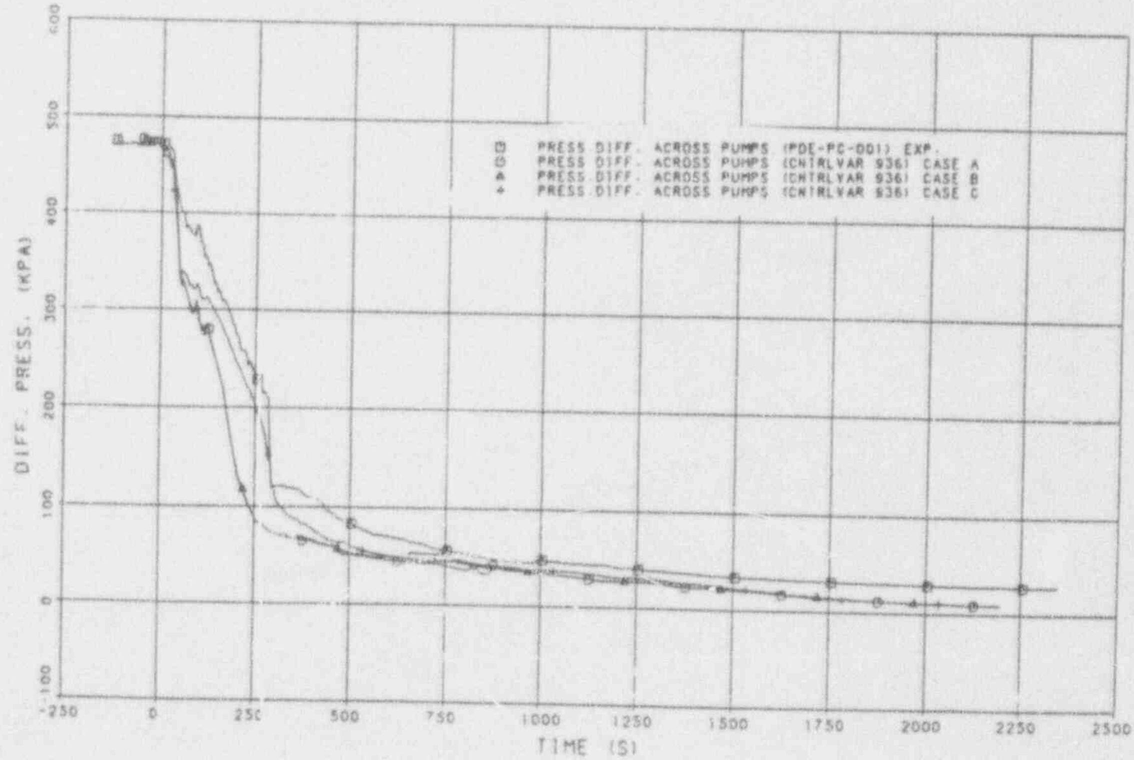


Figure 3.2.18.4 LOFT L3-6 Experiment: Calculated and Measured Reactor Coolant Pump Differential Pressure.

3.2.19 LOFT Small Break LOCE LP-SB-01 Assessment

Reference: P. C. Hall and G. Brown, *RELAP5/MOD2 Calculations of OECD LOFT Test LP-SB-01*, NUREG/IA-0012, January, 1990.

Code version: RELAP5/MOD2, Cycle 36.02

Facility: Loss-of-Fluid-Test Facility (LOFT), Idaho National Engineering Laboratory

Objectives: Evaluate the capability of the code to calculate several thermal-hydraulic responses associated with a SBLOCA; in particular the thermal response of the reactor core during slow core uncovery. The objectives were satisfied.

Major phenomena: Break flow rates, primary and secondary pressure responses, loop flow rates, and fluid densities. The LP-SB-01 experiment was a 1% hot leg SBLOCA.

Code deficiencies:

1. A systematic undercalculation of the break critical mass flow rate.
2. An erroneous calculation of the sudden draining of the hot legs.

Impact of deficiencies:

1. The break flow was undercalculated by about 30% at low quality inlet conditions. Consequently, calculated transient events were late.
2. The erroneous drain rate of the facility hot legs was caused by the vertical stratification model activated in the model upper plenum. Activation of the model suddenly reduced the interphase drag forces at the coupling fluid junction and resulted in sudden, unphysical, hot leg draining behavior.

User guidelines: None.

Base calculation: The base calculation modeled the break nozzle with the liquid and two-phase discharge coefficients equal to 0.93 and 0.81 respectively. Analysis showed the two-phase calculated break mass flow rate to be 30% low.

Sensitivity studies: The base case was modified by increasing the two-phase discharge coefficient to 1.18 until the vapor fraction in the break line equaled 40%; afterwards, the two-phase coefficient was reset to 0.81. This calculation is the basis for most of the report discussion.

Nodalization studies: None.

Summary: The code was assessed using the LP-SB-01 experiment small break loss-of-coolant accident (SBLOCA) data obtained in the LOFT facility. The test was conducted during the Organization for Economic Cooperation and Development phase of the LOFT Program.

LOFT was a 50 MWt integral effect test facility designed to simulate the responses of the major components and system responses of a commercial four-loop PWR during either loss-of-coolant-accidents or operational transients. LOFT contained a half-height nuclear-powered core and was a 1/50-volumetrically scaled system.

The LP-SB-01 experiment was a simulation of a 1% hot leg break in a Westinghouse-type pressurized water reactor. The break was opened at time zero in the experiment. The reactor coolant pumps were tripped early in the experiment. Following completion of the pump coastdown, the core energy was transferred to the secondary by first two-phase natural circulation followed by reflux condensation. Following termination of reflux condensation and a continued decrease in primary system pressure, a balance condition was achieved such that the high pressure injection system injection rate equalled the mass flow rate loss through the break. Thereafter the primary system inventory level and pressure level increased. The test was terminated when the primary pressure reached 2.5 MPa. No core heatup was observed during the experiment.

Overall agreement between the calculation and the experimental data was reasonable. The code systematically undercalculated the break mass flow rate at low quality conditions by about 30% during the early portion of the experiment. Also, the activation of the vertical stratification model in the upper plenum led to an erroneous sudden draining of the loop hot legs.

3.2.20 LOFT Small Break LOCE LP-SB-02 Assessment

Reference: P. C. Hall, *RELAP5/MOD2 Calculations of OECD LOFT Test LP-SB-02*, NUREG/IA-0021, April, 1990.

Code version: RELAP5/MOD2, Cycle 36.04.

Facility: Loss-of-Fluid-Test Facility (LOFT), Idaho National Engineering Laboratory.

Objectives: Evaluate the code's ability to model a small break loss-of-coolant accident experiment. The objective was met.

Major phenomena: Horizontal stratified flow, break mass flow rate, fluid densities, temperatures, and pressures; also vapor pull-through and entrainment.

Code deficiencies:

1. Critical break mass flow model including the upstream break conditions.
2. Calculation of the onset of stratified flow.

Impact of deficiencies: Improperly calculated break mass flow; specifically under two-phase break flow conditions with stratified conditions upstream of the break, too much liquid loss is predicted, but not enough vapor loss is calculated. Consequently, the calculated pressure is high compared to the data and event timings occur late.

User guidelines:

1. Use of the nearly-implicit option will cause code failure.
2. Although the RELAP5 user guidelines advise the user to model a tee with a crossflow junction combined with a very short volume, the author found that such a technique results in a very restrictive Courant limit. The author found that larger volumes will represent the geometry well and also allow faster running times.

Base calculations: The transient is a 1% hot leg SBLOCA with delayed primary coolant pump trip. The comparison between the calculation and data is reasonably good for the first 1200 s, but unsatisfactory afterwards. (Note: The transient lasted for approximately 3000 s.)

Sensitivity studies: The sensitivity calculation was conducted following revision of the horizontal stratification model (internal to RELAP5). Improvement was noted, however further improvement is needed. In addition, sensitivity calculations were done varying the size of the volumes used with the tee/cross flow junction nodalization discussed under "User guidelines," item 2.

Nodalization studies: None.

Summary: The code was assessed using the LP-SB-02 experiment small break loss-of-coolant accident (SBLOCA) data obtained in the LOFT facility. The test was conducted during the Organization for Economic Cooperation and Development phase of the LOFT Program.

LOFT was a 50 MWt integral effect test facility designed to simulate the responses of the major components and system responses of a commercial four-loop PWR during either loss-of-coolant-accidents or operational transients. LOFT contained a half-height nuclear-powered core and was a 1/50-volumetrically scaled system.

The LP-SB-02 experiment was a simulation of a 1% hot leg break in a pressurized water reactor. The break was opened at time zero in the experiment. The reactor coolant pumps trip was delayed. The pumped loop flow was degraded at about 600 s and evidence of flow stratification was observed. However, the pumps maintained loop circulation until about 1300 s shortly after the break was completely uncovered. The pumps were allowed to continue operation until approximately 2900 s. The pump trip caused minor adjustments to the primary inventory distribution but had no significant effect on the break line density and break mass flow rate.

The base case calculation shows reasonable agreement between the measured and calculated primary pressure until about 1200 s. From 1200 to 1900 s there are significant errors in the calculated depressurization rate leading to an overprediction of the pressure late in the transient. The calculated and measured break flow rates differ by as much as 50%. Thus, large cumulative errors were recorded in the system mass inventory. All the difficulties noted above are thought to result from inaccuracies in the horizontal stratification entrainment model.

3.2.21 LOFT Small Break LOCE LP-SB-03 Assessment

Reference: C. Harwood and G. Brown, *RELAP5/MOD2 Calculation of OECD LOFT Test LP-SB-03*, NUREG/IA-0013, January, 1990.

Code version: RELAP5/MOD2, Cycle 36.01.

Facility: Loss-of-Fluid Test Facility, Idaho National Engineering Laboratory.

Objectives: Evaluate the capability of the code to model representative thermal-hydraulic behavior occurring during a small-break LOCA.

Major phenomena: Primary/secondary pressure history, break mass flow rate, primary inventory history, primary fluid densities, fuel rod cladding temperatures, and accumulator injection behavior.

Code deficiencies: None.

User guidelines: None.

Base calculations: The base calculation was performed using the INEL-constructed pretest prediction model. However, a number of changes were made to the INEL model since the original had been built to be used with RELAP5/MOD1. Overall agreement with the test data is reasonable, with all key phenomena correctly predicted in the proper sequence.

Sensitivity studies: None.

Nodalization studies: None.

Summary: The assessment was based on the LP-SB-03 experiment conducted in the LOFT facility. The assessment was performed to evaluate the code's capability to calculate the phenomena present during a small break loss-of-coolant accident experiment.

LOFT was a 50 MWt integral effect test facility designed to simulate the responses of the major components and system responses of a commercial four-loop PWR during either loss-of-coolant-accidents or operational transients. LOFT contained a half-height nuclear-powered core and was a 1/50-volumetrically scaled system.

The LP-SB-03 experiment was a 0.4% cold leg LOCA. The experiment consisted of four periods: (i) Rapid mass depletion - occurring from the break initiation until the reactor coolant pumps were tripped off. This period had single-phase and relatively homogeneous two-phase critical break flow. (ii) Boiloff period - transition of the break mass flow to high quality steam occurs, the core inventory decreases as boiling occurs. Core dryout is observed and the break was isolated. (iii) Cooldown using secondary feed and bleed procedures - initiated when the core temperatures reached 977 K. (iv) Accumulator injection occurred when the primary pressure decreased to the accumulator injection setpoint.

Agreement between the calculated and measured parameters was reasonable since all key phenomena were adequately calculated, although not always within the instrumentation uncertainty bands.

It should be noted that this assessment was conducted in much the same manner as that by Guntay (see Section 3.2.22). Both studies reached much the same conclusions and showed very similar results.

3.2.22 LOFT Small Break LOCE LP-SB-03 Assessment

Reference: S. Guntay, *RELAP5/MOD2 Assessment: OECD-LOFT Small Break Experiment LP-SB-3*, NUREG/IA-0018, April, 1990.

Code version: RELAP5/MOD2, Cycles 33 to 36.01 (Note: only the assessment work concerning Cycle 36.01 was reviewed.)

Facility: Loss-of-Fluid-Test Facility (LOFT), Idaho National Engineering Laboratory.

Objectives: Evaluate the capability of the code to calculate several thermal-hydraulic responses associated with a SBLOCA; in particular the thermal response of the reactor core during slow core uncovery.

Major phenomena: Primary and secondary relationships involving single and two-phase forced-convection and reflux natural circulation, slow boil-off in the core (including radiation heat transfer, dryout, and redryout), single and two-phase break flow, pump performance and degradation, and plant cooldown and recovery using secondary feed-and-bleed.

Code deficiencies: None. But, the author did identify three areas in which the code did not meet his expectations: (a) reflux condensation draining to the core periphery, (b) radiation heat transfer, and (c) fuel stored energy. However, the reviewers noted that items (a) and (b) are beyond the capability of the code, i.e., since RELAP5/MOD2 is a one-dimensional code with no core radiation heat transfer. Further, the reviewers did not find sufficient data in the report to support item (c).

Impact of deficiencies: The code may not accurately calculate the temperatures observed in the core periphery.

User guidelines: Using cross-flow junction connections between the hot- or cold-legs and the reactor vessel may eliminate the artificial elevation differences between these components that can occur if ordinary junctions are used.

Base calculations: The calculation was conducted using as a basis the RELAP5/MOD1 model built at INEL. The model was modified to be used with RELAP5/MOD2. Although some discrepancies were noted between the calculated results and measured data, the code calculated most thermal-hydraulic responses in the proper sequence.

Sensitivity studies: The use of cross-flow junctions for the hot- and cold-leg connections to the vessel were studied.

Nodalization studies: None.

Summary: The assessment was based on the LP-SB-03 experiment conducted in the LOFT facility. The assessment was performed to evaluate the code's capability to calculate the phenomena present during a small break loss-of-coolant accident experiment.

LOFT was a 50 MWt integral effect test facility designed to simulate the responses of the major components and system responses of a commercial four-loop PWR during either loss-of-coolant-accidents or operational transients. LOFT contained a half-height nuclear-powered core and was a 1/50-volumetrically scaled system.

The LP-SB-03 experiment was a 0.4% cold leg LOCA. The experiment consisted of four periods: (i) Rapid mass depletion - occurring from the break initiation until the reactor coolant pumps were tripped off. This period had single-phase and relatively homogeneous two-phase critical break flow. (ii) Boiloff period - transition of the break mass flow to high quality steam occurs, the core inventory decreases as boiling occurs. Core dryout is observed and the break was isolated. (iii) Cooldown using secondary feed and bleed procedures - initiated when the core temperatures reached 977 K. (iv) Accumulator injection occurred when the primary pressure decreased to the accumulator injection setpoint.

Agreement between the calculated and measured parameters was reasonable since all key phenomena were adequately calculated, although not always within the instrumentation uncertainty bands.

It should be noted that this assessment was conducted in much the same manner as that by Harwood and Brown (see Section 3.2.21). Both studies reached much the same conclusions and showed very similar results.

3.2.23 LOFT Small Break LOCE L3-7 Assessment

Reference: E. J. Lee, B. D. Chung, and H. J. Kim, *ICAP Assessment of RELAP5/MOD2, Cycle 36.05 Against LOFT Small Break Experiment L3-7*, NUREG/IA-0031, April, 1990.

Code version: RELAP5/MOD2, Cycle 36.05

Facility: Loss-of-Fluid-Test Facility (LOFT), Idaho National Engineering Laboratory.

Objectives: Evaluate the capability of the code to calculate various thermal-hydraulic phenomena associated with a SBLOCA located in the cold leg.

Major phenomena: The thermal-hydraulic phenomena of interest include the break mass flow, the primary system pressure, the fluid temperatures, and densities, and the fuel-rod temperature.

Code deficiencies: None. But, the authors did identify two areas in which the code did not meet their expectations: (a) the critical mass flow rate at the break was consistently underpredicted for two-phase conditions and (b) the primary system was calculated to depressurize more quickly than measured.

Impact of deficiencies: No deficiencies were identified.

User guidelines: None.

Base calculations: The base calculation was performed using a model nodalization based on the developmental assessment studies described in Ransom, 1985.

Sensitivity studies: Three sensitivity studies were performed. The first sensitivity calculations investigated the effect of varying the break discharge coefficient from 0.9 to 1.2. The second sensitivity study was conducted to study the effect of changing the two-phase pump torque and head multipliers and the third sensitivity study was conducted to explore the effect of an increased high pressure injection pump head.

Nodalization studies: The base calculation nodalization was simplified by reducing the number of cells in the hot legs, the cold legs, the pump outlet, and the upper plenum from two to one.

Summary: The assessment was based on experimental data from the LOFT L3-7 experiment. The objectives of the assessment were to determine the code's capability to calculate small break loss-of-coolant accident related phenomena.

LOFT was a 50 MWt integral effect test facility designed to simulate the responses of the major components and system responses of a commercial four-loop PWR during either loss-of-coolant-accidents or operational transients. LOFT contained a half-height nuclear-powered core and was a 1/50-volumetrically scaled system.

The L3-7 experiment simulated a 0.1% cold leg SBLOCA. The experiment was conducted to evaluate SBLOCA thermal-hydraulic interactions when the high pressure injection system is available. The core did not uncover during the transient.

The base calculation was performed using a model provided by INEL and modified at the Korea Institute of Nuclear Safety (KINS). The base calculation gave a reasonable match to the key experimental parameters.

To study the possibility of matching the data more closely, three sensitivity studies were also performed. These studies examined the effect of: (i) changing the break discharge coefficients over a range from 0.8 to 1.2, (ii) modifying the pump two-phase multiplier, and (iii) the HPIS flow capacity characteristic. The first sensitivity study showed that single and two-phase discharge coefficients of 0.9 gave a better match to the data than the values of 1.0 used in the base calculation. The change in the pump two-phase multiplier was based on an arbitrary decision to increase the value from 0.6 (at a void fraction of 0.5) to 0.95; no change in the results was noted. Finally, the effect of changing the HPIS flow characteristic was studied by performing two cases - one at flow rates greater than the nominal HPIS characteristic and one at flow rates less than the nominal HPIS characteristic; the author's conclusions are not discussed further herein.

One nodalization study was performed by decreasing the number of volumes from 130 to 123, the number of junctions from 136 to 132, and the number of heat structures from 137 to 129. The simplifications were made by reducing the base case nodalization with two volumes to one volume in the following regions: the intact cold leg, the intact hot leg, the pump outlet, the broken loop hot leg, the broken loop cold leg, and the upper core region. Otherwise the nodalization and options that were used were the same as the base case. No obvious differences were obtained in the comparisons between the calculated and measured thermal-hydraulic parameters.

3.2.24 LOFT Loss-of-Feedwater Without SCRAM Experiment L9-3 Assessment

Reference: J. C. Birchley, *RELAP5/MOD2 Analysis of LOFT Experiment L9-3*, NUREG/IA-0058, April, 1992.

Code version: RELAP5/MOD2 Cycle 36.04.

Objectives: Investigation of code capabilities for simulating the response to a loss-of-feedwater anticipated transient without scram. Specifically, decreased steam generator heat removal capability with secondary inventory boil-off, steam generator performance during single phase forced circulation, pressurizer response during insurge and spray, and mass and energy flows through relief valves.

Major phenomena: Degradation of steam generator performance during secondary boil-off, steam generator heat transfer during forced circulation, pressurizer response during periods of insurge and spray operation, and mass and energy flow through relief valves.

Code deficiencies: The code likely overpredicts interphase drag.

User guidelines: The author and the reviewer both suggested that motor valves should only be used when the valve stroke times are well known and the movement is at constant speed. It is recommended that servo or trip valves be used.

Base calculation: The base calculation exhibited excessive primary-to-secondary heat transfer which resulted in early an rapid heat transfer degradation.

Sensitivity studies: Modifications were made to the steam generator model including 1) the bottom boiler and downcomer nodes were divided into two, 2) the downcomer flow area was increased based on data from the facility, 3) flow resistance of the steam generator was reduced to increase the recirculation ratio, and 4) the trip settings were modified to better represent the settings in the experiment. The sensitivity calculation was determined to be a fairly good representation of the test results. However, the steam generator still boiled-off too quickly.

Nodalization studies: None

Summary: This report assesses RELAP5/MOD2 Cycle 36.05 (with Winfrith Cray error corrections) through a comparison of calculated results with LOFT Experiment L9-3.

LOFT was a 50 MWt integral effect test facility designed to simulate the responses of the major components and system responses of a commercial four-loop PWR during either loss-of-coolant-accidents or operational transients. LOFT contained a half-height nuclear-powered core and was a 1/50-volumetrically scaled system.

This experiment simulated the response of a pressurized water reactor to a loss of feedwater event followed by a failure of reactor trip. The experiment was designed to provide a benchmarking tool for the reactor vendors anticipated transient without scram (ATWS) computer codes as required by a proposed NRC rule. The experiment also was designed for evaluation of alternative methods of achieving long-term shutdown without control rod insertion during ATWS events.

The experiment was performed in two phases. The first phase, lasting 600 s, represented the automatic plant action portion of the transient. The second phase simulated the effect of boric acid addition to the primary coolant system such as might be accomplished by operator action. Only the first phase of the transient was simulated with RELAP5. The experiment was initiated from conditions representing a full power steady operation by terminating all feedwater delivery to the steam generators. The reactor trip logic was deactivated, but most other plant systems were assumed operative.

When the feedwater was lost, a boil-off of the steam generator secondary inventory began and steam generator heat transfer began to degrade slowly. The resulting mismatch in the primary system heat balance caused a slight heating of the primary fluid and a pressurizer insurge. This insurge resulted in lifting of the pressurizer PORV and safety valves. When the steam generators had boiled dry, their heat transfer degraded more rapidly and steaming stopped. The increasing primary side temperatures and void fractions caused a significant reduction in the core power.

The LOFT input model was based on one previously used for analyzing LOFT Test L9-4 (see Croxford, et al., 1992; Keevill,1992). A satisfactory agreement was obtained between calculated and measured data for the test initial condition and a short null transient was run to assure steady state convergence.

A preliminary calculation was run to compare the calculated and measured responses over the early portion of the test. Based on this comparison, minor changes were made to the model and a final calculation was performed. These minor changes are summarized as follows:

(1) The bottom nodes in the boiler and downcomer of the steam generator were divided into two, to seek a more gradual degradation in heat transfer during the boiloff.

(2) The flow area in the lower part of the steam generator downcomer was increased in line with engineering data on the facility.

(3) The flow resistance of the steam generator was reduced to increase the recirculation ratio. (This and the previous change were intended to increase the initial steam generator inventory).

(4) The trip settings were adjusted to more closely match the measured conditions at actuation.

The following discussion compares the results of the final calculation with the experiment.

The RELAP5 calculation was judged to be an overall reasonable simulation of the experiment. This transient was considered fairly challenging for the code to predict because a large number of events occurred and setpoints were reached in a short period of time.

The most significant discrepancy between calculation and data was the rate at which the primary-to-secondary heat transfer degraded as the steam generator boiled dry. As was discussed above, this discrepancy is believed to be due to an overprediction of interphase drag that levitated liquid in the steam generator boilers. With this levitation, the outside of the tubes remained wet and heat transfer continued until the dryout was almost complete. As a result the calculated heat transfer degraded more abruptly than did the test data as is shown in Fig. 3.2.24.1.

An additional discrepancy was the rate at which the primary pressure increased during pressurizer insurge. For a given level increase, the calculated pressure increased more than was indicated in the test data. Possible causes for this relatively minor discrepancy are inadequate calculated mixing of spray and steam and an improper initial spray line fluid temperature. The calculated and measured hot leg pressures are compared in Fig. 3.2.24.2.

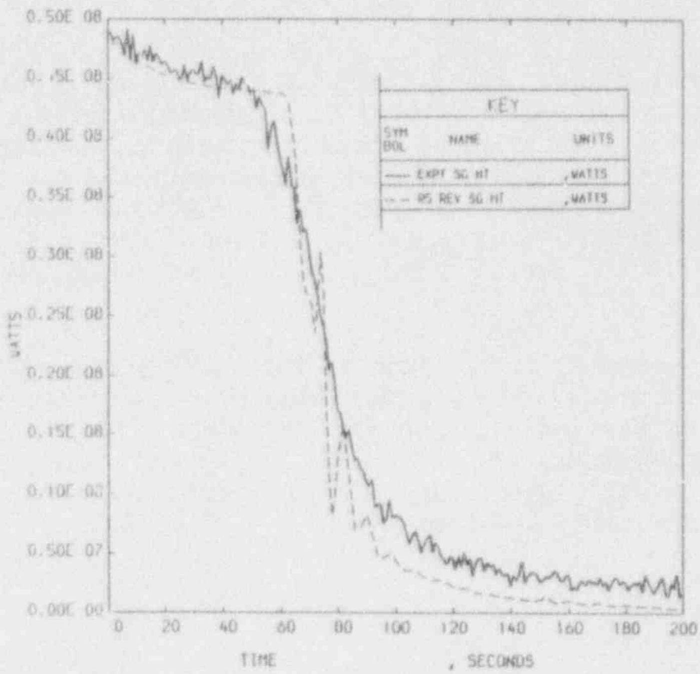


Figure 3.2.24.1 L9-3 Steam Generator Heat Transfer - Revised Model.

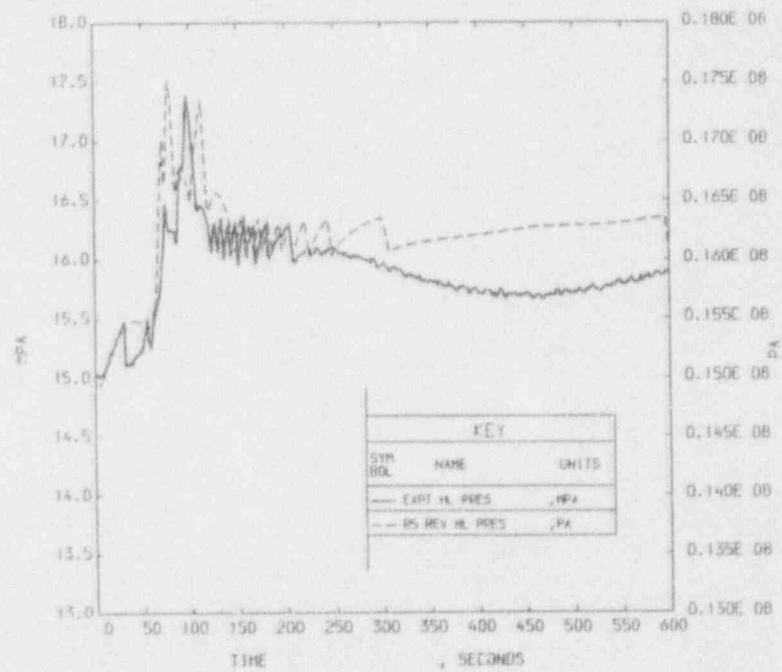


Figure 3.2.24.2 L9-3 Hot Leg Pressure - Revised Model.

3.2.25 LOFT Loss-of-Offsite Power without SCRAM Experiment L9-4 Assessment

Reference: M. B. Keevill, *RELAP5/MOD2 Analysis of LOFT Experiment L9-4*, NUREG/IA-0066, April, 1992.

Code version: RELAP5/MOD2 Cycle 36.05 UK Version E03

Facility: Loss-of-Fluid Test Facility, Idaho National Engineering Laboratory

Objectives: The objective of the study was to evaluate the capability of the code to predict the response to a loss-of-offsite-power transient without scram.

Major phenomena: General thermal-hydraulic phenomena were addressed with specific attention paid to the critical heat flux correlation used by the code.

Code deficiencies: The author observed that the Biasi Critical Heat Flux Correlation was being applied at pressures outside the range of validity.

User guidelines: None

Base calculation: A base calculation was performed using the frozen version of the RELAP5 code. The code generally captured the response observed in the L9-4 transient.

Sensitivity studies: None

Nodalization studies: None

Summary: Post test calculations were performed to measure the ability of RELAP5/MOD2/CY36.05 UK Version E03 to simulate a Loss-Of-Offsite-Power Anticipated Transient Without Trip (LOOP ATWT) experiment conducted at the LOFT facility.

LOFT was a 50 MWt integral effect test facility designed to simulate the responses of the major components and system responses of a commercial four-loop PWR during either loss-of-coolant-accidents or operational transients. LOFT contained a half-height nuclear-powered core and was a 1/50-volumetrically scaled system.

Test L9-4 simulated a LOOP ATWT in which power was lost to the primary coolant pumps. Additionally, the main feed was lost to the steam generators and the control rods failed to insert into the reactor core.

The input model was based on previous work done by Croxford and Harwood. The model was changed to match the boundary conditions and initial conditions used during the experiment.

A reasonable calculation of the transient was obtained using the code, particularly during the initial primary heat-up which is the phase of the transient where departure from nucleate boiling will most likely occur. The primary coolant system remained sub-cooled throughout the transient.

The steam generator boil-down was predicted to occur significantly faster than in the experiment, which subsequently affected the remainder of the calculation. The reason for this is not clear but it is likely that inaccuracies in the input power and primary flow are the main contributory factors. Systematic errors in the calculation of the void fraction in the riser region may also have contributed to an under-prediction of the initial steam generator mass inventory.

Due to the lack of data, the pump coast-down and reactor power had to be specified as boundary conditions. This caused the calculation to be very sensitive to the primary coolant flow rate, to the extent that changing the flow within the measurement uncertainty band had a large effect on primary pressure. It is believed that the inclusion of reactivity feedback modelling would have alleviated this sensitivity.

In this study RELAP5/MOD2 applied the Biasi critical heat flux correlation outside its range of validity. This resulted in calculation of a negative critical heat flux at pressures above 162.5 bar. The coding was modified in Cycle 36.05 to correct this error.

3.2.26 LOFT Loss-of-Feedwater Experiment LP-FW-01 Assessment

Reference: M. G. Croxford, C. Harwood, and P. C. Hall, *RELAP5/MOD2 Calculation of OECD LOFT Test LP-FW-01*, NUREG/IA-0063, April, 1992.

Code version: RELAP5/MOD2 Cycle 36.04

Facility: Loss-of-Fluid Test Facility, Idaho National Engineering Laboratory.

Objectives: The objective of the study was to assess the code capability to capture the feed-and-bleed recovery procedure following a loss-of-feedwater event.

Major phenomena: Boil-off of the steam generator secondary volume and during the feed-and-bleed portion of the transient, the distribution of mass and the primary pressure response were evaluated.

Code deficiencies: The horizontal stratification entrainment model under-predicts the quality of the mass flow entering the surge line.

User guidelines: None.

Base calculation: A base calculation was performed which, in a general sense, predicted the transient response well. However, during the initial period of two-phase discharge from the PORV the system pressure was overestimated.

Sensitivity studies: A sensitivity study was performed using an improved horizontal stratification entrainment model that provided a more accurate prediction of the two-phase discharge through the PORV.

Nodalization studies: The upper head by-pass flow path was replaced by a nozzle by-pass flow path using estimated values for junction area and loss coefficient.

Summary: Post test calculations were performed to measure the ability of RELAP5/MOD2/CY36.04 to simulate a feed-and-bleed recovery procedure following a complete loss-of-feedwater event. The code was used to simulate the Test LP-FW-01 performed at the LOFT experimental reactor under the OECD LOFT program.

LOFT was a 50 MWt integral effect test facility designed to simulate the responses of the major components and system responses of a commercial four-loop PWR during either loss-of-coolant-accidents or operational transients. LOFT contained a half-height nuclear-powered core and was a 1/50-volumetrically scaled system.

Test LP-FW-01 simulated a fault sequence in which there was a complete loss-of-feedwater to the steam generator followed by primary system feed-and-bleed. Feed-and-bleed is where coolant is simultaneously injected by the High Head Safety Injection system and vented through the primary side PORV (Power Operated Relief Valve).

The input model was based on previous work done by Hall and Brown. The input model was modified to include the boundary and initial conditions required to calculate the LP-FW-1 experiment.

A good overall prediction of the experimental transient was obtained using the standard version of RELAP5/MOD2/Cy 36.04. However, the pressure increase during the initial period of two-phase discharge from the power-operated relief valve (PORV) was overestimated, leading to an over-prediction of primary system pressure for the remainder of the transient. With the United Kingdom modified code version, including an improved representation of the entrainment in the hot-leg/surge line connection, a closer agreement within the early re-pressurization period was achieved, leading to an improved primary pressure prediction.

The mass flow rate through the PORV was over-predicted in the latter part of the transient. Also predicted were intermittent surges of liquid flow through the PORV which were not observed in the test. Detailed investigation revealed these errors were probably not due to the physical models in the code. Rather they likely relate to the simplified modelling of the flow of steam in the complex bypass flow paths connecting the cold legs and the upper plenum in LOFT.

Comparison with a previous analysis of the same test (LP-FW-01) using RETRAN-02/MOD2 has shown RELAP5 gives a superior prediction of secondary pressure and pressurizer level in this transient. The improvement is believed due in part to more accurate modelling of the primary-to-secondary heat transfer in the steam generator boil-off phase, in the RELAP5 calculation.

3.2.27 LOFT V Sequence Experiment LP-FP-2 Assessment

Reference: J. J. Pena, S. Enciso, F. Reventos, *Thermal-Hydraulic Post-Test Analysis of OECD-LOFT LP-FP-2 Experiment*, NUREG/IA-0049, April, 1992.

Code version: RELAP5/MOD2 Cycle 36.04 and SCDAP/MOD1/21

Facility: Loss-of-Fluid Test Facility, Idaho National Engineering Laboratory.

Objectives: The objective of the study was to assess the code's capability to simulate the V sequence.

Major phenomena: Transient fuel and control rod thermal and mechanical behavior prior to and during severe core damage were the subjects of the analysis Phenomena included cladding ballooning, control rod melting and relocation, high temperature Zircaloy oxidation in the presence of steam, generation of hydrogen, fuel liquefaction, relocation, and resolidification. Only the thermal-hydraulic behavior prior to core severe damage are of interest herein.

Code deficiencies: Lack of a radiation heat transfer model.

User guidelines: None.

Base calculation: A base calculation was performed using the model developed at INEL and described in Guntay, 1985.

Sensitivity studies: A sensitivity study was performed to evaluate the effect of having no center fuel module (CFM) blockage.

Nodalization studies: None.

Summary: The report presents the results of the thermal-hydraulic posttest analysis of the LP-FP-2 experiment, made by the Spanish FP-2 calculation group using the RELAP5/MOD2 and SCDAP/MOD1 computer codes. The LOFT LP-FP-2 experiment simulated an interfacing-systems loss-of-coolant accident (LOCA), a hypothetical event labeled the V-sequence. This risk-dominant accident sequence represents a significant contribution to the calculated risk associated with Pressurized Water Reactor (PWR) operation. The purpose of the experiment was to provide information on the release, transport, and deposition of fission products and aerosols during a severe core damage event, an accident resulting in fuel rod failure, control rod melting, fuel relocation, and a release of fission products from the UO_2 fuel. During this experiment, the fuel rod temperatures in the Center Fuel Module (CFM) exceeded 2100 K for about 4.5 minutes before test termination temperatures were reached on the exterior wall of the CFM shroud. The experiment simulated the system thermal-hydraulics and core uncovery conditions during fission product release and transport expected to occur in a four-loop PWR from rupture of a Low Pressure Injection System (LPIS) pipe, from initial conditions typical of commercial PWR operations.

The thermal-hydraulic calculation was performed using RELAP5/MOD2/36.04; SCDAP/MOD1/21 was used to model the detailed thermo-mechanical core behavior during the heatup phase of the experiment. This interdependency between the codes is known as the RELAP5-SCDAP passive link.

The RF`AP5/MOD2 and SCDAP/MOD1 input decks were based on those used by the Idaho Nation. Engineering Laboratory (INEL) to prepare the Best Estimate Prediction (BEP) Document. Modifications were made to the RELAP5 input deck as described under item C1. This model was used to calculate the thermal-hydraulic information required for SCDAP for the detailed core thermal response calculations.

The calculated pressure response agreed with the observed data until initiation of the LPIS line break at 221.6 s. The subsequent depressurization rate was initially underestimated until 350 s, and overestimated from 425 s until the closure of the ILCL break at 735.5 s. This anomalous behavior was not well understood. It was postulated in the Quick Look Report that the complicated network of bends in the LPIS line resulted in higher flow resistance under single phase conditions and inhibiting the draining of liquid from the line under two phase conditions. Fluid temperature measurements indicated that the LPIS line was not completely drained until after about 1200 s. In the calculation, the line was completely void after 425 s, and it subsequently vented steam. The pressure discrepancy affected all the comparisons of systems hydraulics and core thermal response beyond 425 s. Even with the revised LPIS line nodalization, the model was still unable to provide a fully satisfactory representation of the LPIS line flow characteristics. It was not clear if the deficiency was a nodalization problem or an error in the RELAP5 critical flow model. Calculational improvement could have been obtained by using different discharge coefficients for the two-phase and single phase flow periods of the LPIS discharge process; this practice is inconsistent with previous experiences using RELAP5/MOD2. After the initial closure of the ILCL break at 735.5 s, calculated depressurization rates agreed well with data. The rate of secondary system depressurization was slightly overestimated because of the differences in primary system pressure and possibly because of some inaccuracy of the steam generator simple leak model.

Direct comparisons of break flow rates were not made. Actual primary system pressure was higher during the heatup and core damage phase (between 1200 and 1750 s) and resulted in higher than calculated break flow. However, calculated LPIS line flow and measured single points were compared for the "critical time period," during single phase vapor flow both indicated and calculated. The steam flow rate was about 0.2 kg/s in both cases. The differences in break flow contribute to differences in core mass flow. Although there was no direct measurement of core mass flow, the experimental steam flow rate was obtained based on core thermal measured data. The resulting CFM mass flow rate was 0.04 kg/s per fuel rod; this value was above the value for which the metal-water reaction is steam-limited. The calculated CFM inlet flow during the damage phase (1250 to 1750 s) is a factor of 5 to 25 lower than the calculated experimental value. This enormous difference in calculated CFM inlet flow cannot be explained in terms of the differences in LPIS line flow. The low calculated CFM inlet flow value can be related to either errors in the calculation of core flow redistribution due to blockages or to phenomena not considered in the calculation (i.e. steam generation due to the slumping of some molten material into the lower plenum), or both.

The calculated collapsed liquid level in the reactor vessel was the same for the CFM and for the average channel in the core. No significant differences were found between these two calculated values. Progression of core uncovery in the center and peripheral fuel assemblies was fairly rapid until ILCL break closure at 735 s; thereafter uncovery progressed very slowly because depressurization was terminated. This caused a sharp reduction in vapor generation rate and a total or partial collapse of froth level in the vessel. During the time the ILCL break was reopened (877.6 to 1021.5 s) the liquid level again decreased rapidly. Afterward, until the end of the transient, core uncovery rate was solely dependent on heat input from the reactor core.

The CFM thermal response was well predicted at the 0.25 m elevation until the time of the first blockage. After cladding ballooning blockage was modeled, the temperature rise rate was overpredicted until the end of the transient. The underprediction of CFM steam flow is believed to have resulted in underprediction of heat transfer coefficient. The observed increase in temperature rise rate at 1700 s was below the metal-water reaction onset temperature, and may have been caused by thermal radiation from material at higher elevations or from material relocation. Neither of these effects is modelled with the RELAP5 code.

At the 0.69 m (27 inch) elevation, good agreement was obtained with the initial heatup rate until the time of reopening the ILCL break (877.6 s) and opening the PORV (882.0 s). The heatup rate then decreased, apparently due to flashing of liquid in the lower plenum. This additional cooling was underpredicted by RELAP5. Therefore, the prediction exceeded the actual temperature prior to initiation of the Metal-Water Reaction (MWR), and MWR onset was predicted early (1225 s). The observed oxidation of zircaloy by steam did not occur until temperature exceeded 1400 K; Cathcart-Pawel MWR onset temperature is 1273 K. Actually, the oxidation heat generation rate equation is valid between 1000 and 1850 K, and there is no particular temperature at which MWR onset occurs. The temperature at which the energy added by the MWR exceeds that lost by the fuel cladding depends on the system boundaries. After about 1550 s, the calculated reaction became steam starved; the experiment showed no evidence of this. Even so, the maximum predicted temperature of 2430 K was very close to the maximum validated experimental data. Calculated cooldown during ECCS injection was much faster than measured at this elevation.

At the 1.07 m (42 inch) elevation, observed initial heatup rate was about 1.3 K/s until 1450 s, after which temperature increase was very rapid because of the MWR. Up until this point the calculation was not too much different. As before, the CFM steam flow was underpredicted, resulting in higher initial temperatures but a steam-starved MWR reaction. At this elevation, the cooldown rate during quench was accurately predicted.

Calculated and measured peripheral cladding temperatures were in excellent agreement at the 10-in elevation until about 1700 s. At this time, thermocouples near the outside of the shroud, particularly at lower elevations, began a rapid temperature rise that was attributed to shunting of the thermocouple leads, which passed through a high temperature area. At the 0.66 m (26 inch) elevation, agreement was excellent until the time of PORV opening (882 s), which introduced core steam cooling that was underpredicted by the model. The agreement at the 45-inch elevation was remarkable. The calculated temperatures at the outer wall of the shroud between 27 and 42 inches were low because of the lack of a thermal radiation model in RELAP5.

The relationship between center and peripheral fuel rod and shroud temperatures agreed well with the experiment. As a result, the time above 2100 K in the center bundle as predicted by the code (279 s) closely matched the actual time of 270 s.

Considering the known limitations in the capability of RELAP5/MOD2 to model core thermal response during a severe accident, the calculated core temperature excursion reproduced the experiment reasonably well. The major problem was the underprediction of primary system pressure, believed to be dominated by differences in LPIS line flow characteristics. Calculated and measured core uncovery processes were in very close agreement. Global core thermal response was reasonably well calculated despite the lack of radiation and material relocation models. Measured and calculated core heatup rates prior to onset of rapid oxidation are in overall agreement. The underprediction of core cooling flow was related to the difference in the break flow rate which occurred because of the discrepancy in the pressure response prediction.

After onset of rapid oxidation, the calculation significantly underestimates the heatup rate in the upper CFM because of steam starvation. The differences were attributed to errors in the calculation of core flow redistribution due to blockages or to phenomena not considered in the calculation (i.e. steam generation due to the slumping of some molten material into the lower plenum), or both.

As noted, the calculated CFM blockage had a significant effect on predicted heatup behavior following the onset of MWR. The conclusion was that the underprediction of CFM steam flow was due to overcalculation of flow blockage. A sensitivity study was therefore performed for a case with no CFM flow blockage.

The general LOFT system response was not affected by this modification: i.e., primary and secondary system pressures, loop densities, break flows, and core liquid level were not significantly changed. The difference in CFM flow rate was approximately twice that of the base case. The difference in mass flow through the peripheral channels did not significantly modify the heatup process in those assemblies, but the higher CFM flow rate dramatically altered the temperature excursion for locations within the shroud. Both maximum temperatures and heatup rates were in much closer agreement with the experiment. It was therefore concluded that the core flow redistribution following blockage is one of the most important uncertainties associated with the RELAP5/MOD2 simulation.

It is expected that the calculational uncertainties in the amount and timing of blockages will be significantly improved by use of the integrated RELAP5/SCDAP code, when this tool becomes available.

3.2.28 LOBI Small Break LOCA Experiment BL02 Assessment

Reference: A. H. Scriven, *Analysis of LOBI Test BL02 (Three Percent Cold Leg Break) With RELAP5 Code*, NUREG/IA-0036, March, 1992.

Code version: RELAP5/MOD2, Cycle unknown.

Facility: LOBI in Ispra, Italy.

Objectives: Evaluate the capability of the code to calculate the behavior of a SBLOCA integral experiment. The objectives were satisfied.

Major phenomena: Critical break mass flow rate, liquid entrainment and vapor pull-through.

Code deficiencies: Because the data uncertainty bands were so large, code deficiencies could not be accurately identified. However, suspected code deficiencies are:

1. Liquid entrainment/vapor pull-through model does not interact with critical break flow model to give correct upstream conditions at break.
2. The core void fraction was incorrectly calculated. This deficiency is attributed to the code interphase drag model.

Impact of deficiencies: Inaccurate break flow calculation.

User guidelines: None.

Base calculation: The base calculation gave an overprediction of the critical break mass flow rate. The liquid entrainment/vapor pull-through model gave inadequate predictions of the upstream break flow conditions. In addition, the model didn't calculate the loop seal clearing phenomena well and the calculated core void fraction was overpredicted by 30 to 40%. Finally, although the experiment showed core heatup dur ng early core uncovery and late core uncovery, the calculation only showed heatup during the late core uncovery.

Sensitivity studies: Several sensitivity calculations were conducted to study the effect of (a) locating the bypass in two different locations, and (b) changing the bypass flow area. In addition, cross-flow junctions were added to model the connection between the loop and vessel. The net result was a calculated loop seal clearing that was too early. (Note: The calculated loop seal clearing time for the base calculation also appears to be too early.)

Nodalization studies: None.

Summary: The assessment was performed by using the LOBI experiment BL02 data. Two calculations were completed; the first was a pretest calculation and the second was post test.

LOBI is an electrically-heated PWR simulator located in Ispra, Italy. The facility is reminiscent of the Semiscale facility that once operated at the INEL. The experimental facility features a broken loop, an unbroken loop, complete high pressure injection and accumulator systems, and break simulation.

The test, a 3% small break loss-of-coolant accident (SBLOCA) located in the cold leg, was initiated from an unsteady condition. So the measured data uncertainty is larger than usual. However no uncertainty bands of the measured data were given. A brief synopsis of the experimental phenomena shows the primary system depressurized to saturation conditions, held at 80 bar until the break uncovered, and then started to depressurize rapidly again. Both loop seals cleared during the test; core depression was limited to approximately 1.5 m. Thus, rod heatup occurred only above the nozzle elevations. HPIS flow quenched the rods and restored core inventory before the accumulators were activated.

The code reasonably predicted the system depressurization and integrated break mass loss. The pretest calculation (RELAP5/MOD1) predicted too much core level depression and hence the rod heatups were calculated to be too low. The post-test calculations were better with approximately correct level depression and clad temperature profiles.

The author was not able to determine the quantity of bypass flow present in the experiment. Since the bypass directly influences the pressure buildup in the core, the core level depression is also affected. The author concluded that more knowledge about the LOBI facility would have been beneficial in the performance of the subject calculations.

It was shown that RELAP5 tended to overpredict both the single-phase and two-phase break flows. The break flow multipliers were reduced to 0.85 for both phases, which resulted in good agreement. The liquid entrainment and vapor pull-through model used in RELAP5 was blamed for the original overprediction.

RELAP5 did not predict the loop seal behavior very well. As stated earlier, in the test, the intact loop seal cleared followed by the broken loop seal. RELAP5 predicted the broken loop seal clearance but never cleared the intact loop seal. At least part of the reason for this was RELAP5's inability to correctly calculate the draining of the steam generators. It was concluded that the counter-current flow modeling in RELAP5 should be further studied.

It was found that RELAP5 overpredicts the interphase drag in low voidage flow regimes such as bubbly and slug flows. The problem seems to be particularly apparent in rod bundle geometries.

Another problem in the RELAP5 calculations was the insufficient downcomer penetration. Previous LOBI tests have shown significant penetration. The lesser penetration in the code calculation results in less subcooling in the downcomer, hence more voiding when that liquid reaches the core. This problem contributes to the code overprediction of the core level depression. Attempts to remedy this problem by using a split downcomer input model failed due to code errors.

The author concluded that the differences between the experimental results and the code results were due partly to lack of knowledge about facility bypass flow and partly due to code deficiencies. In particular, RELAP5 needs further work in the areas of two-phase break flow modeling, counter-current flow modeling, high pressure injection mixing, and interphase drag modeling.

3.2.29 LOBI Loss-of-Feedwater Transient Experiment ST-02 Assessment

Reference: A. H. Scriven, *Pre- and Post-Test Analysis of LOBI MOD2 Test ST-02 (BT-00) with RELAP5 MOD1 and MOD2 (Loss of Feedwater)*, NUREG/IA-0061, April, 1992.

Code version: RELAP5/MOD2 Version 36.04

Facility: LOBI MOD2 Facility

Objectives: Evaluation of the code capability to simulate thermal-hydraulic behavior following a loss of main feedwater event in a pressurized water reactor. Specifically, the evaluation of code capability to capture secondary boil-off and feed-and-bleed recovery procedures.

Major phenomena: Single-phase liquid forced convection on the primary side of the heat exchanger, boil-off of the secondary inventory, and finally, feed-and-bleed heat removal from the primary system.

Code deficiencies:

1. The code does not contain a method by which the heat transfer from heat structures to fluid volumes is modified to account for a mixture level residing within the fluid volume
2. The code is deficient in simulating the void fraction for stagnant pools.

User guidelines: Valves should be modeled in as much detail as is possible relative to the expected valve behavior.

Base calculation: A base calculation using the RELAP5/MOD2 code was not performed for this study. A pre-test calculation using RELAP5/MOD1 was used for this purpose.

Sensitivity studies: Sensitivity studies were performed to investigate primary-to-secondary heat transfer during a decreasing secondary inventory condition. The studies also addressed the feed-and-bleed mode of energy removal from the primary system.

Nodalization studies: As noted above, nodalization modifications were made to evaluate the impact on code calculation capability for the above conditions in the LOBI experiments.

Summary: This report assess RELAP5/MOD2 Version 36.04 through a comparison of calculated results with LOBI experiment ST-02 (BT-00). The experiment simulated the response of a pressurized water reactor to a loss of main feedwater, followed by a reactor trip, a termination of all auxiliary feedwater, boil-off of the steam generator secondary inventory, and operator recovery involving a primary-side feed-and-bleed cooling procedure.

A pre-test calculation was not performed with RELAP5/MOD2; rather one was performed and documented using RELAP5/MOD1. A RELAP5/MOD1 model of the LOBI facility was then converted for RELAP5/MOD2. A special effort was made to incorporate a split-downcomer model in the upper region of the reactor vessel to better simulate possible multi-dimensional effects where bypassed steam mixes with cold ECC fluid. The split-downcomer model had to be abandoned because of calculational difficulties that led to run termination.

So much difficulty was encountered in attempting to directly compare a base case RELAP5/MOD2 simulation of the test with the experimental data, that the performance of a "base case" was skipped. Instead, the experiment response was subdivided into phases and sub-phases and satisfactory agreements of the calculated and experimental data were obtained for each in chronological order. In this summary, the difficulties encountered are first discussed, followed by discussions of code performance (where it could be adequately compared with data) and a global comparison of the final code calculation with the experiment data.

Experimental Difficulties

Problems with the initial test conditions were encountered. The main steam valve appeared to be closed immediately at the beginning of the test rather than at 1.5 s as specified in the test plan. On the secondary side, one of the steam generator levels was measured incorrectly due to controller malfunction and that generator thus began the experiment with too low a level. Timing of steam generator boil-off was found to be significantly affected by this error.

The test plan specified complex operations using the pressurizer and steam generator secondary relief valves. During the test these valves were controlled in a manner other than was planned, and some data regarding their control during the test were lost. The code/data comparisons were adversely affected by these anomalies; and as a result the data proved more useful for determining how the test was run than as benchmark experimental data against which the code may be compared. The list of uncertainties for the valve operation include the discharge rates, the valve flow areas as functions of the stem positions, the settings and responses of the electronic valve controllers, and the timings and discharges of the backup valves.

The LOBI primary and secondary side heat losses were found to be poorly characterized. For a long-term transient such as this, the heat loss anomalies were found to be controlling at times.

A spurious primary-to-secondary side leak in the broken loop steam generator appears to have been present during the test and this adversely affected the draining time after boil-off of that steam generator.

Code Performance

RELAP5 was shown to well predict the onset of degraded primary-to-secondary heat transfer as the steam generator levels fell. The code was also shown to predict the thermal stratification expected in the bottom of the pressurizer during periods of insurge and outsurge.

RELAP5 performance in predicting the virtually complete loss of primary-to-secondary heat transfer at completion of the boil-off was shown to be adversely affected by a basic RELAP5 modeling assumption. Specifically, it is not possible to partition heat transfer from a heat structure to a fluid volume as being above or below a mixture level. As a result, excessive wall-to-fluid heat transfer is calculated as a mixture level falls through a fluid volume and heat transfer is not degraded until an extremely voided condition is present. The user's only recourse is to use finer nodalization such that the error introduced is acceptable. However a nodalization study is needed to determine acceptability and the resulting model can be expensive to use because of the added nodes.

The method of attaining an adequate simulation of full-power primary-to-secondary heat transfer by setting the heated equivalent diameter on the secondary-side of the U-tubes to the minimum tube-to-tube spacing may adversely affect the calculation when the secondary side is stagnant. The method permits an adequate representation of both the secondary side pressure and the primary-to-secondary heat transfer and is believed to compensate for the lack of understanding of the actual flow mechanisms in the steam tube bundle region. However, when the secondary-side recirculation is lost, such as was caused by the boil-off in this experiment, then the primary-to-secondary heat transfer appears to be overpredicted. Further work is recommended in this area.

The previously documented deficiency regarding the overprediction of level swell in pool boiling situations was shown to affect the pressurizer PORV flow during the feed-and-bleed portion of the experiment. Because the code overpredicted the level swell, the pressurizer level tended to be higher than the data showed. As a result the liquid content of the PORV flow was too high.

Global Comparison

An overall comparison between the final RELAP5/MOD2 calculation and the experimental data is provided by the primary- and secondary-side pressure responses in Fig. 3.2.29.1.

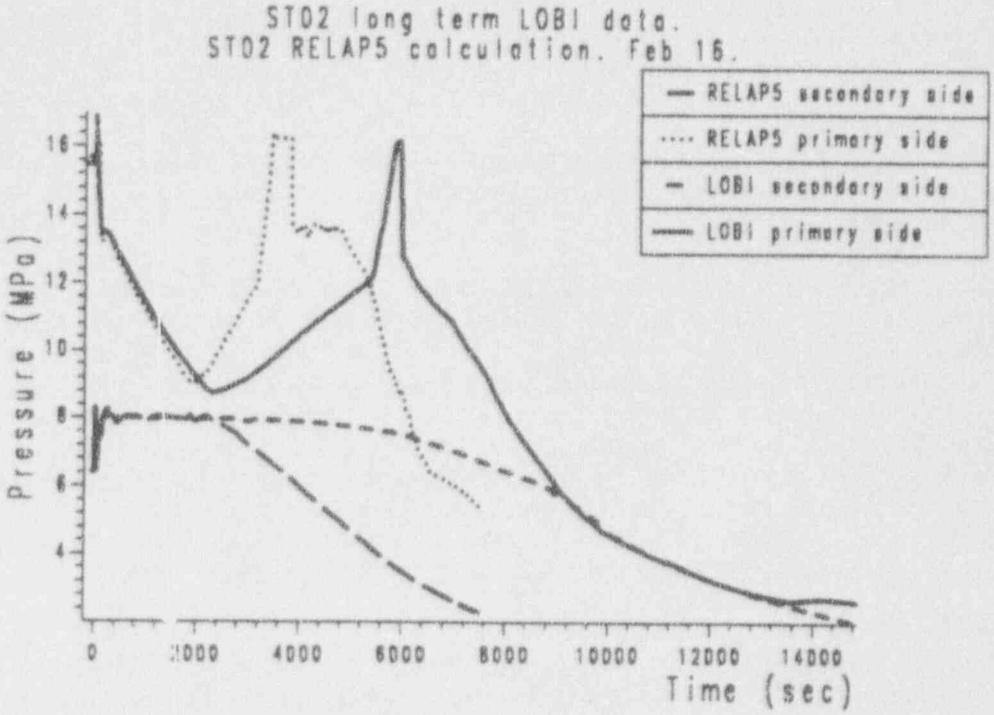


Figure 3.2.29.1 Global comparison: Post-test Calculation.

3.2.30 FIX-II Experiment 5061 Assessment

Reference: J. Eriksson, *Assessment of RELAP5/MOD2, Cycle 36.04 Against FIX-II Guillotine Break Experiment No 5061*, NUREG/IA-0016, July, 1989.

Code version: RELAP5/MOD2, Cycle 36.04.

Facility: FIX-II in Nykoping, Sweden.

Objectives: Evaluate the code's capability to simulate the integral behavior of the FIX-II facility during a 200% break simulation. In particular, the code's ability to calculate the break mass flow and the system depressurization were to be examined in some detail. The objectives were met.

Major phenomena: Break mass flow rate, primary system depressurization, coolant temperatures, heater-rod cladding temperatures and rod thermal behavior.

Code deficiencies: None.

User guidelines: None.

Base calculation: When the base calculation was done, it was determined that the system initial mass was probably underestimated. The author attributed the mass discrepancy to an under calculation of the condensed liquid addition to the facility system volume from the separator/condenser component. However, the reviewers did not find sufficient evidence to support the author's belief.

Sensitivity studies: Two sensitivity studies were conducted to study the effect of: (a) increasing the system initial mass, and (b) a different pump outlet restriction. Study (a) showed a marked improvement in the agreement between the break mass flow rate, the distribution of the flow from both branches of the break, and the system depressurization rate. Study (b) showed a decrease in agreement with the data and thus was not discussed in great detail in the report.

Nodalization studies: The base case model was renodalized in the steam separator, the downcomer, and the broken leg volumes (on the downcomer side) to study the effect of nodalization density. Worse agreement was obtained with the data than shown by the base calculation. However, no explanation was given by the author.

Summary: The assessment was performed using the Test 5061 data obtained in the FIX-II facility.

The FIX-II facility is a 1/777-volume scaled model of the Oskarshamn-II boiling water reactor nuclear power plant. The plant is an external recirculation pump design. The facility contained a full-length electrically-heated core bundle simulator. The facility has 6x6 core bundles instead of 8x8 bundles, as found in the reference plant. No emergency core cooling systems were mounted in the FIX-II facility - it was built for conducting blowdown experiments only.

Test 5061 simulated the occurrence of a 200% break in one of the plant's recirculation lines. The experimental blowdown period lasted for approximately

27 s. after the break valves were opened.

The assessment calculations were performed using a RELAP5 model constructed by Studsvik. Four calculations were performed to study the combined sensitivity of break flow and system depressurization to the initial coolant mass, junction options, and break discharge line nodalization. The break mass flows, especially during two-phase, low-quality conditions, were generally undercalculated at the same time system depressurization was overcalculated. The author concluded that this discrepancy resulted from an undercalculation of the initial coolant mass and that this was caused by an undercalculation of the fall velocity of the spray droplets in the separator/condenser steam atmosphere.

For the base calculation the measured initial water level was used. The calculated system pressure decreased significantly more quickly than in the experiment. At the same time, the calculated break mass flow rate was undercalculated compared to the data. The calculation predicted core heatups at all elevations - generally at times later than measured in the experiment.

As a result of the author's analysis of the base calculation, the system initial mass was surmised to be too low. The calculation was redone by including an additional 16 kg of mass in the model. In general, better results were obtained since both the calculated system depressurization and the break mass flow rates matched the data better.

The reviewers concluded that the author's evidence supporting his contention that the model had less mass than the experiment was inconclusive. Hence, the results of the sensitivity calculations were not examined further.

3.2.31 FIX-II Experiment 3027 Assessment

Reference: J. Eriksson, *Assessment of RELAP5/MOD2, Cycle 36, Against FIX-II Split Break Experiment No. 3027*, NUREG/IA-0005, September, 1986.

Code version: RELAP5/MOD2, Cycle 36.04.

Facility: FIX-II in Nykoping, Sweden.

Objectives: Evaluate the ability of the code to simulate the thermal-hydraulic events associated with an intermediate (31% break in the pump outlet to lower plenum inlet piping) loss-of-coolant accident in boiling water reactors.

Major phenomena: Break mass flow, system pressure response, coolant inventory, core pressure drop, fluid densities, fluid temperatures, and core heater rod temperatures.

Code deficiencies: None. But three discrepancies between the data and code calculation were identified; the code did not match the measured (a) core pressure drop, (b) heater rod dryout times, and (c) initial fluid inventory of the facility. However, the reviewers did not find sufficient evidence in the report to allow identification of any code deficiencies.

Impact of deficiencies: No deficiencies were identified.

User guidelines: None.

Base calculations: Many of the parameters identified as major phenomena were calculated to be close to the data. However, the calculated heater rod dryout occurred later than indicated by the measured data. The heatup rates were comparable to the measured data.

Sensitivity studies: Three sensitivity studies were done to examine the influence of (a) increasing the critical break mass flow rate, (b) increasing the initial quantity of system inventory, and (c) removing some heat structures as passive heat sources. Each sensitivity calculation was conducted including the input change used in the previous calculation. The results of the sensitivity studies with respect to defining code deficiencies was inconclusive.

Nodalization studies: None

Summary: The assessment was performed using the Test 3027 data obtained in the FIX-II facility.

The FIX-II facility is a 1/777-volume scaled model of the Oskarshamn-II boiling water reactor nuclear power plant. The plant is an external recirculation pump design. The facility contained a full-length electrically-heated core bundle simulator. The facility has 6x6 core bundles instead of 8x8 bundles, as found in the reference plant. No emergency core cooling systems were mounted in the FIX-II facility - it was built for conducting blowdown experiments only.

Test 3027 simulated the occurrence of a 31% break (31% of the flow area of one line) in the line from the second recirculation pump to the lower plenum. The author labeled this break a "split break." The total transient time was about 80 s.

The assessment calculations were performed using a RELAP5 model constructed by Studsvik. Four calculations were performed to study the calculation's sensitivity to break mass flow, initial coolant mass, and passive heat structures.

The base calculation gave a lower core pressure drop than measured. The author contended this result occurred because the code did not properly simulate the strong dependence of the vapor fraction on wall friction. Heater-rod dryout times were calculated by the code to occur later than measured, especially at elevations above the core midplane. This, the author contended, resulted because a multiplier is applied to the critical heat flux correlation. In addition, the code under calculated the initial fluid system mass by approximately 30 kg. During the steady-state operation, liquid coolant is sprayed into the condenser and plays an important role in the heat removal process. The code uses the vertical slug flow regime to calculate the spray droplet fall velocity of 1.2 m/s. However, based on vendor data, the fall velocity is only about 0.8 m/s. This difference, the author contends, results in the under calculation of the initial fluid content. Review and evaluation of the supporting evidence however, indicated that additional evidence and/or analysis is needed to support the above observations.

The reviewers concluded that the author's evidence supporting his contention that the model had less mass than the experiment was inconclusive. Hence, the results of the sensitivity calculations were not examined further.

3.2.32 FIX-II Experiment 3051 Assessment

Reference: J. Eriksson, *Assessment of RELAP5/MOD2, Cycle 36.04 Against FIX-II Split Break Experiment No. 3051*, NUREG/IA-0029, October, 1989.

Code version: RELAP5/MOD2 Cycle 36.04.

Facility: FIX-II in Nykoping, Sweden.

Objectives: Study the ability of the code to simulate the thermal-hydraulic phenomena associated with a SBLOCA in a BWR recirculation loop.

Major phenomena: System pressure response, break mass flow rate, fluid temperatures and densities, and heater-rod cladding temperature response.

Code deficiencies: None.

User guidelines: None.

Base calculations: Differences between the calculated break mass flow rate and the system depressurization were noted. The differences were attributed to (a) a calculated break mass flow rate that is larger than the measured values, (b) inadequately calculated core and downcomer inventory, and (c) calculated heat transfer coefficients that show a larger variation with time than the measured values. However, the evidence given for the above three differences was not sufficient in the reviewer's opinion to allow a conclusive statement concerning code deficiencies.

Sensitivity studies: Two sensitivity studies were conducted: (a) The discharge coefficient, for subcooled blowdown, was decreased from 1.0 to 0.76 and (b) the heat transfer coefficients for the outside surface of various heat structures were changed to decrease the core inlet temperature. Both sensitivity study calculations gave better agreement with the data than did the base calculation.

Nodalization studies: None.

Summary: The assessment was performed using the Test 3051 data obtained in the FIX-II facility.

The FIX-II facility is a 1/777-volume scaled model of the Oskarshamn-II boiling water reactor nuclear power plant. The plant is an external recirculation pump design. The facility contained a full-length electrically-heated core bundle simulator. The facility has 6x6 core bundles instead of 8x8 bundles, as found in the reference plant. No emergency core cooling systems were mounted in the FIX-II facility - it was built for conducting blowdown experiments only.

Test 3051 simulated the occurrence of a 10% break (10% of the flow area of one line) in the line from the second recirculation pump to the lower plenum. The author labeled this break a "split break." The total transient time was about 140 s.

The assessment calculations were performed using a RELAP5 model constructed by

Studsvik. Three calculations were performed to study the calculation's sensitivity to break mass flow and passive heat structure.

The base calculation showed a larger than measured critical mass flow at the break early in the transient during subcooled blowdown (for the first 43 s). Thereafter, reasonable agreement was shown between the data and the calculation. The measured critical break flow was based on the accumulation of the coolant mass in their blowdown catch tank. The piping from the break to the catch tank was initially full of liquid and thus may have significant error, especially during the early portion of the transient. Consequently, the significance of the mismatch between the calculation and measurement early in the transient is unclear.

Comparisons between the calculated and measured differential pressures over the length of the core and downcomer regions showed the calculated differential pressures were low for both regions. The author attributed the differences to be caused by an undercalculation of the region mass and also a nonrepresentative calculation of the frictional pressure losses. However, it was not clear to the reviewers that frictional pressure losses were considered when converting the pressure drop data to mass inventory. Consequently, the error associated with these measurements is unknown.

The author evaluated the heat transfer coefficients present during the experiment. His comparison between the measurement and the calculation showed the calculated values to have a larger variation with time than the RELAP5 calculation. Since the measured temperature differences were small, the uncertainty of the measured values may be significant. No definite explanation of the differences between the calculation and the measurements were offered by the author.

Based on the lack of information regarding the experimental uncertainty, the reviewers concluded that code deficiencies could not be deduced from the observed differences between the calculation and the measurement.

3.2.33 Semiscale Large Break LOCE S-06-3 Assessment

Reference: K. S. Liang, L. Kao, J. L. Chiou, L. Y. Liao, S. F. Wang, and Y. B. Chen, *Assessment of RELAP5/MOD2 Using Semiscale Large Break Loss-of-Coolant Experiment S-06-3*, NUREG/IA-0046, April, 1992.

Code version: RELAP5/MOD2 Cycle 36.04.

Facility: Semiscale facility located at the Idaho National Engineering Laboratory.

Objectives: Study the ability of the code to simulate the thermal-hydraulic phenomena associated with a LBLOCA in the 1/1700-scale Semiscale facility. Other specific objectives were to determine the effect on the overall system response of nodalization changes in the following regions of the base case model: (i) the pressurizer, (ii) radial flow connections between the average and hot fuel channels, (iii) the number of axial heat slab intervals used in the two-dimensional reflood calculation, (iv) the core, (v) cross-flow junctions at the vessel inlets, and (vi) not using the reflood model.

Major phenomena: Thermal-hydraulic phenomena representative of the blowdown, refill, and reflood phases of a LBLOCA.

Code deficiencies: The code cannot simulate CCFL observed in the downcomer.

User guidelines: None.

Base calculations: The base case calculation was completed using a model built at INEL.

Sensitivity studies: A sensitivity study was conducted to determine whether the code provided better calculated behavior with or without the reflood model activated.

Nodalization studies: Studies were performed to evaluate the effect of changing the nodalization in the pressurizer, the core, using cross-flow junctions at the vessel-to-cold leg connections, and the axial heat slab intervals in the core for the reflood model.

Summary: Data from the large break loss-of-coolant accident (LBLOCA) simulation experiment S-06-3, conducted in the Semiscale facility, was used to assess the code.

The experiment was conducted in the MOD-1 configuration of the Semiscale facility. The system included an active coolant loop which represented three coolant loops in a prototype pressurized water reactor. A simulated loop was used to represent an affected loop. The MOD-1 facility was scaled from the LOFT facility and was used as a scoping facility to help in the design and planning of the LOFT program. The experiment was a simulation of a 200% offset shear LBLOCA conducted with 75% of the maximum core power.

Analysis of the base calculation showed that the code generally captured the observed phenomena. Specific exceptions to this statement follow. During the blowdown phase of the transient the code predicted a later draining of the pressurizer which was attributed to inappropriate interaction between the liquid and vapor phase in the pressurizer and to inappropriate interaction with the heat structures in the pressurizer. However, the overall blowdown response was in good agreement with the observed response.

During the refill phase of the transient the code was not able to capture the CCFL related behavior in the downcomer associated with the hot wall delay that occurred in the MOD-1 system during high pressure injection system injection. In the facility, HPIS liquid was bypassed to the break due to steam generation in the downcomer causing holdup of the liquid. The code does not have a CCFL model. The failure to capture the hot wall delay phenomena resulted in a much earlier refill of the downcomer, lower plenum, and beginning of reflood. During the reflood phase of the transient the code predicted excessive amounts of liquid entrainment as the core reflood progressed resulting in excessive precursory cooling prior to surface quench.

Several sensitivity studies were performed to investigate the minor shortcomings in the code for the simulation of the Semiscale S-06-3 experiment. The areas of sensitivity studies are listed below:

1. Pressurizer modelling: the pressurizer model was reduced from 13 axial volumes to 5. The result was essentially the same as in the base calculation with the best agreement with experimental data occurring with 13 axial volumes.

2. Radial connection between core average and hot channels: the cross-flow connections used between the average and hot channels in the base calculation were eliminated. The observed difference in peak cladding temperature was small and only on the hot rods. The effect on the reflood calculation was noticed in a 6 to 10 s delay in the rod quench. The reviewer noted that to realistically represent the Semiscale core region cross-flow junctions should be used as the average and hot channels can freely exchange fluid.

3. Effect of the number of axial heat slabs on a two-dimensional reflood calculation: the heat slab axial maximum interval was varied from the base calculation of 8 to 32 and 2. In each case the observed response was similar. The review notes that the sensitivity results suggest that a threshold value may exist and that the threshold value is near 8. The results for the case of 8 and 32 are very similar.

4. Effect of the number of axial core hydraulic volumes: the number of axial core nodes was varied from the 11 used in the base calculation to 22 and 5. Differences were observed primarily when the number of nodes was decreased to 5. The reviewer suggests that the differences observed in the sensitivity were primarily due to the effect of averaging that enters into the calculation when 5 nodes are used as opposed to 11 or 22 to represent the thermal-hydraulic response to a LBLOCA.

5. Use of cross-flow junctions at reactor vessel entrances: the normal

junctions used in the base calculation at vessel penetrations were changed to cross-flow junctions to investigate the effect of momentum transfer on the system response. The observed differences were noted as insignificant.

6. Investigation of the axial conduction effects on the reflood heat transfer in the core: the code reflood package was disabled and the calculation was altered to investigate the effect of axial conduction in the reflood heat transfer process that leads to cladding quench. The determination was that axial conduction has minimal effect on the reflood heat transfer response.

The major conclusion of this assessment study was that the code was unable to accurately calculate the delivery of ECCS liquid to the lower plenum due to the lack of a CCFL model. The hot-wall delay observed in the experiment was not calculated. In other respects the code did a reasonable job of calculating the experimental thermal-hydraulic behavior.

3.2.34 Semiscale Small Break LOCA Experiment S-LH-1 Assessment

Reference: r. C. Hall and D. R. Bull, *Analysis of Semiscale Test S-LH-1 Using RELAP5/MOD2*, NUREG/IA-0064, April, 1992.

Code version: RELAP5/MOD2 Cycle 36.05, Winfrith version E03

Facility: Semiscale Mod 2-C

Objectives: Evaluation of the code capability to simulate thermal-hydraulic behavior during a small break loss-of-coolant accident in a pressurized water reactor. Specifically, the phenomena associated with liquid hold-up in the steam generator tubes.

Major phenomena: Primary and secondary side pressure response, break mass flow rate, collapsed liquid levels in the reactor vessel, steam generator tubes, and coolant pump suction legs, core heat transfer, draining of the team generator U-tubes, and clearing of the loop seals.

Code deficiencies: The code was noted as not able to characterize the mixture level behavior in the core during core level depression and boil-off events.

User guidelines: Code users should carefully consider leakage when modeling steam valves.

Base calculation: The base calculation was performed using RELAP5/MOD2 Cycle 36.05, Winfrith version E03. The calculation showed that in general the code predicted liquid distribution reasonably well. However, the code was not able to provide the correct distribution of coolant in the core region during the core uncovery phase.

Sensitivity calculation: A sensitivity study was performed to determine the leakage in the steam valve after the closure occurred. The study indicated that a leakage area of about 0.16% of full-open area explained the differences between the calculated and measured results.

Nodalization calculations: A nodalization study was performed to evaluate multi-dimensional core effects.

Summary: The code (RELAP5/MOD2 Cycle 36.05 - Winfrith Version E03) was assessed using the Semiscale Experiment S-LH-1 data.

The S-LH-1 experiment simulated the response of a pressurized water reactor to a small break loss-of-coolant accident with a break flow area equal to 5% of the cold leg cross section. The experiment was designed to investigate the effects on the core level of liquid hold-up in the steam generator U-tubes. As the primary coolant system drains, voiding at the top of the U-tubes interrupts natural loop circulation. It is the period following this interruption and before clearing of a loop seal that is of most interest. During this period, differences in the draining rates of the upflow and downflow sides of the U-tubes provide a static head effect that depresses the core level. Draining of the upflow side is opposed by steam flow from the core while in the downflow side it is aided by the steam flow. The difference between the upflow and downflow side inventories creates a static head that tends to decrease the core level.

The model used was derived from one originally created by INEL for simulating the Semiscale S-LH test series. For the calculations reported here the Ardron-Bryce break offtake model was used. Single- and two-phase break discharge coefficients of 0.9 were used.

Three hundred seconds of transient time was used to drive the model to a satisfactory steady state at the test initial conditions. The time dependent volumes controlling the pressure were then removed and an additional 50 s of transient calculation was performed prior to opening the break for the test simulation. Reasonable agreement was obtained between the calculated and measured initial test conditions.

The calculated depressurizations of the primary and secondary systems were somewhat slower than measured. As a result, many actions keyed to the attainment of low pressure setpoints were delayed in the calculation as compared to the test. These discrepancies were found to be caused by a finite steam outlet valve leakage following its closure. No leakage was modeled in the RELAP5 base case calculation. A sensitivity study indicated the actual valve leakage was equivalent to that through 0.16% of the full-open valve area. Once their cause is understood, these discrepancies are seen as not particularly important to the experiment prediction.

Reasonable agreement was indicated between the calculated and measured break mass flow rates. Some differences were noted between 50 and 175 s, in the low-quality break flow region, but these were considered minor.

The RELAP5 calculation included significant steam generator U-tube liquid hold-up. Generally, about 1 m too much liquid holdup is calculated by the code, and the calculated hold-up is less prolonged, than is indicated in the measured data. The calculated collapsed liquid levels in the broken loop pump suction leg compared poorly with the test data. This anomaly was caused by differences in the calculated and measured loop seal clearance behavior; this difference is however of little practical importance to the overall prediction.

Comparison between the calculated and measured reactor vessel downcomer and core collapsed levels showed the code prediction to have reasonable agreement with the measured data during the loop seal clearing portion of the transient. A loop seal was cleared at 175 s after which the core level initially recovered and then declined again due to boil-off of the coolant. The declines in the calculated downcomer and core collapsed levels during the boil-off period were not as rapid as in the experiment data.

A comparison of representative calculated and measured fuel rod temperatures showed minimal agreement (Fig. 3.2.34.1). At first, the poor comparison was suspected to be due to a lack of adequate radial core nodalization in the model for defining the core mixture level during a period of reflux natural circulation cooling (see Fig. 3.2.34.2). The calculation showed the presence of less liquid in the core region at the time of loop seal clearing (175 s) but more liquid in the core region 370 s to 750 s indicating the code's inability to distribute the primary inventory correctly. A nodalization study was performed with the core model subdivided into two uncoupled channels. This was an attempt to simulate expected multidimensional behavior with steam exiting the center of the core bundle and returning condensate entering the core at its periphery. Renodalized core results showed the opposite of those originally attained during the period of loop seal clearing; the code predicted much more heatup than was observed in the experiment. However, the calculation still showed more primary inventory in the core during the core boiloff period. The conclusion was that it is likely not a core multidimensional effect that caused the poor fuel rod temperature prediction, but rather simply a lack of adequate axial core nodalization and a deficient interphase drag model. Based on this study, it is recommended that, for calculations where tracking of the core mixture level and resulting fuel rod heatup are expected to be important phenomena, a code employing a very fine axial mesh be used for these determinations. RELAP5 output may be used as boundary condition input for this side calculation.

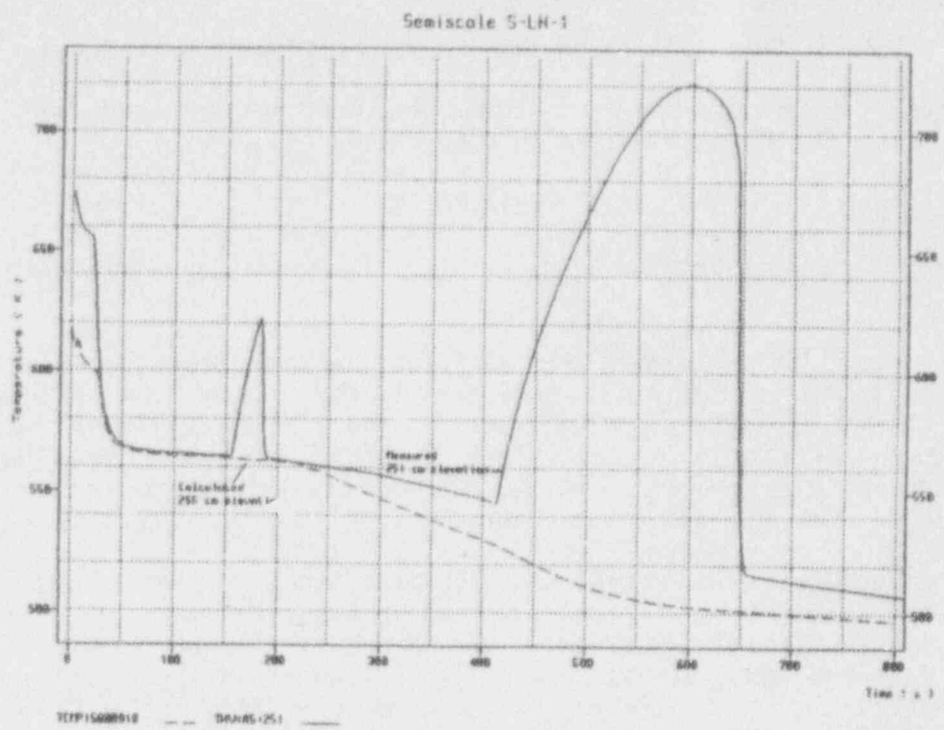


Figure 3.2.34.1 Measured and Calculated Heater Rod Temperatures (Base Case 1).

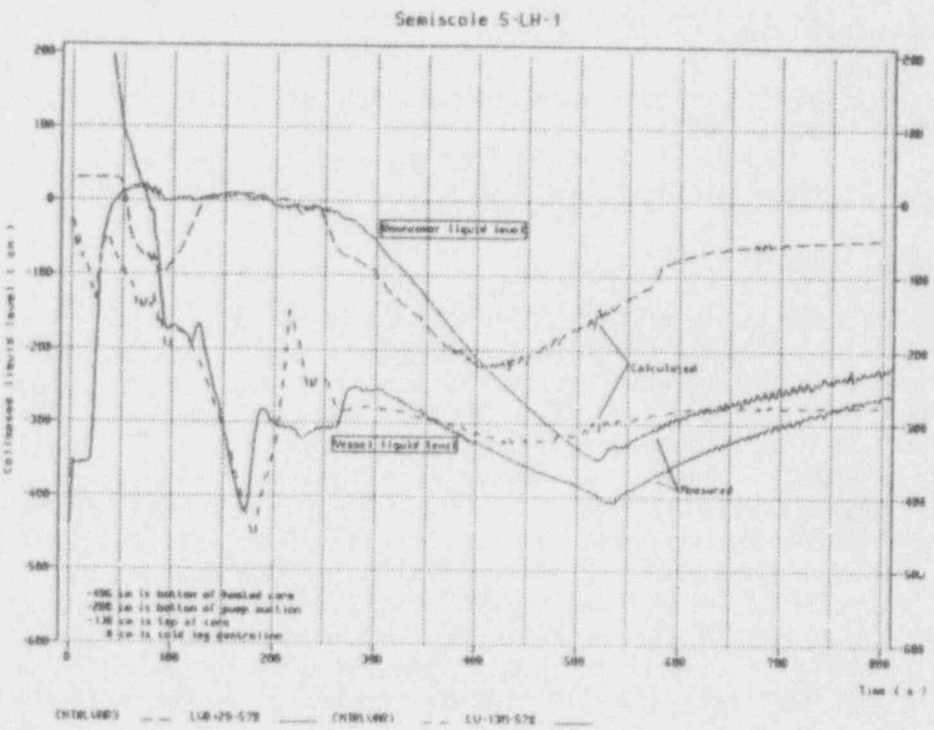


Figure 3.2.34.2 Vessel and Downcomer Collapsed Liquid Levels (Base Case 1).

3.2.35 Semiscale Small Break LOCA Experiment S-LH-2 Assessment

Reference: P. Brodie and P. C. Hall, *Analysis of Semiscale Test S-LH-2 Using RELAP5/MOD2*, NUREG/IA-0065, April, 1992.

Code version: RELAP5/MOD2 Cycle 36.05.

Facility: Semiscale facility located at the Idaho National Engineering Laboratory.

Objectives: The assessment was performed to determine code capabilities for simulating thermal-hydraulic behavior during a small break loss-of-coolant accident in a pressurized water reactor. The S-LH-2 experiment simulated a cold leg break (with an area equivalent to 5% of the cold leg flow area) in a pressurized water reactor. Of particular interest in this experiment was the hold-up of liquid in the steam generator tubes and its effect on the core level.

Major phenomena: Comparisons of calculated and experimental data pertinent to small break loss-of-coolant accidents in pressurized water reactors are shown. These comparisons include primary- and secondary-side pressures, break mass flow rate, fuel rod temperatures, and collapsed levels in the reactor vessel, steam generator tubes, and main coolant pump suction legs. The primary phenomena addressed in the report regard the draining of the steam generator U-tubes, the clearing of the loop seals, and the resulting hydrostatic effects upon the core level.

Code deficiencies: The code was unable to predict the core mixture level behavior during the boil-off event following clearing of the loop seal.

User guidelines: None.

Base calculations: The base case calculation was performed using the model described in Loomis, 1985.

Sensitivity studies: None.

Nodalization studies: None.

Summary: The code assessment (RELAP5/MOD2 Cycle 36.05 - Winfrith Version E03) was based on the Semiscale Experiment S-LH-2 data. This experiment simulated the response of a pressurized water reactor to a small break loss-of-coolant accident with a break flow area equal to 5% of the cold leg cross section. The experiment was designed to investigate the effects on the core level of liquid hold-up in the steam generator U-tubes. As the primary coolant system drains, voiding at the top of the U-tubes interrupts natural loop circulation. It is the period following this interruption and before clearing of a loop seal that is of most interest. During this period, differences in the draining rates of the upflow and downflow sides of the U-tubes provide a static head effect that depresses the core level. Draining of the upflow side is opposed by steam flow from the core while in the downflow side it is aided by the steam flow. The difference between the upflow and downflow side inventories creates a static head that tends to decrease the core level.

The model used was derived from one originally created by INEL for simulating the Semiscale S-LH test series, and was the same model (except for upper reactor vessel bypass modeling) as was used for a CEGB assessment calculation for Test S-LH-1 (see Section 3.2.34). For the calculations reported here the Ardron-Bryce break offtake model was used. Single- and two-phase break discharge coefficients of 0.9 were used. Tests S-LH-1 and S-LH-2 were identical, except for the upper reactor vessel bypass. In S-LH-1 this bypass was 0.9% of core inlet flow and in S-LH-2 it was 3%. This difference was accounted for in the model by reducing the loss coefficient for the bypass path from 2.57 to 0.125.

Three hundred seconds of transient time was used to drive the model to a satisfactory steady state at the test initial conditions. The time dependent volumes controlling the pressure were then removed and an additional 50 s of transient calculation was performed prior to opening the break for the test simulation. Generally good agreement was obtained between the calculated and measured initial test conditions.

The calculated depressurizations of the primary and secondary systems were somewhat slower than measured. As a result of the slower calculated depressurization, actions keyed to the attainment of low pressure setpoints were delayed in the calculation as compared to the test. These discrepancies were noted in Section 3.2.34 to be caused by a finite steam outlet valve leakage following its closure. No leakage was modeled in the RELAP5 calculation. A sensitivity study in the S-LH-1 assessment indicated the actual valve leakage was equivalent to that through 0.16% of the full-open valve area.

A reasonable agreement was indicated between the calculated and measured break mass flow rates.

Comparison of the calculated and measured reactor vessel downcomer and core collapsed levels showed the code predictions were in reasonable agreement with the measured data. Minimal agreement was obtained with the measured fuel rod temperatures. Again, as discussed in Section 3.2.34, the poor comparison is believed to have been caused by a lack of adequate axial core nodalization and deficiencies in the code's interphase drag model. It was recommended that, for calculations where tracking of the core mixture level and resulting fuel rod heatup are expected to be important phenomena, a code employing a very fine axial mesh be used for these determinations. RELAP5 output may be used as boundary condition input for this side calculation.

3.2.36 Semiscale Small Break LOCA Experiment S-NH-1 Assessment

Reference: E. J. Lee, B. D. Chung, and H. J. Kim, *RELAP5 Assessment Using Semiscale SBLOCA Test S-NH-1*, Korea Institute of Nuclear Safety.

Code version: RELAP5/MOD2 Cycle 36.04

Facility: Semiscale Mod 2-C located at the Idaho National Engineering Laboratory, Idaho Falls, ID.

Objectives: Evaluation of the code capability to simulate thermal-hydraulic behavior during a small break loss-of-coolant accident in a pressurized water reactor.

Major phenomena: Primary and secondary side pressure response, break mass flow rate, core thermal-hydraulic response, and the primary mass inventory distribution.

Code deficiencies: The break mass flow calculation was deficient (the integrated mass flow was underpredicted), the calculated primary mass distribution did not match the data, and no core heatup was calculated.

User guidelines: The authors and the reviewers indicated that the break area and the break discharge coefficients can be modified to obtain the correct break mass flow. However, no guidelines for making generic changes have been identified.

Base calculation: The base calculation was performed using a RELAP5 model of the Semiscale MOD2C facility provided by INEL.

Sensitivity calculation: Sensitivity studies were performed to examine the effect of changing the break flow area, using the downward break junction option, and the homogeneous break junction option.

Nodalization calculation: None.

Summary: The code was assessed using the Semiscale S-NH-1 experimental data.

The Semiscale MOD-2C facility was a 1/1705 volumetrically-scaled two-loop model of a 3411 MWt four-loop pressurized water reactor. The facility had an electrically-heated core. Semiscale consisted of a pressure vessel with simulated reactor internals and an external downcomer. The simulated core consisted of a 5x5 array (23 heated) of rods. The broken loop Type III steam generator had an external downcomer designed to measure the riser fluid density together with two inverted U-tubes. The intact loop steam generator contained six inverted U-tubes.

The S-NH-1 experiment was a simulation of a 0.5% small break loss-of-coolant accident in the cold leg.

The base calculations showed reasonable agreement with the experimental data. The authors noted several deficiencies based on the comparison between the data and the calculation:

1. The integrated break mass flow was underpredicted. A sharp transition from two-phase to single-phase vapor break flow was calculated.

2. An unrealistic calculation of liquid moving into the downcomer rather than to the break, as shown be the data, was noted.

3. Core heatup was not calculated even though core heatups were observed in the experiment.

4. Liquid holdup, observed in the experiment, was not observed in the calculation.

Deficiency 2 was caused by the interphase drag model. Deficiency 4 was not surprising since RELAP5/MOD2 does not have an countercurrent flow limiting model. The cause of deficiency 3 is not clear, however it probably was a direct result of deficiency 4.

Sensitivity studies were performed to examine the effect of changing the break flow area, using the downward break junction option, and the homogeneous break junction option. The authors increased the break flow area by 17% to compensate for the undercalculated break mass flow observed in the base calculation. The only notable result of these sensitivity calculations was observed for the case with the increased break area; a better match to the break mass flow data was observed and also a core heatup was calculated.

3.2.37 Semiscale Steam Line Break Experiment S-FS-1 Assessment

Reference: J. M. Rogers, *An Analysis of Semiscale Mod-2C S-FS-1 Steam Line Break Test Using RELAP5/MOD2*, NUREG/IA-0052, March, 1992.

Code version: RELAP5/Mod2 Cycle 36.05

Facility: Semiscale Mod-2C

Objectives: The assessment was performed to determine code capability to simulate the thermal-hydraulic behavior during a steam line break accident in a pressurized water reactor.

Major phenomena: Blowdown of the steam generator secondaries, peaking and degradation of the primary-to-secondary heat transfer, and cooldown and shrinkage of the primary-side fluid.

Code deficiencies: The interphase drag package was identified as deficient.

User guidelines: The reviewer suggested that the steam outlet valve be modeled using a restricted junction to facilitate convergence on a satisfactory hot standby steady state condition. With this method, the steam outflow is continuous at the desired rate and the solution is not perturbed by the opening and closing of a valve.

Base calculation: Difficulties were encountered during the hot standby plant conditions. The conditions were achieved by cracking the turbine bypass valve open slightly to eliminate valve cycling. During the transient calculation a satisfactory intact steam generator secondary pressure response was not obtained in the baseline calculation.

Sensitivity studies: A sensitivity calculation was performed to investigate the effect of intact loop steam generator boundary conditions specification. Specifically, the base case calculation failed to predict the secondary pressure response. The pressure was controlled by a time dependent volume in a sensitivity calculation to determine the effect of secondary pressure on the remaining system response.

Nodalization studies: Two steam generator separator nodalization schemes were evaluated. The two schemes differed in the elevation of the nodes comprising the separator region.

Summary: This report assesses RELAP5/MOD2 Cycle 36.05 (with Winfrith Cray error corrections) through a comparison of calculated results with Semiscale Experiment S-FS-1. This experiment simulated the response of a pressurized water reactor to a double-ended offset shear of one steam line. The experiment was designed to investigate the cooldown effects on the primary system resulting from the blowdown of the steam generator secondary. The experiment also included additional phases, investigating plant recovery; these are not covered in this report.

When the steam line ruptures, the steam generator secondary side blows down and secondary inventory is accelerated through the separator to the break. If modeling of the separator behavior is incorrect, then the secondary-side inventory and the primary-to-secondary side heat transfer responses also will be incorrect. In the experiment, both the broken and intact loop steam generators blow down initially (because they are connected through a common steam line header). The blowdown of the intact steam generator is terminated, via closure of the main steam isolation valves, when a low secondary-side setpoint pressure is attained.

The model used was derived from one originally created by EG&G Idaho for simulating the Semiscale S-LH test series. For the calculations reported here the separator region was renoded; two different separator nodings were investigated. The nodalization that provided the best match with the experimental data was the "B" nodalization appearing in Fig. 3.2.37.1.

The author encountered difficulties in obtaining a satisfactory steady state simulation of the hot standby plant initial conditions. The difficulty arises because the hot standby feedwater and steam flow rates are much lower than at full power. The steam flow tends to cycle the turbine bypass steam outlet valve open and closed and the solution is perturbed by the valve cycling. An alternate method used by the reviewer in similar applications is to "crack" the turbine bypass valve open slightly to pass a small steady steam flow; this eliminates the valve cycling and the difficulties it causes for obtaining a hot standby steady state.

Difficulty was also encountered obtaining a satisfactory intact steam generator secondary pressure response during the transient calculation. This difficulty was circumvented in some of the "final" calculations by controlling the intact generator pressure directly with a time dependent volume. This pressure response is sensitive to the modeling of the steam lines and their connections between the two steam generators; apparently the model was deficient in this area (but was not mentioned as being so by the author).

Three "final" calculations were performed using combinations of steam generator nodalization and constraint of the intact loop steam generator secondary-side pressure. The calculation with the "B" steam generator nodalization and with the intact loop steam generator secondary pressure controlled by a time dependent volume was considered the best prediction of the experimental data. This simulation was termed the "best estimate" calculation and the following discussion summarizes its comparison with the test data.

The course of the "best estimate" calculation is described as follows. When the steam line ruptures the secondary-side pressures decline immediately. Level swell in the boiler regions overwhelms the separator and liquid passes out the break. The primary-to-secondary side heat transfer increases greatly. The primary-side is rapidly cooled, its fluid shrinks, and the pressurizer level declines. The intact steam generator is isolated, terminating its blowdown, when the secondary pressure falls below a setpoint pressure. After that time the intact steam generator heat removal rate declines rapidly. The broken steam generator blowdown continues and its separator function is restored. As the secondary-side inventory is depleted, the heat transfer is degraded.

The broken loop steam generator secondary pressure comparison is shown in Fig. 3.2.37.2 and the broken loop cold leg temperature comparison is shown in Fig. 3.2.37.3. The pressure comparison is reasonable; the divergence is believed to be caused by an overprediction of interphase drag in the calculation. This overprediction resulted in carryout of too much liquid during the early portion of the calculation and a resulting deficit of secondary inventory at later times. The inventory deficit resulted in degrading primary-to-secondary heat transfer sooner in the calculation and a calculated primary-side cooldown rate that was lower than the test data.

Regardless of the poor comparison of cold leg temperatures, the calculated and measured primary-side pressure and pressurizer level responses generally were reasonable.

A firm comparison of calculated and measured steam generator heat removal rates was not possible because of inadequacies in experiment data acquisition. The trends of these calculated and measured data show the same trends. However, the calculation appears to be deficient in the period when the primary-to-secondary heat transfer degrades because its liquid is depleting. The calculated response tends to decline in discrete steps, caused by the successive dryout of the model's boiler region hydrodynamic cells. This behavior was not apparent in the measured data and the difference appears to be an artifact of the noding scheme. A relatively fine nodalization was employed, but this comparison indicates it is likely not fine enough to adequately match the measured data during the dryout phase of the transient.

The main conclusion of the assessment is that, overall, the calculation is a reasonable simulation of the experiment. However, the carryover of liquid during the first few seconds of the calculation was too great, resulting in an early degradation of primary to secondary heat transfer.

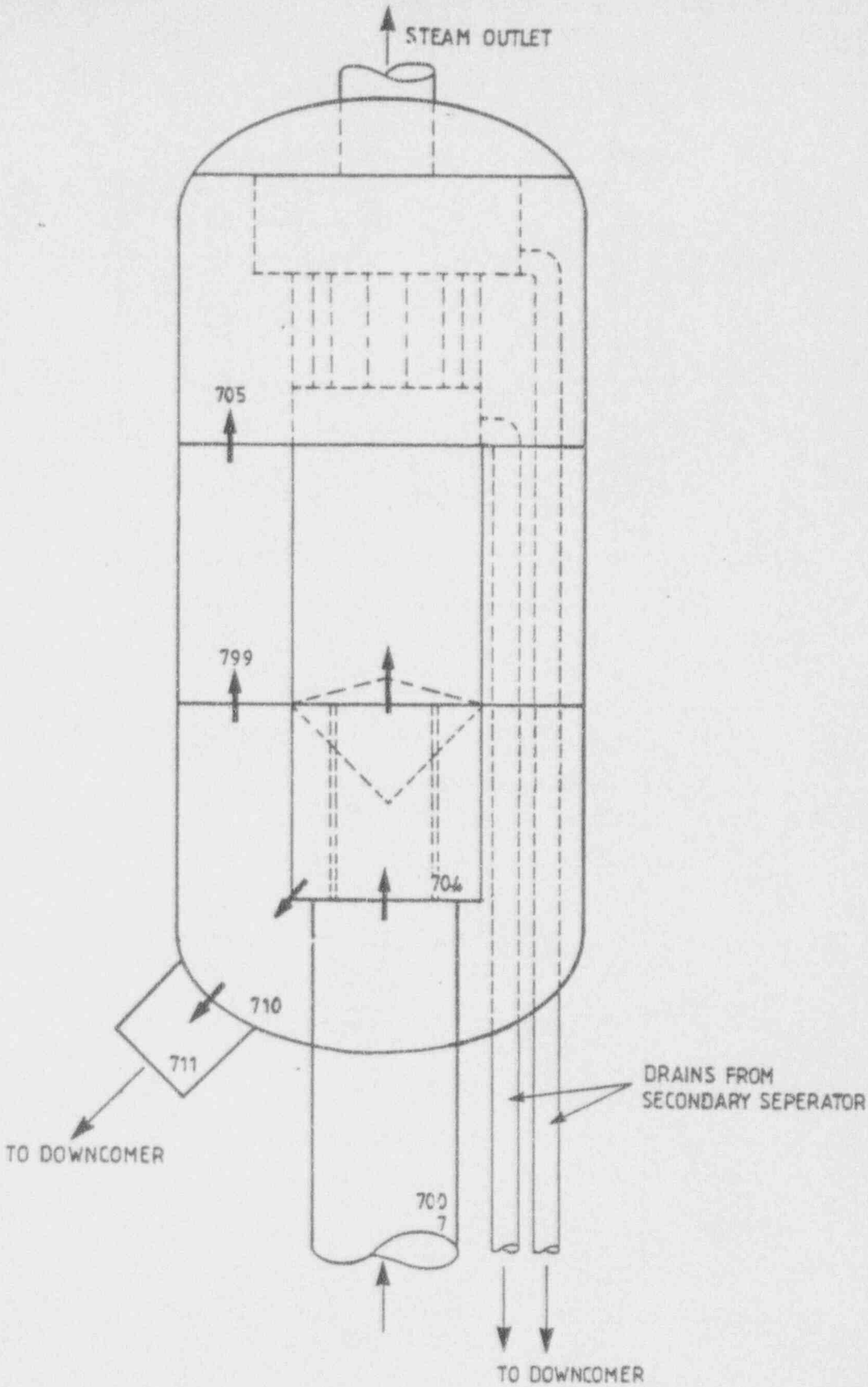


Figure 3.2.37.1 Semiscale Steam Dome and Separator Model "B".

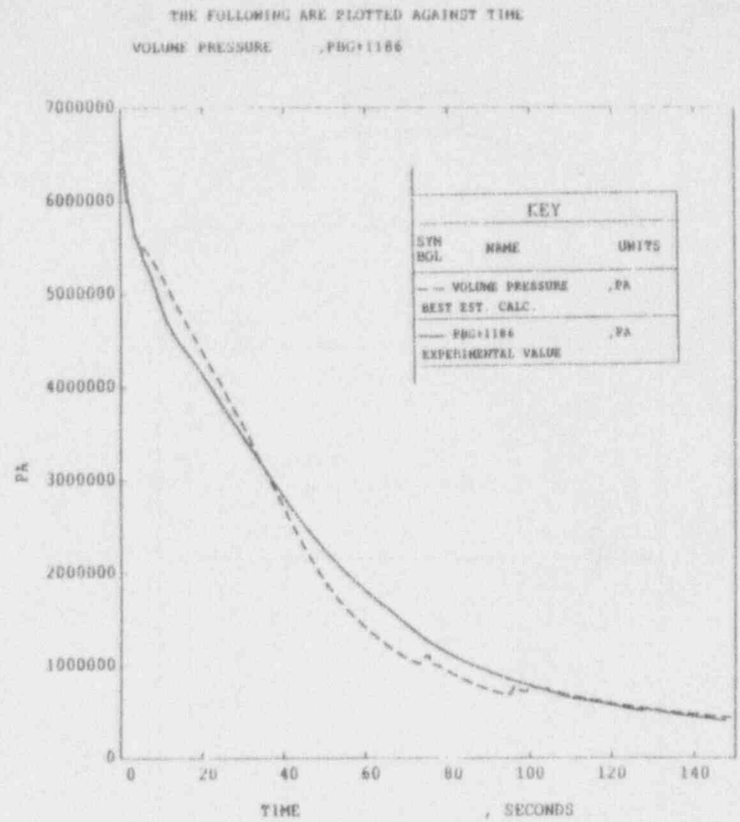


Figure 3.2.37.2 Broken Loop Steam Generator Pressure.

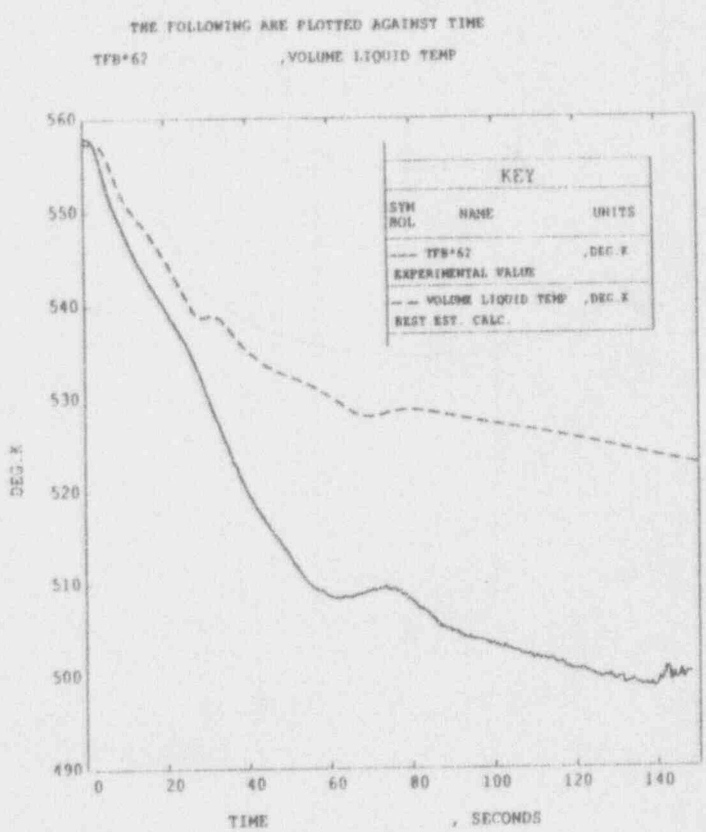


Figure 3.2.37.3 Broken Loop Cold Leg Temperature.

3.2.38 REWET-II Natural Circulation Experiment Assessment

Reference: J. Hyvarinen, T. Kervinen, *Assessment of RELAP5/MOD2 Against Natural Circulation Experiments Performed with the REWET-III Facility*, NUREG/IA-0059, April, 1992.

Code version: RELAP5/Mod2 Cycle 36.04

Facility: REWET-III integral effects test facility, Technical Research Centre of Finland

Objectives: Evaluation of the single-phase and two-phase natural circulation capability of the RELAP5 code using data from the REWET-III facility.

Major phenomena: Natural circulation of the primary coolant at decay power levels in the core. Both single-phase and two-phase natural circulation were addressed in the assessment.

Code deficiencies: The steady state search algorithm within RELAP was deemed inadequate since steady state was predicted while the thermodynamic state was still changing slowly. Also, a number of property table failures occurred in volumes nearly full of air with very small amounts of steam and liquid present.

User guidelines: None.

Base calculation: A base calculation was performed using the frozen version of the code. The code was not able to reproduce the single-phase flow oscillations that were observed in the experiment. However, for the two-phase conditions the code was able to duplicate the flow oscillations observed in the experiment although the code predicted a more rapid dampening of the oscillations.

Sensitivity studies: Sensitivity calculations were performed to determine the appropriate junction loss coefficients so that the calculated system flow rate would match the test data. The omission of hot leg environmental losses during the two-phase test was evaluated. It was shown that oscillations in flow and pressure would be predicted if the losses were not modeled.

Nodalization studies: None

Summary: Natural circulation experiments in the REWET-III facility were used for assessment of RELAP5/MOD2. Assessments were done for single- and two-phase conditions because both can be important for transient/accident heat removal in LWRs. The REWET-III facility was primarily designed to investigate natural circulation phenomena in VVER-440 reactors. It was scaled to a VVER-440 PWR, at a 1:2333 ratio based on power to coolant volume. Elevations are 1:1 except for the reactor upper head. The model represents the vessel and one loop, including loop seals and a horizontal steam generator. Tube bundle scaling was 1:2333 based on heat transfer surface area. The main design principle was the accurate simulation of rod bundle geometry and the primary system elevations.

The test series was a matrix of 22 experiments, which were designed to characterize natural circulation phenomena under a variety of conditions and to generate a data base for computer codes used to predict full-scale plant behavior. Parameters varied were heater power, primary water inventory, and noncondensible gas content. Primary pressure was not regulated during the experiments. Instead, it was allowed to achieve equilibrium as a function of heater power and water inventory. The secondary side of the steam generator simulator was operated at saturated conditions and atmospheric pressure. The pressurizer was not used.

Two experiments were used to assess RELAP5/MOD2 cycle 36.04: a single phase, 20 kW (3% decay heat), and a two-phase, 80% water inventory, 30 kW. The main results reported were the stationary final states into which the natural circulation process converges. The assessment objective was to reproduce these final states as closely as possible; less attention was paid to transition behavior.

RELAP5/MOD2 successfully predicted the main parameters of the experiments. Differences between calculated and measured values were within the error band of the measurement. The single-phase experiment showed relatively large mass flow oscillations that could not be reproduced. The information provided was not sufficient to allow identification of a modeling problem or a code deficiency. In the two-phase case, the calculation successfully predicted the oscillations, but they were damped more rapidly than in the experiment. Removing ambient losses from the hot leg resulted in calculated oscillations of higher amplitude and longer duration. The oscillations were of low frequency and not of numerical origin.

Solution scheme instabilities were noted in the steam generator secondary volumes for any requested time step value greater than one-fourth to one-third the Courant limit. The instability was attributed to the low L/D ratio of the secondary volumes and to the stability of wall-to-fluid heat transfer.

In the two-phase case, the code predicted significant voiding in the hot leg. Condensation to the piping walls was calculated to occur, and only single-phase liquid flow was present above the loop seal region. The fact that steam bubbles do not penetrate the hot leg loop seal was not well understood; the authors recommended additional analyses with various filling ratios and modified nodalizations to further examine the phenomenon.

Pressure losses in various parts of the facility must be accurately known for accurate prediction of natural circulation conditions, and the calculation is sensitive to environmental heat losses.

Problems were noted with property table failures in volumes nearly completely air-filled, with minute amounts of steam and liquid. Also, the code predicted steady state conditions when parameter values were still changing slowly. An alternate the steady state algorithm was suggested in which terms containing time derivatives are set to zero.

4.0 DISCUSSION: IMPACT OF IDENTIFIED CODE DEFICIENCIES

As summarized in Sections 3.1 and 3.2, a number of code deficiencies were identified. The following discussion is focused on how each deficiency affects the code's analysis capability.

Code deficiencies are not always apparent when performing an analysis. A code deficiency is most apparent if it is part of the models used to calculate the dominant phenomena during a transient. However if the deficiency affects models that calculate phenomena of lesser importance, the presence of the deficiency may or may not be obvious. It is conceivable that some code deficiencies would never be noticed by a particular group of analysts if their analyses only focused on a limited class of transients.

The deficiencies are listed in Table 4.1 and categorized according to whether their effect is most important during the steady-state analysis (SS), large break LOCAs (LBLOCA), small break LOCAs (SBLOCA), or operational transients (OT). If a deficiency is important in all classes of transients it is labeled as such. Thus the categories used in Table 4.1 are SS, L, S, O, and A for steady-state, LBLOCAs, SBLOCAs, operational transients, and all transient classes respectively.

The deficiencies are discussed from the perspective of the three major types of transients plus the steady-state calculations that are generally analyzed using the RELAP5 code. Sections 4.1, 4.2, 4.3, and 4.4 describe the affect of the deficiencies on steady-state calculations plus LBLOCA, SBLOCA, and operational transient analysis respectively.

4.1 EFFECT OF DEFICIENCIES ON STEADY-STATE CALCULATIONS

The first stage in performing any transient calculation is to match the facility pretest steady-state conditions. To do so the experimental system state conditions in both the primary and the secondary systems are matched as closely as possible. However several code deficiencies have been identified that prevent the analyst from matching the system conditions exactly and in fact may cause difficulties during the transient portion of the analysis as well.

Three code deficiencies affecting steady-state have been noted. The first two deficiencies are centered on obtaining the correct primary to secondary energy transfer and the correct secondary inventory distribution respectively. The first deficiency is caused by the code's incorrect steady-state heat transfer calculation and the second deficiency is caused by the interfacial friction model. The third deficiency is a numerical algorithm shortcoming that causes the code to assume steady-state has been reached when it has not.

4.1.1 Incorrect Steady-State Energy Transfer

This error is encountered when an analyst attempts to model a balanced integral system at a defined steady-state power condition. For example, if a full-scale plant transient is going to be studied with an initial condition

Table 4.1 - Deficiencies Identified by ICAP RELAP5/MOD2 Code Assessments

Code Deficiency	Source(s)[a]	Applicable Transient[b]
Accumulator model	3.2.15	L
Steady-state algorithm	3.2.38	SS
Interfacial friction	3.1.5, 3.1.7, 3.1.8, 3.2.1, 3.2.2, 3.2.3, 3.2.9, 3.2.12, 3.2.17, 3.2.29, 3.2.33, 3.2.34, 3.2.35, 3.2.37	A,SS
Vapor pull-through/liquid entrainment	3.1.2, 3.2.20, 3.2.26	S
Critical heat flux	3.1.6, 3.2.10, 3.2.25	L
Critical flow modeling	3.1.3, 3.1.4, 3.2.9, 3.2.19, 3.2.20, 3.2.36	A
Return to nucleate boiling	3.2.10, 3.2.11, 3.2.12	L
Onset of horizontal stratification	3.2.20	S
Vertical stratification model	3.1.7, 3.2.19	S
Incorrect steady-state heat transfer	3.2.4	SS
Subcooled boiling model	3.1.9	O
Condensation	3.2.1, 3.2.7, 3.2.11	A
Radiation heat transfer model	3.2.27	L

[a] The identified deficiencies are described in the assessments summarized in the following sections.
[b] A = all, L = LBLOCA, O = operational transient, S = SBLOCA, SS = steady-state calculation.

equal to 100% rated power, the analyst will not be able to calculate the exact primary system steady-state conditions measured in the plant without adjusting the secondary system. The deficiency is noted specifically in Gerth, 1992. This deficiency has been present in the code for many years and is described in the *User's Guidelines*, pages 5-7 and 5-8 (Fletcher and Schultz, 1992) as follows:

> It often is difficult to obtain a satisfactory agreement with steam generator full-power conditions. The difficulty arises because the heat transfer coefficient calculated on the outside surface of the steam generator tubes is based on general vertical-pipe correlations rather than correlations that account for the swirling flows present within the tube bundle region. The swirling flow pattern results because horizontal baffles in the boiler direct the flow back and forth across the tube bundle instead of allowing the flow to proceed axially (vertically upward) through the boiler. The effect of this discrepancy is that tube heat transfer is understated by the code, resulting in excessively high calculated primary coolant temperatures (the temperatures increase until the core heat is driven across the tubes). Since the source of the calculated error is understood (i.e., a general heat transfer correlation is not appropriate for this application), it is recommended that the modeler "adjust" the heat transfer on the outside of the tubes to remedy the discrepancy.
>
> The recommended adjustment is to reduce the input heated equivalent diameter on the heat structure cards for the outer tube surface....

The best discussion of the effect of the various parameters on the primary to secondary energy transfer under steady-state conditions is given in Putney and Preece, 1991.

4.1.2 Interfacial Friction

Interfacial friction modelling capabilities are important when an analyst is performing the steady-state calculations for a pressurized water reactor because the secondary inventory level and mass are affected. If the analyst matches the calculated secondary inventory level and the steady-state recirculation ratio with the measured parameters, then the total secondary inventory will be undercalculated. The degree of undercalculation is a function of the power level.

Usually the exact quantities of secondary inventory present in the SGs are not known since only the liquid levels in the downcomer are measured. Since the secondary void fraction profile and fluid density variations are generally unknown there can be a large uncertainty on the quantity of secondary inventory present in integral effects experiments and especially in full-scale plants.

Putney and Preece, 1991 conducted an exhaustive study to determine the status of the code concerning initial secondary inventory. Their findings, based on the analyses for the Wolf Creek and the Sizewell 'B' plants, show that for Sizewell 'B' the secondary inventory is underpredicted by 10% when the recirculation ratio is correct, and that the secondary inventory is calculated correctly if the recirculation ratio is overcalculated for the Wolf Creek plant. No similar analyses are included in the assessments summarized herein.

4.1.3 Steady-State Algorithm

Hyvarinen, 1992 noted that the code has a tendency to terminate calculations performed in the "STEADY-ST" mode before actually reaching a steady-state condition. For this reason, many analysts simply perform their steady-state calculations in the transient mode. When the transient mode is used, the length of the steady-state calculation is determined by the user rather than by the code's internal algorithm. (Unfortunately, when the transient mode is used to perform steady-state calculations hundreds of calculation seconds are required for the thermal gradients to stabilize.)

4.2 EFFECT OF DEFICIENCIES ON LBLOCA TRANSIENT ANALYSIS

The normal progression of a LBLOCA, initiated when the reactor system is at full power, begins with the event causing a pipe break and moves through loss of the vessel inventory to core uncovery and core dryout. During the course of these events the various emergency core cooling systems are initiated. As a result the reactor vessel lower plenum first refills and ultimately the core refloods. Once the core is cooled and the primary system has reached a steady-state condition, the portion of the transient with the greatest potential for causing an extended core heatup is complete.

A LBLOCA transient analysis for an integral system consists of three phases: blowdown, refill, and reflood. In the following subsections the discussion is grouped according to the code deficiency that has been identified. When necessary the LBLOCA transient phase is described.

Once the steady-state portion of the analysis is complete and a model that properly represents the experimental or plant steady-state condition has been produced, the transient portion of the analysis begins. When the break is opened and blowdown is initiated, important RELAP5 models that have been identified as deficient include: the critical flow model, the interfacial friction model, the critical heat flux model and the model describing return to nucleate boiling (both included in the RELAP5 heat transfer package).

The critical flow model and the interfacial friction model are the most frequently cited problem areas in the code. Also, the code's capability to calculate critical heat flux (CHF) and return to nucleate boiling, phenomena calculated by the RELAP5 heat transfer package, have been cited as deficient a number of times (see Table 4.1). Each of the above deficiencies will be discussed in the following three subsections.

4.2.1 Critical Flow Model

The critical flow model, aside from being one of the most frequently cited deficient models, is also the most misunderstood model. Since critical flow is a function not only of the upstream thermodynamic state conditions but also the discharge opening geometry, there is quite often a great deal of uncertainty associated with critical flow modelling.

Only two of the seven assessments that cite the critical flow model as being deficient are directly applicable to LBLOCAs. The assessments performed using the Marviken and CUMULUS separate effects experimental data will be discussed in this section.

<u>Marviken Experiments</u>: The Marviken test facility was used to conduct both critical flow tests (CFT) and jet impingement tests (JIT). In both sets of tests the same vessel was used. The principal difference between the two experiments was that single-phase liquid or two-phase fluid exhausted through the discharge nozzle in the CFTs (Rosdahl and Caraher, 1986b) but only steam exhausted through the exhaust nozzle in the JITs since the vessel was fitted with a standpipe that rose from the inlet of the nozzle test section (located at the vessel bottom) to an elevation above the liquid free surface.

The simulations of the JIT11 test (saturated steam flow) overpredicted the experimental discharge flow rate by 20 to 25 percent when using only a junction to model the break. Explicitly representing the nozzle region by up to five computational cells had little effect on the computed results. It was thus concluded that when simulating saturated steam critical flow with RELAP5, a discharge coefficient of about 0.8 should be used. Furthermore, short lengths of pipe, i.e., L/D<4, should not be explicitly modeled.

The simulations of the subcooled portion of the CFT21 test overpredicted the experimental critical flow rates by 18 to 20 percent when the nozzle was not explicitly modeled. A discharge coefficient of approximately 0.85 was required to match the data using this technique. However, if the nozzle was explicitly modeled, the code would underpredict the measured flow results; applying a discharge coefficient greater than unity did little to improve the computed results, but would greatly increase the computational times. Consequently, it was concluded that short discharge regions, i.e., nozzles, should not be explicitly modeled.

For low quality two-phase flow, the calculation showed reasonable agreement with the gamma densitometer measurements, but overpredicted the critical flow (by up to 30 percent) when based on the vessel differential pressure measurements. Since the actual fluid state in the vessel is probably between the two measurements, it was concluded that a discharge coefficient between 0.80 and 0.95 should be used. (Note: However, when the discharge coefficient was reduced from 1.0 to 0.85 for the saturated flow condition, only an 8% flow reduction was realized. Consequently, upon further study, it was found that the code logic contains a feedback between the break sonic velocities and the local equilibrium quality.)

<u>CUMULUS Experiments</u>: Six critical flow tests were conducted on a SEBIM valve in the CUMULUS facility (Stubbe and Vanhoenacker, 1990). Two of the experiments used superheated steam at pressures of 17 MPa and temperatures 21 to 27 K above saturation. The other four experiments were conducted using subcooled water at pressures ranging from 7 MPa to 3 MPa with 43 to 84 K subcooling. The data from two experiments yielded a calibration showing a valve discharge coefficient of 0.794 for the superheated steam flow and 0.778 for subcooled water discharge.

A simple model, consisting of 6 BRANCH components, one VALVE component, and two time dependent volumes was constructed to represent the CUMULUS facility with the SEBIM valve mounted. The above discharge coefficients were input to the model. The resulting calculations showed excellent agreement with the data; the calculation deviated from the data by a difference that is less than the known data uncertainty.

During the course of performing the above calculations, three code deficiencies were noted: (i) A mismatch exists between imposed initial temperature and the initial input processing for time dependent volumes. The mismatch is as large as 1 K for superheated steam conditions. (ii) Flow rate oscillations for superheated steam conditions. (iii) The critical mass flow rate is more sensitive to the vapor superheat than indicated by the perfect gas law.

4.2.2 Interfacial Friction

Fourteen assessment studies cited interfacial drag as a code deficiency for RELAP5/MOD2. Of the fourteen, five are specific to LBLOCAs (Bang, et al., 1990; Croxford and Hall, 1989; Kao, et al., 1992; Liang et al., 1992; Richner, et al.). Bang et al. and Kao, et al. are calculations based on the integral system LOFT L2-5 experiment. Croxford and Hall describe an assessment study based on the THETIS data. Richner et al. describes an assessment study based on NEPTUN data. Finally, Liang, et al., summarizes an assessment based on the Semiscale S-06-3 LBLOCA experiment.

<u>LOFT LOCE L2-5</u>: The LOFT loss-of-coolant experiment (LOCE) L2-5 was conducted to investigate the LBLOCA-induced core thermal-hydraulic behavior when the primary coolant pumps do not have a normal coastdown but instead were tripped off within 1 s after initiation of the transient. Since the pumps were not connected to their flywheels, the pump coastdowns were uncharacteristically short (Nalezny, 1983).

Both Bang et al. and Kao et al. showed calculated results that displayed oscillations (the data were smooth) in the intact hot leg mass flow during reflood (after 40 s). Bang et al. attributed the oscillations to a deficient interfacial friction model in the code and showed that with code updates produced by Aksan at the Paul Scherrer Institute (Switzerland), designed to better model interfacial friction in the core bundles, the oscillatory behavior was greatly reduced. Although Bang et al. have stated that the improvement obtained using the PSI updates indicate the interfacial friction model is deficient, the evidence is not conclusive.

<u>THETIS</u>: The assessment input model consisted of the test section with time dependent volumes attached to the test section ends. The time dependent volumes provided the coolant boundary conditions. Two models were used: one with a 24 fluid cell PIPE component to model the test section and one with a 6 fluid cell PIPE component. The 6 fluid cell PIPE component was used to provide a point of comparison with the normal nodalization approach used by analysts when modeling a typical power plant. (Most integral system calculations are done using approximately 6 cell cores.)The code showed good agreement, using the finely nodalized model, with the mixture level boildown

rate data at 4.0 MPa (see Figure 4.2.2.1) and at 2.0 MPa. However, at pressures below 1.0 MPa the boildown rate was considerably overpredicted (see Figure 4.2.2.2). The code was found to have a tendency to over-predict the void fraction below the two-phase mixture level, with errors increasing with decreasing pressure. At 4.0 MPa the maximum error in calculated two-phase mixture density is -33%; the maximum error at 1.0 MPa is -46% at experimental void fractions of 0.4.

Comparison between the above errors and the work of Ardron and Clare showed qualitative and quantitative similarities. Ardron and Clare's conclusions,based on their studies of the interphase drag model, were that for conditions pertinent to boil-off the models would tend to over predict void fraction, with the error increasing as the pressure decreased. Their work showed similar percentage errors for boiloff rates at pressures of 4.0 MPa and 1.0 MPa as those found by Croxford and Hall. Thus, based on Ardron and Clare's rigorous work, Croxford and Hall attributed the code's inability to match the THETIS data to the interphase drag model deficiencies.

The assessor's nodalization studies, based on the 6 volume core, revealed:

- o Good agreement between the calculations and data at pressures above 2.0 MPa.
- o Calculated oscillations (see Figure 4.2.2.3) predicted during the steady-state conditions simulated prior to boildown.

Investigations of the steady-state oscillations led to the discovery that the vertical stratification model was periodically triggered. However, oscillations were damped once the boiloff calculation began.

<u>Marviken JIT 11 Level Swell Experiment</u>: The vessel was filled to the 10.2 m elevation with nearly saturated liquid; the remaining part of the vessel and the standpipe were filled with saturated steam. The initial pressure in the vessel was 5.0 MPa.

The test was initiated by breaking the rupture disks. Because of the standpipe configuration, only steam flowed from the vessel. Differential pressures were recorded at various elevations in the vessel, thus allowing a measurement of fluid density profiles. Discharge mass flow rate was also measured. The experiment was terminated when the pressure in the vessel reached 1.9 MPa.

After the rupture disks broke, bulk flashing occurred in the liquid. The level of the resulting two-phase mixture rose rapidly and reached a maximum height of about 18 m - the top of the standpipe - within 15 s. The mixture level declined slowly thereafter, receding to near the 14 m elevation by the time the test ended (80 s). For elevations below the 13 m height the differential pressure measurements remained fairly constant over the 15 to 80 s time period; indicating a fairly-steady void fraction.

Code simulations were made using 20, 40, and 100 nodes to model the annular region in the vessel below the top of the standpipe. The experimental mass

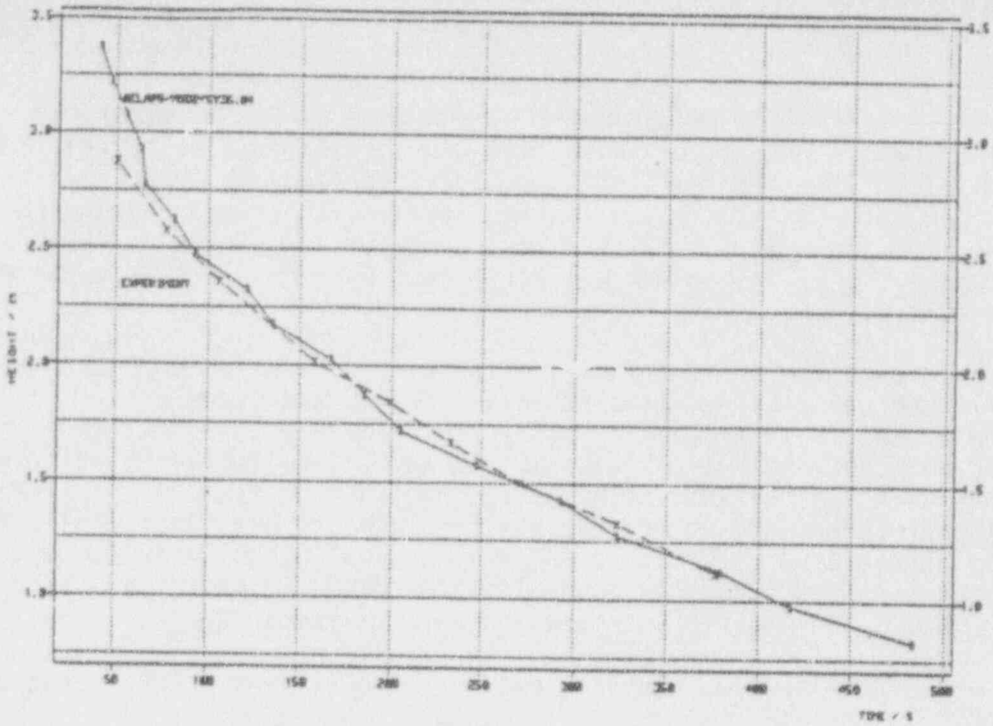


Figure 4.2.2.1 THETIS Boildown Facility: Calculated and Measured Dryout Histories at Pressure = 4.0 MPa.

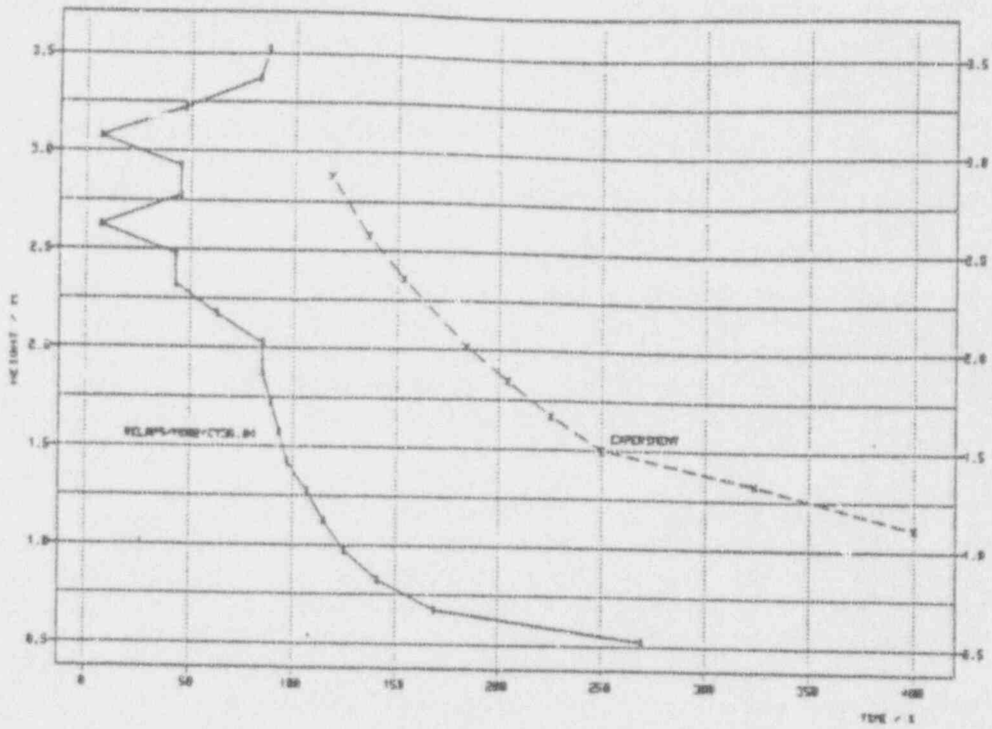


Figure 4.2.2.2 THETIS Boildown Facility: Calculated and Measured Dryout Histories at Pressure = 0.2 MPa.

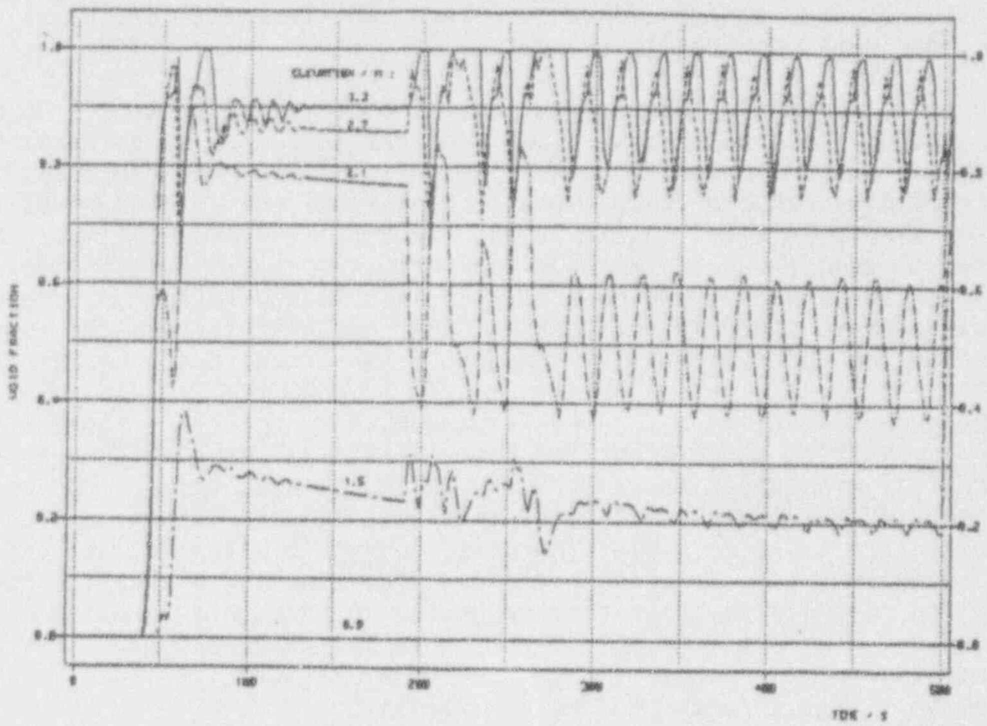


Figure 4.2.2.3 THETIS Boildown Facility: Calculated and Measured Test Section Void Fraction Histories at Pressure = 0.5 MPa.

flow rate was used as a boundary condition. Differential pressures calculated by the code were compared to the data.

The 20 node and the 40 node simulations showed similar results. Both calculations indicated that the code underpredicted the void fraction of the swollen two-phase mixture for elevations below 9.28 m and overpredicted the void fraction for higher elevations. Examples of the 20 node results are shown in Figures 4.2.2.4, 4.2.2.5, and 4.2.2.6. The results imply that the interfacial drag force in the code fell off too rapidly with increasing void fraction. Consequently the code carried less liquid to the upper elevations than was carried there in the test and the code also allowed the liquid to drain from the upper elevations somewhat faster than it did in the test.

The 100 node simulation was characterized by very erratic differential pressure histories which were, for some elevations, much different from the differential pressure histories of the 20 and 40 node cases. Moreover, the 100 node simulation was found to be sensitive to time step size - changing the step size form 0.1 s to 0.05 s (material Courant limit = 0.12 s) produced large changes in void fraction profiles. The behavior of the 100 node simulation is believed to be related to the interphase drag model in the code - its strong dependence on void fraction in the bubble-to-slug transition region; its explicit connection to the numerical solution; and its algorithm for damping large changes in computed values.

Time step studies, using the 20 node model revealed that the code calculation was sensitive to time step size during the time period (0 to 30 s) when the level swelled to its maximum height.

The 20 and 40 node simulations have demonstrated that the code can give a fairly accurate simulation of the void fraction profile associated with level swell in large vessels. Improvements in the interphase drag model could lead to better simulations but would probably sacrifice the generality of the present model.

The 100 node simulation has demonstrated that the code solution does not converge with increasing nodalization - at least not for time step sizes allowed by the automatic time step control algorithm. An example of the nonconvergent behavior is shown in Figure 4.2.2.7. The results suggest that at some level of nodalization the numerical models for smoothing locally computed variables begin to overshadow the physics.

<u>NEPTUN</u>: Of the seven NEPTUN experiments simulated using the code, the measured data recorded during the high and medium reflooding rate experiments were reasonably matched by the code calculations but the measured data from the low reflooding rate experiments were not.

Richner et al. noted the code's interphase friction correlations were obtained from tube experiments. But since the fluid interphase friction in tubes is greater than the interphase friction for flow through rod bundles (Bestion, 1985), the implementation of a bubbly/slug flow interphase friction correlation for rod bundles taken from the CATHARE code greatly improved the

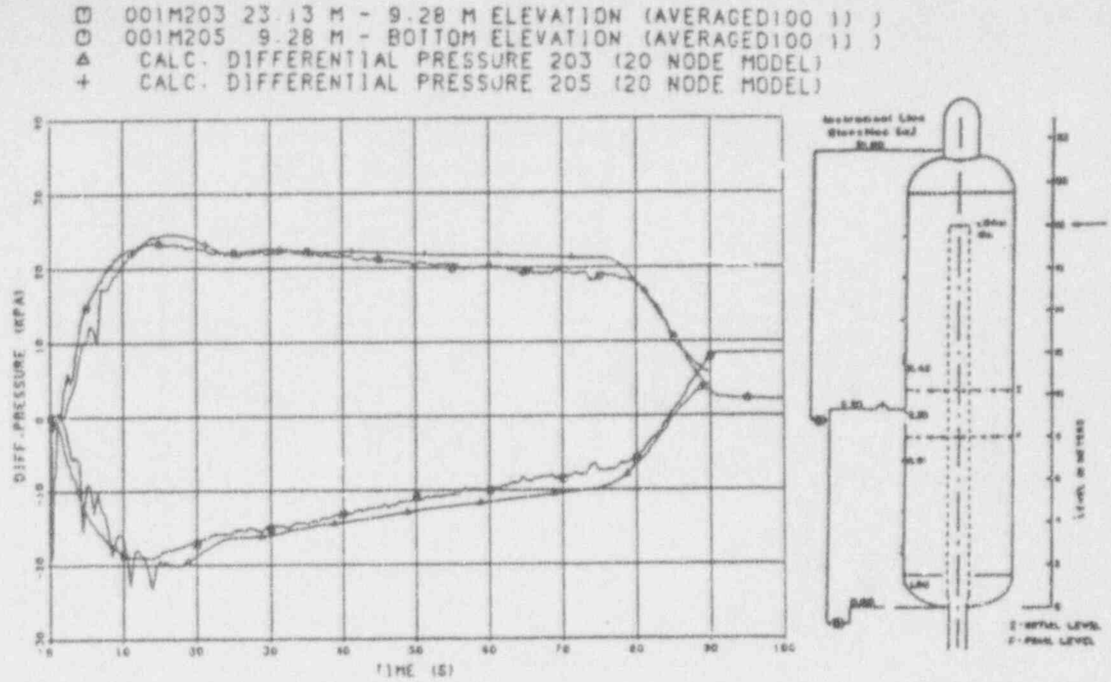


Figure 4.2.2.4 Marviken JIT-11 Experiment: Calculated and Measured Differential Pressures at Top and Bottom of Vessel.

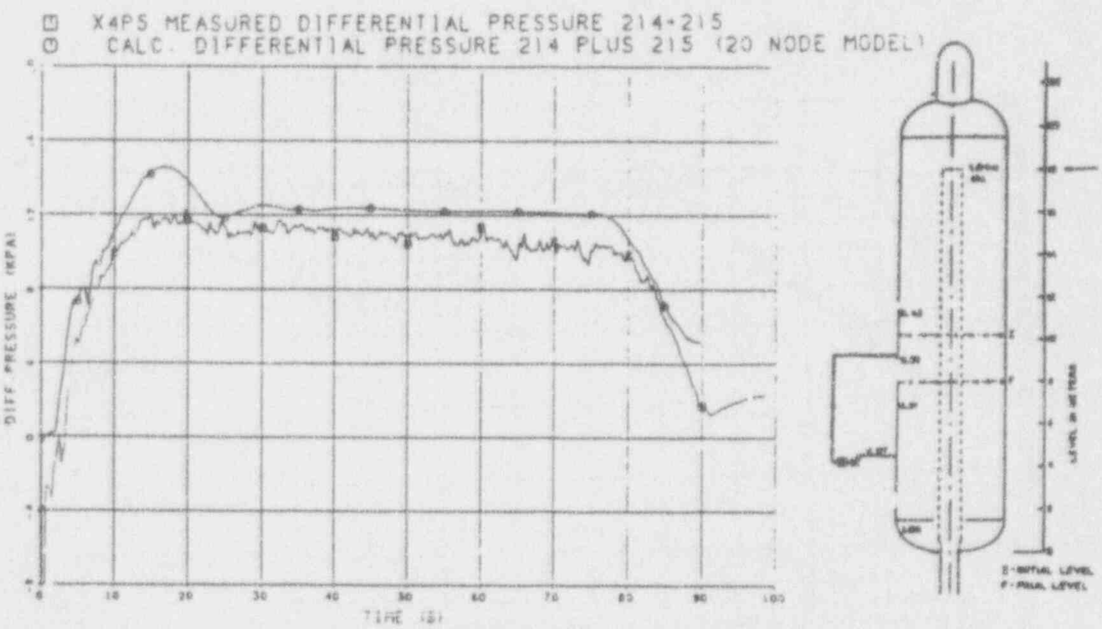


Figure 4.2.2.5 Marviken JIT-11 Experiment: Calculated and Measured Differential Pressure Over 4.97 m to 9.73 m Span.

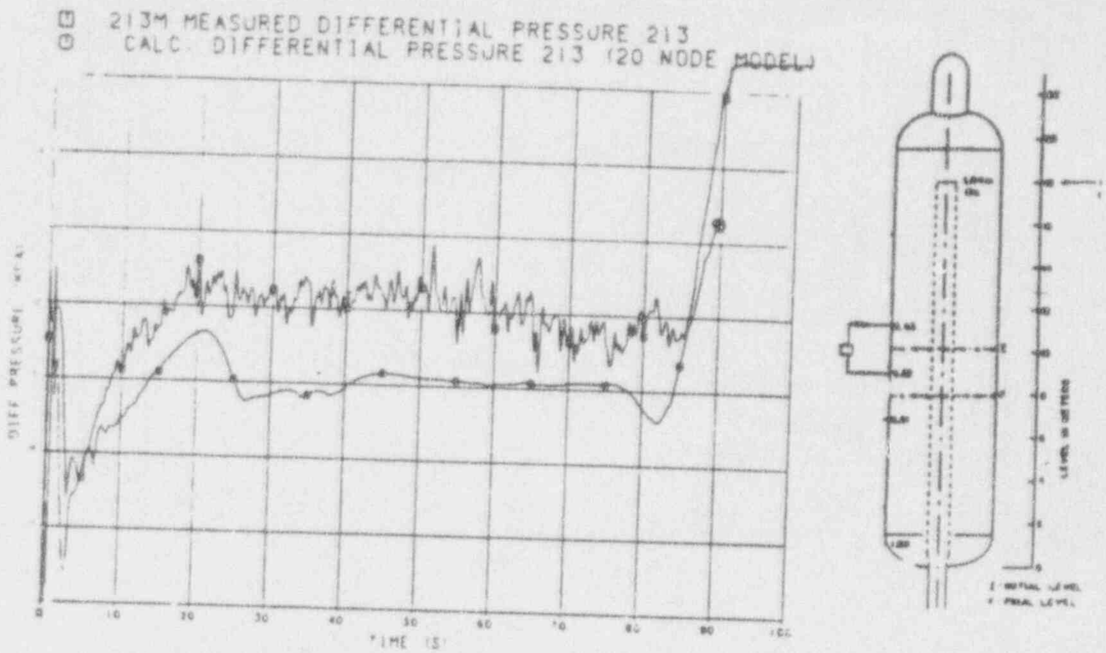


Figure 4.2.2.6 Marviken JIT-11 Experiment: Calculated and Measured Differential Pressure Over 9.23 m to 11.43 m Span.

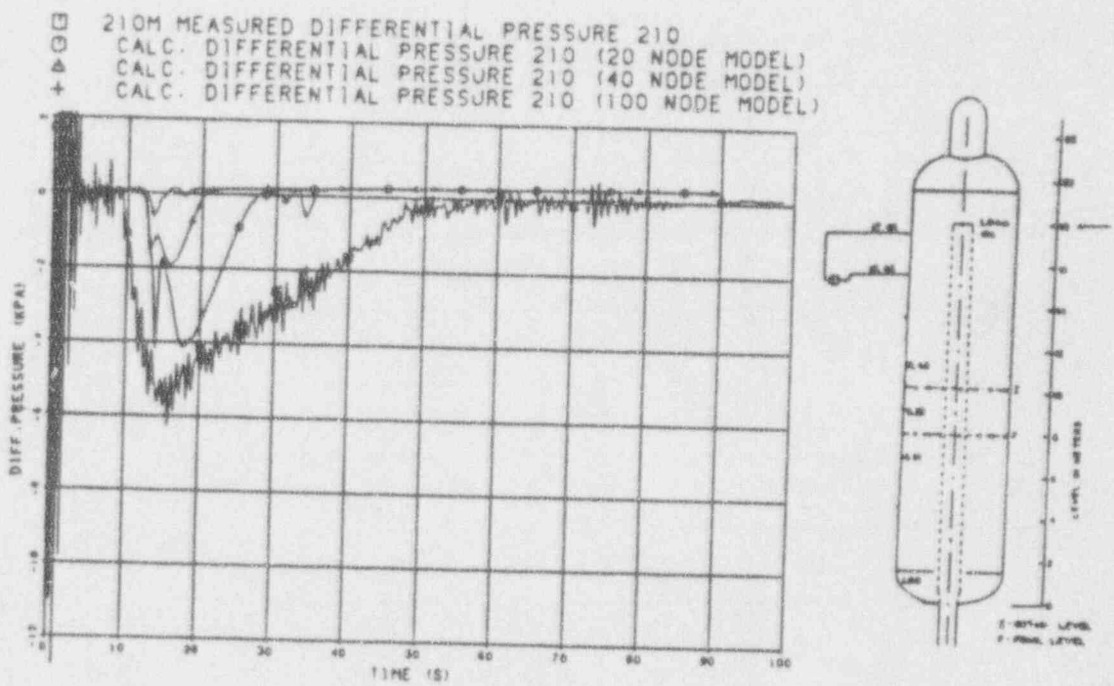


Figure 4.2.2.7 Marviken JIT-11 Experiment: Calculated and Measured Differential Pressure Over 15.45 m to 17.45 m Span.

calculation of the quantity of water in the test section during reflood at low flooding rates.

Semiscale S-06-3 Experiment: The S-06-3 experiment was conducted as part of a pretest evaluation prior to conducting the first LOFT experiment. For the most part the code did a reasonable job of matching the data. However, the code was unable to calculate the emergency core cooling inventory downcomer bypass observed during the experiment (Liang et al.). This result was expected since the code had previously been advertised as a SBLOCA and operational transient evaluation tool.

4.2.3 Heat Transfer Package

The ability to calculate the thermal behavior of facility fuel rods or fuel rod simulators during LBLOCAs is one of the most important functions of the analysis code. However, it is important to remember that the heat transfer package does not function independently of the other code models such as the interfacial friction and the field equations. Consequently, it is often quite difficult to separate deficiencies into their component parts. For example, if core dryout and heatup are experienced during a particular transient and the code calculation does not show the same behavior, the analyst must identify whether the fluid dynamics were calculated correctly while the heat transfer was not or whether all the models involved are incorrect to one degree or another. The assessment process is complicated because the system fluid behavior and heat transfer form a feedback system that is often quite complex and furthermore changes as a function of the phenomena being evaluated.

Five assessments identified various portions of the heat transfer package as deficient. Two of the assessments (Sjoberg and Caraher; Bang et al., 1990) identified deficiencies in the critical heat flux (CHF) modelling applicable to LBLOCA analysis and three of the assessments (Bang et al., 1990; Bang, et al., 1992; Kao, et al.) identified deficiencies related to core heatup following dryout and also the ability of the code to calculate return to nucleate boiling.

Royal Institute of Technology: Experiments were conducted to study the critical heat flux and dryout heatup characteristics of an electrically-heated tube with a round cross-section. The resulting assessments were reported by Sjoberg and Caraher.

Post-dryout heat transfer experiments were conducted at the Royal Institute of Technology in Stockholm, Sweden. The experimental test section was a 7 m long, 1.5 cm diameter heated tube. Experimental pressures ranged from 3 to 20 MPa; mass fluxes ranged from 500 to 2000 kg/m^2-s; heat fluxes ranged from 10 to 125 W/cm^2; and inlet subcooling ranged from 7 to 13 K. Five hundred ten experiments were conducted, for the above variable range, with three different test sections. For each test, a constant, axially uniform heat flux was imposed on the tube. For nearly all the experiments, nucleate boiling was sustained over the lower 3 m of the test tube.

A code model was constructed to represent the above facility using 47 vertical fluid volumes that were simulated using a PIPE component. The boundary conditions were simulated using time dependent volumes at either end and a time dependent junction at the PIPE component inlet. The PIPE component volumes were constructed using coarse nodalization for the lower 3 m while cell lengths of 10 cm were used above the 3 m elevation. A constant, axially uniform heat flux was imposed on the tube model that matched the test heat flux. An insulated boundary condition was imposed upon the outside edge of the tube wall while the inner edge received its boundary condition - a heat transfer coefficient and temperature sink - from the code's heat transfer package. Once the code simulation reached a steady-state, the calculated axial temperature distribution along the tube was compared to data.

The first series of code simulations showed minimal agreement with the data since the code predicted CHF to occur much farther downstream than the data indicated. A subsequent comparison of the Biasi CHF correlation (used in RELAP5/MOD2) to 177 of the post-dryout experiments showed that the mean difference between the measured and Biasi CHF was -60.8%, the negative sign indicating that the Biasi heat flux was greater than the actual heat flux. As a result, the next series of calculations was done with an updated version of the code for which CHF occurred at the measured location. Thus, the assessment objective, i.e., studying post-CHF heat transfer, was achieved.

Using the updated code, the temperature distributions more than 30 cm downstream of the CHF point were reasonably calculated. In all but one calculation the differences between the measured and calculated temperatures were less than 10%. A typical example, of the 21 figures shown in Sjoberg and Caraher, is shown in Figure 4.2.3.1. The calculated and measured temperatures as a function of axial position for a mass flux equal 2000 kg/m^2-s and a pressure equal to 7 MPa for Run 261 are shown in Figure 4.2.3.1. In general, the code underpredicted the temperature for the higher pressure experiments, i.e., greater than 10 MPa, and overpredicted the temperatures for the lower pressure experiments.

In the regions less than 30 cm downstream of the CHF point, large differences between measured and calculated temperatures were evident. In this region, the axial temperature gradient is very large, e.g., 200 to 300 K over a 10 to 20 cm length. The differences between calculated and measured temperatures were, in some cases, attributed to the discreetness of both the code model and the measurements. In other cases, the differences were due to transition boiling occurring in the experiment, but not in the calculation.

Nodalization studies showed that if CHF were calculated to occur at the proper location, then a reasonable comparison between the calculation and data could be achieved with only a 14 equally-sized node input model. Such a model is the typical nodalization for an integral experiment.

<u>LOFT L2-3 and L2-5 Experiments</u>: The L2-3 and L2-5 experiments were conducted to study the LOFT system thermal-hydraulic behavior with the reactor coolant pumps (RCPs) configured to simulate the expected plant equipment behavior (L2-3) and to simulate an unrealistically fast coastdown (L2-5) of the RCPs. As such, during the L2-3 experiment the RCPs tripped early in the transient and

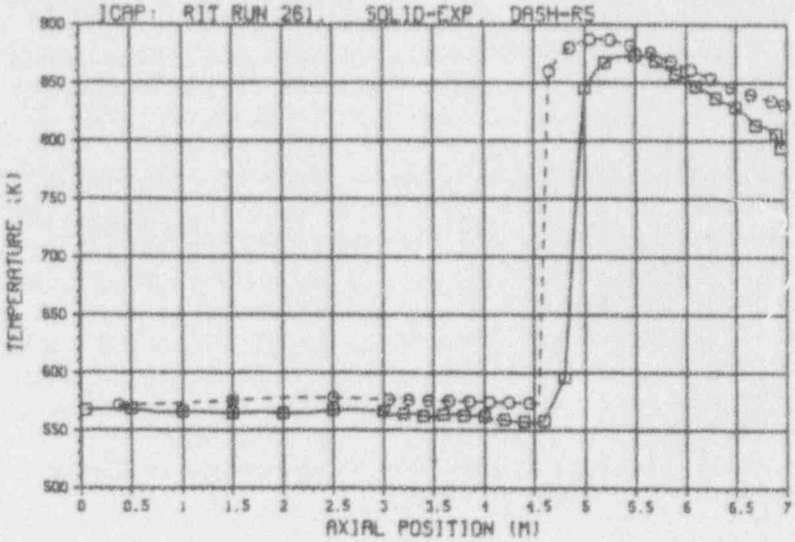


Figure 4.2.3.1 Royal Institute of Technology Dryout Experiment 261: Calculated and Measured Heater Rod Temperature and Axial Position.

then decreased their rotational speeds as a function of their inertia (including the pump and flywheel) and the fluid behavior adjacent to the impeller. On the other hand, during the L2-5 experiment the RCPs tripped early in the transient and decreased to nearly zero rotational speed very quickly. The resulting assessment calculations were reported by Bang et al, 1992 for LOFT L2-3, Bang et al., 1990 for LOFT L2-5, and Kao et al for LOFT L2-5.

CHF was observed in the first few seconds after the beginning of both transients. However, during the L2-3 experiment the core quickly rewet and resumed nucleate boiling due to the presence of core mass flow induced by the RCP coastdown (a process identified as "blowdown quench" by some analysts).The LOFT L2-3 experiment assessment analysis, performed by Bang et al., 1992 demonstrated that the code heat transfer package could not calculate the early core rewet observed in the experiment (see Figure 4.2.3.2) with any reliability. The few locations at which early core rewet was calculated, the time of subsequent dryout was calculated too late, the magnitude of core heatup was undercalculated, and the calculated core rewet was too early compared to the data (see Figure 4.2.3.3).

4.3 EFFECT OF DEFICIENCIES ON SBLOCA TRANSIENT ANALYSIS

Whereas LBLOCA-analysis is generally focused on the 200% double-ended break located in the cold leg of a PWR operating at full-power and usually with a minimum of ECCS equipment available, SBLOCAs under investigation encompass a huge category of scenarios that vary as a function of break size, break location, break orientation, system state, and ECCS equipment availability.

The ten SBLOCA-related assessments included herein were performed using data from LOFT, the LSTF, LOBI, and Semiscale. Eight focused on the system behavior following breaks in the cold leg while the remaining two were performed for SBLOCAs in the hot leg. Specifically, SBLOCA assessments were performed for 5% cold leg breaks in the LSTF and Semiscale, a 3% cold leg break in the LOBI facility, 2.5% cold leg breaks in the LOFT facility with pumps on and pumps off, 1% hot leg breaks with pumps on and off in the LOFT facility, a 0.5% cold leg break in Semiscale, and 0.4% and 0.1% cold leg breaks in the LOFT facility.

Following completion of the steady-state analysis phase of each assessment, the SBLOCA studies were initiated by simulating the presence of the break. The system in each of the break sizes depressurized relatively quickly to a primary pressure slightly greater than the opening setpoint of the steam generator safety relief valves. The primary pressure remained at this level until finally the break flow became two-phase and the depressurization continued to levels below that of the low pressure injection system. Just prior to the renewed depressurization achieved when two-phase discharge began, the primary system underwent loop seal clearing with concurrent core liquid level depression.

Important phenomena inherent to SBLOCAs that highlighted RELAP5/MOD2 code deficiencies were: interfacial drag (including counter current flow limiting), critical flow (including vapor pull-through and liquid entrainment), the onset

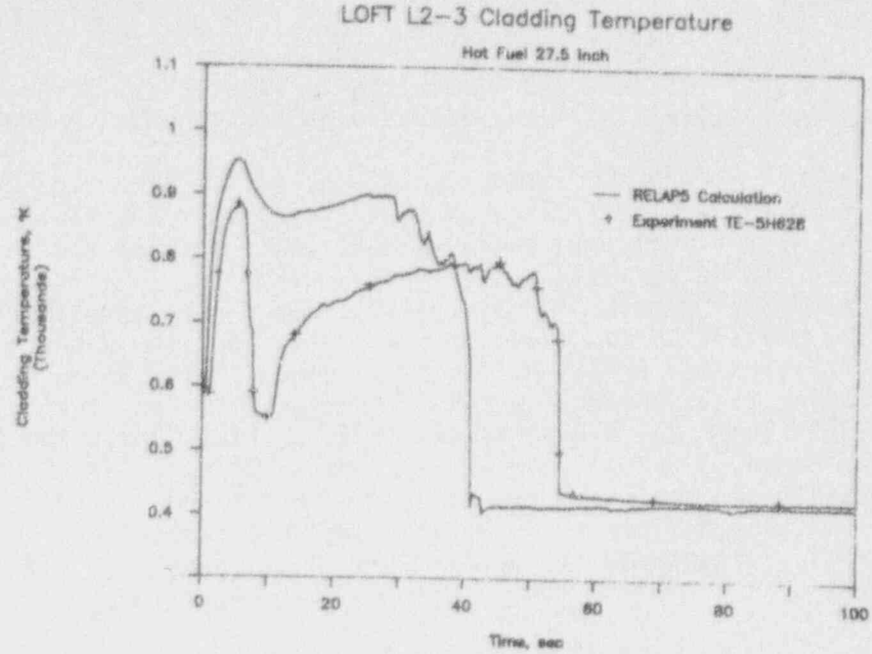


Figure 4.2.3.2 Comparison of cladding temperature at 27.5 inches of hot fuel between the base case calculation and the experiment.

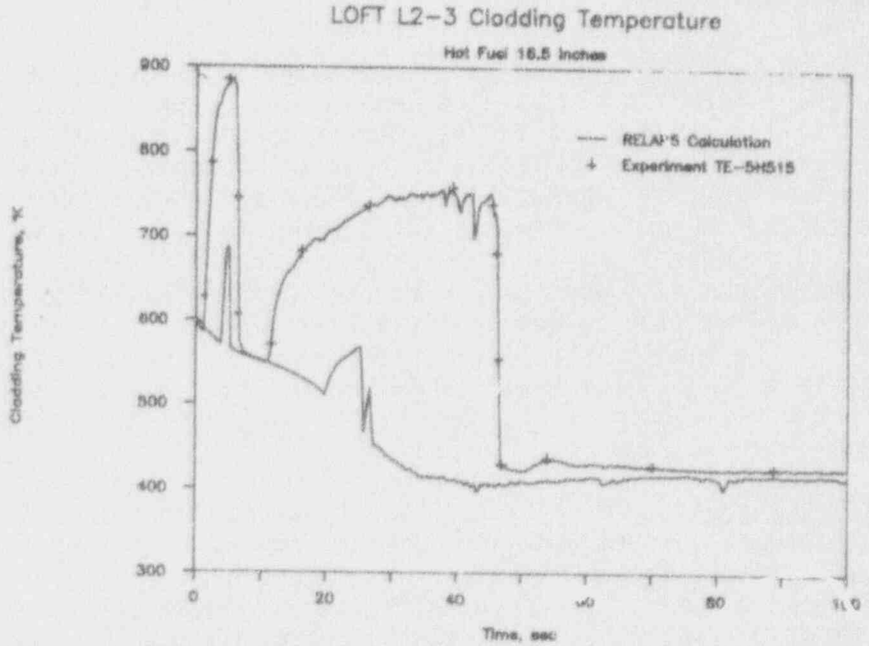


Figure 4.2.3.3 Comparison of cladding temperature at 16.5 inches of hot fuel between the base case calculation and the experiment.

of horizontal stratification, and simulation of vertical stratification. The above four deficient areas will be discussed in Sections 4.3.1 through 4.3.4.

Of the forty-eight RELAP5/MOD2 assessments, twelve isolated code deficiencies specific to SBLOCAs were identified in five areas. Six of the assessments identified the interfacial drag model as deficient (Rosdahl and Caraher, 1986a; Croxford and Hall, 1987; S. Lee, Chung, and Kim, 1991; Scriven, 1992b; Hall and Brown, 1991; Brodie and Hall, 1992), two of the assessments identified the critical break flow vapor pull-through/liquid entrainment model as deficient (Ardron and Bryce; Hall, 1990), seven of the assessments identified the critical break flow model as being deficient (Rosdahl and Caraher, 1986b; Stubbe and Vanhoenacker, 1990; S. Lee, Chung, and Kim, 1991;Hall and Brown, 1990; Hall, 1990; Scriven, 1992a; E. Lee, Chung, and Kim), one assessment identified the model indicating the onset of horizontal stratification to be deficient (Hall,1990), and two assessments indicated the vertical stratification model to be deficient (Hall and Brown, 1990; Croxford and Hall, 1989).

4.3.1 Interfacial Friction

Six of the assessments identified the interfacial drag model as deficient (Rosdahl and Caraher, 1986a; Croxford and Hall, 1987; S. Lee, Chung, and Kim, 1991; Scriven, 1992b; Hall and Brown, 1991; Brodie and Hall, 1992). Two of these assessments (Rosdahl and Caraher, 1986a; Croxford and Hall, 1987) were discussed in some detail in the previous chapter (Section 4.2) and consequently will not be discussed further here. The remaining four assessments were based on integral effects experimental data recorded in the LSTF (S. Lee, Chung, and Kim, 1991), LOFT (Scriven, 1988), and Semiscale (Hall and Brown, 1991; Brodie and Hall, 1992).

LSTF 5% Cold Leg SBLOCA: During the loop seal clearing phase of the transient (for SB-CL-18, the International Standard Problem 26), when the core liquid level is pushed to its minimum (at 140 s), the primary inventory liquid holdup in the steam generator U-tubes and plena is a direct function of the countercurrent flow limiting. Because RELAP5/MOD 2 has no specific CCFL modelling capability, the code relied on its interfacial friction model alone.

Although the code showed good agreement with the SG differential pressure across the U-tube upflow side for much of the transient prior to loop seal clearing at 140 s, the code's interfacial friction model overcalculated the quantity of liquid heldup in the steam generator and consequently calculated a minimum core liquid level below that shown by the data when the minimum core liquid level was observed (see Fig. 4.3.1.1).

LOFT 2.5% Cold Leg SBLOCAs: The LOFT L3-5 and L3-6 experiments were conducted to evaluate the integral system behavior considering the affect of the reactor coolant pumps off versus on respectively. The interfacial drag model was judged to be deficient because the code calculated a frothy-liquid mixture in the steam generator U-tubes in contrast to the separated primary inventory distribution deduced from the data.

Semiscale 5% Cold Leg SBLOCAs: The Semiscale S-LH-1 and S-LH-2 experiments,

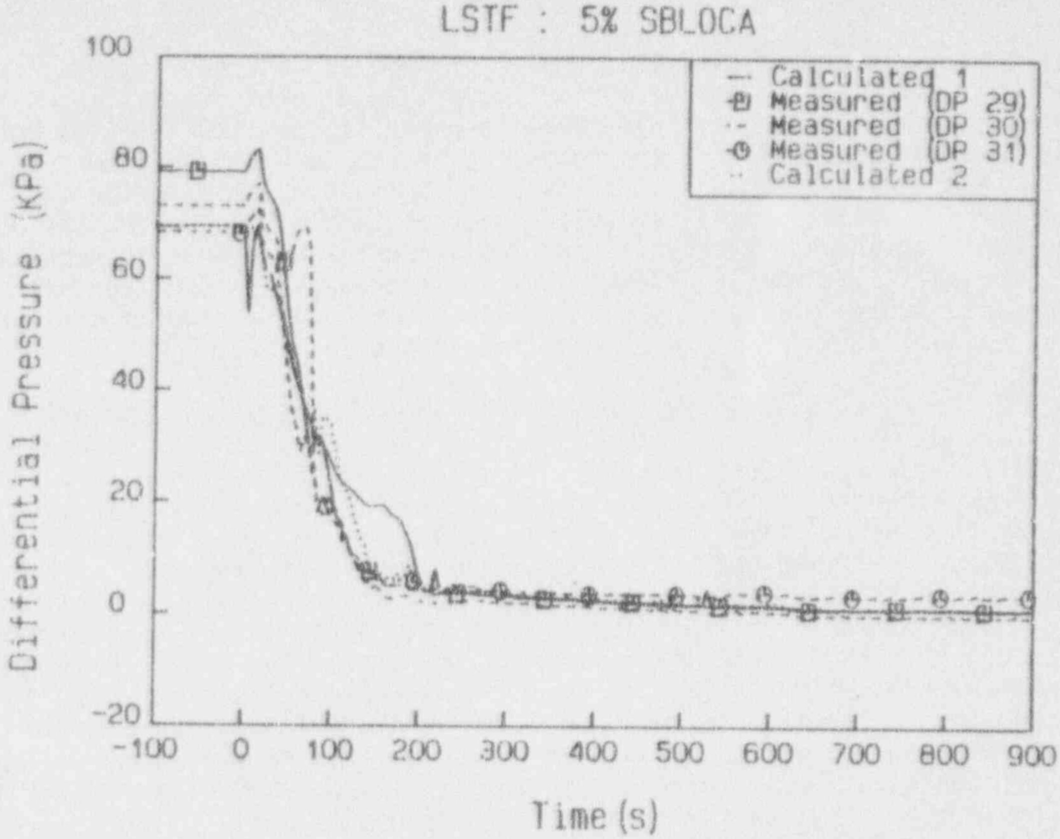


Figure 4.3.1.1 SG-B Inlet - Tube Top Differential Pressure.

conducted to evaluate the affect of two upper head to downcomer bypass levels on the loop seal clearing and concurrent core liquid level depression, both showed core heatups during the core boiloff period (after 180 s). The code calculation indicated more primary inventory in the core and the downcomer (see Fig. 4.3.1.2) than observed in the experiment for both experiments. Because the calculated primary inventory distribution did not match the data and the code has shown evidence of miscalculating the two-phase interphase drag for boiloff (the THETIS experiments) and blowdown (the Marviken JIT experiments), it is believed the code's interphase drag is responsible for the calculation/data mismatch in these experiments too.

4.3.2 Critical Break Flow (Vapor Pull-Through/Liquid Entrainment)

The critical flow model deficiencies were summarized in Section 4.2.1. However, one aspect of modelling critical break flow, not important in simulating LBLOCAs but very important for simulating smaller SBLOCAs, is an accurate vapor pull-through/liquid entrainment model. In essence, if a break is small enough to allow the primary fluid inventory to develop a stratified flow condition upstream of the break for an extended period of time then vapor pull-through when the liquid level is above the break flow area elevation, liquid entrainment when the liquid level is below the break flow area elevation, and combinations of vapor pull-through and liquid entrainment for the upstream liquid level as a function of the break area elevation are all important considerations in determining the primary depressurization and mass inventory loss history for the transient scenario.

Although RELAP5/MOD2 has a vapor pull-through/liquid entrainment model, one of the early findings of the code assessment effort was that the code's model is deficient. The work done by Ardron and Bryce gives the clearest indication of the deficiency. Further evidence is provided by the integral effects experimental assessment based on the LOFT LP-SB-02 test (Hall, 1990).

Ardron and Bryce Separate Effects Assessment: Ardron and Bryce quickly note that an accurate model to calculate entrainment through an off-take branch from a horizontal stratified flow regime condition is essential for a best-estimate transient analysis code. Accurately calculating the break flow will depend, in part, on accurately calculating the entrainment through the break. Similarly, a best estimate code must accurately calculate entrainment into the surge line for transients that discharge through the pressurizer relief valves.

Using data documented by Smoglie,1984; Maciaszek and Menponteil, 1986; Shrock et al., 1986; and Anderson and Benedetti, 1986 obtained with air-water and steam-water two-phase flows in off-take branches connected to large diameter horizontal pipes, the assessors showed that the code calculated break flow qualities that are generally too low. Figure 4.3.2.1 is a comparison of the code calculated flow quality in relation to the liquid depth in the horizontal, centered off-take branch at system pressures of 0.7 MPa and 7.0 MPa. The comparison is especially poor for an upward oriented off-take branch (see Figure 4.3.2.2). However, even downward oriented off-take branches with stratified flow are calculated to have flow qualities that are too low with respect to the data (see Figure 4.3.2.3).

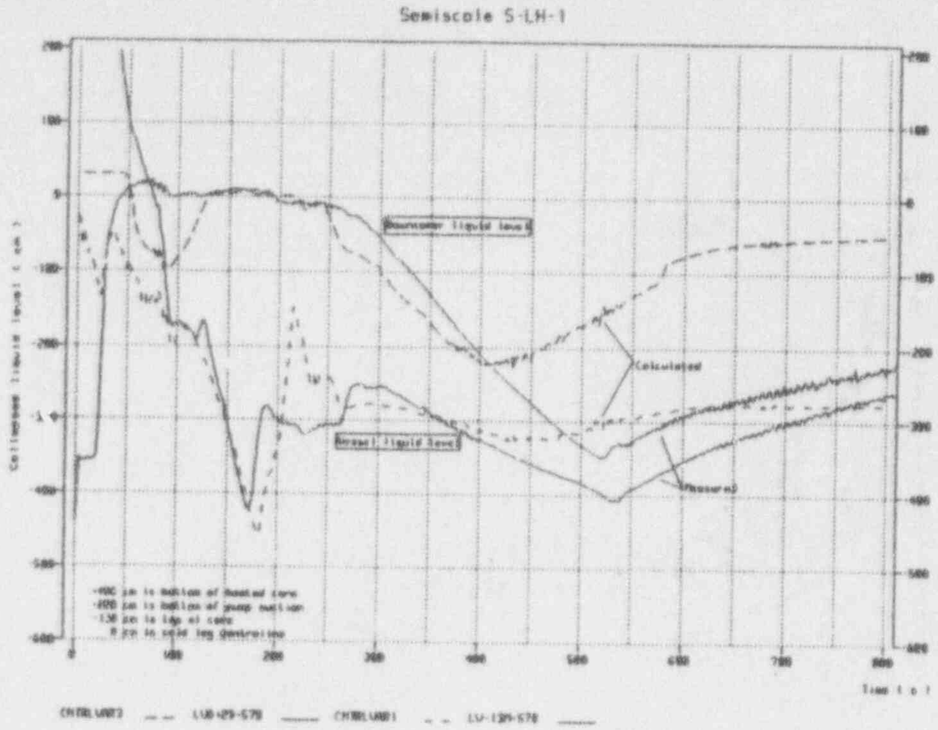


Figure 4.3.1.2 Vessel and Downcomer Collapsed Liquid Levels (Base Case 1).

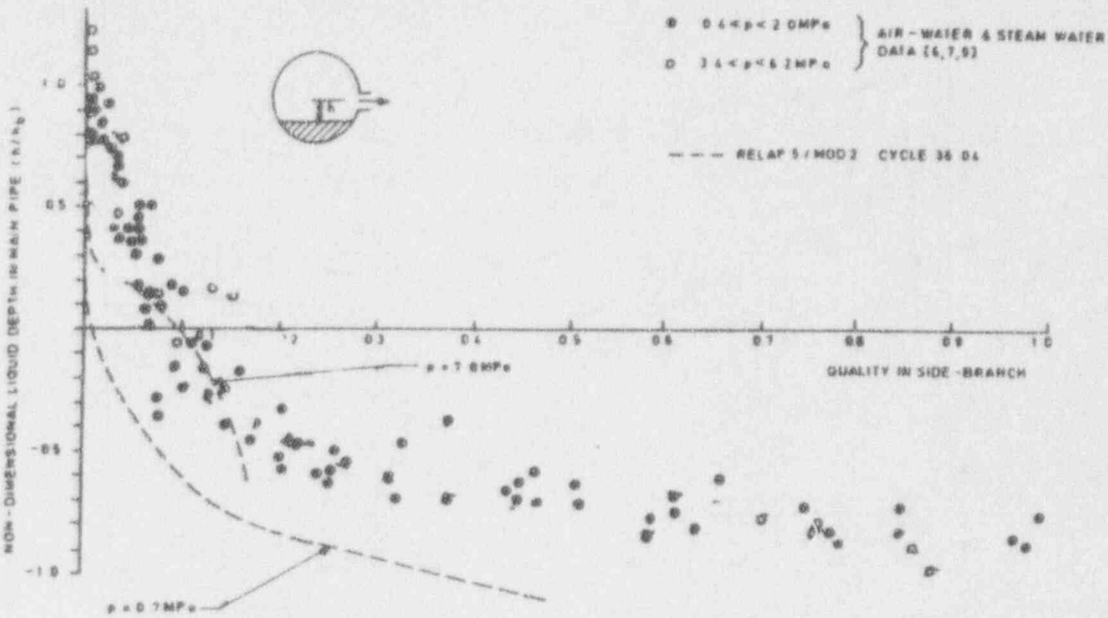


Figure 4.3.2.1 Ardron and Bryce: Calculated and Measured Discharge Flow Quality and Liquid Depth for Horizontal Side Branch.

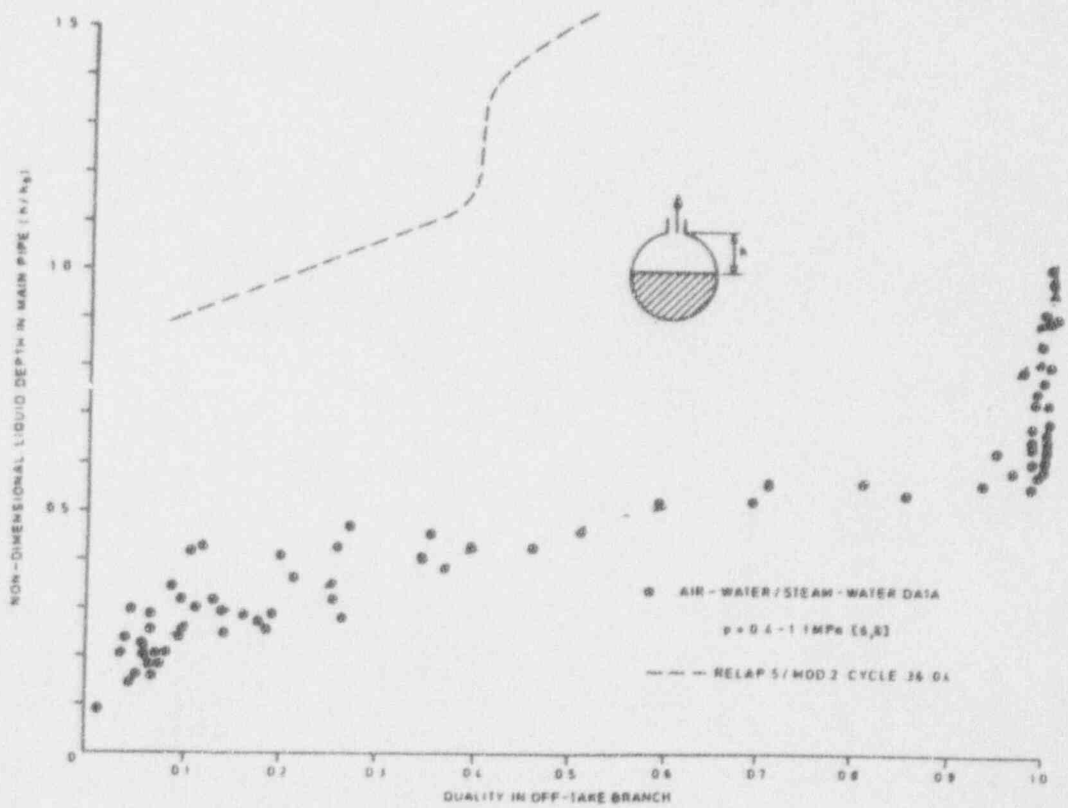


Figure 4.3.2.2 Ardron and Bryce: Calculated and Measured Discharge Flow Quality and Liquid Depth for Upward Oriented Offtake Branch.

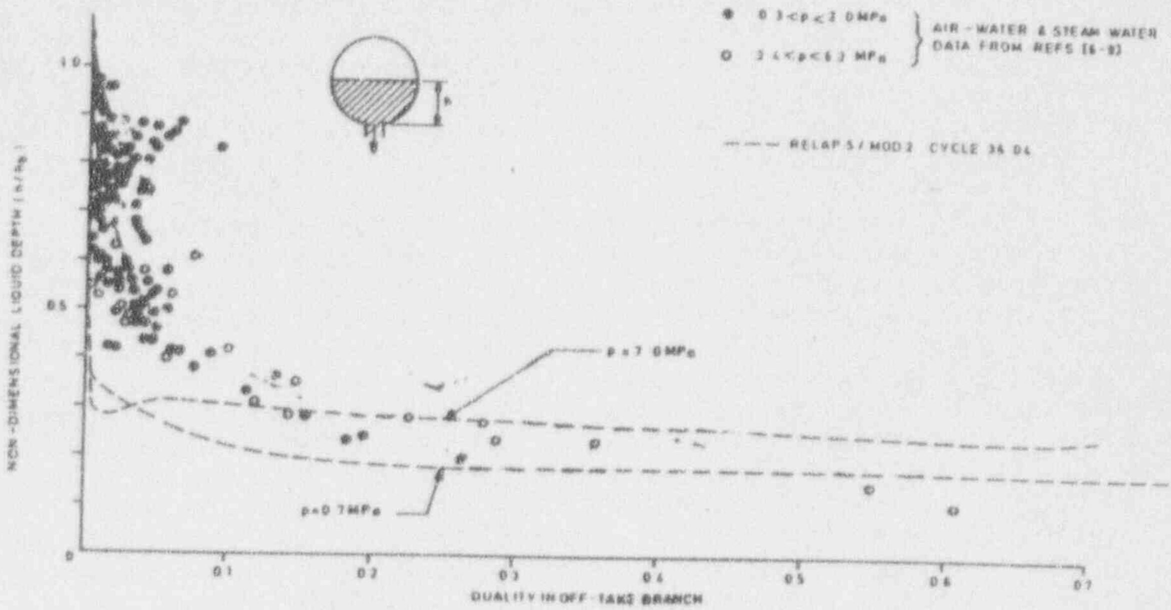


Figure 4.3.2.3 Ardron and Bryce: Calculated and Measured Discharge Flow Quality and Liquid Depth for Downward Oriented Offtake Branch.

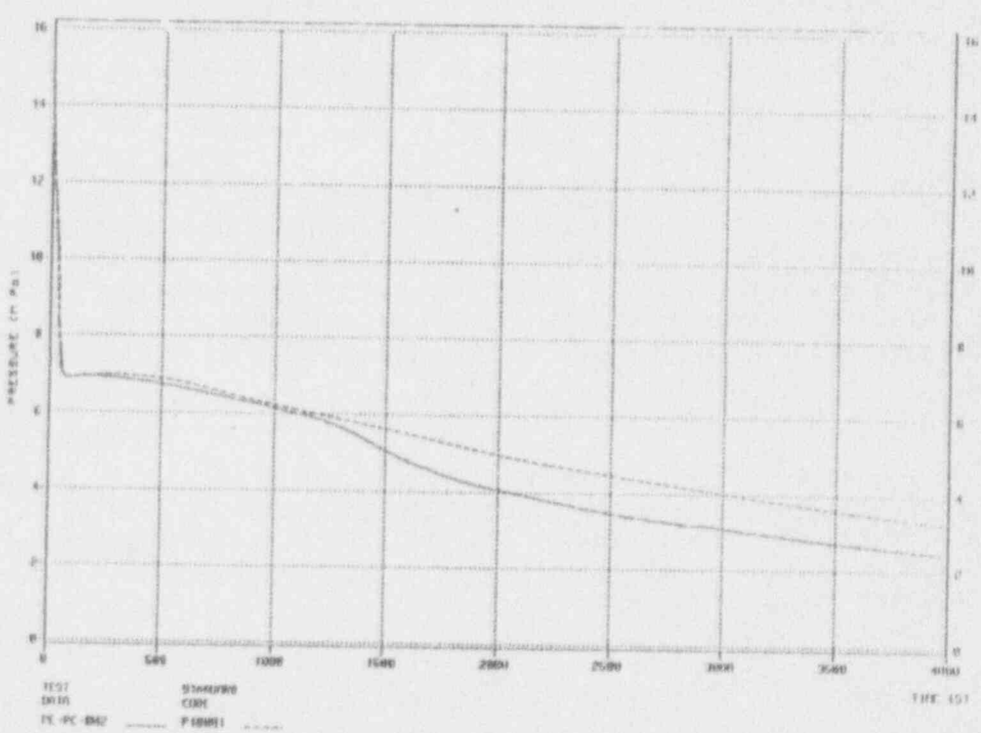


Figure 4.3.2.4 LOFT LP-SB-02 Experiment: Calculated and Measured Primary Pressure.

LOFT LP-SB-02 Experiment: The LP-SB-02 experiment, conducted in the LOFT facility, was a 1% SBLOCA located in the hot leg with HPI and the reactor coolant pumps left on. The LP-SB-02 experiment was a comparison experiment to LP-SB-01 that had the same initial and boundary conditions except that the pumps were shut off. The LP-SB-02 experiment, following break opening at time zero, included HPI injection beginning at 42 s. At about 600 s the pump head degraded sharply and there was evidence of flow stratification in the hot leg. However, the pumps maintained flow circulation around the loop until 1290 s.

The code calculation showed reasonable agreement between the measured and calculated primary pressure (see Figure 4.3.2.4) until about 1200 s. During the period from 1200 to 1900 s there are significant errors in the calculated depressurization rate leading to an overprediction of the pressure late in the transient. The calculated and measured break flow rate differ by about 15% (see Figure 4.3.2.5) for the first 800 s during subcooled blowdown. Beyond that time the code overestimated the rate by as much as 50% as saturated and two-phase blowdown occurred. This leads to large cumulative errors in the system mass inventory (see Figure 4.3.2.6). Comparisons of the measured hot leg and break line densities to the code calculation show that the code overpredicted the break line density throughout the transient even though the hot leg density was only slightly overpredicted during the subcooled blowdown phase and substantially underpredicted during the two-phase blowdown phase (see Figures 4.3.2.7 and 4.3.2.8).

Using code modifications to better simulate the horizontal stratification upstream of the break, Hall showed that a better match to the data resulted (see Figure 4.3.2.6). However, he noted that further work was required to define (i) the transition to stratified flow in geometries resembling a PWR hot leg and (ii) the flow quality in an offtake pipe connected to a larger horizontal pipe with two-phase mass fluxes on the order of 1000 kg/m^2-s.

4.3.3 Onset of Horizontal Stratification

The Hall, 1990 assessment of the LOFT LP-SB-02 experiment, described in Section 4.3.2, also provided evidence that RELAP5/MOD2 is unable to accurately calculate the onset of horizontal stratification. Hall observed that even though turbine meters located in both the top and bottom of the hot leg indicated stratified flow at 1100 s, the code did not calculate the onset of horizontally stratified flow until after the pumps were shut off at 2853 s.

4.3.4 Vertical Stratification Model

Transition to the vertically stratified flow regime in a cell is triggered when the difference in void fraction between the cell above and below is greater than 0.5 and the volume average mixture mass flux is less than the Taylor bubble rise velocity mass flux (Ransom et al, 1985). Two of the assessment studies noted switches to the vertical stratified flow regime that produced incorrect results.

Hall and Brown, using the LOFT LP-SB-01 experimental data, found that when the code switched to the vertical stratification model in the lower component

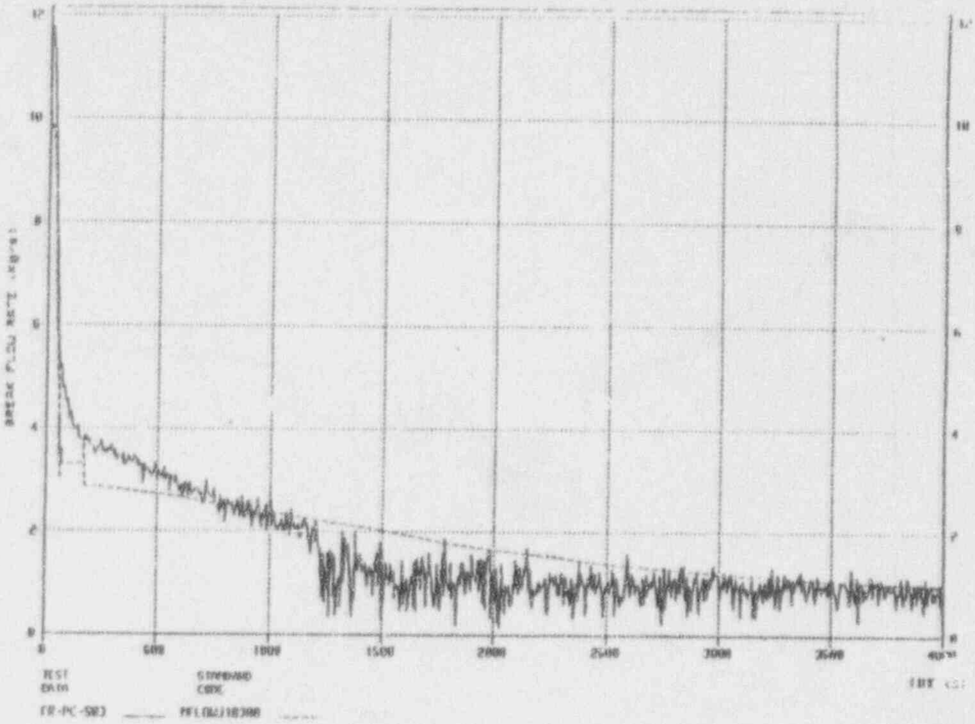


Figure 4.3.2.5 LOFT LP-SB-02 Experiment: Calculated and Measured Break Mass Flow Rate.

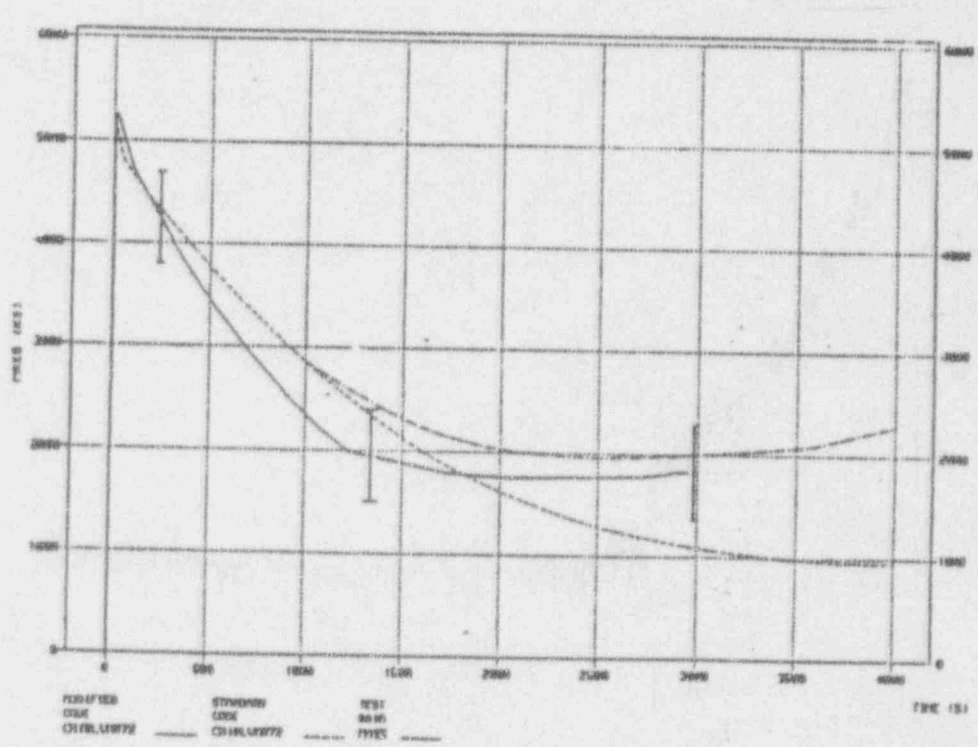


Figure 4.3.2.6 LOFT LP-SB-02 Experiment: Calculated and Measured Primary Coolant System Inventory.

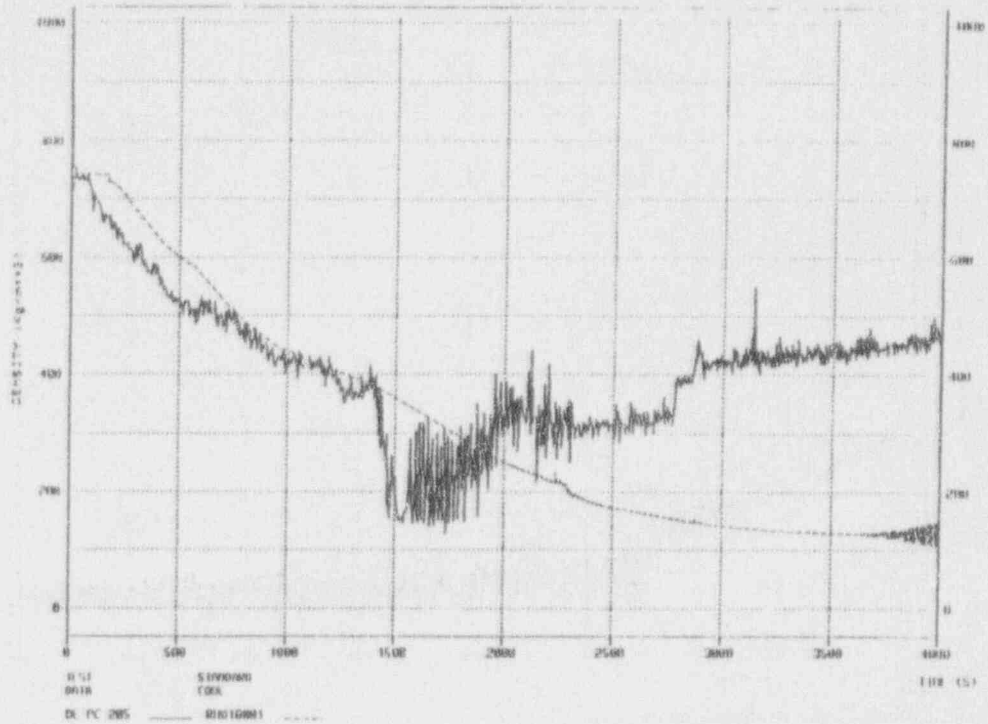


Figure 4.3.2.7 LOFT LP-SB-02 Experiment: Calculated and Measured Hot Leg Density.

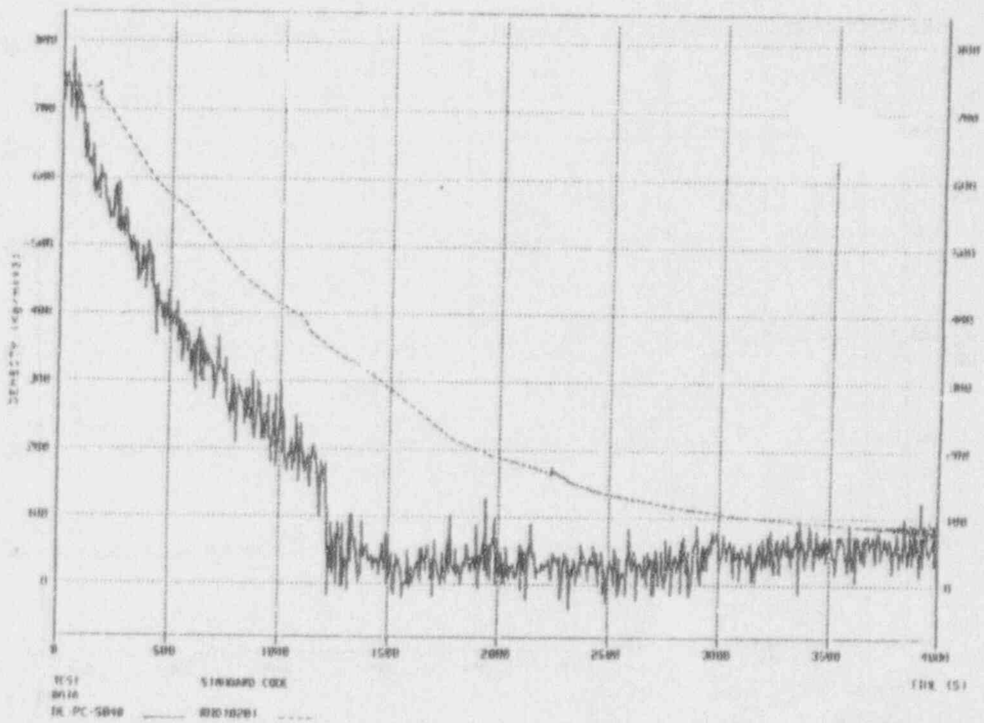


Figure 4.3.2.8 LOFT LP-SB-02 Experiment: Calculated and Measured Break Line Density.

volume of a two volume, vertically-oriented representation of the LOFT upper plenum the upper volume drained such that the hot legs also drained (see Figure 4.3.4.1 at 1080 s). Sudden draining was calculated since the interphase drag forces were suddenly reduced as the vertical stratification flow regime was simulated. Consequently, the calculated hot leg conditions, after 1080 s, showed a mixture density of 30 kg/m^3 whereas the data showed densities of nearly 300 kg/m^3.

Another instance of the vertical stratification model giving incorrect results was noted by Croxford and Hall in their assessment studies using the THETIS data. As briefly discussed in Subsection 4.2.2, and shown in Figure 4.2.2.3, oscillatory behavior was calculated by the code when the authors tried to set the problem steady-state conditions prior to boildown. The assessors found that when the vertical stratification model was disabled, the oscillations were eliminated.

4.4 IMPACT OF CODE DEFICIENCIES ON OPERATIONAL TRANSIENT ANALYSIS

Operational transients include such scenarios as the loss of feedwater, loss of offsite power (LOOP), anticipated transient without scram (ATWS), and many others. Evaluations of these transients are conducted as part of the pre-commissioning phase of a plant's licensing process and consequently there are a number of plant experimental data sets available.

Many of the operational transients can be characterized as "gentle" events, especially compared to LOCA transients, because the transient time scale is long and because changes in plant conditions are small. Often operational transients last several thousands of seconds.

In general the code has a highly-respected capability to calculate the behavior of a plant during an operational transient, but even so code deficiencies have been identified. Deficiencies identified during operational transient analyses include interphase drag (Stubbe and Vanhoenacker, 1992; De Vlaminck, Deschutter, and Vanhoenacker, 1992; Scriven, 1990), vapor pull-through-liquid entrainment (Croxford, Harwood, and Hall, 1992), critical heat flux (Keevill, 1992), subcooled boiling model (Brain, 1992), and condensation (Rouel and Stubbe, 1992). Deficiencies in each of these five areas will be discussed in the next five subsections.

4.4.1 Interphase Drag

Deficiencies in the interphase drag model were responsible for misrepresenting the void fraction profiles in two regions of PWR systems during operational transients: the steam generator (SG) secondaries and the pressurizer. Stubbe and Vanhoenacker, 1992 and De Vlaminck, Deschutter, and Vanhoenacker, 1992 identified the interphase drag model as being the cause of incorrect calculations of the SG secondary inventory liquid level during the manual loss of load test and the November 22, 1985 reactor trip event at the Doel-4 Nuclear Power Station. Scriven, 1990 identified the interphase drag model as being the cause of excessive calculated liquid swell in the pressurizer; as a consequence, the power-operated-relief-valve (PORV) flow was too large.

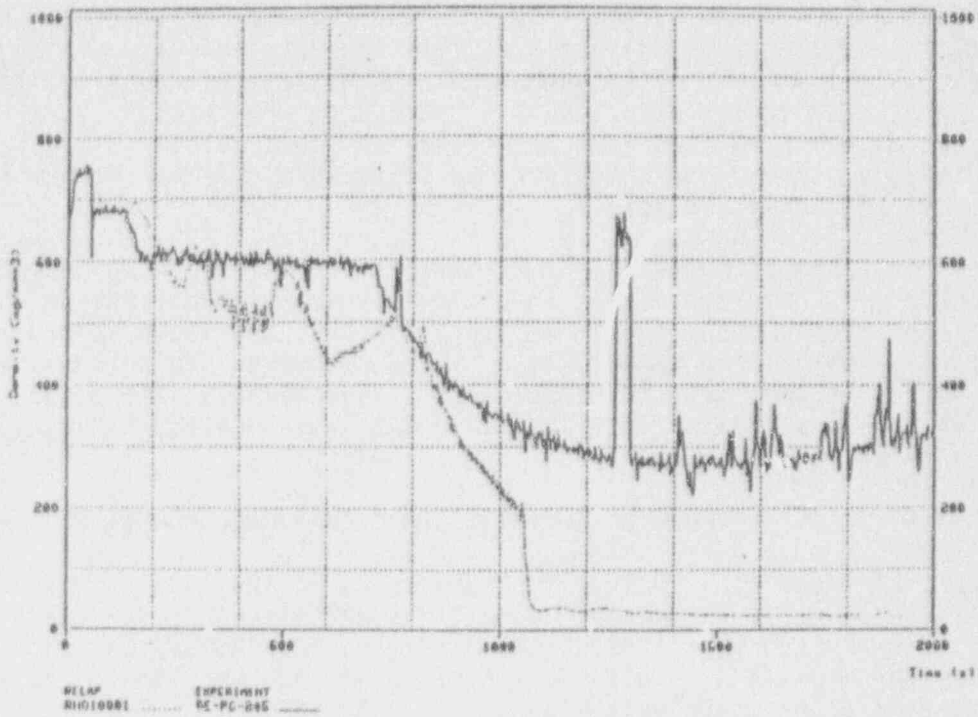


Figure 4.3.4.1 LOFT LP-SB-01 Experiment: Calculated and Measured Hot Leg Density.

Doel-4 Operational Transients: The interphase drag model tends to calculate too much drag between the liquid and vapor phases for the SG steady-state conditions. The net result is an incorrect distribution of the secondary liquid inventory that affects the calculated mixture level and the calculated readings for the narrow range water level. The net result is seen in the comparison of the code calculation and the measured data for the narrow range water level meter for the plant as shown on Fig. 4.4.1.1. Initially the calculated water level was set to equal that of the measured narrow range water level. However, after 40 s the plant power output decreased and the calculated narrow range measurement was found to be substantially less than the measured narrow range measurement. Following analysis of the results, including a number of parametric calculations, the authors found that the measured values could be matched by increasing the secondary inventory by 10 %. Thus, the authors concluded the code gave an unrealistic level swell at steady-state conditions due to an unrealistically large interfacial drag.

LOBI Loss-of-Feedwater Analysis: The analysis completed by Scriven, 1990 concluded, by deductive reasoning, that the code overcalculated the level swell in the pressurizer. Scriven's reasoning was based on the coupling between the calculated primary pressure and the PORV flow discharge quality. He observed that because a significant percentage of the calculated PORV flow was liquid, the primary pressure did not markedly decrease. Thus, the calculated pressurizer mixture level resulted in more primary system mass loss through the PORV than experienced by the experimental system. However, it should be noted that the observation by Scriven could be affected also by the mixture quality of the flow moving from the hot leg to the pressurizer through the surge line. Analyses completed by Croxford, Harwood, and Hall, 1992 imply that the problem may also be partially caused by a faulty calculation of the surge line mass flow resulting from an overcalculation of the liquid entrainment (see Subsection 4.4.2).

4.4.2 Vapor Pull-Through and Liquid Entrainment

Croxford, Harwood, and Hall, 1992 assessed the RELAP5/MOD2 code by using the LOFT loss-of-feedwater experiment LP-FW-01. Because the experiment investigated the system behavior when the PORV was latched open by the operator after reactor scram, the measured PORV mass flow data was available for assessment.

Following evaluation of the authors' base case calculations, it was found that the average density of the pressurizer PORV mass flow was significantly overcalculated (see Fig. 4.4.2.1). The overcalculation was postulated to be caused by an overcalculation of the quantity of hot leg liquid entrainment for mass moving from the hot leg to the surge line. To investigate the effect of a more accurate vapor pull-through/liquid entrainment model, the authors installed the vapor pull-through/liquid entrainment model currently implemented in RELAP5/MOD3 (see Ardron and Bryce) and recalculated the transient. The revised calculational results showed a marked improvement (see Fig. 4.4.2.2).

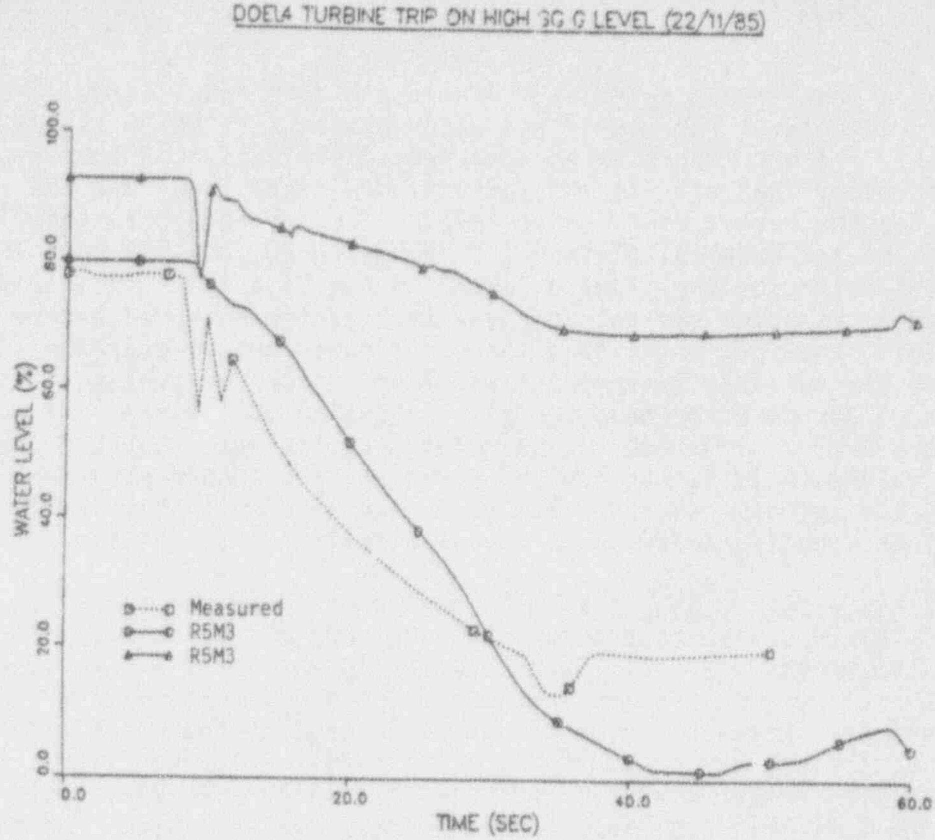


Figure 4.4.1.1 Doel 4 Power Plant: Calculated and Measured Narrow Range Water Level and Calculated Wide Range Water Level.

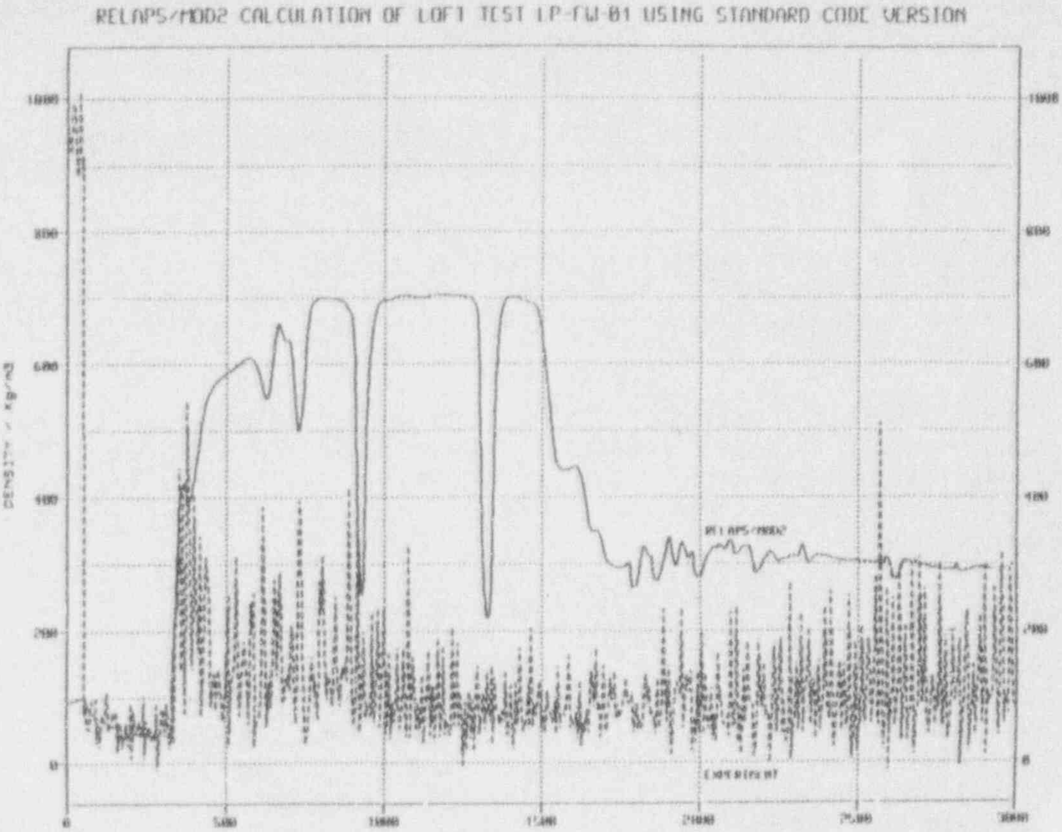


Figure 4.4.2.1 LOFT-LP-FW-01: Measured and Calculated Pressurizer PORV Flow Density for Base Calculation.

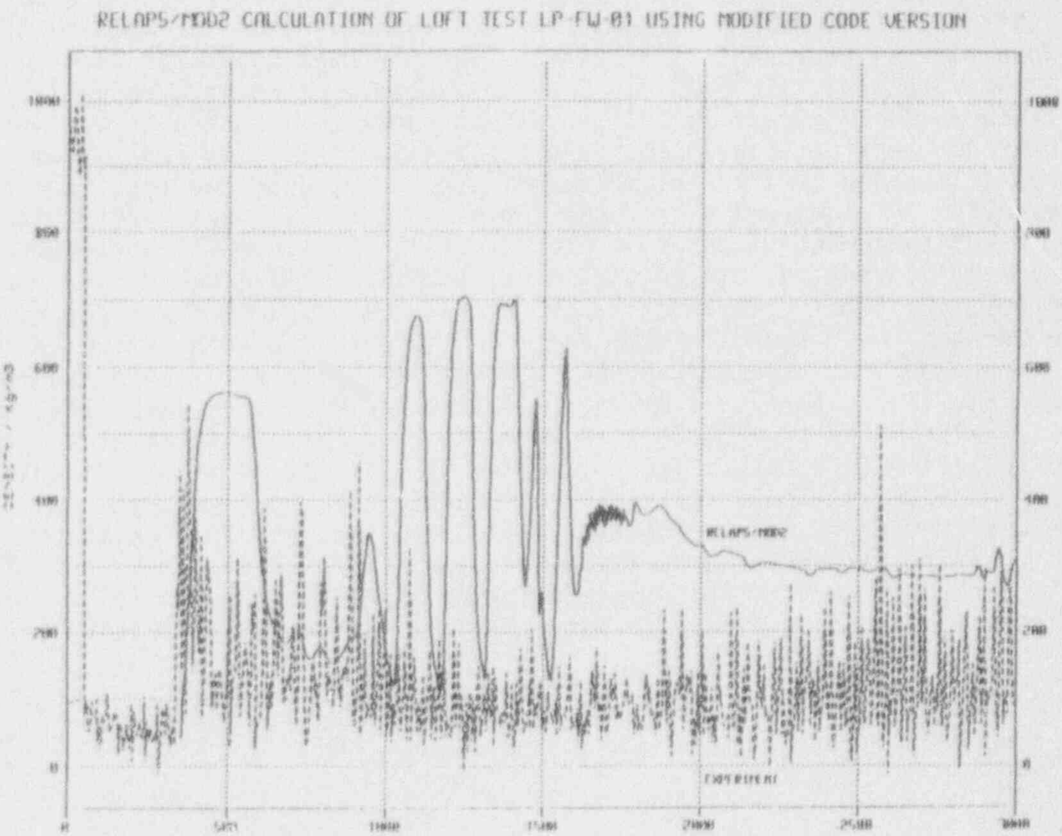


Figure 4.4.2.2 LOFT LP-FW-01: Measured and Calculated Pressurizer PORV Flow Density using RELAP5/MOD2 with Ardron/Bryce modifications.

4.4.3 Critical Heat Flux

The code assessment based on the LOFT L9-4 experiment, performed by Keevil, 1992, showed that the Biasi critical heat flux correlation was used outside its range of validity.

The LOFT L9-4 experiment (see Subsection 3.2.25) simulated a loss of offsite power in which power was lost to the primary coolant pumps. In addition, the main feed water was lost to the steam generators and the control rods failed to scram.

Keevil's base calculation revealed that unexpected pressure spikes were occurring due to an erroneous calculation of fuel rod dryout. Further investigation showed the Biasi critical heat flux correlation was being applied at pressures outside its range of validity and was giving negative CHF predictions.

4.4.4 Subcooled Boiling Model

During an anticipated transient without scram (ATWS) accident in a pressurized water reactor, the core power is not tripped and the resulting heat-load mismatch causes the primary coolant system to be significantly overpressurized. The peak pressure attained will be sensitive to the volume of vapor produced within the primary coolant system as a result of subcooled nucleate boiling phenomena.

The code was assessed against data from two series of high-pressure, steady-state, subcooled boiling experiments (see Subsection 3.1.9). The experiments were modeled with the RELAP5 code using pipe components to represent the test sections. The RELAP5-calculated volume void fractions in each of the cells were plotted against test section height for the experiments for a variety of inlet mass flow rates and fluid subcoolings. For cases of high subcooling (Egen Runs 7, 15, and 16), the code was shown to overpredict the void fraction near the test section inlet. Because the heat addition process was continuous along the tube length, the overprediction of void at the test section inlet resulted in a general overprediction of void at all test section locations. This disagreement is shown in Fig. 3.1.9.2. To evaluate the effect that the code deficiency in void prediction at high-subcooling would have for an ATWS event, the code-calculated channel-average void fraction was compared with the upper and lower bounds of the experimental data for each of the tests. This comparison indicated that RELAP5 systematically overpredicted the channel-average void fraction. However, in five of the twelve cases the calculation fell within the experimental uncertainty band, and in three other cases the calculation fell only slightly outside it. In the worst case a void fraction error of 6% was indicated. This agreement was considered reasonable, and the study concluded that the code may be used with reasonable confidence to calculate the subcooled nucleate boiling void fraction during PWR ATWS sequences. However, the code's behavior for this case should be remembered and used in conjunction with future assessments of the subcooled boiling model.

4.4.5 Condensation

Rouel and Stubbe, 1992 found that excessive condensation was calculated when feedwater injection was initiated. The presence of excessive condensation caused the calculated secondary system pressurization to temporarily halt and depart from the measured data. The secondary pressure measurement showed a smooth pressurization.

5.0 INTEGRATING THE ICAP ASSESSMENT STUDIES

The assessment studies described in the previous two sections, have a number of uses. Of course, each assessment study gives insights concerning the code's strengths and weaknesses with regard to the specific assessment data set. But, just as importantly, the implications of the defined code strengths and weaknesses guide code users in knowing what kinds of problems the code can best analyze and what kinds of problems the code will probably give marginally acceptable or even totally unacceptable results.

The objective of this section is to: (i) list the problem signature of each major code deficiency and then (ii) relate the deficiency to PWR transient behavior. Items (i) and (ii) are summarized in Table 5.1.

Table 5.1 Code Deficiencies: Problem Signatures and Effect on LWR Analysis

Deficiency/Status	Problem Signature	Impact of Deficiency on LWR Analyses
Counter-current flow-limiting (CCFL)/ MOD2 does not have a CCFL model; interphase drag model used instead.	1. Since CCFL was simulated using interphase drag that was too large, non-representative liquid hold- is calculated. 2. Vessel/loop inventory incorrect - too much liquid in SG.	1. Unphysical liquid holdup in SGs and other locations. 2. Misleading inventory distribution between parts of vessel and loops. 3. Core liquid level depression during SBLOCA can be unrepresentatively too deep.
Interfacial friction in bubbly-slug flow regime.	1. Too much level swell. 2. Incorrectly calculated vertical void fraction profile; void fraction too low at lower elevations and too high at upper elevations.	1. Vessel water level too high. 2. Core has tendency remain wetted.

Table 5.1 Code Deficiencies: Problem Signatures and Effect on LWR Analyses

Deficiency/Status	Problem Signature	Impact of Deficiency on LWR Analyses
Vapor pull-through/liquid entrainment in horizontal pipe.	1. Incorrect critical mass flow rate. 2. Incorrect horizontal pipe fluid conditions.	1. Incorrect event timings. 2. Affects vessel and loop inventory distribution.
Critical heat flux	1. Calculated surfaces remain well-cooled while data shows heatup. 2. Location of CHF unknown.	1. Heated surface temperatures under predicted. 2. Axial temperature profile underpredicted.
Critical flow modeling	o For flow geometries that induce nonequilibrium critical flow, mass flow rate substantially underpredicted.	1. All events occur late. 2. Incorrect mass distribution.
Inception of vertical stratification.	o Early draining of hot leg and vessel upper plenum.	o Poor loop and vessel mass distribution.

5.0 CONCLUSIONS AND OBSERVATIONS

During the course of conducting and reviewing the assessment studies that have been discussed in the previous sections:

- Nine major code deficiencies (including various facets of each deficiency) were identified.
- A better understanding of the code assessment matrix needed to assess RELAP5/MOD3 has been obtained, since MOD3 was created to eliminate many of the shortcomings of MOD2.
- The ICAP assessments showed that MOD2 is a very effective tool for analyzing a great number of problems, even though some deficiencies are present.

PART II: SUMMARY OF RELAP5/MOD3 CODE ASSESSMENTS

7.0 SUMMARY OF ASSESSMENT RESULTS AND APPLICABILITY OF CODE

The RELAP5/MOD3 Version 5M5 code was "frozen" and released in March, 1990 such that only error corrections could be inserted thereafter. The code was "frozen" to allow the world-wide user community to identify code deficiencies in a stable code version that did not change with time as each deficiency was found.

7.1 HISTORY OF CODE

As summarized in Weaver et al., 1989:

> Prior to the formation of the RELAP5/MOD3 improvement consortium, the members of ICAP performed assessment calculations using "frozen" versions of the RELAP5/MOD2 computer program. The results of these assessment calculations were sent to the INEL (described in Part I) for the correction of code errors and the evaluation of code deficiencies. In accordance with the rules of ICAP, code errors were corrected with the issuance of a new "frozen" code version, while code deficiencies were logged and remained uncorrected in the several "frozen" versions of RELAP5/MOD2. As the list of code deficiencies grew and with the desire of the USNRC and ICAP members to extend the mission of the RELAP5/MOD2 to include the analysis of large break LOCAs, the RELAP5/MOD3 code improvement program was developed and initiated in the spring of 1988.

The areas in the RELAP5/MOD2 code that were improved or changed to create RELAP5/MOD3, version 5M5 are listed below (items 1 through 17) together with a short description of the new model's capabilities (Riemke, 1989, 1990). Of these items, the first 9 were modified in response to the RELAP5/MOD2 deficiencies identified at least in part by ICAP assessment studies (see Section 2.2, pages 8 and 9 of this report).

1. Counter-current-flow-limiting (CCFL) model - Added to enable the code to simulate CCFL behavior through geometrically complex flow passages such as the core upper tie plate.

2. Interfacial friction in wetted wall bubbly/slug flow regime - Added to specifically improve the calculated interfacial friction for rod bundles and large diameter pipes. The model is based on the relationship between void fraction and interfacial friction for steady, fully-developed conditions. The model is applicable to the full range of geometries and flow conditions encountered in PWR safety analysis.

3. Vapor pull through and liquid entrainment in horizontal pipe offtakes - Added to improve the code's capability to calculate the correct flow through horizontal pipe offtakes as a function of the upstream pipe liquid level.

4. Critical heat flux (CHF) - The Biasi CHF correlation was replaced with a Groeneveld CHF correlation-based lookup table.

5. Condensation in horizontal pipes - Model introduced to calculate the processes of mixing the emergency core coolant (ECC) liquid with steam and steam condensation in the vicinity of the ECC injection point.

6. Horizontal stratification inception criterion - Based on a recommendation from the Japan Atomic Energy Research Institute (Kukita,

et al., 1987), the Taitel-Duckler transition criteria was modified to test the relative velocity between the phases against the transition criterion rather than the vapor velocity alone. In addition, junction based interfacial drag was added and programmed to use donor void fraction.

7. Reflood heat transfer - The convection heat transfer logic and correlations for reflood surfaces were removed and the same correlations are now used for all surfaces.

8. Critical flow modeling - The transition from the subcooled choking model to the two-phase choking model was smoothed and a discontinuity in the two-phase choking model was removed.

9. Inception of vertical stratification - The model logic for detecting the inception of vertical stratification was modified to be that used in the TRAC-BWR code (Riemke, 1989). Also, the vertical stratification model was modified to be consistent with the junction-based interfacial drag model (see item 6).

10. Metal-water reaction simulation - The Cathcart-Pawel model was added to the code to enable analysts to calculate the extent of core fuel rod metal-water reaction during large-break LOCA transient simulation.

11. Fuel mechanical model - The Powers and Meyer correlations were added to the code to calculate cladding deformation for large break LOCA analysis.

12. Radiation heat transfer modeling - A radiation model was incorporated into the code to estimate the radiation energy exchange between a surface and its surroundings.

13. Non-condensible gas modeling - The code's capability to track non-condensible gas movement in the system was enhanced to simulate a pure noncondensible medium. Also, the pressure-temperature iteration scheme for evaluating the thermodynamic state of the steam/noncondensible mixture was replaced with a pressure-energy iteration.

14. Downcomer penetration and emergency core coolant bypass - Simulation of these phenomena were enhanced by including the Wallis correlation for annular films (Schneider, 1991).

15. Enhanced code portability - The ease of adapting the code for use on other computer platforms was enhanced primarily so analysts can more easily use workstations of various designs.

16. Code speedup - The code's calculational speed was increased by vectorizing a portion of the coding used on the Cray.

The RELAP5/MOD3 Version 5M5 code, including the above improvements, was released to ICAP members in March, 1990.

7.2 CODE DEFICIENCIES

RELAP5/MOD3 code deficiencies, isolated in version 5M5, are (Slater, 1992):

a. <u>Phase appearance and disappearance</u>: In the presence of noncondensibles water property errors and unrealistically low temperatures at phase interfaces sometimes result. When noncondensibles are not present water property errors sometimes result.

b. Equation of state: The code allows fluid to remain in a metastable state.

c. Critical point condensation: Thermodynamic state errors are calculated and excessive condensation is often calculated, even when the ECCMIX component is used.

d. Numerics: Results from different computers are different; run time is unexpectedly slow; oscillations.

e. Containment modelling: Boundary condition and continuity equation errors.

f. Accumulator model: Unrealistically low temperature and unphysical flow spikes.

g. Counter current flow limiting (CCFL): The code allows flow limiting in forbidden region.

h. Interphase drag: Interphase drag for some scenarios is incorrect.

Following definition of the above deficiencies and as work undertaken by the ICAP community progressed, several additional deficiencies were identified:

1. Incorrect channel flow fluid depth: The code often cannot calculate the correct fluid depth for stratified channel flow (S. Lee and Kim, 1992).

2. Critical flow: The code's critical flow model continues to give results that fall outside the data uncertainty band for some transient scenarios (S. Lee, Chung, and Kim).

3. Incorrect calculation of vertical pipe draining: Under some conditions, the limits of which have not been rigorously defined, the code will not realistically calculate the process of draining a vertical pipe; voids are calculated to pass into lower cells before the upper cells are fully drained (Roth, Choi, and Schultz, 1992).

4. Undercalculated wall temperatures following dryout: Following CHF the code may undercalculate the wall surface temperature. The calculated temperature magnitude and distribution may not match the data (Nilsson, 1990).

5. ECCMIX component: The ECCMIX component overcalculates the condensation rate for some SBLOCA scenarios and causes the code to fail (Roth, Choi, and Schultz, 1992).

6. Chen transition boiling criteria: The Chen transition boiling criteria has been shown by Analytis, 1992 to be a factor causing large amplitude heat transfer coefficient oscillations and thus a factor causing code calculation instability under some conditions.

The following two sections discuss reviews of ICAP assessment reports that have identified deficiencies g, h, and 1 through 5. Deficiency 6 was reported at the First CAMP Specialist's Meeting held at Villigen, Switzerland in June, 1992 and is by the author's own admission a "preliminary assessment." Consequently a final assessment report is not available.

7.3 CODE ASSESSMENT MATRIX

The RELAP5/MOD3 code assessment matrices were defined by the ICAP members and are shown in Figs. 7.3.1 and 7.3.2 for LBLOCAs and SBLOCAs. Both matrices are based simply on the assessment studies performed by ICAP members and were defined using the phenomena of importance for each transient type as listed by the Committee on the Safety of Nuclear Installations (CSNI) of the Task Group on the Status and Assessment of Codes for Transients and Emergency Core Cooling Systems of the Principal Working Group No. 2 on Transients and Breaks for the Organization for Economic Cooperation and Development (see CSNI, 1987). Phenomena represented by data in a particular experimental data set are indicated by cross-referencing the test facility versus the phenomena (see Fig. 7.3.1 - Matrix I). If the experimental data set contained good data for a phenomena of interest, then a filled-in circle is shown, for example the ECC bypass and penetration data recorded in the UPTF facility are good. If on the other hand, the data is of limited usefulness due to large uncertainty bands or other reasons, then an open circle is shown. Finally, if the data do not contain information on a particular phenomena, then a dash is shown.

Because of the more extensive data set available for SBLOCA-related phenomena, correlations are also shown between the "test type" and the phenomena as well as the test type versus the test facility systems tests.

Matrix I

CROSS REFERENCE MATRIX FOR LARGE BREAKS IN PWRs

- Phenomena versus test type
  - ● simulated
  - ○ partially simulated
  - – not simulated
- Test facility versus phenomena
  - ● suitable for code assessment
  - ○ limited suitability
  - – not simulated
- Test type versus test facility
  - ● simulated
  - ○ partially simulated
  - – not simulated

Phenomena	Test Facility: Test Type: Blowdown	Reflood	Refill	Sep. Effects: RIT	UPTF
Break flow	–	–	–	–	–
Phase separation	–	–	–	–	–
Mixing and condensation during injection	–	–	–	–	–
Core wise void + flow distribution	–	–	–	–	–
ECC bypass and penetration	–	●	●	–	●
CCFL (UCSP)	–	–	–	–	–
Steam binding (liquid carry over, etc.)	–	–	–	–	–
Pool formation in UP	–	–	–	–	–
Core heat transfer incl. DNB, dryout, RNB	–	●	●	○	●
Quench front propagation	–	–	○	–	○
Entrainment (Core, UP)	–	–	–	–	–
Deentrainment (Core, UP)	–	–	–	–	–
1- and 2-phase pump behavior	–	–	–	–	–

Important test parameter
- leak location/leak size
- pumps off/pumps on
- cold leg injection/ combined injection

Figure 7.3.1 Code Assessment Matrix: LBLOCAs.

JEN-MATRIX1P

Matrix II

CROSS REFERENCE MATRIX FOR SMALL AND INTERMEDIATE LEAKS IN PWRs

- Phenomena versus test type
  - ● simulated
  - ○ partially simulated
  - – not simulated
- Test facility versus phenomena
  - ● suitable for code assessment
  - ○ limited suitability
  - – not simulated
- Test type versus test facility
  - ● simulated
  - ○ partially simulated
  - – not simulated

Test Facility

		Test Type			System Tests			Sep. Effects Tests				
		Stationary test addressing energy transportation on primary side	Stationary test addressing energy transp. on sec. side	Small leak w/o HPIS overfeeding, secondary side necessary	BETHSY 1:100	LSTF 1:50	SEMISCALE 1:1600	Northwestern	KAERI	KREISING	ROYAL Inst. of Tech.	UPTF
Phenomena	Natural circulation in 1-phase flow, primary side	●	●	●	●	●	●	–	–	–	–	–
	Natural circu.ation in 2-phase flow, primary side	●	●	●	●	●	●	–	–	–	–	–
	Reflux condenser mode and CCFL	●	●	●	●	●	○	–	●	–	–	●
	Asymetric loop behavior	–	–	●	–	–	○	–	–	–	–	–
	Leak flow	–	–	●	●	●	●	–	–	●	–	–
	Phase separation without mixture level formation	●	–	●	●	●	○	–	–	–	–	○
	Mixture level and entrainment in vertic. comp. s.g.**	●	●	●	●	–	–	–	–	–	–	–
	Mixture level and entrainment in the core	●	–	●	●	○	○	–	–	–	–	–
	Stratification in horizontal pipes	●	●	●	●	●	○	○	–	–	–	●
	ECC-mixing and condensation	–	–	–	–	–	–	○	–	–	–	–
	Loop seal clearance	–	–	●	●	●	●	–	–	–	–	–
	Pool formati n in UP/CCFL(UCSP)	●	●	○	○	○	○	–	–	–	–	–
	Core wide void and flow distribution	●	●	○	○	○	○	–	–	–	–	–
	Heat transfer in covered core	●	●	○	●	●	●	–	–	–	–	–
	Heat transfer in partially uncovered core	●	●	○	●	●	○	–	–	–	○	–
	Heat transfer in SG primary side	●	●	●	●	●	○	–	–	–	–	–
	Heat transfer in SG secondary side	○	●	●	●	●	○	–	–	–	–	–
	Pressurizer thermohydraulics	–	–	–	–	–	–	–	–	–	–	–
	Surgeline hydraulics	–	–	–	–	–	–	–	–	–	–	–
	1- and 2- phase pump behavior	–	–	–	–	–	–	–	–	–	–	–
	Structural heat and heat losses***	●	●	●	○	○	○	–	–	–	–	–
	Noncondensible gas effects	–	–	–	–	–	–	–	–	–	–	–
	Phase separ. in T-junc. and effect on leak flow	–	–	●	○	○	○	–	–	–	–	–
Test Facility System Tests	BETHSY	●	●	●								
	LSTF	–	–	●								
	SEMISCALE	–	–	●								

* volumetric scaling
** secondary side
*** problem for scaled test facilities
**** for intermediate leaks phenomena included in large break reference matrix may be also important

Figure 7.3.2 Code Assessment Matrix: SBLOCAs.

JEN-MATRIX2E

8.0 SYNOPSES AND EXECUTIVE SUMMARIES OF THE ICAP ASSESSMENTS

The results of ten code assessment studies are summarized. Six of the studies are separate effects experiment assessments and four of the studies are integral effects experiment assessments. The summaries of the separate effects experiments and the integral effects experiments are described in the following two sections.

8.1 SEPARATE EFFECTS ASSESSMENTS

The separate effects assessments that have been reviewed include studies completed to review the capability of the RELAP5/MOD3 version 5M5 code to calculate: (i) critical heat flux - Section 8.1.1, (ii) piping loads during safety and relief valve discharge - Section 8.1.2, (iii) downcomer penetration for LBLOCAs - Section 8.1.3, (iv) countercurrent flow in PWR hot legs - Section 8.1.4, (v) direct contact condensation during horizontal stratified flow - Section 8.1.5 and (vi) counter current flow limiting - Section 8.1.6.

8.1.1 CHF Correlations and Dryout

Reference: L. Nilsson, *Assessment of RELAP5/MOD3 Against Twenty-Five Post-Dryout Experiments Performed at the Royal Institute of Technology*, STUDSVIK/NS-90/93, October, 1991.

Code version: RELAP5/MOD3, Version 5M5.

Facility: Separate effects facility at the Royal Institute of Technology, Stockholm, Sweden

Objectives: To assess the code's ability to calculate heater-rod thermal response during core dryout.

Major phenomena: The time and location of CHF are presented in conjunction with axial heater temperature distributions.

Code deficiencies: The code always underpredicts the axial heater temperature magnitude and quite often gives an unrepresentative axial temperature distribution above the CHF point.

Impact of deficiencies: The code may undercalculate a heater rod temperature following CHF and may also give the improper axial temperature distribution. Thus the code probably would not give a conservative estimate of the heater rod behavior based on the evidence presented in this assessment.

User guidelines: Nodalization studies, performed with the heated tube divided into cell lengths of 0.05 m, 0.25 m and 0.5 m, showed the case with the longest cell length gave the same result as the more finely nodalized cases. Most integral effects calculations have core nodalizations with cell lengths of 0.5 m.

Base calculations: Calculations were done for each of the 25 experiments. The model was built by Studsvik personnel originally to complete the RELAP5/MOD2 assessment (Sjoberg and Caraher, 1986). The calculations showed reasonable agreement with the measured CHF location using the Groeneveld lookup table method. The calculated heater temperature, as a function of axial elevation, was generally low compared to the data as summarized under "Code Deficiencies."

Sensitivity studies: None.

Nodalization studies: Nodalization studies were performed to examine the change in the calculated results when the heated tube test section cell lengths were increased from 0.05 m to first 0.25 m and then 0.5 m (a decrease in the model cell number from 47 to first 22 cells and then 11 cells). The 0.5 m cell length was chosen because many integral system nodalizations model the core with 0.5 m cell lengths. The author found: "A nodalization study showed little effect of the number of axial fluid cells in the range 11 to 47 cells on the wall temperature."

Summary: Assessment of RELAP5/MOD3 has been made against twenty-five post-dryout experiments conducted at the Royal Institute of Technology (RIT) in Stockholm. A similar assessment has earlier been performed of RELAP5/MOD2 (Sjoberg and Caraher, 1986).

In the experiments, wall temperatures were measured on an electrically heated 7 m long tube with in inner diameter of 14.9 mm. The tube was cooled by upward flow of water with mass fluxes from 500 to 2000 kg/m^{2}s. The cases selected for this assessment covered pressures ranging from 3 to 14 MPa, heat fluxes from 400 to 1060 kW/m^2 and inlet subcoolings from 8.5 to 12 K.

The RELAP5 model for the tube was comprised of 47 axial fluid cells. A fine nodalization was employed for the upper part of the test section where post-dryout conditions took place. Time dependent pressure, temperature and flow boundary conditions were imposed to simulate the fluid entering the test section. The region downstream of the test section was modeled by a time dependent pressure boundary condition. The axial heat flux distribution was uniform. An insulated boundary condition was imposed upon the outside edge of the tube wall while the inner edge received its boundary condition - a heat transfer coefficient and a temperature sink - from the RELAP5 heat transfer package. Once the RELAP5 simulations reached steady state the axial temperature distribution along the tube was compared to experimental values.

The new method for calculating critical heat flux in RELAP5/MOD3 uses a four-dimensional table with the heat flux as a function of pressure, mass flux, and steam quality based upon "Groeneveld's CHF Lookup Table." This procedure replaces Biasi's correlation in RELAP5/MOD2. After interpolation has been done in the table, eight multipliers are imposed on the CHF value to account for effects of diameter, bundle geometry, heat flux distribution etc. All of the analyzed RIT data were within the range of the MOD3 tables.

The results of the RELAP5/MOD3 assessment shows improvement in the CHF prediction compared to MOD2 (see Section 3.1.6). The prediction is generally non-conservative, i.e. the calculated dryout position falls in most cases downstream of the actual measured point (see Fig. 8.1.1.1). The code's calculation of CHF is judged to be reasonable.

The code's calculation of the postdryout heat transfer results in an underprediction of the temperature as a function of axial length and also gives an axial temperature profile different than measured. Whereas the difference between the calculated and measured temperature profile may be indicative of a code deficiency, the root cause of such a deficiency and the effect of the new coding on calculating the temperature behavior of a light water reactor core fuel rod have not been identified.

Nodalization studies with longer test section cells show that increasing the length from 0.10 to 0.5 m did not remarkably impair the temperature predictions. The larger cell length, typical in nuclear reactor simulations, led of course to less resolution for the calculated axial dryout point and for regions with steep temperature gradients. However the net result was the same. (Note: The axial power profile used for this experiment was constant as a function of length. Since the axial power profiles characteristic of typical nuclear fuel rods are not constant as a function of length, further nodalization studies should be performed to verify this result for typical plant fuel configurations.)

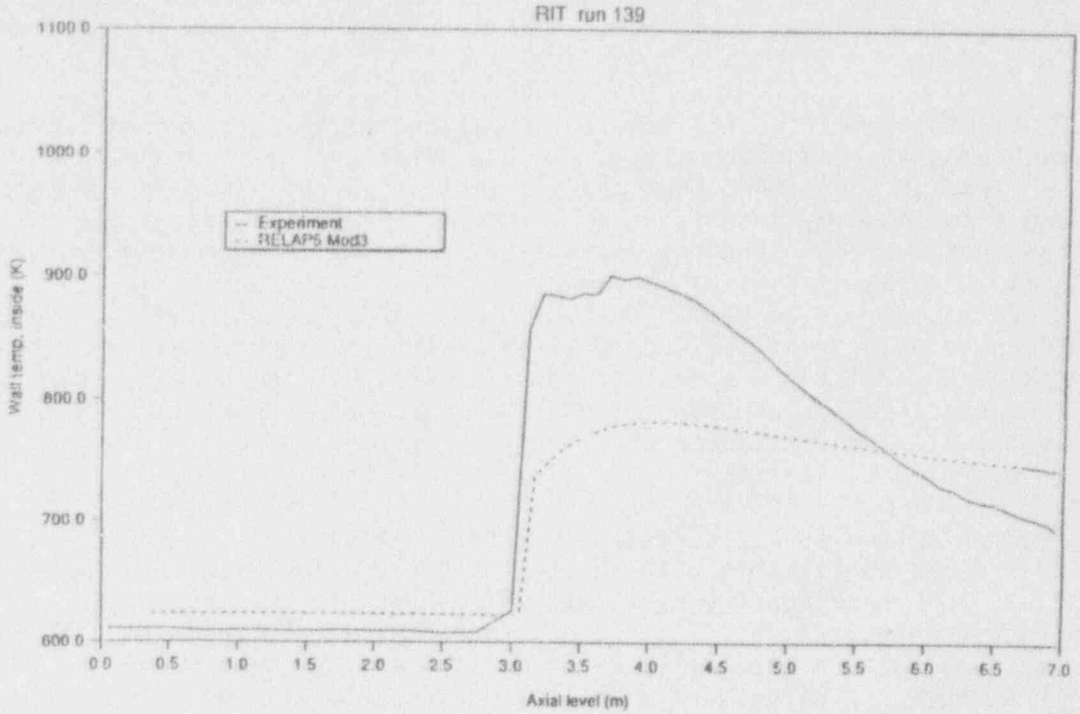


Figure 8.1.1.1 Inner wall temperatures for run 139 p=14.00 MPa, G=1970.5 kg per sqm-s delta-t=9.8 K, heat flux=757 per sqm.

8.1.2 Discharge Piping Hydrodynamic Loads

Reference: E. J. Stubbe, L. Vanhoenacker, and R. Otero, *RELAP5/MOD3 Assessment for Calculation of Safety and Relief Valve Discharge Piping Hydrodynamic Loads*, TRACTEBEL, October, 1990.

Code version: RELAP5/MOD3 Version 5M5.

Facility: Combustion Engineering's Kreising Development Laboratory at Windsor, CT.

Objectives: Assess the potential of RELAP5/MOD3 to predict safety and relief valve discharge piping hydrodynamic loads, propose modelling guidelines for calculating piping loads, and highlight the effect of various physical models present in RELAP5 on calculating piping loads.

Major phenomena: Rapid acceleration of a liquid slug in a loop seal and the eventual establishment of two-phase steady flow in the discharge piping. The liquid slug was accelerated by the source steam upon the rapid opening of a safety valve at the downstream end of the slug. Significant initial temperature gradients (623 K to 423 K over 2.74 m) in liquid slug and rapid depressurization caused annular-mist flow to evolve in 0.3 s. Two-phase choked discharge, interphase drag, heat transfer to pipe walls, and two-phase flow at abrupt expansions were considered.

Code deficiencies: The authors have stated that the code calculates the peak loads to occur at later times than the measured values due to an undercalculation of the coupling between the liquid and vapor phases. That is, the authors contend that because the interphase drag is underestimated for this application the calculated liquid slug velocities are less than those in the experiment and thus the times of peak loading are delayed. The reviewers have not finished the confirmatory calculations required to verify the author's point.

Impact of code deficiencies: The calculated liquid slug velocity is less than measured and the peak loads are thus delayed.

User guidelines: The authors made use of user's guidelines provided in a tudy completed earlier by the Electric Power Research Institute (see Langerman, 1983) using RELAP5/MOD1 for calculating safety valve discharge piping hydrodynamic loads. The authors of this assessment study followed, modified, and added to the EPRI guidelines as follows:

EPRI Guidelines

1. The length of control volumes must be between 0.15 m and 0.3 m for a correct slug and pressure front tracking calculation.
2. The time step must be limited externally to the material Courant limit of approximately 0.2 m/s.
3. The no-choking option must be imposed at all the junctions downstream from the test valve.
4. Cold water loop seals (<373 K) should be located initially downstream from the test valve.

Modified EPRI Guideline

5. Heat transfer to pipe walls is not required for computing the hydrodynamic loads on the discharge piping due to water loop seal discharges.

New Guidelines

6. Include the orientation (horizontal or vertical) for pipes downstream of the valve.
7. Use the two-velocity option for the valve junction.

Base calculation: The modelling guidelines for RELAP5/MOD1 were used to develop the model for the base calculation. The model nodalization was that suggested by Langerman, 1982.

Sensitivity studies: Eight sensitivity studies were performed to investigate the variation in the transient loads and pressure with respect to code options considering (i) interfacial friction, (ii) junction horizontal flow stratification, (iii) abrupt area changes, (iv) heat slab modeling, (v) pipe orientation, and (vi) choke modeling below the valve.

Nodalization studies: None.

Summary: The assessment calculations described in the above report address the code's capability to calculate transient hydrodynamic loads on safety and relief valve discharge piping. The experimental data of a hot water loop seal discharge through a Crosby valve was used for comparison. This test is a part of the EPRI Safety and Relief Valve Test Program (see EPRI Valve Test Program Staff, 1982) conducted at Combustion-Engineering's Kreising Development in Windsor, Connecticut, USA.

The study objectives were:

1. Assess the potential of RELAP5/MOD3 to predict safety and relief valve discharge piping hydrodynamic loads.

2. Propose modeling guidelines for piping loads.

3. Highlight the effect the different physical models in RELAP5 version have on calculating piping loads.

Fig. 8.1.2.1 shows a schematic of the test facility. The mass in the loop seal is about 9 kg. With the valve initially closed, the temperature of the water slug varies from 623 K (the saturation temperature at tank pressure to 423 K. Upon opening the valve, the water loop seal discharge is followed by steam discharge. The elevated temperature and the sudden depressurization promote the formation of two-phase flow. The valve outlet piping was divided into four segments: horizontal, vertical downwards with an area increase at mid-length, long horizontal, and short vertical upwards discharging into the atmosphere. Hydrodynamic loads were measured for each segment. Extremely rigid supports were designed and pipe oscillations in the pipe section's perpendicular direction were restricted so the measured loads directly reflect the transient fluid loads. Other measurements include fluid temperature at the valve inlet, pressure at the valve discharge, and flow rate through the venturi.

The thermal-hydraulic phenomena addressed is rapid acceleration of a liquid slug in a loop seal, the subsequent steam discharge, and the eventual establishment of two-phase steady flow in the discharging piping. The liquid slug is accelerated by the source steam upon the rapid opening of a safety valve at the downstream end of the slug. Significant initial temperature gradients (623 K to 423 K over 2.74 m) in the liquid slug and rapid depressurization cause annular-mist flow to evolve in about 0.3 s. Two-phase choked discharge, interphase drag, heat transfer to pipe walls, and two-phase flow at abrupt expansions were considered.

Although the RELAP5 code does not calculate the hydrodynamic loads, no code changes or postprocessing has been performed to obtain the loads. Instead, the loads were computed by RELAP5's control blocks which were put together by Tractebel's preprocessor code TROPIC. The hydrodynamic load for a pipe segment was calculated by a momentum balance and the load was equated to the time rate of change of momentum of the fluid (liquid and vapor) inside the pipe segment. A blowdown force was added for the last segment that opens to the atmosphere.

A base case model and eight sensitivity cases were reported. The code options tested were: interface friction model, horizontal stratification at junctions, sudden expansion and contraction, heat slab modelling, vertical orientation of pipe segments, choking model downstream, and phase velocity option. Peak loads (positive and negative) for all the cases were tabulated for comparison with test data and with RELAP5/MOD1 results.

The nodalization used was basically the same as the one used in a previous study (Langerman, 1983) so no nodalization study was performed.

Figs. 8.1.2.2 and 8.1.2.3 show typical comparisons between the measurement and calculations for pressure and hydrodynamic loads. The calculations were performed using RELAP5/MOD1 (results provided by the Intermountain Technologies Inc. and Tractebel) and RELAP5/MOD3 Version 5M5 (result provided by Tractebel). In general, the pressure was underestimated by RELAP5/MOD3. However, the maximum peak load and to a lesser extent, the negative load were reasonably estimated by RELAP5/MOD3, although the computed peak loads were delayed. No temperature and mass flow rate comparisons were given.

The authors concluded that:

1. For liquid loop seal discharges the effect of heat transfer to pipe walls need not be modeled for a correct evaluation of liquid discharge loads.

2. RELAP5/MOD3 underestimates the coupling between the liquid and vapor phases, producing a lower liquid slug velocity than in the experiment. Although the maximum values for the loads are quite comparable to the measurements, the loads are delayed.

3. The inclusion of a transition zone between the subcooled and two-phase flow regimes produces a characteristic two-bump valve mass flow that is reflected on the loads of the downstream piping.

4. Some of the modelling guidelines established in a previous RELAP5/MOD1 assessment are modified and expanded as follows: (a) heat transfer to pipe walls needs not to be included for computing hydrodynamic loads for water loop seal discharges, (b) the orientation (horizontal or vertical) of the discharge piping should be included, (c) the two-velocity option for the valve junction should be used.

5. With all the modeling guidelines followed, the calculated "positive" forces agree with the measured values. A margin of 10 percent covers all experimental points. The calculated "negative" forces exceed the measured values except for one pipe segment where the measured forces exceed the calculated ones by 80 percent. This exception is probably due to the lower stiffness of supports for this pipe segment. With a suitable margin to estimate the negative forces, the RELAP5/MOD3 results are acceptable.

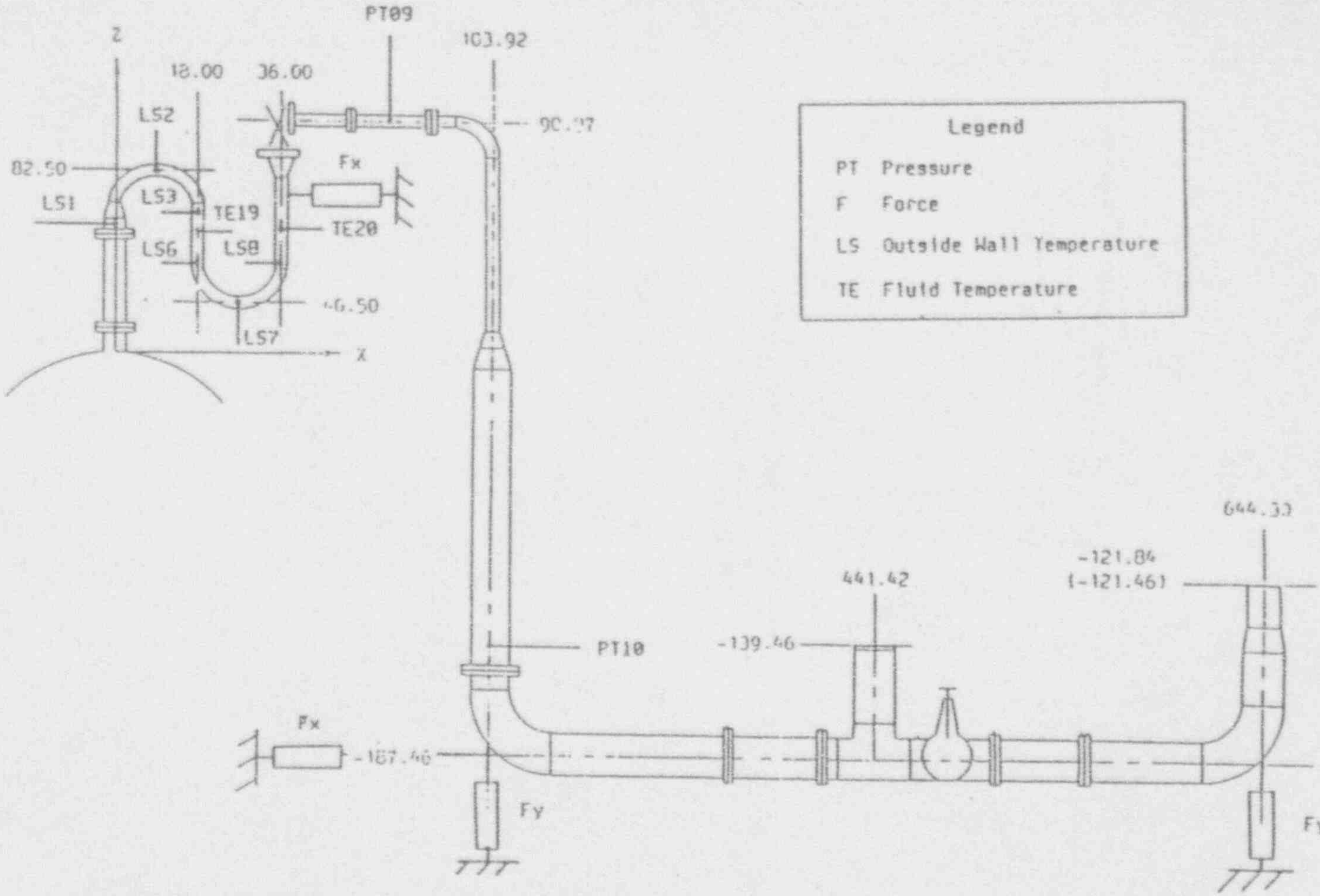


Figure 8.1.2.1 Dimensions and measurement locations for the Crosby valve tests.

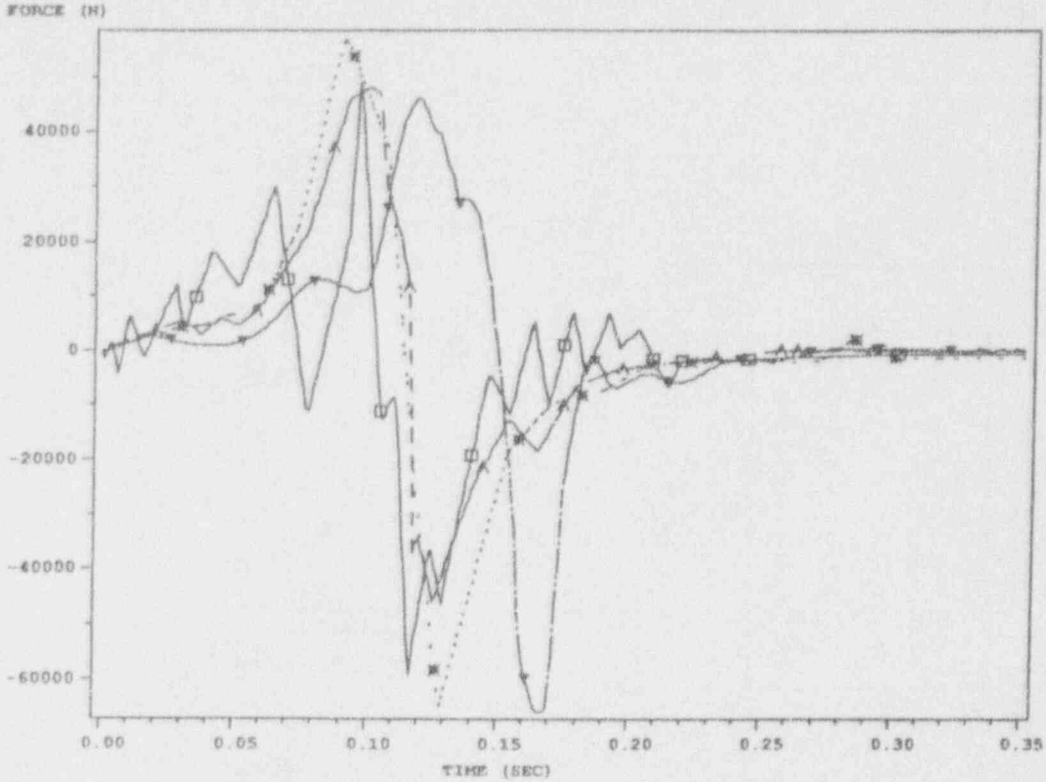


Figure 8.1.2.2 CE Test No. 917 Case 4 load on segement 2.

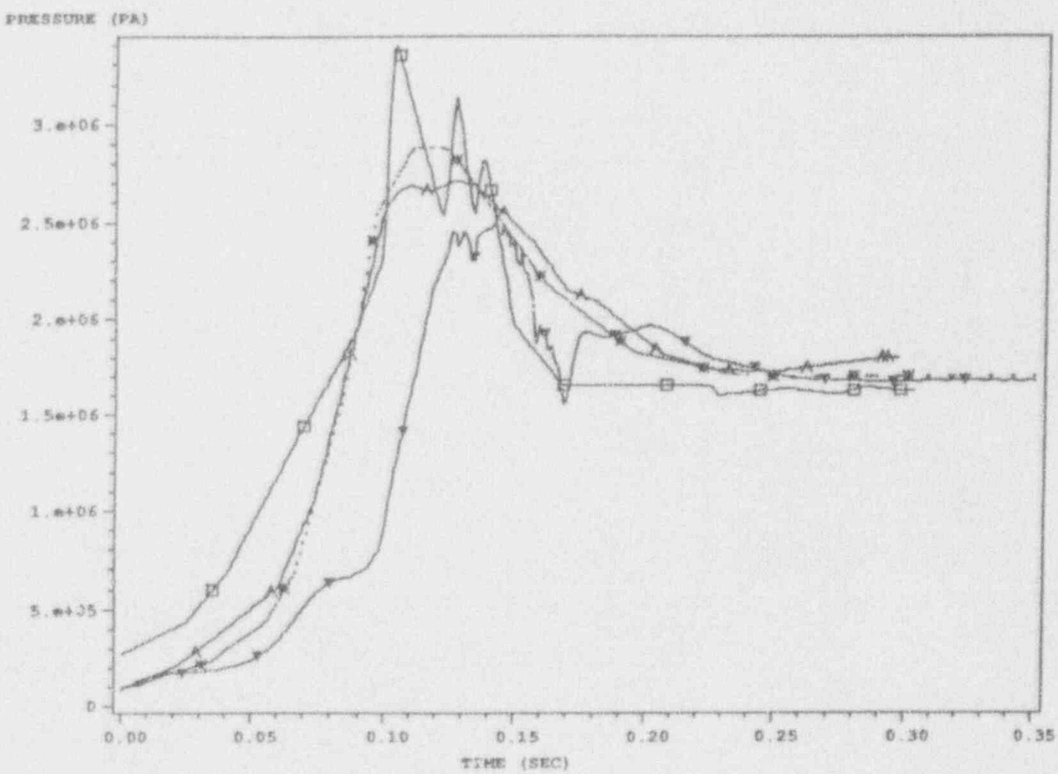


Figure 8.1.2.3 CE Test No. 917 Case 4 Pressure at PT09.

8.1.3 Downcomer Penetration Studies During LBLOCAs

Reference: K. Schneider, *Assessment of RELAP5/MOD3/V5M5 Against Upper Plenum Test Facility Test No. 6, Run 131, 132, 135, and 136 Downcomer Countercurrent Flow*, Siemens AG, April, 1991.

Code version: RELAP5/MOD3 Version 5M5.

Facility: Upper Plenum Test Facility (UPTF) located at Mannheim, Germany adjacent to the coal-fired power station Grosskraftwerk Mannheim which supplies the experimental steam.

Objectives: Assess the ability of RELAP5/MOD3 version 5M5 to model emergency core cooling system water downcomer penetration and lower plenum refill during end-of-blowdown and refill phases of a double-ended break in the cold leg of a typical PWR.

Major phenomena: The phenomena addressed were ECC bypass and ECC lower plenum penetration during the end-of-blowdown and refill phases of a LBLOCA.

In a LBLOCA in the cold leg of a PWR, the coolant in the primary system is rapidly expelled through the break. The pressure in the primary system decreases as this blowdown progresses. Slightly subcooled ECC is injected into the cold legs of the three intact loops when the primary system pressure decreases to a set value. At this time, the steam from the core moves upwards through the downcomer and out through the break.

The upflow of steam in the downcomer inhibits the flow of ECC into the downcomer. As a result, some or all of the ECC bypass the downcomer and discharge through the break directly. As the upflow of steam diminishes, the bypass diminishes accordingly and ECC is eventually fully delivered to the lower plenum.

Previous scale tests have shown that the ECC penetrates into the downcomer and reaches the lower plenum at a certain steam upflow rate before the blowdown ends. This steam/water countercurrent flow phenomenon was created at full scale in the UPTF for a range of steam flow rates. It was observed that during the initial ECC injection, only a small portion of the ECC entered the downcomer while filling of the cold legs took place. Among the three intact loops, the ECC accumulation was the greatest in the leg adjacent to the broken loop due to higher local steam flux. Strong heterogeneous and multi-dimensional flow exists in the downcomer. The ECC delivery to the lower plenum was intermittent.

Code deficiencies: The author identified, from the base case results, that the code overestimated the ECC bypass and, correspondingly, underestimated the lower plenum liquid inventory. The discrepancies were caused by an overprediction of liquid entrainment by the steam upflow in the downcomer. However, it is not clear that the author was using the "best" or ideal nodalization for simulating this problem. The reviewers believe that further nodalization studies are necessary.

Users guidelines: The author found significant sensitivity of the model results to nodalization. The author's and reviewer's investigations suggested the following user guidelines for this type of problem:

1. A split downcomer nodalization has to be applied in order to account for the strongly inhomogeneous flow distribution in the downcomer.
2. Loss coefficients of 500 for the azimuthal junctions between the two halves of the split downcomer should be used for both forward and reverse flow. The artificially large coefficients damp out oscillations but do not significantly affect the ECC bypass and ECC downcomer penetration.
3. An axial division of the lower plenum into two nodes with equal volume is required during the filing of the lower plenum.
4. The ECC injection ports should be modeled by branches instead of ECC mixer components.

<u>Base calculation</u>: The model was created by Siemens personnel and was used to perform calculations of UPTF Test No. 6, Runs 131, 132, 133, 135, and 136.

<u>Sensitivity studies</u>: Calculations were performed to examine the effect on the facility transient behavior of: (1) the rate of steam injection to the core simulator, (2) the frictional pressure drop for the azimuthal flow in the downcomer, (3) the modelling options for ECC injection port.

<u>Nodalization studies</u>: Studies were performed to identify the proper nodalization for the flow in the downcomer and in the lower plenum. This was necessitated by the fact that the flow is multidimensional at these locations.

<u>Summary</u>: The study assesses the ability of RELAP5/MOD3/5M5 to model the extent of ECC downcomer penetration and lower plenum refill during the end-of-blowdown and refill phases of a double ended break in the cold leg of a typical PWR.

In a LBLOCA in the cold leg of a PWR, the coolant in the primary system is rapidly expelled through the break. The pressure in the primary system decreases as the blowdown progresses. Slightly subcooled ECC was injected into the cold legs of the three intact loops when the primary system pressure decreases to a set value. During this time, the flow path of the steam is up the downcomer and out the break.

The steam upflow in the downcomer inhibits the downward flow of ECC into the downcomer. As a result, some or all of the ECC bypass the downcomer and discharge through the break directly. As the steam upflow diminishes, the bypass diminishes accordingly and ECC is eventually fully delivered to the lower plenum.

Previous scale tests have shown that the ECC penetrates into the downcomer and reaches the lower plenum at a certain steam upflow rate before the blowdown ends. This steam/water countercurrent flow behavior is the focus of this code assessment study.

The data used for this assessment came from the Upper Plenum Test Facility (UPTF) Test No. 6, Runs 136, 133, 132, 131, and 135. The facility was constructed and operated by SIEMENS-KWU, Germany. It is a full scale model of the cooling system of a four-loop 1300 MWe PWR. UPTF includes the reactor vessel, the downcomer, the lower plenum, the core simulator, the upper plenum, and the loop and steam generator simulators. The core simulator is a principal feature of the UPTF facility. It uses a steam-water injection system to create realistic flow distribution in the core region. Located in Mannheim, Germany, it uses the adjacent coal-fired power station Grosskraftwerk Mannheim as the source of steam.

Test No. 6 is a quasi-steady state test designed to investigate the ECC downcomer penetration and lower plenum refill behavior at different steam rates. Figure 8.1.3.1 shows the relevant portion of the test facility. The intact loops and the break valve on the broken hot leg were closed. Steam was injected into the core simulator and was forced to flow upwards through the downcomer and out through the break. ECC was then injected into the cold leg of the intact loops. Five test runs were made, each had a different but constant rate of steam injection to the core simulator.

During the initial ECC injection in the test, only a small portion of the ECC entered the downcomer while filling of the cold legs took place. Among the three intact loops, the ECC accumulation was the greatest in the leg adjacent to the broken loop because of the higher local steam flux. Strong heterogeneous and multidimensional flow existed in the downcomer. The ECC delivery to the lower plenum was intermittent.

The base case nodalization scheme is shown in figure 8.1.3.2. Two assumptions were used in establishing this nodalization: 1) not the entire facility, but only components relevant to Test No. 6 need to be nodalized, and 2) represent the downcomer by a "split downcomer" nodalization. The rationale behind this nodalization was that Test No. 6 was a separate effect test, and thus only a small number of components need detailed representation in the model. The "split downcomer" assumption was motivated by the multi-dimensional nature of the flow in the downcomer.

The base case results show that the RELAP5/MOD3 severely overestimated the ECC bypass, overestimated the liquid entrainment by the steam in the lower plenum, and underestimated the equilibrium level of liquid inventory in the lower plenum. The discrepancies were greater at higher steam injection rates.

To increase the downcomer penetration prediction, the author investigated the effects of: 1) frictional pressure drop for the azimuthal flow in the downcomer, 2) spatial partition of the two flow channels for the downcomer, 3) nodalization of the lower plenum, and 4) modeling options for ECC injection port. Combining the desirable features identified in the sensitivity studies, RELAP5/MOD3 simulated an additional case (case B). Much improved results were obtained. A total of 45 RELAP5/MOD3 simulations were made.

Figures 8.1.3.3 and 8.1.3.4 compare the history of lower plenum mass inventory between the base case, case B, and the experimental data. Figure 8.1.3.3 is for Run 136, which has the lowest steam injection rate. Figure 8.1.3.4 is for Run 135 where the steam injection rate is the highest. Figure 8.1.3.5 compares the average steam upflow versus ECC downflow plots between the base case and the experiment. Figure 8.1.3.6 shows the same comparison between case B and the experiment.

The authors concluded that:

1. Calculations using RELAP5/MOD3/5M5 showed significant sensitivity of the results to input modeling.
2. RELAP5/MOD3 can yield good agreement with the UPTF Test 6 experiment if proper nodalization scheme and suitable code options are used.
3. The key elements of the best modeling for countercurrent flow phenomenon in the downcomer are:

a. A split downcomer nodalization has to be applied.

b. In the azimuthal junctions of the downcomer, a value of 500 for the loss coefficient is appropriate.

c. The nodalization of the lower plenum is crucial if liquid is accumulated in that area. Although the nodalization of the lower plenum by only one node showed better agreement when the lower plenum was almost empty, an axial division of the lower plenum into two nodes is required if the filling of the lower plenum continues.

4. Calculations using the above guidelines agreed with the test results of UPTF Test No. 6. The delay of penetration by the filling of the loops was correctly predicted. The extent of ECC downcomer penetration in the early phase of refill is satisfactory. During the final phase of refill, the liquid inventory in the lower plenum is sufficiently correct.

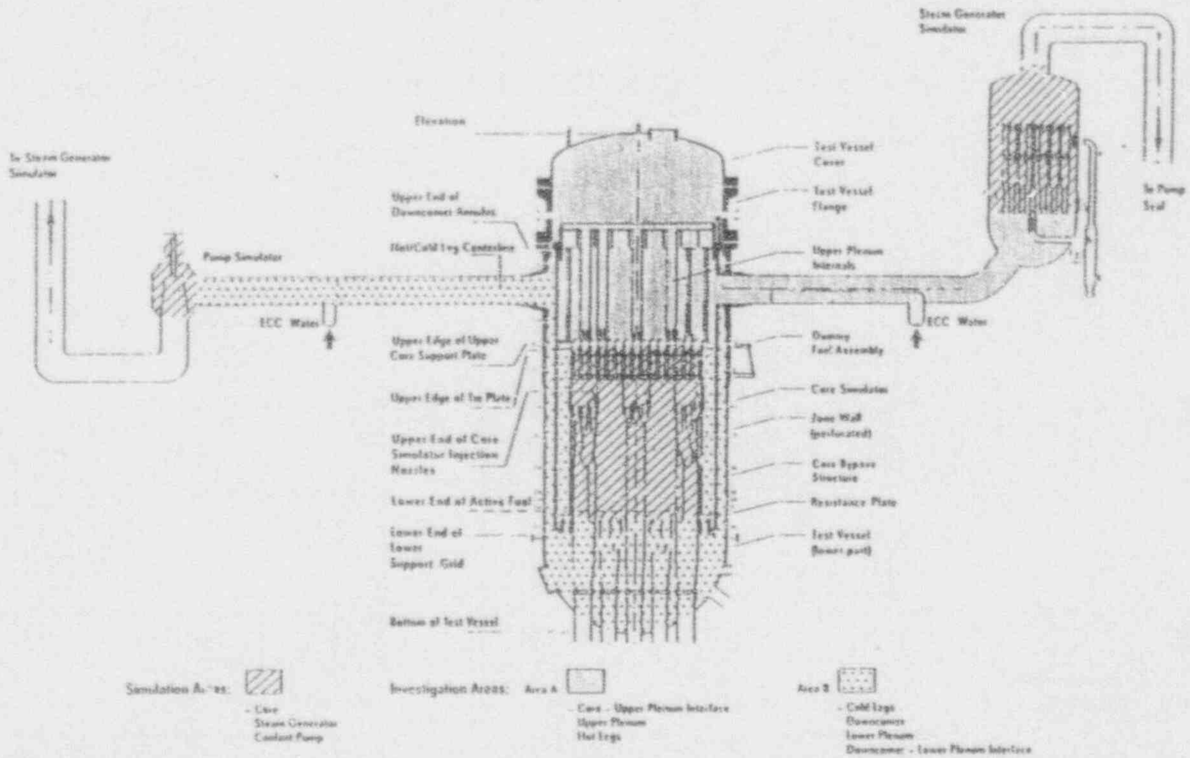


Figure 8.1.3.1 UPTF investigation and simulation areas.

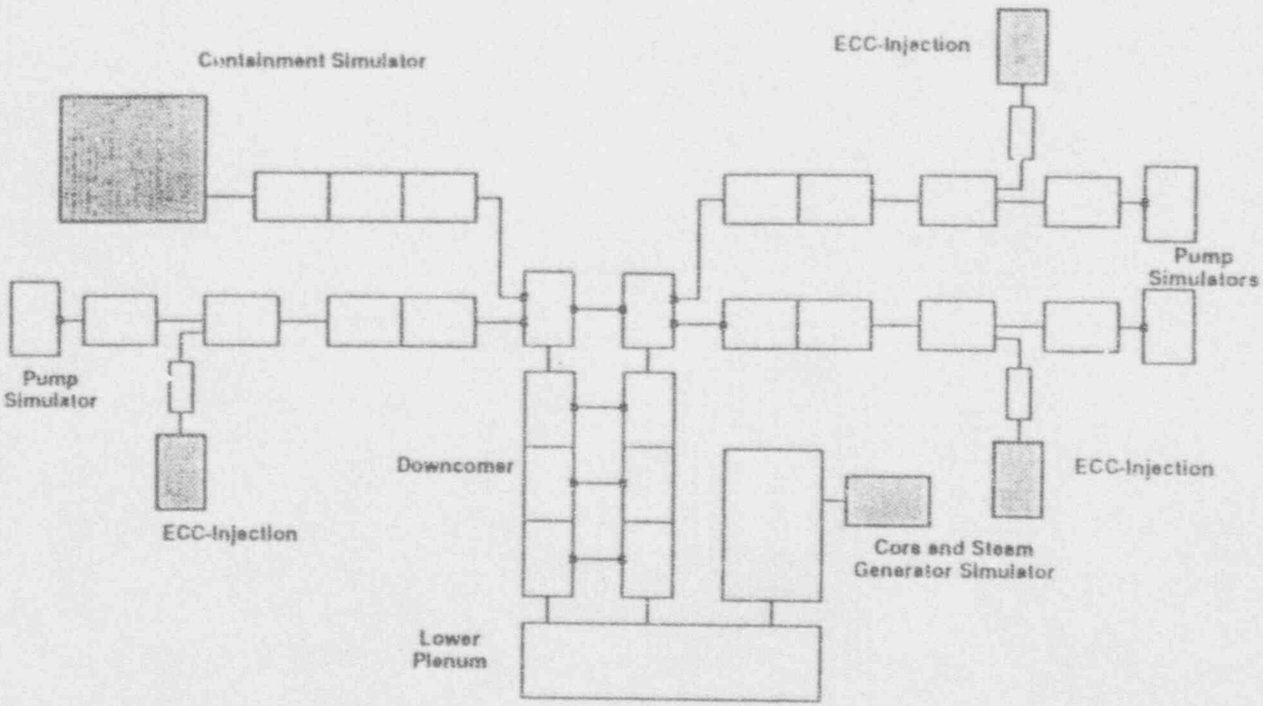


Figure 8.1.3.2 Nodalization diagram for UPTF Test 6 (Base case).

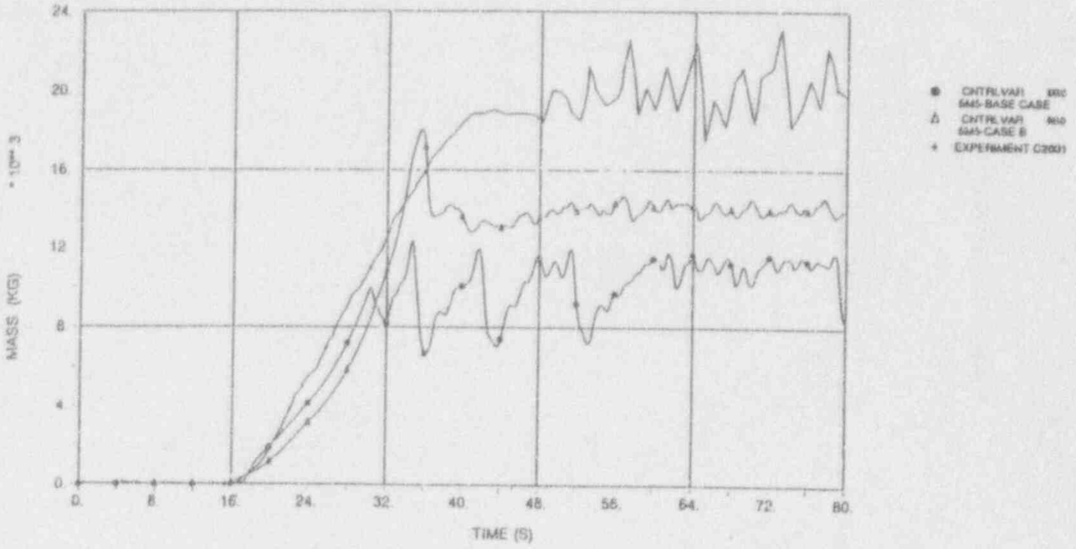


Figure 8.1.3.3 UPTF Test No.6 Run 136: Downcomer Countercurrent Flow Test Mass Inventory in Lower Plenum.

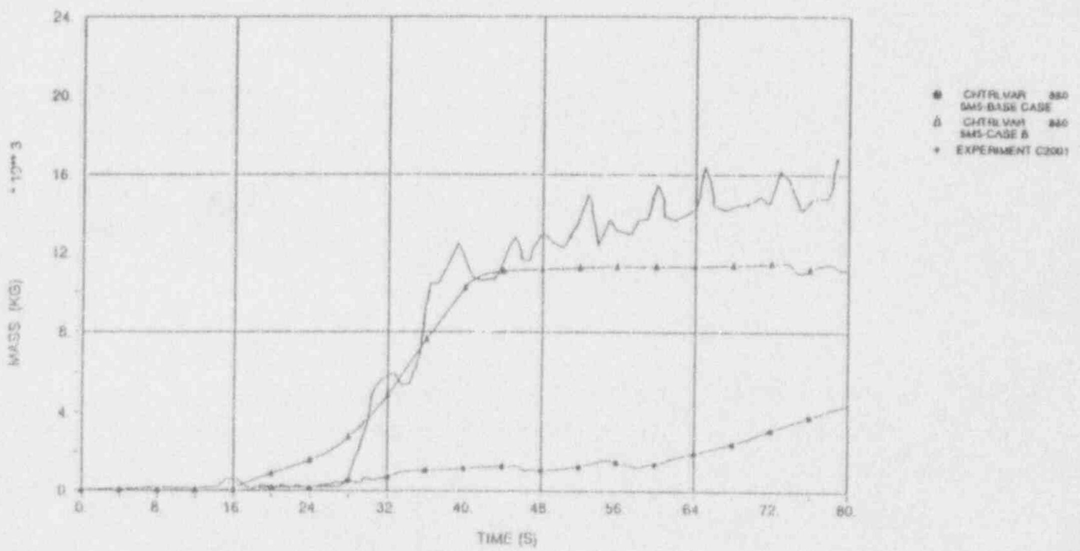


Figure 8.1.3.4 UPTF Test No.6 Run 135: Downcomer Countercurrent Flow Test Mass Inventory in Lower Plenum.

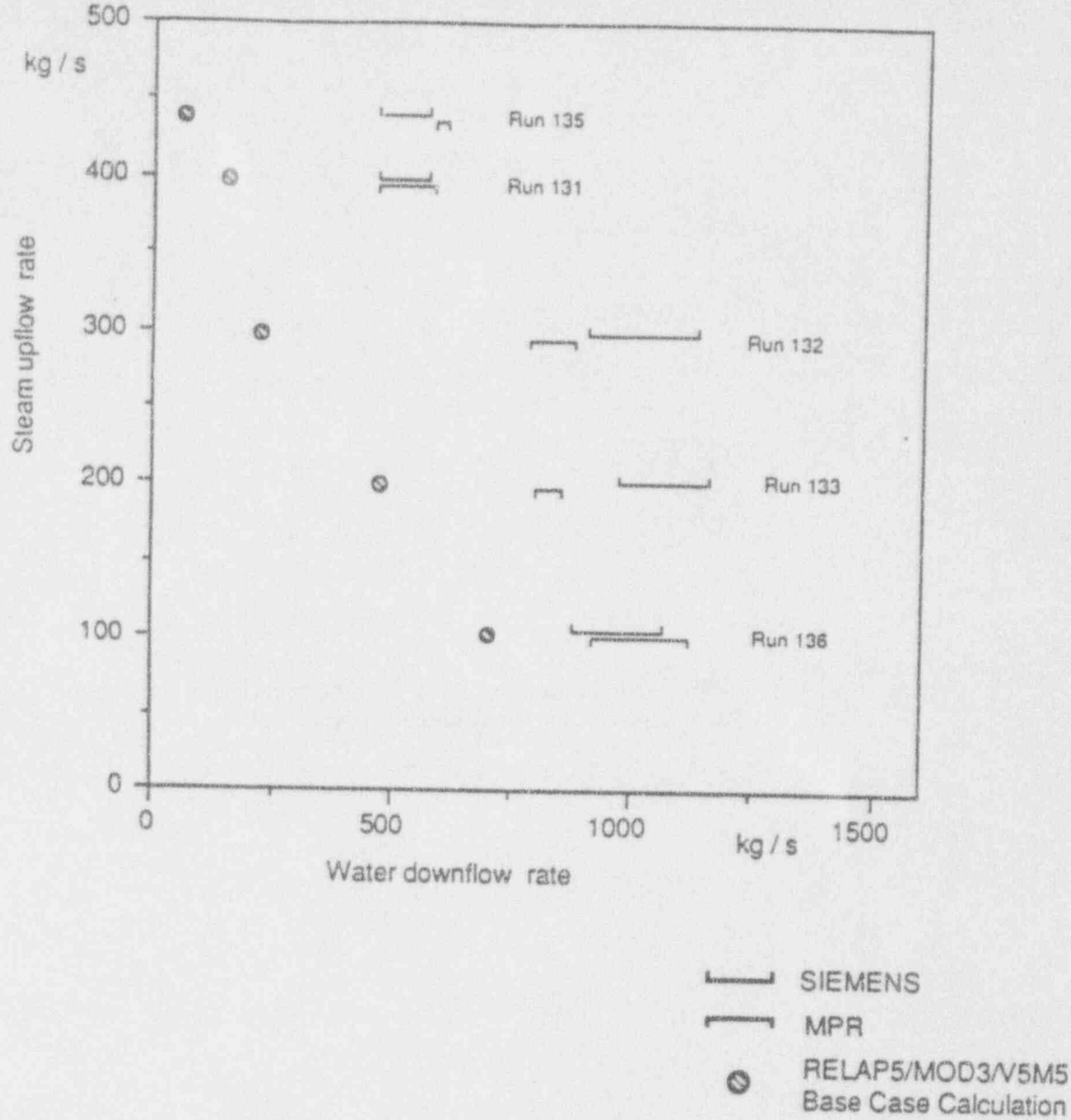


Figure 8.1.3.5 Results of the base case calculations: Average downflow rates in DC versus injected steam.

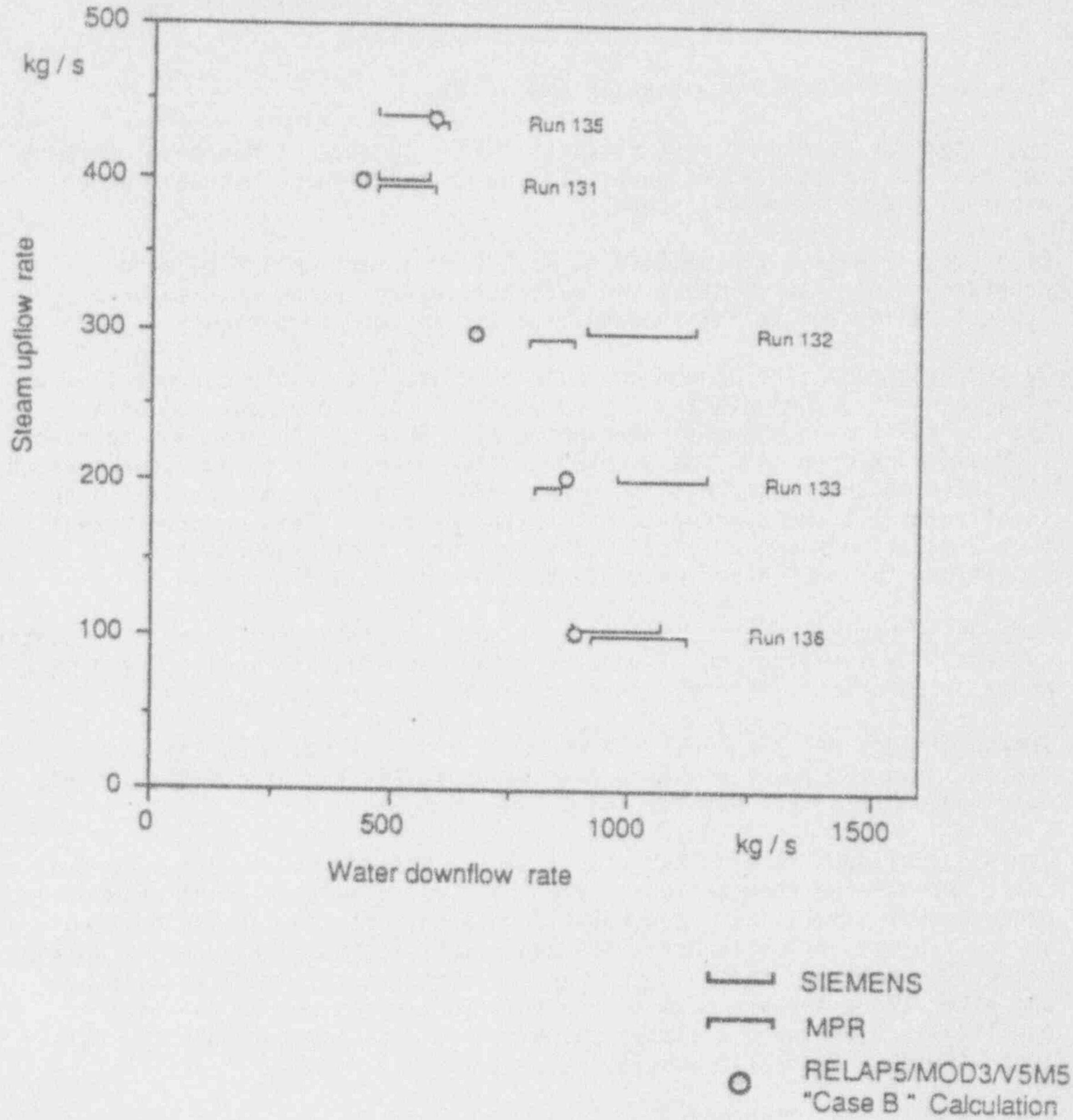


Figure 8.1.3.6 Results of "Case B" - calculations: Average downflow rates in DC versus injected steam.

8.1.4 Countercurrent Flow in PWR Hot Leg

Reference: F. Curca-Tivig, *Assessment of RELAP5/MOD3/V5M5 Against the UPTF Test No. 11 (Countercurrent Flow in PWR Hot Leg)*, KWU E412/91/E1002, Siemens AG, March, 1991.

Code version: RELAP5/MOD3 Version 5M5.

Facility: Upper Plenum Test Facility (UPTF) located at Mannheim, Germany adjacent to the coal-fired power station Grosskraftwerk Mannheim which supplies the experimental steam.

Objectives: Assess the ability of RELAP5/MOD3 version 5M5 to model countercurrent flow of steam and saturated water in the hot leg of a typical PWR during reflux condensation and reflood conditions.

Major phenomena: The phenomenon addressed was the countercurrent flow of steam generator condensate and core steam in the broken hot leg of a typical PWR during the boil-down phase of a SBLOCA. The condensate tends to drain back down into the pressure vessel. The core steam flows through the steam generator and out the break. The steam flow may partially or totally inhibit the condensate return to the core. This countercurrent flow limitation was addressed at two pressure levels representing conditions for SBLOCA reflux condensation and reflood.

Code deficiencies: Unphysical results were obtained when using the counter current flow limiting (CCFL) model for the 1.5 MPa test series. The CCFL model is deficient. (A programming error may be present.)

Impact of code deficiencies: If the CCFL model is not used the code will overestimate the water downflow rate by up to 35 percent for the 1.5 MPa data and up to 43 percent for the 0.3 MPa data.

Users guidelines: The author stated that a hot leg model with 9 control cells between the reactor vessel and the steam generator inlet chamber is adequate for simulating steam-water countercurrent flow in the hot leg during typical reflux condensation conditions if the CCFL model is working properly. However, the evidence for recommending this nodalization over the other three appears weak to the reviewers since one of the other nodalizations produced a closer match to the data (even though the cell length to diameter ratio was less than 1.)

Base calculation: The model was created by Siemens personnel and was used to perform calculations of UPTF Test No. 11, Runs 30 to 45.

Sensitivity studies: Calculations were performed to examine the effect on the facility transient behavior of: (1) the system pressure level, (2) using the CCFL option at the middle of the hot leg riser, (3) the flooding curve slope when the CCFL model was activated, and (4) the abrupt area change model for the Hutze to pipe transition.

Nodalization studies: Four nodalization studies were performed to evaluate the number of cells that should be used to model the system hot leg.

Summary: This report assesses the ability of RELAP5/MOD3/5M5 to model countercurrent flow of steam and saturated water in the hot leg of a typical PWR. Countercurrent flow of steam generator condensate and core steam in the broken hot leg takes place during the boil-down phase of a

SBLOCA. The condensate tends to drain back down into the pressure vessel. The core steam flows through the steam generator and out the break. The steam flow may partially or totally inhibit the condensate return to the core. This countercurrent flow limitation was addressed at two pressure levels representing conditions for SBLOCA reflux condensation and reflood.

The data used for this assessment came from the Upper Plenum Test Facility (UPTF) Test No 11, Runs 30 to 45 (16 runs total). The facility was constructed and operated by SIEMENS-KWU, Germany. It is a full scale model of the cooling system of a four-loop 1300 MWe PWR. UPTF includes the reactor vessel, the downcomer, the lower plenum, the core simulator, the upper plenum, and the loop and steam generator simulators. The core simulator is a principal feature of the UPTF facility. It uses a steam-water injection system to create realistic flow distribution in the core region. Fig. 8.1.4.1 shows the main components and dimensions of the test facility.

Test No. 11 was a quasi-steady state, separate effect test. It was designed to investigate the conditions for the countercurrent flow in the hot leg. Saturated water was fed into the inlet plenum of the UPTF water separator which simulates the steam generator in the broken loop hot leg. The injected water tends to drain down to the core. Saturated steam at various flow rates was introduced via the core simulator system. All vent valves, pump simulators, and all emergency core coolant system valves were closed so that the injected steam was forced to flow opposite to the injected water in the hot leg and out through the break. Two test series were conducted. During the 1.5 MPa test series, the hot leg break valve was partially open and a bypass valve was used to maintain the system pressure. During the 0.3 MPa test series, the break valve was fully open and the containment simulator pressure was kept at 0.3 MPa.

Figure 8.1.4.2 shows the base case nodalization. RELAP5/MOD3 simulations were made to establish the result sensitivity with respect to: 1) system pressure level (0.3 MPa versus 1.5 MPa), 2) imposition of the CCFL option at the middle of the hot leg riser, 3) the slope of the flooding curve when the CCFL option was used, and 4) the Abrupt Area Change Model for the Hutze to pipe transition. Figure 8.1.4.3 compares the predicted flooding curves with the experimental ones.

The authors concluded that:

1. Without using the new CCFL model, RELAP5/MOD3/5M5 overestimated the mass rate of water downflow by as much as 35 percent (1.5 MPa runs) and 43 percent (0.3 MPa runs). At complete liquid carry over and for the 1.5 MPa runs, RELAP5 predicted a steam mass flow rate of 46 kg/s as compared with 40.2 kg/s in the experiment. For the 0.3 MPa runs, RELAP5 predicted steam flow rate of 21.3 kg/s at complete liquid carry over while the experimental value was 20.5 kg/s.
2. Very good agreement with the 1.5 MPa experimental data (which are relevant for SBLOCA conditions) could be obtained by using the code's new CCFL option at a junction in the middle of the inclined part (riser) of the hot leg. The flooding correlation used was of the Wallis type with an intercept C of 0.664 and a slope m of 1. With this CCFL correlation, the 40.2 kg/s steam mass flow rate at complete liquid carry over was calculated by the code.
3. Using the same CCFL correlation for the simulation of the 0.3 MPa test series, which is typical for reflood conditions, RELAP5/MOD3 underestimated the steam mass flow rate by 44 percent at complete

liquid carry over.

4. A hot leg model with 9 control volumes between the reactor vessel and the steam generator inlet chamber is adequate for simulating steam-water countercurrent flow in the hot leg during typical reflux condensation conditions. However, the evidence for recommending this nodalization over the other three appears weak to the reviewers since one of the other nodalizations produced a closer match to the data (even though the cell length to diameter ratio was less than 1.).
5. An unphysical result was obtained when simulating the 1.5 MPa test series (runs 36 - 45 of the UPTF Test No. 11) using a CCFL correlation of the Wallis type with an intercept C of 0.644 and a slope m of 0.8. A programming error in the CCFL model of the RELAP5/MOD3/5M5 was suspected.

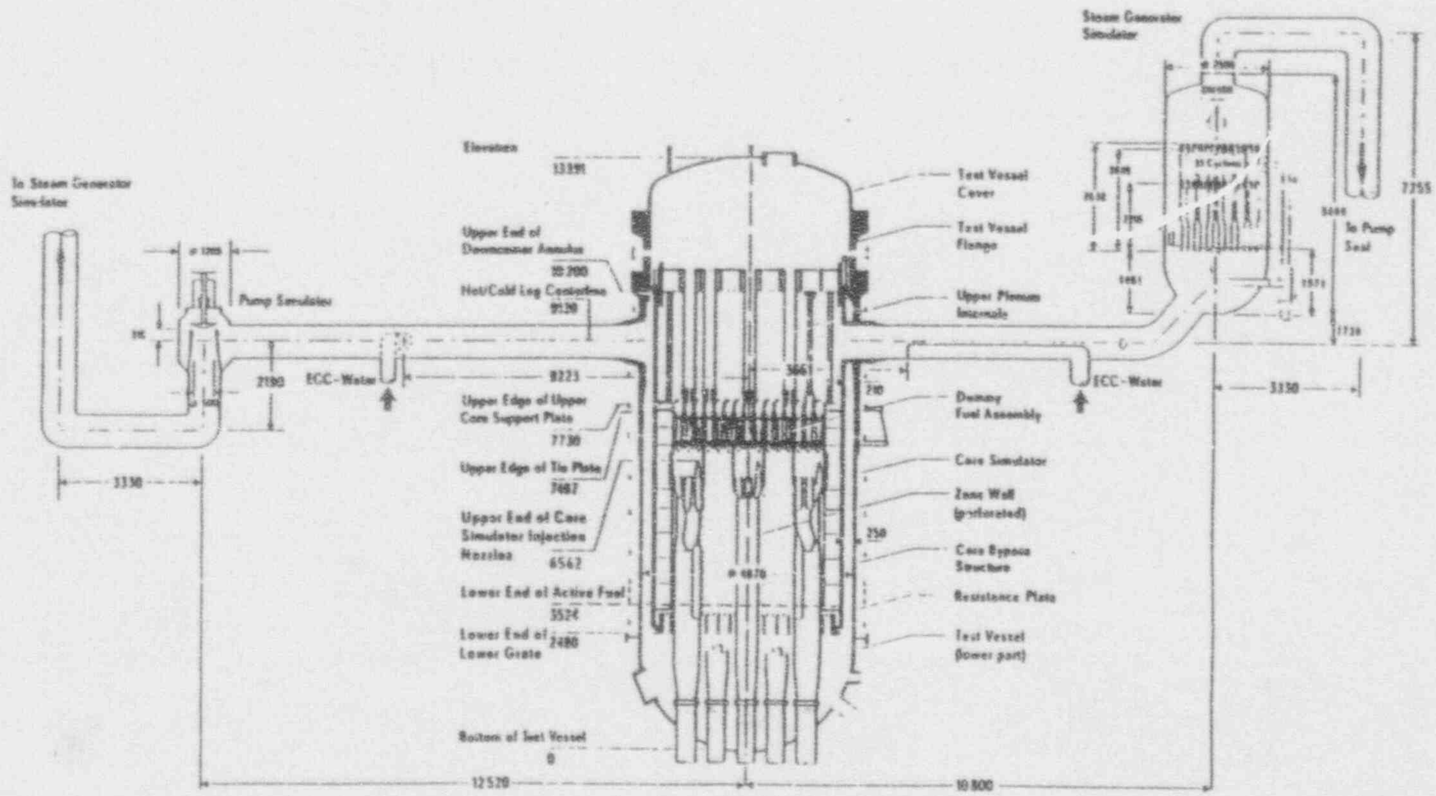


Figure 8.1.4.1 Major dimensions of UPTF-Primary System.

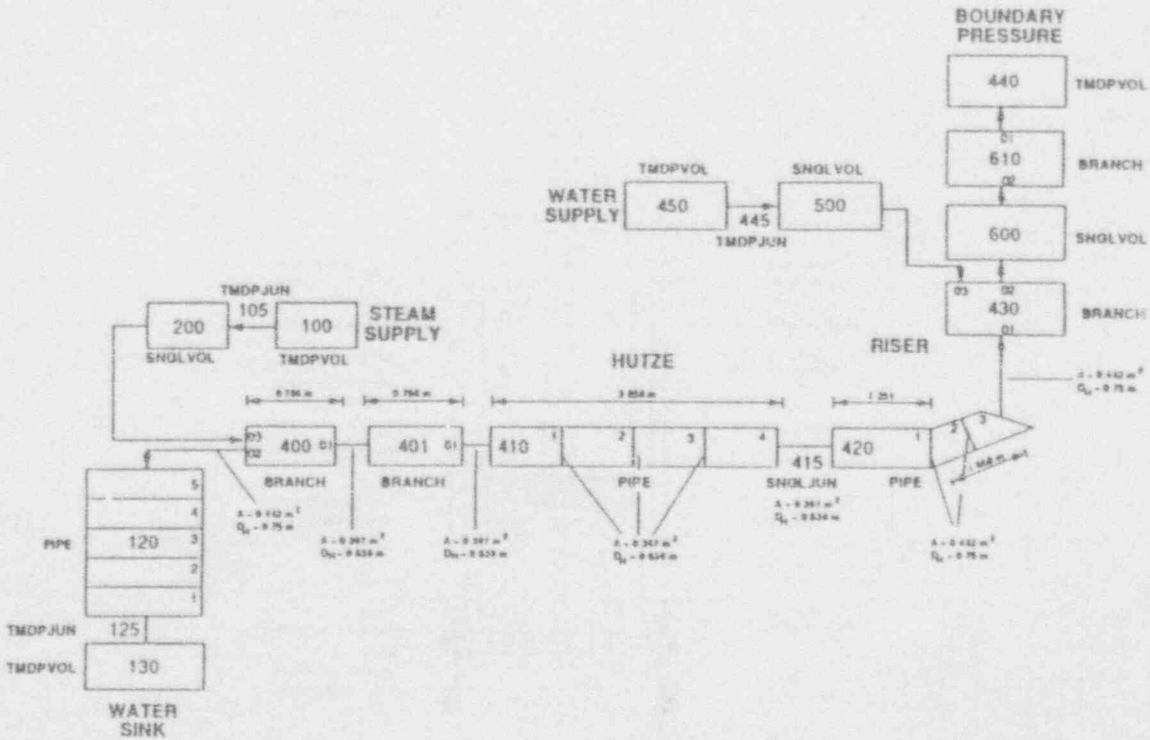


Figure 8.1.4.2 RELAP5 nodalization 4 (Basic Nodalization) for UPTF Test No. 11.

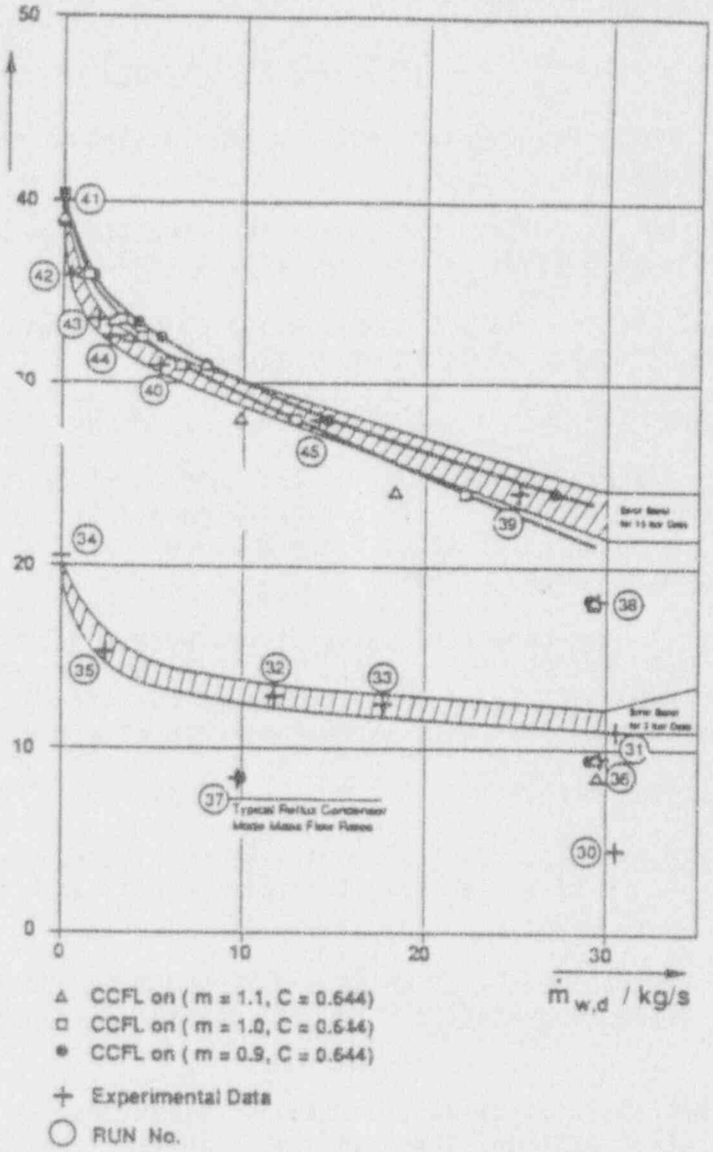


Figure 8.1.4.3 Experimental and predicted flooding curves for 1.5 MPa (Nodalization No. 4) study on code sensitivity to the slope m of the CCFL correlation.

8.1.5 Assessment of Direct Contact Condensation

Reference: S. Lee and H. J. Kim, *RELAP5 Assessment on Direct-Contact Condensation in Horizontal Cocurrent Stratified Flow*, NUREG/IA-0077, April, 1992.

Code version: RELAP5/MOD2 Cycle 36.04 and RELAP5/MOD3 version 5M5.

Facility: Horizontal rectangular test channel located at Northwestern University, Chicago, IL.

Objectives: Assess the code's capability to calculate the proper condensation rate on a liquid stratified flow interface.

Major phenomena: Direct-contact condensation on the liquid-steam interface between horizontal cocurrent steam-water flow.

Code deficiencies: The code usually undercalculated the liquid fluid depth.

User guidelines: A coarse nodalization was sufficient to calculate the experimental condensation rate. The original nodalization with 10 nodes, each representing a length of 16 cm, gave the same results as a more detailed nodalization with 20 nodes.

Base calculation: Four base-case calculations were performed, based on runs 253, 259, 279, and 293, using RELAP5/MOD2/36.04 and RELAP5/MOD3 version 5M5. The parametric studies focused on the effect of various water flow/steam flow combinations with a constant channel water level.

Sensitivity studies: None.

Nodalization studies: The original nodalization with 10 nodes, each representing a length of 16 cm, gave the same results as a more detailed nodalization with 20 nodes.

Summary: Both RELAP5/MOD2 Cycle 36.04 and RELAP5/MOD3 version 5M5 were assessed using steam condensation rate data generated at Northwestern University.

The experimental facility was composed of a rectangular channel that represented the test section, steam and water inlet plena, and a water tank. The water line was a closed loop while the steam line was built to provide steam to the test section. The channel was 1.6 m long, 0.3 m wide, and was 0.06 m deep. Uniform flow was assured by constructing large plena that assured low plenum velocities. The tests were performed at atmospheric pressure with steam flow rates ranging from 0.04 kg/s to 0.16 kg/s, water flow rates ranging from 0.2 kg/s to 1.45 kg/s, and water inlet temperatures ranging from 25 C to 50 C. The injected steam was slightly superheated. The condensation data was obtained by measuring the water flow rate at incremental positions along the channel length.

The test section was nodalized by using a PIPE with 10 cells (each 16 cm long). The code calculations of the condensation rates were in reasonable agreement with the data. However, differences were observed between the calculated channel water depth and the local heat transfer coefficient particularly for cases with a wavy interface. A nodalization study was conducted by increasing the test section cell from 10 to 20. No difference in the calculated condensation rates were observed.

8.1.6 Assessment of Counter Current Flow Limiting

Reference: S. Cho, N. Arne, B. D. Chung, and H. J. Kim, *Assessment of CCFL Model of RELAP5/MOD3 Against Simple Vertical Tubes and Rod Bundle Tests*, NUREG/IA-0192, (to be published).

Code version: RELAP5/MOD3 version 5M5.

Facility: Vertical tube and rod bundle flow experiment at Korea Atomic Energy Research Institute (KAERI) at Taejon, Korea.

Objectives: Evaluate the code's ability to model counter current flow limiting phenomena (CCFL) in tubes and rod bundles.

Major phenomena: CCFL phenomena including flooding characteristics, the onset of liquid mixing, and two-phase pressure drop when CCFL is present.

Code deficiencies: The code's CCFL model logic will allow liquid downflow after the gas upflow velocity has exceeded the flooding point. This deficiency may be due to a coding error. Other deficiencies suggested by the authors are still being investigated.

User guidelines: None.

Base calculation: The models used to simulate the experiments were constructed by KAERI personnel. Base case calculations were performed both with CCFL options and without CCFL options for the simple vertical tubes and the rod bundle tests.

Sensitivity studies: Calculations were performed for the simple vertical tube geometry by varying the tube inside diameter and observing the different flooding characteristics.

Nodalization studies: The effect of finer nodalization on calculating CCFL was investigated.

Summary: The study is an assessment of the code's CCFL model performed by comparing its results to experimental data from a simple vertical tube test and rod bundle tests conducted at KAERI.

The experimental facility is composed of a test section, water and air supply system, and measurement system. The test section consists of a lower plenum, a channel, and an upper plenum. The 3x3 tube array in the channel has the same geometrical dimensions of the typical 17x17 PWR fuel bundle. The experimental data was taken from the experiments of the simple vertical tubes and 3x3 rod bundle test section; type-1 had no spacer grid, type-2 had one spacer grid, and type-3 had two spacer grids.

The base case calculations were performed for a simple vertical tube and rod bundle tests. A pair of calculations were performed with and without the CCFL option for each experiment. The calculational results with the CCFL option for the vertical tubes gave reasonable agreement with the data. The code calculated flooding for the rod bundles but did not match the data. Investigation of the differences between the calculation and the data is continuing.

The sensitivity calculations, conducted for different tube diameters without the CCFL model invoked, showed flooding to occur before CCFL was observed in the experiment for small diameter tubes, but showed flooding to occur after CCFL was observed in the experiment for large diameter tubes. When the CCFL model was invoked, the calculation showed a reasonable match to the data. However, when using the RELAP5 CCFL model to simulate a Kutateladze characteristic unphysical results were obtained: liquid downflow was allowed when the upward gas superficial velocity exceeded the flooding point.

The nodalization studies showed that coarse nodalization simulated the CCFL phenomena as well as finer nodalization.

8.2 INTEGRAL EFFECTS ASSESSMENTS

Integral effects assessments were completed for four SBLOCAs and a number of natural circulation experimental conditions. Two of the SBLOCA assessments were performed using BETHSY data, one using Semiscale data, and one using the Large Scale Test Facility. The natural circulation assessment was performed using BETHSY data.

8.2.1 Large Scale Test Facility (LSTF) 5% Cold Leg SBLOCA Experiment

Reference: S. Lee, B. D. Chung, and H. J. Kim, *RELAP5 Assessment Using LSTF Test Data SB-CL-18*, NUREG/IA-00095 (to be published).

Code version: RELAP5/MOD3 version 5M5

Facility: The ROSA-IV Program's Large Scale Test Facility (LSTF), Tokai, Japan.

Objectives: Evaluate the code's ability to simulate the important phenomena occurring during a SBLOCA such as break critical flow, loop seal clearing and core uncovery, and core heatup.

Major phenomena: Critical flow, countercurrent flow limiting (CCFL), loop seal clearing and core uncovery, core heatup, stratified two-phase in the horizontal legs, vessel inventory boiloff, and vessel refill due to accumulator injection.

Code deficiencies: The two-phase critical flow rate was undercalculated prior to loop seal clearing while the single-phase steam critical flow rate was overcalculated after loop seal clearing.

Impact of code deficiencies: Timing of the calculated events was shifted from the measured event times.

User guidelines: By doubling the number of loop seal cells from nine to eighteen and the number of primary steam generator U-tube cells from eight to sixteen the calculation showed better agreement with the data.

Base calculation: The base calculation was performed using the nodalization recommended by the INEL. Deficiencies were detected in the performance of the code's critical flow model.

Sensitivity studies: None.

Nodalization studies: The affect of increasing the number of cells in the volumes representing the loop seals and the steam generator U-tubes was briefly examined by increasing the number of cells by a factor of two. The new calculational results showed better agreement with the data.

Summary: The code was assessed using the experimental data obtained during the SB-CL-18 experiment conducted in the LSTF. The experiment was conducted to investigate the thermal-hydraulic mechanisms responsible for the early core uncovery, including the manometric effect due to an asymmetric coolant holdup in the steam generator upflow and downflow side during the 5% cold leg small break loss-of-coolant accident (SBLOCA). The simulation capability of the code of the phenomena occurring during the SBLOCA is the subject of the report.

The LSTF is a 1/48 volumetrically scaled nonnuclear model of a Westinghouse type 3423 MWt four loop PWR. The facility is designed to simulate SBLOCAs (up to 10%) and operational transients at the same high pressures and temperatures as the reference PWR. The LSTF has two equally sized loops that differ only in the possible break geometries and in the presence of a pressurizer in one of the loops. The 1064 electrically-heated rods and the 104 unheated rods are used to simulate the 17x17 fuel assembly of the PWR core. The design scaling compromise is the 10 MW maximum core power limitation, 14% of the scaled reference PWR rated power. Each steam generator (SG) with 141 full-sized U-tubes in a scaled secondary volume is designed considering the steady-state flow to be 14% of the scaled reference PWR SG flow.

The baseline calculations show good agreement with the experimental data in predicting thermal-hydraulic phenomena. The authors, however, point out several differences regarding the evolution of phenomena and affecting the timing order. Specific deficiencies noted by the authors are as follows:

1. The calculated break flow rates show some discrepancy with experimental data in RELAP5/MOD3 version 5M5. Underestimation of the two-phase break flow resulted in an insufficient mass discharge from the primary system prior to the loop seal clearing. Overpredicted vapor phase break flow caused a fast primary mass loss and an earlier accumulator injection following loop seal clearing.

2. The code did not calculate complete loop seal clearing.

A nodalization study, performed to evaluate the effect of doubling the number of cell volumes representing the loop seals and the steam generator U-tubes, produced better agreement with the data.

In conclusion, the code can predict the major phenomena occurring during a 5% cold leg break LOCA although some deficiencies in predicting the break flow and loop seal clearing were noted.

8.2.2 BETHSY Natural Circulation Assessment

Reference: P. A. Roth and R. R. Schultz, *Analysis of Reduced Primary and Secondary Coolant Level Experiments in the BETHSY Facility Using RELAP5/MOD3*, EGG-EAST-9251, July, 1991.

Code version: RELAP5/MOD3 Version 5M5

Facility: Boucle d'Etudes THermohydrouliques SYsteme (BETHSY) facility located at the Centre d'Etudes Nucleares de Grenoble (CENG) in Grenoble, France.

Objectives: (i) Describe the BETHSY facility single-phase natural circulation, two-phase natural circulation, and reflux condensation characteristics and (ii) assess the RELAP5/MOD3 code using these data.

Major phenomena: Single-phase natural circulation, two-phase natural circulation, and reflux.

Code deficiencies: None.

User guidelines: Through a series of nodalization studies, the following user guidelines were identified:

1. Doubling the number of nodes representing the steam generator (SG) U-tubes from the base case value of 8 to 16 did not produce any change in the model's ability to predict the onset of reflux at system power levels ranging from 1 to 5 percent of rated scaled power.
2. Core bypass regions should be nodalized to have a complementary nodalization structure to that of the core region.
3. If an experiment or plant secondary system condition considering a low secondary liquid level must be simulated, a fine nodalization must be used to accurately capture the primary-to-secondary energy transfer.

Base calculation: The base calculations were performed using the nodalization originally specified by the INEL and thereafter modified by considering the output of a number of nodalization studies. Although a number of nodalization studies were performed, only one base calculation was performed for Test 4.1a-TC and Test 5.1a.

Sensitivity studies: None.

Nodalization studies: Nodalization studies were performed to investigate the best way to nodalize the SG U-tubes, the vessel lower plenum, the vessel upper plenum, the pump (BETHSY has a pump configuration unique to the Framatome PWRs),the SG secondaries, and the core bypass region.

Summary: This report documents the assessment of RELAP5/MOD3 Version 5M5 using the data from BETHSY facility experiments 4.1a-TC and 5.1a. These experiments were designed to study the phenomena that occur during single-phase natural circulation, two-phase natural circulation, and reflux at various secondary conditions. Single-phase natural circulation was studied as a function of the primary mass inventory level, the secondary mass inventory level and the core power. Two-phase natural circulation and reflux condensation were studied as a function of the primary and secondary mass inventory levels.

The BETHSY facility is a 1/100 volume-scaled simulator of a 2775 MWt 3 loop

Framatome pressurized water reactor (PWR). The BETHSY facility has 3 identical loops, with the exception of a pressurizer being mounted in one loop, and has the same component heights as it's Framatome counterpart.

The objectives of the Test 4.1a-TC and 5.1a analysis effort were to (i) describe the BETHSY facility single-phase natural circulation, two-phase natural circulation, and reflux condensation characteristics and (ii) assess the RELAP5/MOD3 code using these data. Objective (i) was reached by correlating the single-phase natural circulation, two-phase natural circulation, and reflux condensation data with primary and secondary inventory levels. These data were readily available since Test 4.1a-TC Parts 1 and 2 were conducted by maintaining the secondary liquid level at a constant value (Part 1: rated secondary level and Part 2: secondary level 1 m above the tube sheet) while draining the primary inventory incrementally; and Test 5.1a Part 1 was conducted by maintaining the pressurizer liquid level at a predetermined value while draining the secondary liquid level incrementally. As a reference, Test 5.1a Part 2 was conducted by maintaining the primary mass flow rate at a constant value using the reactor coolant pumps while draining the secondary liquid level.

Objective (ii) was reached by constructing a RELAP5/MOD3 model of the BETHSY facility and performing calculations of the Tests 4.1a-TC and 5.1a using the initial and boundary conditions defined by the BETHSY experimentalists. It should be noted that since these calculations were done prior to and just when the RELAP5/MOD3 code was being released, the BETHSY model nodalization was changed a number of times prior to reaching a final nodalization.

Following the data analysis, the RELAP5/MOD3 code assessment was conducted. Tests 4.1a-TC and 5.1a were simulated using fundamentally the same procedure used by the experimentalists. However, the drain periods and steady-state periods were in general shorter than used in the experiments.

RELAP5/MOD3 was shown to reasonably simulate single-phase natural circulation, two-phase natural circulation, and reflux condensation. The calculated natural circulation primary mass flow rate was consistently greater than the measured values for single-phase natural circulation and two-phase natural circulation at or near the peak mass flow rates in Test 4.1a-TC and Test 5.1 - Part 1. Such a difference can be easily explained by the additional frictional pressure loss sustained in the experiment by the bulk primary mass flow, together with the postulated reverse flow contribution, moving through a reduced number of SG U-tubes.

The minimum calculated two-phase primary natural circulation mass flow rates matched those of the data very closely. The mass inventory level at which reflux condensation began was about 50 % both in the code calculation and in the experimental data.

The code was found to be able to calculate the CCFL that occurred during the latter portion of Test 4.1a. Unfortunately the code calculated a considerable oscillation in the vapor velocity which caused CCFL to cease periodically and prevented a buildup of mass in the SG U-tube's upflow sides.

The code's capability to calculate secondary riser mass distribution was found reasonable for both parts of Test 5.1a. In addition, the code gave a reasonable calculation of the secondary inventory degree of superheat for these tests.

8.2.3 BETHSY SBLOCA Assessment

Reference: P. A. Roth, C. J. Choi, and R. R. Schultz, *Analysis of Two Small Break Loss-of-Coolant Experiments in the BETHSY Facility Using RELAP5/MOD3*, EGG-NE-10353, July, 1992.

Code version: RELAP5/MOD3 Versions 7o and 7q.

Facility: Boucle d'Etudes THermohydrouliques SYsteme (BETHSY) facility located at the Centre d'Etudes Nucleares de Grenoble (CENG) in Grenoble, France.

Objectives: (i) Gain greater understanding of the phenomena which occur during a small break loss-of-coolant accident, and (ii) assess the RELAP5/MOD3 version 7 code using these data.

Major phenomena: System depressurization rate, subcooled and saturated break flow, core heat transfer as indicated by the measured cladding temperature, loop seal clearing, core boiloff, vapor pull-through/entrainment simulation.

Code deficiencies: Three code deficiencies were identified:

1. The counter-current flow limiting (CCFL) model was found to contain an error.
2. The ECCMIX component was found to calculate excessive condensation compared to the data for the 0.5 % SBLOCA experiment. Also use of this model caused the code to fail.
3. Under some conditions, the limits of which have not been rigorously defined, the code will not realistically calculate the process of draining a vertical pipe; voids are calculated to pass into lower cells before the upper cells are fully drained.

User Guidelines: Two user guidelines were identified:

1. To more accurately calculate the process of draining a vertical pipe, vertical regions such as the loop seals should be nodalized to 6 to 8 cells in the vertical region upstream of the pump.
2. Break nozzles with length-to-diameter ratios greater than 10., used for SBLOCA experiments, can be modelled to produce calculated results that match their calibration data by sizing their volumes to give the same Courant limit as the remainder of the model while maintaining a length-to-diameter (hydraulic) ratio equal to that of the hardware. Using such a technique, the break volume is large enough to allow a reasonable time step for the calculation and the length-to-diameter ratio will allow the nozzle frictional pressure loss to be accurately calculated.

Base calculation: The baseline calculations were made using the model developed and tested at INEL. The results showed reasonable agreement with the experimental data.

Sensitivity studies: Sensitivity studies were performed to investigate: (i) the behavior of the ECCMIX component and (ii) the effect of using and not using the CCFL model for the 5 % SBLOCA (Test 6.2 TC).

Nodalization studies: Nodalization studies were performed to investigate the calculated draining process from vertical pipes. When the model was found to give unrealistic results using the usual coarse loop seal nodalization, the number of cells was approximately doubled. Reasonable results were obtained with the finer nodalization.

Summary: Data from Tests 9.1b and 6.2 TC, conducted in the Boucle d'Etudes THermohydrouliques SYsteme (BETHSY) facility in Grenoble, France, describe the phenomena observed during two small break loss-of-coolant accident (SBLOCA) experiments conducted in the facility. Since the two tests were performed from different starting conditions, used different sized break nozzles, and assumed different failures and operator action, they exhibited somewhat different phenomena. Thus it is interesting and relevant to analyze both tests. The Committee on the Safety of Nuclear Installations (CSNI) approved test 9.1b to be used as the experimental basis for the International Standard Problem number 27.

The BETHSY facility is a 1/100 volume-scaled simulator of a 2775 MW, 3 loop Framatome pressurized water reactor (PWR). The BETHSY facility has 3 identical loops, with the exception of a presssurizer being mounted in one loop, and has the same component heights as its Framatome counterpart. BETHSY was designed to be able to study asymmetric phenomena which can occur in a large number of accident scenarios. Hot legs and cold legs were constructed to preserve the pipe length to root pipe diameter scaling between the reference plant and BETHSY.

Test 9.1b/ISP27 involved a 0.5 % (2-inch) cold leg break without available high pressure safety injection. Initially the core operated at 10 % scaled power while the pumps ran at full scaled flow. Reactor scram and a lengthy pump coastdown were used. An operator action was simulated by depressurizing the secondary when core thermocouples detected significant heat up. The experiment continued through accumulator injection and low pressure safety injection (LPSI). No nitrogen was injected into the system. Auxiliary feedwater was used to maintain the secondary level above the top of the steam generator U-tubes. This test showed several major phenomena: (a) single- and two-phase flow through a break nozzle, (b) pump operation during two-phase flow, (c) primary and secondary depressurization, (d) natural circulation and reflux cooling, (e) loop seal clearing. (f) core boiloff, (g) accumulator injection, and (h) LPSI injection.

Test 6.2 TC involved a 5.0 % (6 inch) cold leg break without available high or low pressure safety injection. Initially the core operated at 10 % scaled power and the pumps ran at reduced flow to obtain a realistic primary temperature distribution. Reactor scram was followed by a rapid shutdown of the primary pumps. Accumulator injection began and was terminated before nitrogen entered the system. The transient was terminated when unmitigated core heatup began. No auxiliary feedwater was used but the U-tubes remained covered. This test showed several major phenomena: (a) single- and two-phase flow through a break nozzle, (b) primary system depressurization, (c) natural circulation and reflux cooling, (d) loop seal clearing, (e) core boiloff, and (f) accumulator injection.

The objectives of the 9.1b/ISP-27 and 6.2 TC test analysis efforts were to (i) gain greater understanding of the phenomena which occur during a small break loss-of-coolant accident, and (ii) assess the RELAP5/MOD3 version 7 code using these data. Objective (i) was reached by evaluating the time progression of several critical parameters for which experimental data were gathered during the tests. Objective (ii) was reached by constructing a

RELAP5/MOD3 model of the BETHSY facility and performing calculations of the tests 9.1b/ISP-27 and 6.2 TC using the initial and boundary conditions defined by the BETHSY experimenters.

Following the data analysis, the RELAP5/MOD3 version 7 code assessment was conducted. Using the baseline models, the two tests were simulated following the same scenario that occurred during the tests. The RELAP5/MOD3 simulations of the two tests showed reasonable agreement with experimental data including loop seal clearing and simulation of the vapor pull-through/entrainment phenomena experienced during the 0.5% SBLOCA experiment. The emergency core cooling mixer (ECCMIX) component was found to be inappropriate for use with the 0.5 % break model but acceptable for use with the 5.0 % break model. A mismatch between the calculated and measured primary inventory distributions at various times during the two transients indicated a likely problem with the interphase drag model. An error was identified in the counter-current flow limitation (CCFL) model which has led to a code correction that will appear in RELAP5/MOD3 Version 8.

8.2.4 Semiscale Small Break LOCE S-NH-1

Reference: E. J. Lee, B. D. Chung, and H. J. Kim, *RELAP5 Assessment Using Semiscale SBLOCA Test S-NH-1*, Korea Institute of Nuclear Safety.

Code version: RELAP5/MOD3 Version 5M5

Facility: Semiscale Mod 2-C located at the Idaho National Engineering Laboratory, Idaho Falls, ID.

Objectives: Evaluation of the code capability to simulate thermal-hydraulic behavior during a small break loss-of-coolant accident in a pressurized water reactor.

Major phenomena: Primary and secondary side pressure response, break mass flow rate, core thermal-hydraulic response, and the primary mass inventory distribution.

Code deficiencies: None.

User guidelines: None.

Base calculation: The base calculation was performed using a RELAP5 model of the Semiscale MOD2C facility provided by INEL.

Sensitivity calculation: None.

Nodalization calculation: None.

Summary: The code was assessed using the Semiscale S-NH-1 experimental data.

The Semiscale MOD-2C facility was a 1/1705 volumetrically-scaled two-loop model of a 3411 MWt four-loop pressurized water reactor. The facility had an electrically-heated core. Semiscale consisted of a pressure vessel with simulated reactor internals and an external downcomer. The simulated core consisted of a 5x5 array (23 heated) of rods. The broken loop Type III steam generator had an external downcomer designed to measure the riser fluid density together with two inverted U-tubes. The intact loop steam generator contained six inverted U-tubes.

The S-NH-1 experiment was a simulation of a 0.5% small break loss-of-coolant accident in the cold leg.

Although the authors identified the RELAP5/MOD3 base calculation as giving reasonable results, not enough evidence was shown to merit that classification in the reviewers' opinion.

9.0 IMPACT OF IDENTIFIED CODE DEFICIENCIES AND NODALIZATION STUDIES

The ten assessment reports discussed in Section 8 are not sufficient to show a complete picture of the code's capabilities. Indeed, because the code has continued to change following release of Version 5M5, a complete picture of Version 5M5's capabilities will never be recorded.

In October, 1992, RELAP5/MOD3 Version 80 will be completed and independent studies will begin. Differences between Versions 5M5 and 80 reflect correction of some of deficiencies a through h as listed in Section 7.2. Thus not all of the assessment results obtained in the assessment reports given in Section 8 are applicable to Version 80.

9.1 IMPACT OF IDENTIFIED CODE DEFICIENCIES

Based on the RELAP5/MOD3 assessment studies summarized above, Version 80 includes the following deficiencies: (i) The ECCMIX component overcalculates the condensation rate for some SBLOCA scenarios and causes the code to fail. (ii) The code will not calculate the correct fluid depth for stratified channel flow. (iii) The code's critical flow model continues to give results that fall outside the data uncertainty band for some transient scenarios. (iv) Under some conditions, the limits of which have not been rigorously defined, the code will not realistically calculate the process of draining a vertical pipe; voids are calculated to pass into lower cells before the upper cells are fully drained. (v) Following CHF the code may undercalculate the wall surface temperature. The calculated temperature magnitude and distribution may not match the data. These deficiencies are discussed individually in the following five subsections.

<u>ECCMIX Component</u>: The ECCMIX component was originally built to simulate the steam condensation on emergency core cooling system (ECCS) inventory that occurred during LBLOCAs.

Tests using the ECCMIX component to model ECC injection during a 0.5% SBLOCA showed the code to fail during the calculation due to excessive condensation rates (Roth, Choi, and Schultz, 1992). When the ECCMIX component was used in the model calculating a 5% SBLOCA code failure did not occur but the calculated condensation rates did not differ markedly from those obtained when the ECCMIX component was not used at all. Finally, work done by Choi, Ban, and S. Lee to assess the code using the LOFT L2-5 LBLOCE indicated that excessive condensation rates were calculated by the code. Thus, even though a complete assessment picture is not available on the performance and capability of the ECCMIX component, the first assessments indicate the ECCMIX component calculates excessive condensation under some conditions.

From the perspective of the user, further assessments are required to quantify the performance and capability of the ECCMIX component. Based on the above assessments, if the ECCMIX component is used in a model, the user should specifically watch out for excessive condensation in the calculation and should view the component as a possible source of code failure.

<u>Calculation of Liquid Depth During Channel Flow</u>: Often, during the course of a SBLOCA, stratified flow conditions occur particularly during reflux. Under some conditions the liquid channel flow rates may become large enough to exceed the critical Froude number and produce a hydraulic jump. Unfortunately the code does not have the capability to account for a critical Froude number condition.

The work done by S. Lee and Kim, 1992 modelled steam condensation on liquid in a horizontal rectangular cross-section channel. The steam and liquid flowed cocurrently. S. Lee and Kim's work showed the calculated liquid channel depth sometimes missed the measured value by as much as 50%. Sometimes the calculated depth was too low and sometimes it was too high. Thus, further study is required to identify the cause of the calculation/measurement mismatch.

Calculating the correct liquid depth under stratified conditions is important in obtaining the correct system mass distribution and in determining the correct upstream conditions for SBLOCAs, particularly when the liquid entrainment/vapor pull-through model is activated.

Critical Flow Model: The deficiency identified by S. Lee, Chung, and Kim showed that the code's critical flow model undercalculated the two-phase break flow prior to loop seal clearing and overcalculated the break flow following loop seal clearing. The net effect of the calculational-measurement mismatch is to cause the calculation to miss the timing of important events in the transient.

Draining of Vertical Pipes: The loop seals were finely nodalized (Roth, Choi, and Schultz, 1992) to more accurately calculate the loop seal clearing process during SBLOCAs. Prior to using a finer nodalization, Roth found that voids appeared in model cells located below cells that had not fully drained. The reason for the code's miscalculation of the draining phenomena is not clear, but if applicable models are not finely nodalized to calculate loop seal clearing a poor simulation of the process may result.

Low Calculated Wall Temperatures During Film Boiling: Nilsson's analysis of the Royal Institute of Technology's post-dryout experiments showed that the code usually undercalculated the wall temperatures of the test section for these experiments. This result is similar to results shown in the developmental assessment calculations. Comparison of the data and the calculations of Bennett's heated tube experiments (see Fig. 2.2-43 of Volume 3, Carlson, et al., 1990) at intermediate mass fluxes showed the same trend. This trend is important to remember since it shows the code is not conservative under all conditions.

9.2 IMPACT OF NODALIZATION STUDIES

Nodalization studies were conducted by Nilsson, 1991; Schneider, 1991; Curca-Tivig, 1991; S. Lee and Kim, 1992; Cho, et al.; S. Lee, Chung, and Kim; and Roth, Choi, and Schultz, 1992. The net results of their studies are given in Section 8. In some cases nodalization studies were done to investigate whether the results given by a particular code model would change under various conditions (Cho, et al). In other cases nodalization studies were done to determine fixes to a code deficiency (Roth, Choi, and Schultz, 1992; S. Lee, Chung, and Kim). The resulting nodalizations are discussed below:

Finely Nodalized Loop Seals: Researchers involved in two assessments found that a finely nodalized loop seal would produce better agreement between the data and the resulting calculation. S. Lee, Chung, and Kim investigated the effect of doubling the number of loop seal cells on the calculation of loop seal clearing. Their efforts led them to nodalize the loop seals with 18 cells. Roth found that nodalizing the loop seal with 19 cells resulted in better agreement between the data and the calculation.

Simulation of CHF Location: Nilsson, 1991 determined that changing the test section nodalization for the Royal Institute of Technology's post dryout experiments from 0.05 m long cells to 0.5 m long cells did not affect the code's predicted location of CHF (within the accuracy of the assumed cell length).

Modelling ECC Penetration and Bypass for LBLOCAs: Schneider, 1991 tried a number of different nodalizations to represent the Upper Plenum Test Facility (UPTF) lower plenum and downcomer for the multi-dimensional ECC penetration and ECC bypass calculations. The various geometry configurations that were investigated by Schneider are not given in his report. However Schneider's final recommended nodalization is shown.

Condensation on Horizontally Flowing Liquid Stream: S. Lee and Kim, 1992 performed a nodalization study to investigate the effect of decreasing the size of the model cells by one half. No important differences were noted when the calculational results were compared to the data.

Counter Current Flow Limiting (CCFL): Cho, et al, investigated the effect of decreasing the size of the model cells on the code's calculation of CCFL. Cho, et al. did not report how they changed their nodalization - so even though they reported little change in the code's CCFL calculation with a change in nodalization, the reviewers could not form a definitive conclusion concerning their study.

Primary System Nodalization Studies: While performing the single-phase and two-phase natural circulation and reflux assessments, Roth investigated a number of nodalization possibilities (see Roth and Schultz, 1991, Appendix A). His observations include:

1. A more finely nodalized primary nodalization for the steam generator (SG) U-tubes than four cells up and four cells down (see Fletcher and Schultz, 1992, pages 5-3 through 5-5) does not enhance the code's ability to calculate reflux behavior. However, if the analyst must simulate secondary boiloff with some precision, the nodalization should be defined to include short cell heights in the lower regions of the secondary volumes to better simulate low liquid levels as complete secondary inventory boiloff is approached.

2. The system pressure vessel should be nodalized to have:
   a. The core bypass cells divided in the same fashion as the core region.
   b. Cross flow junctions should be used at the entrance and exit to the vessel from the cold and hot legs respectively.

3. Additional specific nodalizations are recommended for experimental systems that do not have a separator in the secondary and that have pumps with weir exit configurations. Since these nodalization guidelines are specific to the BETHSY facility and Framatome reactors the reader is referred to Roth and Schultz, 1991 if further information is desired.

10.0 CONCLUSIONS AND OBSERVATIONS

During the course of conducting and reviewing the assessment studies that have been discussed in the previous sections:

- Seven code deficiencies/errors were isolated:

  1. The CCFL model in Version 5M5 contains a programming error.

  2. The interphase drag between liquid slugs and steam is undercalculated.

  3. The ECCMIX component overcalculates the condensation rate for some SBLOCA scenarios and causes the code to fail.

  4. In general, the code will not calculate the correct fluid depth for stratified channel flow.

  5. The code's critical flow model continues to give results that fall outside the data uncertainty band for some transient scenarios.

  6. Under some conditions, the limits of which have not been rigorously defined, the code will not realistically calculate the process of draining a vertical pipe; voids are calculated to pass into lower cells before the upper cells are fully drained.

  7. Following CHF the code may undercalculate the wall surface temperature. The calculated temperature magnitude and distribution may not match the data.

- A better understanding of the code development effort needed to correct future versions RELAP5/MOD3 has been obtained.

- The ICAP assessments showed that MOD3 is a very effective tool for analyzing a great number of problems, even though some deficiencies are present.

PART III: SUMMARY OF TRAC-BF1 CODE ASSESSMENTS

11.0 SUMMARY OF ASSESSMENT RESULTS AND APPLICABILITY OF THE CODE

The TRAC-BF1 code (sometimes called G1J1) is described in Weaver, et al, 1986. The code was released in January, 1987 and was "frozen" to allow the world-wide user community to identify code deficiencies in a stable code version that did not change with time as each deficiency was found. Thus, the code was not a "moving target" for thermal-hydraulic modelers and code users.

11.1 HISTORY OF CODE

As summarized in Weaver et al., 1986, the TRAC-BF1 differed from TRAC-BD1 in that BF1 was (i) designed to run faster using a material Courant-limit violating numerical solution for all one-dimensional components, and (ii) included improved model formulations to allow simulations of operational transients and anticipated transients without scram (ATWS).

Thus, additional modeling capability included: (i) a one-dimensional neutron kinetics model, (ii) an improved interfacial heat transfer model, (iii) an improved interfacial shear model, (iv) a condensation model for stratified vertical flow, (v) an implicit turbine model, and (vi) an improved control system logic and solution method.

11.2 CODE DEFICIENCIES

TRAC-BF1 code deficiencies are numerous and have led to some forty-one updates thus far. Because of limited funding none of the forty-one updates were implemented in the code as of 1991. The forty-one updates are designed to correct problems (Schultz, 1990) in numerous subroutines in the code. The problems of greatest concern to the user are deficiencies in the SEPARATOR/DRYER component, deficiencies that prevent use of the code's fast numerics, deficiencies in the water packing algorithm and the mass error calculation, deficiencies that prevent the user from restarting the code properly, and problems that cause the user to have an incorrect problem initialization.

12.0 SYNOPSES AND EXECUTIVE SUMMARIES OF THE ICAP ASSESSMENTS

The results of four code assessment studies are summarized (Aksan, Stierli, and Analytis, 1992; Castrillo, Navarro, and Gallego, 1991a; Castrillo, Gomez, and Gallego, 1991b; Crespo and Fernandez, 1991). A fifth assessment, performed by Akimoto, et al., 1990 was not reviewed because it is based on using the TRAC-BF1 reflood model in the TRAC-PF1 code.

The first of the assessments is based on a separate effects experiment conducted in the NEPTUN facility designed to study the boiloff characteristics of the TRAC code. It should be noted that the assessment conducted by Aksan, Stierli, and Analytis was performed using the TRAC-BD1 Version 22 code which differs from TRAC-BF1 code as outlined in Section 11. Although the modifications to the interfacial friction model that distinguish the TRAC-BD1 Version 22 code from TRAC-BF1 may change the conclusions of the assessment somewhat, Aksan, Stierli and Analytis's study is included because it is the only work available in the form of an ICAP assessment report.

Integral effects assessments were completed for three operational transients. The transients studied were: (i) a feedwater pump trip transient at the Cofrentes Nuclear Power Plant, see Castrillo, Navarro, and Gallego, 1991, (ii) a main steam line isolation valve closure transient at the Santa Maria de Garona Nuclear Power Plant, see Crespo and Fernandez, 1991, and (iii) a turbine trip transient at the Cofrentes Nuclear Power Plant, see Castrillo, Gomez, and Gallego, 1991.

The TRAC-B code assessment matrix was defined by the ICAP members and is shown in Figs. 12.1 for operational transients. The matrix is based simply on the assessment studies performed by ICAP members and was defined using the phenomena of importance for each transient type as listed by the Committee on the Safety of Nuclear Installations (CSNI) of the Task Group on the Status and Assessment of Codes for Transients and Emergency Core Cooling Systems of the Principal Working Group No. 2 on Transients and Breaks for the Organization for Economic Cooperation and Development (see CSNI, 1987). Phenomena represented by data in a particular experimental data set are indicated by cross-referencing the test facility versus the phenomena (see Fig. 12.1). If the experimental data set contained data of limited usefulness due to large uncertainty bands or other reasons, then an open circle is shown. If the data do not contain information on a particular phenomena, then a dash is shown.

Phenomena	Operational BWRs
Natural circulation in one-and two-phase flow	-
Collapsed level behavior in downcomer	-
Core thermal hydraulics	○
Valve leak flow	-
Single phase pump behavior	○
Parallel channel effects and instabilities	-
Nuclear thermahydraulic feedback including spatial effects	○
Nuclear thermahydraulic instabilities	-
Downcomer mixing	-
Boron mixing and distribution	-
Steam line dynamics	○
Void collapse and temp. distribution during pressurization	○
Critical power ratio	-
Rewet after DNB at high press. and high power incl. high core flow	-
Structural heat and heat losses	-

Figure 12.1 Code Assessment Matrix: Operational Transients

JEN-MATRIX8D

12.1 Boil-Off Experiment Assessment

Reference: S. N. Aksan, F. Stierli, and G. Th. Analytis, *Boil-Off Experiments with the EIR-NEPTUN Facility: Analysis and Code Assessment Overview Report*, NUREG/IA-0040, March, 1992.

Code version: TRAC-BD1 Version 22

Facility: NEPTUN facility, located at the former Swiss Federal Institute for Reactor Research (EIR) at Villigen, Switzerland. The Swiss Federal Institute of Reactor Research is now a part of the Paul Scherrer Institute (PSI).

Objectives: Study the hydraulics and heat transfer associated with cooling of fuel rods in a pool of water without external circulation at low and intermediate pressure. These conditions represent those expected during a small break loss-of-coolant accident (SBLOCA) followed by core uncovery. Specific objectives were to obtain good data, assess the ability of analysis codes to calculate the appropriate behavior, and to evaluate the measurement accuracy and cooling influence of externally mounted surface thermocouples.

Major phenomena: Phenomena present during boil-off of a liquid water pool covering the core, simulating that expected during a SBLOCA with no circulation. Specifically the phenomena addressed were the two-phase interfacial shear during bubbly-slug flow and the heat transfer mechanism to single-phase vapor (measured and calculated after termination of rod power).

Code deficiencies: Four deficiencies were noted:

1. The calculated liquid carryover during boil-off was excessive.
2. The calculated CHF was too early.
3. The calculated cladding temperatures were 8 to 15 K low during nucleate boiling.
4. The calculated heat transfer to single-phase vapor was excessive.

Impact of code deficiencies: The code will not be able to calculate the correct thermal-hydraulic behavior during boildown of a liquid pool covering the core during a SBLOCA with decay power.

User guidelines: None.

Base calculation: The base calculation was performed using a model built at EIR, however no information concerning the model was provided.

Sensitivity studies: The authors performed additional calculations after modifying the code's interfacial shear correlation and the heat transfer selection logic. However, no sensitivity calculations (defined as recalculations using other available code options) were performed.

Nodalization studies: None.

Summary: The code was assessed using data obtained from the NEPTUN facility during boil-off of a stagnant pool of liquid water covering an electrically heated rod bundle. The rod bundle consisted of 33 rods of 1.68 m heated length with a chopped cosine axial power distribution. Data from five tests with different values of rod bundle power, liquid subcooling, and overpressure were compared with TRAC calculations.

Comparison of the calculated and measured collapsed liquid level indicated the code was calculating a significantly lower liquid level. Thus the codes were calculating too much liquid expulsion from the rod bundle and an early time for CHF and generally an early occurrence of nucleate boiling. Comparison of measured rod surface temperature with TRAC calculation after rod power had been turned off indicated the calculated heat transfer rate was too high for the single-phase vapor mechanism.

The interfacial shear correlations in the frozen versions of the code for bubbly and slug flow were replaced by a correlation developed for rod bundles and used in the CATHARE code. With this modification the code calculations provided close agreement with rod surface temperature and collapsed liquid level data obtained at about 0.5 MPa pressure. One comparison of TRAC calculations with data obtained at about 0.1 MPa only partially eliminated the original discrepancy.

Figs. 12.1.1 and 12.1.2 show the before and after TRAC calculations compared with base test data for rod surface temperature and collapsed liquid level at about 0.5 MPa. Figs. 12.1.3 and 12.1.4 show the same comparisons with data for about 0.1 MPa.

The frozen code version was also modified to change the selection logic for picking the heat transfer correlation for application to vapor. The frozen version selected the largest coefficient computed for turbulent forced flow, free convection, and forced laminar flow. The modification used the largest coefficient computed for turbulent forced flow and free convection.

Fig. 12.1.5 shows the effect of modifying the heat transfer mechanism as stated above. The figure shows that the slopes of the computed and measured temperature are the same after the peak temperature occurs.

These comparisons indicate the code modifications result in the correct calculations of the rod bundle thermal-hydraulic behavior for coolant boildown of a rod bundle at decay heat levels and one low pressure (about 0.5 MPa). One comparison made with data and calculations at atmospheric pressure did not indicate good agreement. Thus, the modifications are not valid for a range of pressure.

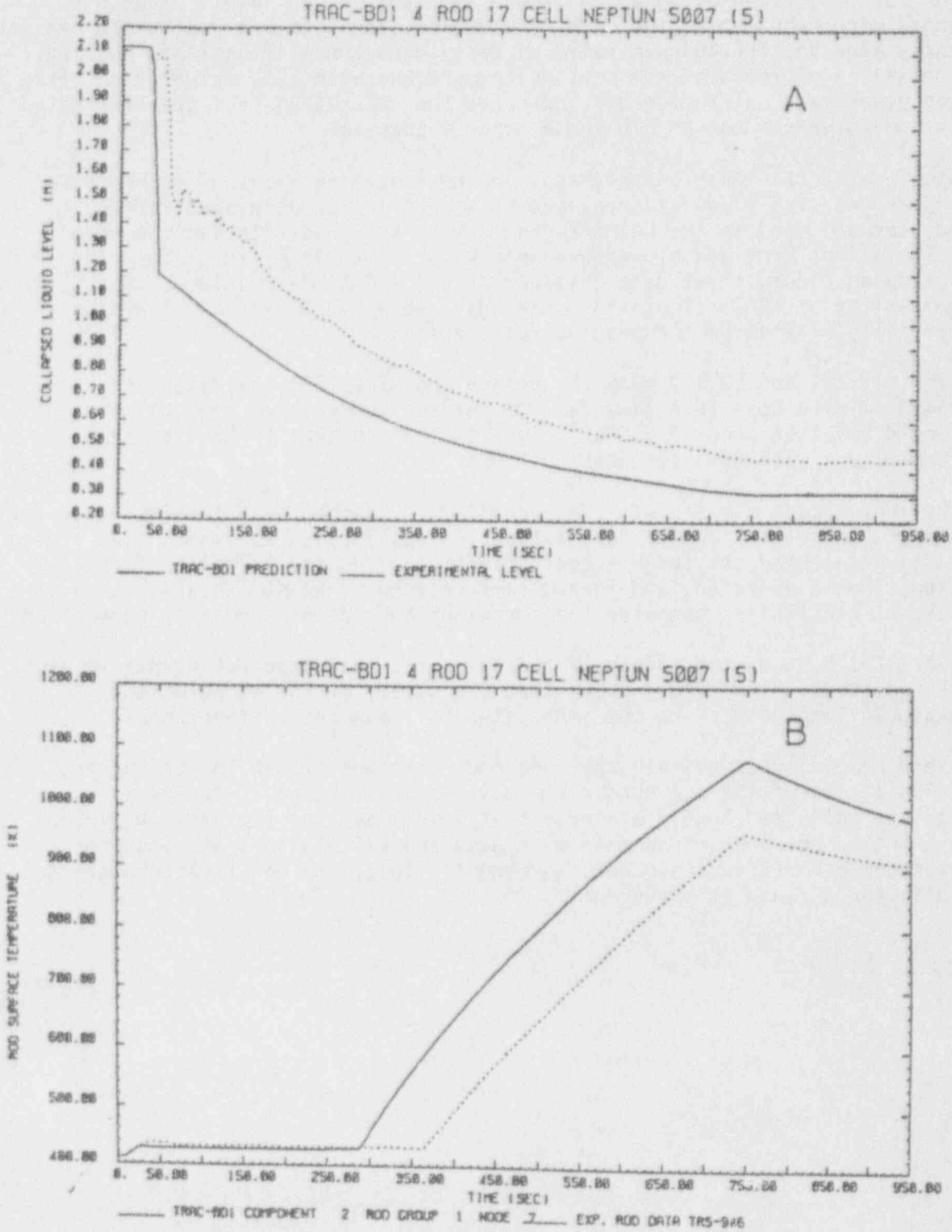
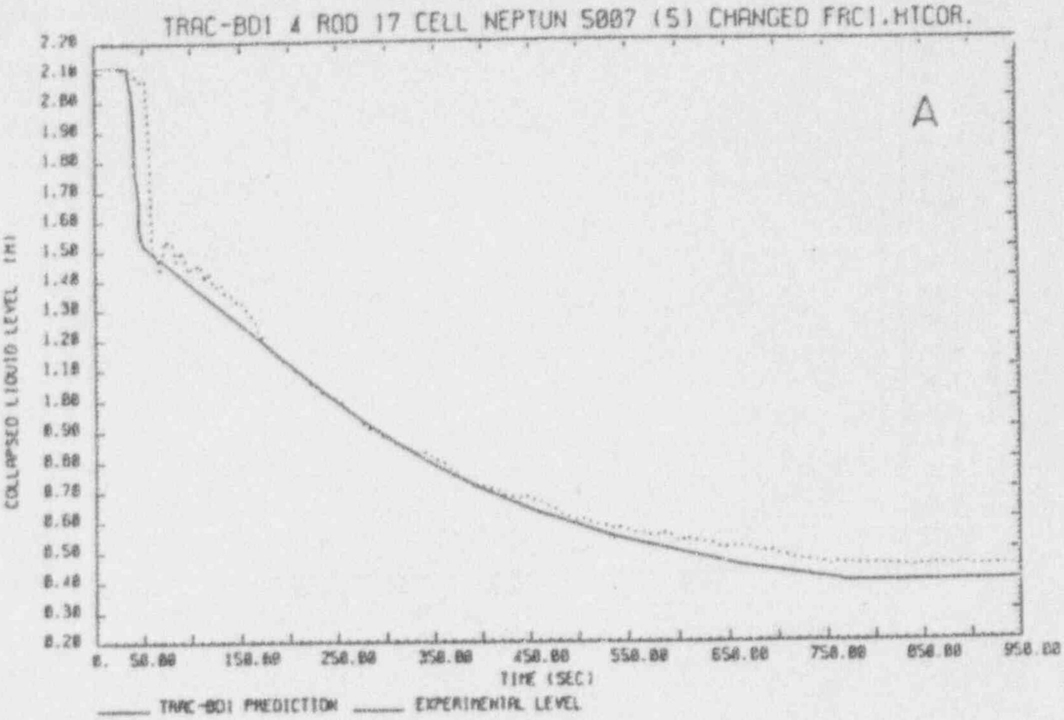


Figure 12.1.1 Comparison of measured and calculated collapsed liquid level and peak axial power level rod surface temperature histories, using frozen version of TRAC-BD1 for NEPTUN boil-off experiment 5007.



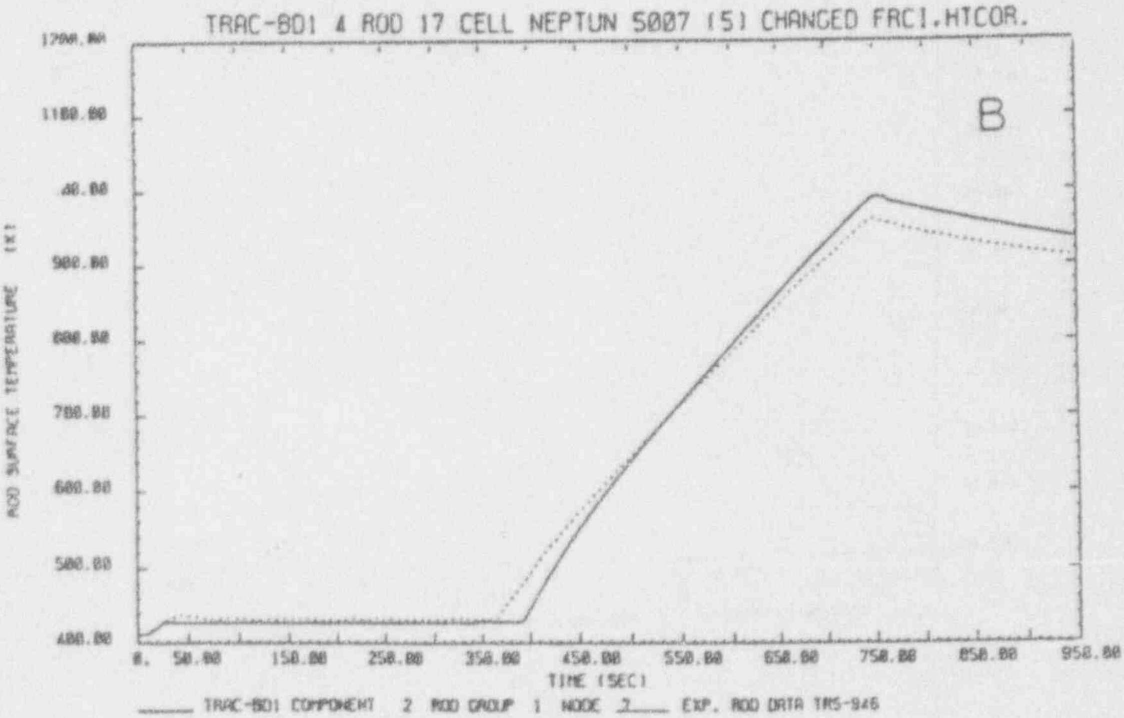


Figure 12.1.2 Comparison of measured and calculated collapsed liquid level and peak axial power leve rod surface temperature histories, using modified version of T[illegible]AC-BD1 for NEPTUN boil-off experiment 5007.

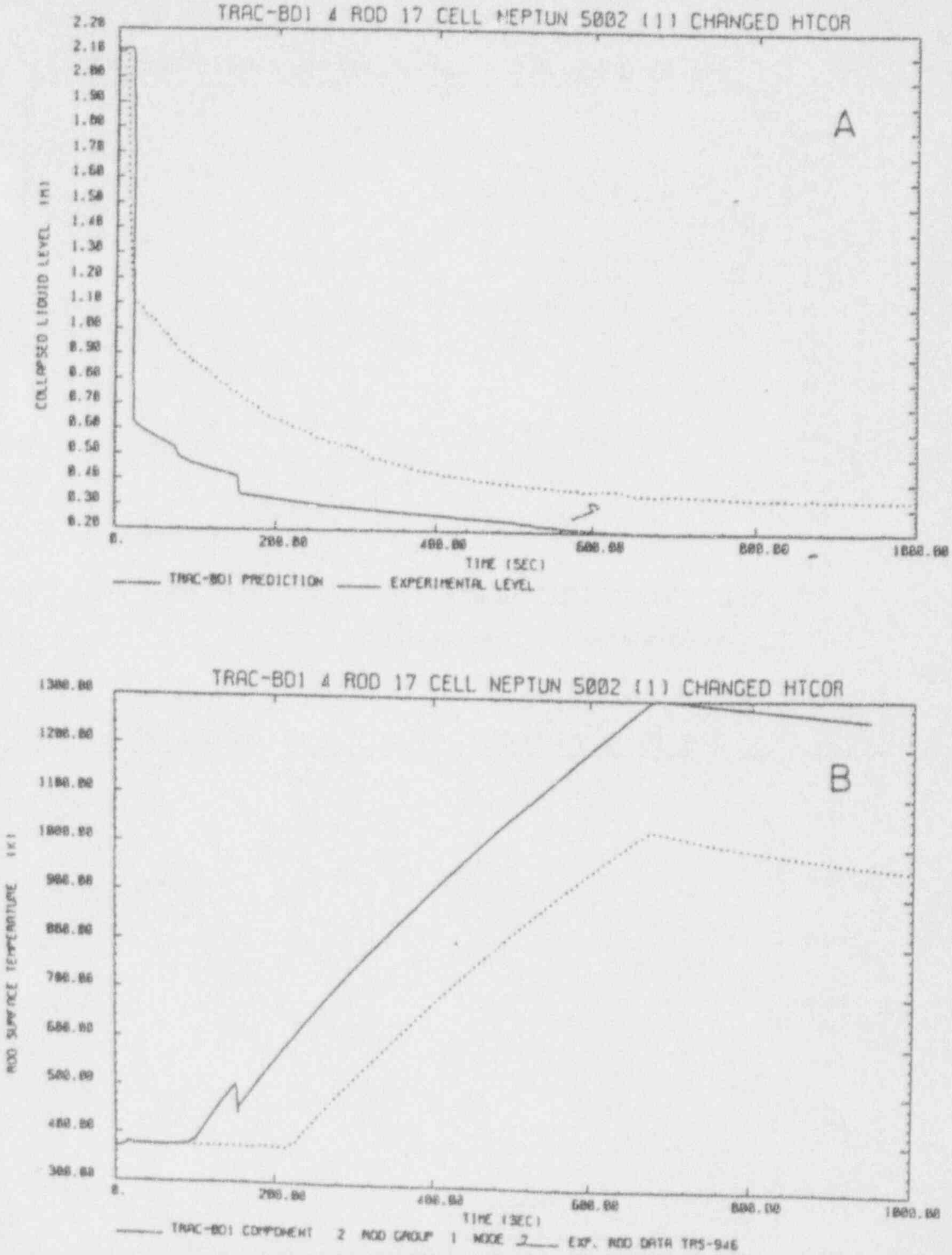
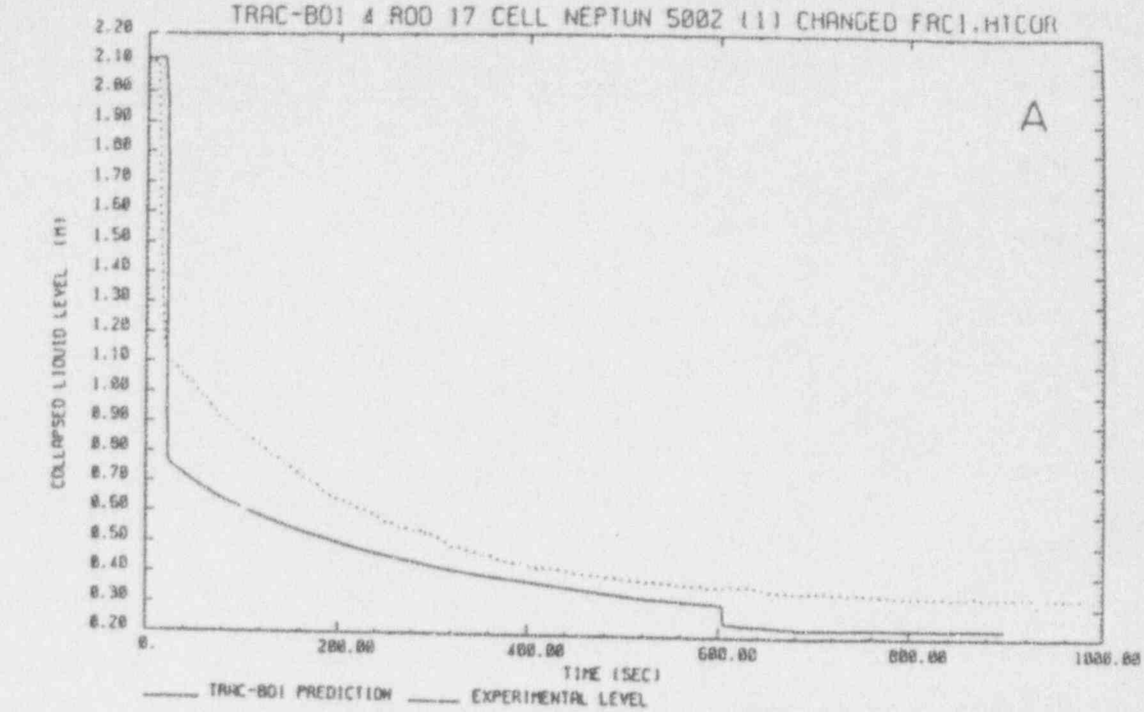


Figure 12.1.3 Comparison of measured and calculated collapsed liquid level and peak axial power level rod surface temperature histories, using frozen version of TRAC-BD1 for NEPTUN boil-off experiment 5002.



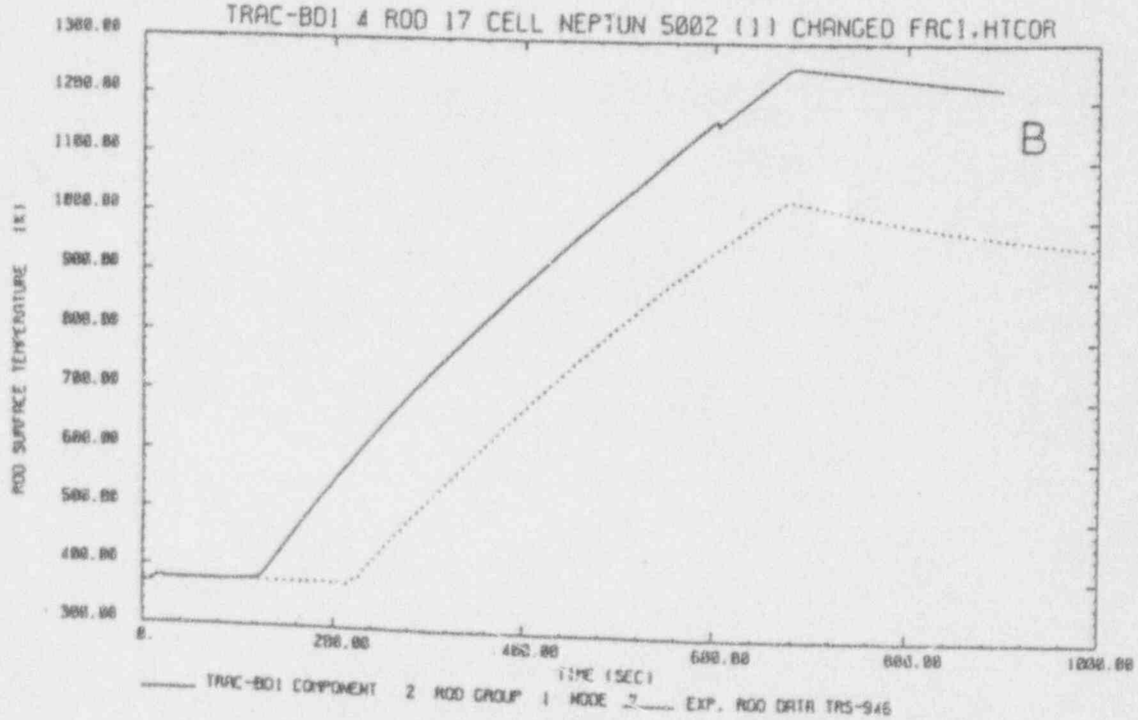


Figure 12.1.4 Comparison of measured and calculated collapsed liquid level and peak axial power level rod surface temperature histories, using frozen version of TRAC-BD1 for NEPTUN boil-off experiment 5002.

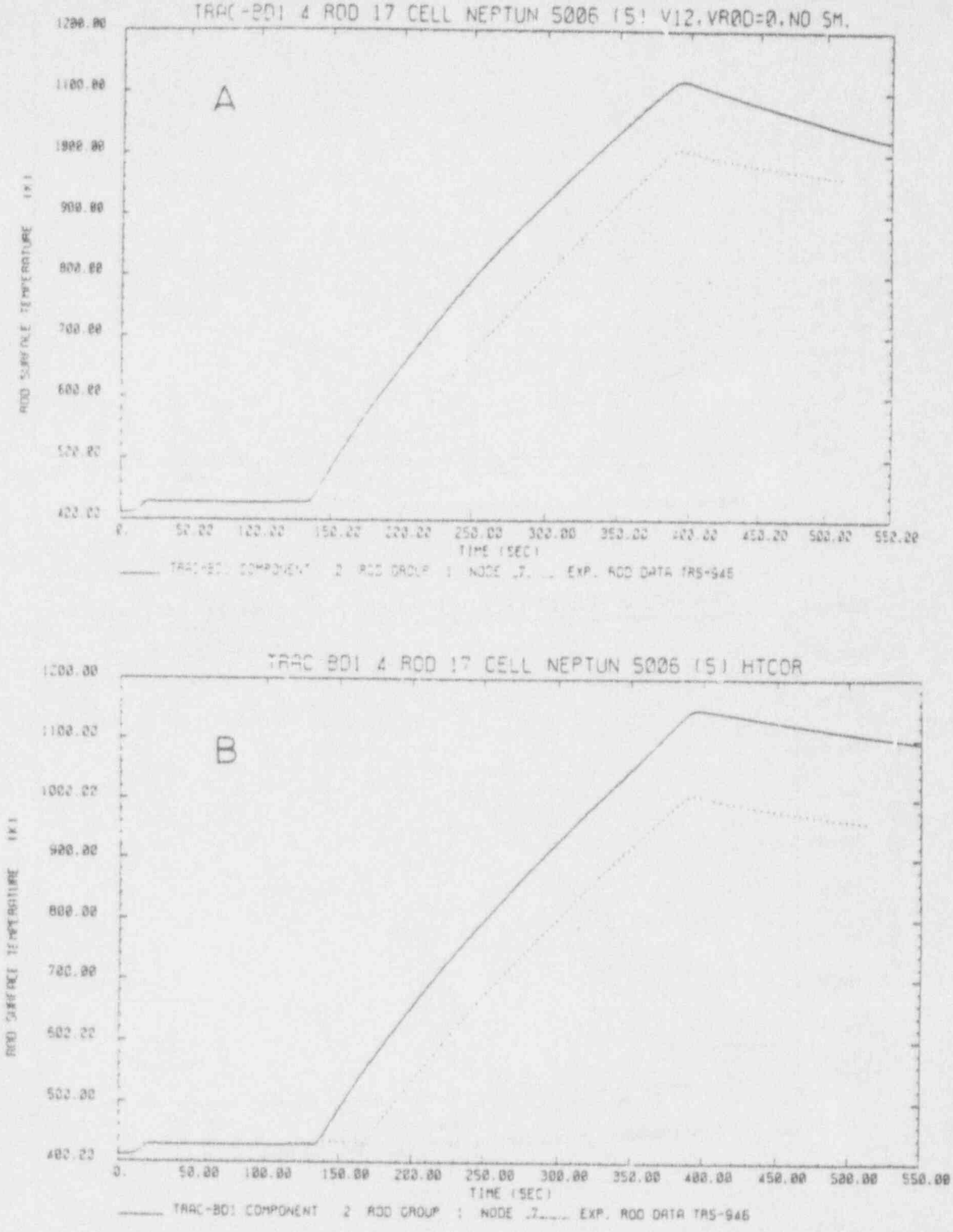


Figure 12.1.5 Rod surface temperature histories at peak axial power level in NEPTUN boil-off experiment 5006 calculated by TRAC-BD1, (a) frozen version, (b) modified for steam cooling.

12.2 Feedwater Pump Trip Transient at Cofrentes Nuclear Power Plant

Reference: F. Castrillo, A. G. Navarro, and I. Gallego, *Assessment of the "One Feedwater Pump Trip Transient" in Cofrentes NPP with TRAC-BF1*, ICSP-CO-TURFW-T, February, 1991.

Code version: TRAC-BF1 (G1J1)

Facility: Cofrentes Nuclear Power Plant, located near Valencia, Spain.

Objectives: The objective of the assessment was to determine the capability of the TRAC-BF1 code to simulate the one-feedwater pump trip transient when the plant is at nominal conditions.

Major phenomena: Dynamic level tracking, core neutronic feedback, recirculation and jet pump performance under normal operating conditions.

Code deficiencies: The mechanistic steam separator model was found to be difficult to use for this transient.

User guidelines: When simulating the vessel water level, it is important to consider the water level shift between regions inside and outside the dryer skirt due to the pressure drop across the dryer.

Base calculation: The base calculation was performed using a model built by Unidad Electrica, S.A. The vessel was simulated with four ring, one azimuthal section, eight level model with two recirculation loops and one representative steam line. The appropriate plant control systems were simulated. Core kinetics were simulated using the code's point kinetics capability.

Sensitivity studies: Sensitivity calculations were performed to determine the best way to represent the water level shift that is present in the plant between the inside and outside of the dryer skirt. However, the calculations were not included in the study.

...dalization studies: None.

The code was assessed using startup test data from the Cofrentes ... Plant. The transient simulated the manual trip of one

... General Electric BWR/6 plant with a nominal core ... that has been in commercial operation since 1985. ... driven feedwater pumps.

... was to determine the capability of the ... feedwater pump trip transient when the

... Electrica, S.A. based on plant ... ting RETRAN model. The model was ... thal section, eight level pressure ... one representative steam line. The ... imulated.

... ed and the point kinetic option ... Reactivity coefficients were obtained from a perturbation study, based on a three-dimensional simulator, performed at the steady-state condition.

The one feedwater pump trip transient was run in the plant to study the capability of the plant to avoid reactor trip by reducing the plant power level to be consistent with the operating characteristics of the one remaining turbine driven feedwater pump.

Simulation of the transient using TRAC-BF1 was undertaken to assess the capability of the code and model to simulate the dynamic level tracking, core neutronic feedback, recirculation and jet pump performance under normal operating conditions.

The assessment transient was 150 s long and included all the key phenomena that occurred during the operational transient experiment. The sensed plant water level was the most critical plant variable that required simulation. It was found that the code did a reasonable job of simulating the transient plant water level (see Fig. 12.2.1 - Note: the traces labeled Measurement A and Measurement B are data from two independent instrumentation channels.). Comparison between the calculated and measured values of other variables: the feedwater flow rate, the steam flow rate, the recirculation and core flow rates, the core power level, and the system pressure level, showed the code did a reasonable job of simulating the transient.

12.2 Feedwater Pump Trip Transient at Cofrentes Nuclear Power Plant

Reference: F. Castrillo, A. G. Navarro, and I. Gallego, *Assessment of the "One Feedwater Pump Trip Transient" in Cofrentes NPP with TRAC-BF1*, ICSP-CO-TURFW-T, February, 1991.

Code version: TRAC-BF1 (G1J1)

Facility: Cofrentes Nuclear Power Plant, located near Valencia, Spain.

Objectives: The objective of the assessment was to determine the capability of the TRAC-BF1 code to simulate the one-feedwater pump trip transient when the plant is at nominal conditions.

Major phenomena: Dynamic level tracking, core neutronic feedback, recirculation and jet pump performance under normal operating conditions.

Code deficiencies: The mechanistic steam separator model was found to be difficult to use for this transient.

User guidelines: When simulating the vessel water level, it is important to consider the water level shift between regions inside and outside the dryer skirt due to the pressure drop across the dryer.

Base calculation: The base calculation was performed using a model built by Unidad Electrica, S.A. The vessel was simulated with four ring, one azimuthal section, eight level model with two recirculation loops and one representative steam line. The appropriate plant control systems were simulated. Core kinetics were simulated using the code's point kinetics capability.

Sensitivity studies: Sensitivity calculations were performed to determine the best way to represent the water level shift that is present in the plant between the inside and outside of the dryer skirt. However, the calculations were not included in the study.

Nodalization studies: None.

Summary: The code was assessed using startup test data from the Cofrentes Nuclear Power Plant. The transient simulated the manual trip of one feedwater pump.

The Cofrentes plant is a General Electric BWR/6 plant with a nominal core thermal power of 2894 MWt that has been in commercial operation since 1985. The plant has two turbine driven feedwater pumps.

The objective of the assessment was to determine the capability of the TRAC-BF1 code to simulate the one-feedwater pump trip transient when the plant is at nominal conditions.

The model was created by the Unidad Electrica, S.A. based on plant drawings, specifications, and an existing RETRAN model. The model was designed to have a four ring, one azimuthal section, eight level pressure vessel with two recirculation loops and one representative steam line. The appropriate plant control systems were simulated.

The plant steady-state condition was obtained and the point kinetic option was used to simulate core neutronic feedback. Reactivity coefficients were obtained from a perturbation study, based on a three-dimensional simulator, performed at the steady-state condition.

The one feedwater pump trip transient was run in the plant to study the capability of the plant to avoid reactor trip by reducing the plant power level to be consistent with the operating characteristics of the one remaining turbine driven feedwater pump.

Simulation of the transient using TRAC-BF1 was undertaken to assess the capability of the code and model to simulate the dynamic level tracking, core neutronic feedback, recirculation and jet pump performance under normal operating conditions.

The assessment transient was 150 s long and included all the key phenomena that occurred during the operational transient experiment. The sensed plant water level was the most critical plant variable that required simulation. It was found that the code did a reasonable job of simulating the transient plant water level (see Fig. 12.2.1 - Note: the traces labeled Measurement A and Measurement B are data from two independent instrumentation channels.). Comparison between the calculated and measured values of other variables: the feedwater flow rate, the steam flow rate, the recirculation and core flow rates, the core power level, and the system pressure level, showed the code did a reasonable job of simulating the transient.

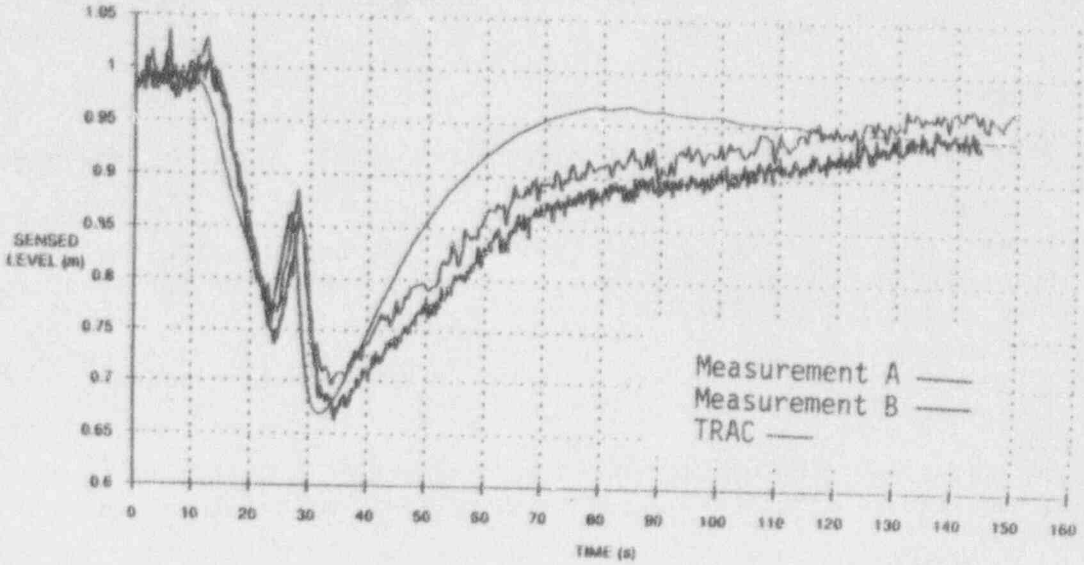


Figure 12.2.1 Cofrentes One Feedwater Pump Trip Transient. Measured water level.

12.3 Main Steam Line Isolation Valve Closure at Nuclear Power Plant

Reference: J. Crespo and R. A. Fernandez, *Assessment of MSIV Full Closure for Santa Maria de Garona NPP Using TRAC BF1 (GIJI)*, ICSP-GA-MSIV-1, June, 1991.

Code version: TRAC-BF1 (GIJI)

Facility: Santa Maria de Garona Nuclear Power Plant at Burgos Province, Spain.

Objectives: Determine the capability of the TRAC-BF1 code to simulate a spurious main steam line isolation valve closure event when the plant is operating at nominal conditions.

Major phenomena: System pressure behavior, dynamic level tracking, core mass flow rate, and feedwater flow rate.

Code deficiencies: The mechanistic separator model is difficult to use, but is required to simulate the phenomena present in this transient scenario.

User guidelines: None.

Base calculation: The base calculation was performed using the nodalization specified by Nucleanor and the University of Cantabria.

Sensitivity studies: None.

Nodalization studies: Three nodalization studies were performed. However no detailed discussion was given to compare them with the baseline calculation.

Summary: The code was assessed using data recorded during a spurious main steam line isolation valve closure event at the Santa Maria de Garona Nuclear Power Plant.

The Santa Maria de Garona plant is a General Electric BWR/3 plant with a nominal core thermal power of 1380 MWt that has been in commercial operation since 1971. The plant has four steam lines with three relief valves (RVs), two safety-relief valves (SRVs), and seven safety valves (SVs). Each steam line has two isolation valves.

The objective of the assessment was to determine the capability of the TRAC-BF1 code to simulate a spurious main steam line isolation valve closure event when the plant is operating at nominal conditions.

The model was created by Nucleanor, S.A. (the utility operating the plant) and the University of Cantabria. The model was designed to have a three ring, one azimuthal section, ten level pressure vessel with one representative recirculation loop and one representative steam line. The steam separator was assumed to be a perfect steam separator, that is all liquid was separated from the separator exhaust steam. The appropriate plant control systems were simulated.

The plant steady-state condition was obtained and the point kinetic option was used to simulate core neutronic feedback. Reactivity coefficients were obtained from data provided by ENUSA.

The spurious main steam line isolation valve closure transient occurred when the plant was operating at 100% power and 89% core flow. During the first 60 s of the transient: (i) no relevant manual actions took place, (ii) the isolation condenser system was not activated, and (iii) the feedwater and recirculation systems continued to operate. The MSIVs closed in about 3 s. The scram was initiated when the valves were in the 90% open position. The resulting pressure increase caused the RVs to open.

Simulation of the transient using TRAC-BF1 was undertaken to assess the capability of the code and model to simulate the system pressure behavior, dynamic level tracking, core mass flow rate, and feedwater flow rate.

The assessment transient was 60 s long and qualitatively agreed with the system behavior. Although the calculated system pressure showed reasonable agreement with the data for the first 15 s, the calculated pressure decreased at a more rapid rate than the data for the next 5 s (see Fig. 12.3.1). The authors contend the difference between the calculation and the data was caused by more efficient condensation of the system steam in the calculation due to the presence of the ideal steam separator. Realistically the code's separator model should allow steam carryunder and liquid carryover to accurately simulate the system pressure. The effect of using an ideal steam separator, as indicated by the system pressure, is also present in other comparisons between the calculation and the data.

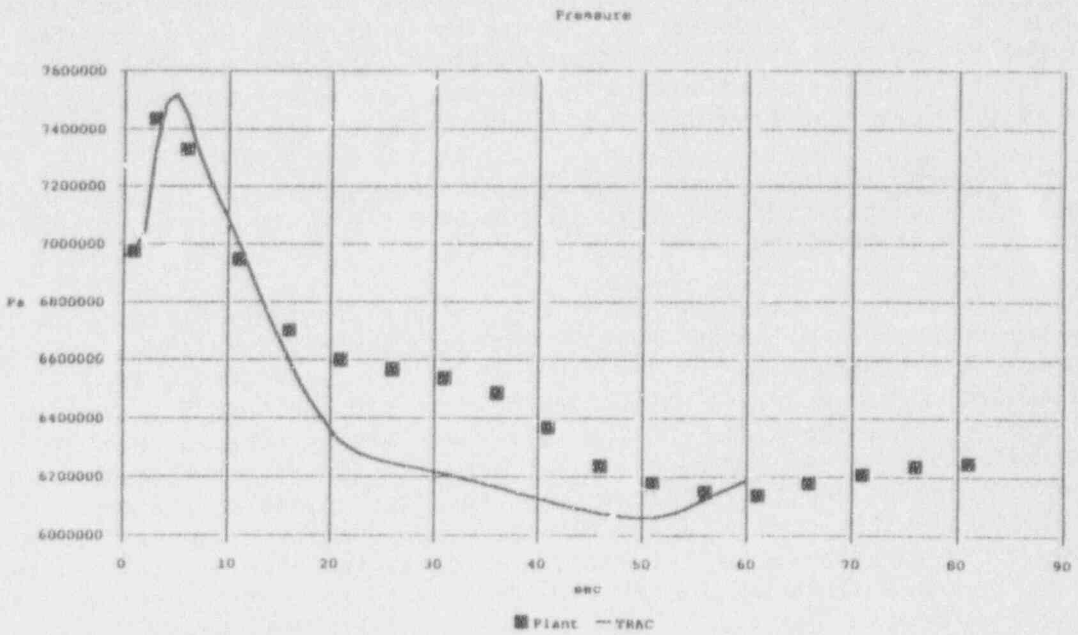


Figure 12.3.1 Santa Maria de Garona MSIV Closure: Reactor Vessel Pressure.

12.4 Turbine Trip Transient at Cofrentes Nuclear Power Plant

Reference: F. Castrillo, A. Gomez, and I. Gallego, *Assessment of the "Turbine Trip Transient" in Cofrentes NPP with TRAC BF1*, ICSP-CO-TTRIP-T, June, 1991.

Code version: TRAC-BF1 (G1J1)

Facility: Cofrentes Nuclear Power Plant, located near Valencia, Spain.

Objectives: Determine the capability of the TRAC-BF1 code to simulate the manual turbine trip transient when the plant is at 70% of nominal power.

Major phenomena: Dynamic level tracking, core neutronic feedback, recirculation and jet pump performance under normal operating conditions.

Code deficiencies: The mechanistic steam separator model was found to be difficult to use.

User guidelines: None.

Base calculation: The base calculation was performed using a model built by Unidad Electrica, S.A. The vessel was simulated with four ring, one azimuthal section, eight level model with two recirculation loops and one representative steam line. The appropriate plant control systems were simulated. Core kinetics were simulated using the code's point kinetics capability.

Sensitivity studies: Sensitivity calculations were performed "tune" the model to match the measured data. Detailed records of the changes required to "tune" the model are not presented.

Nodalization studies: None.

Summary: The code was assessed using data recorded during a manual trip of the main turbine.

The Cofrentes plant is a General Electric BWR/6 plant with a nominal core thermal power of 2894 MWt that has been in commercial operation since 1985. The plant has two turbine driven feedwater pumps.

The objective of the assessment was to determine the capability of the TRAC-BF1 code to simulate the manual turbine trip transient when the plant is at 70% of nominal power.

The model was created by the Unidad Electrica, S.A. based on plant drawings, specifications, and an existing RETRAN model. The model was designed to have a four ring, one azimuthal section, eight level pressure vessel with two recirculation loops and one representative steam line. The appropriate plant control systems were simulated.

The plant steady-state condition was obtained and the point kinetic option was used to simulate core neutronic feedback. Reactivity coefficients were obtained from a perturbation study, based on a three-dimensional simulator, performed at the steady-state condition.

The manual turbine trip transient was a startup test conducted during the plant commissioning effort. The test was initiated by a manual trip of the main turbine. Following receipt of the closure signal, the reactor scrammed and the recirculation pumps were programmed to operate at low speed.

Simulation of the transient using TRAC-BF1 was undertaken to assess the capability of the code and model to simulate the dynamic level tracking, core neutronic feedback, recirculation and jet pump performance under normal operating conditions.

The assessment transient was 45 s long and included all the key phenomena that occurred during the operational transient experiment. The transient was initiated by the closure of the turbine stop valves. The measured downcomer water level was considered to be the most critical variable requiring simulation. To produce the best simulation of the downcomer water level, the vessel model volumes, flow areas, fuel gap conductivities, and the dryer pressure loss coefficient were "tuned." Using this procedure a reasonable match between the data and the calculation was achieved.

13.0 CONCLUSIONS AND OBSERVATIONS

During the course of conducting and reviewing the assessment studies that have been summarized in the previous section it was noted that the mechanistic steam separator model was difficult to use (the authors found the mechanistic separator to be failure-prone),but is important in simulating a plant's behavior during operational transients. As a result of having only an ideal steam separator some of the analysts have "tuned" the input models to match the plant data.

Through the efforts of ICAP members it is apparent that the mechanistic steam separator model should be one of the first corrections made to the code. Thus a better understanding of the code development effort needed to correct future versions TRAC-BF1 has been obtained.

In addition, Aksan, Stierli, and Analytis, 1991 noted four other deficiencies specific to the TRAC-BD1 Version 22 code:

1. The calculated liquid carryover during boil-off was excessive.
2. The calculated CHF was too early.
3. The calculated cladding temperatures were 8 to 15 K low during nucleate boiling.
4. The calculated heat transfer to single-phase vapor was excessive.

At this writing the degree to which the above deficiencies are applicable to TRAC-BF1 is not known.

Even though TRAC-BF1 has a number of deficiencies, the code has been shown to be a useful tool for simulating some important plant transients.

14.0 REFERENCES

Abdollahian, D., J. Healzer, E. Janssen, and C. Amos, 1982, *Critical Flow Data Review and Analysis*, EPRI NP-2192, January.

Adams, J. P., D. L. Batt, and V. T. Berta, 1986, "Influence of LOFT PWR Transient Simulations on Thermal-Hydraulic Aspects of Commercial PWR Safety," *Nuclear Safety*, Vol. 27, No. 2, April-June.

Akimoto, H., A. Ohnuki, Y. Ake, and Y. Murao, 1990, *Assessment of TRAC-BF1 1D Reflood Model with CCTF and SCTF Data*, JAERI-memo 02-137, May.

Aksan, S. N., F. Stierli, and G. Th. Analytis, 1987 *Boil-Off Experiments with the EIR-NEPTUN Facility: Analysis and Code Assessment Overview Report*, Paul Scherrer Institute, EIR-Bericht Nr. 629, September.

Analytis, G. Th., 1992, "A Comparative Study of the Post-CHF Wall Heat Transfer Packages of the RELAP5 Codes and Preliminary Assessment of Model Changes in RELAP5/MOD3/v7j," First CAMP Meeting, June 24-26, 1992, Paul Scherrer Institute, Villigen, Switzerland.

Analytis, G. Th., and M. Richner, 1986, *Implementation and Assessment of a New Bubbly/Slug Interfacial Friction Correlation in RELAP5/MOD2/36.02*, Paul Scherrer Institute, TM-32-86-10, January.

Anderson, J. L., and R. L. Benedetti, 1986, *Critical Flow Through Small Pipe Breaks*, EPRI Report EPRI NP-4534, May.

Ardron, K. H., and W. M. Bryce, *Assessment of Horizontal Stratification Entrainment Model in RELAP5/MOD2*, NUREG/IA-0039 (copies can be obtained from Ms. Nancy Larson, INEL).

Ardron, K. H., and A. J. Clare, 1987, *Assessment of Interphase Drag Correlations in the RELAP5/MOD2 and TRAC-PF1/MOD1 Thermal Hydraulic Codes*, Central Electricity Generating Board Report GD/PE-N/557 (Revised), March.

Arne, N., S. Cho, and S. H. Lee, 1990, *Assessment Study of RELAP5/MOD2 Computer Code Against the Net Load Trip Test Data from Yong-Gwang Unit 2*, Korea Electric Power Co., January.

Bang, Y. S., S. Y. Lee, and H. J. Kim, 1990, *Assessment of RELAP5/MOD2 Cycle 36.04 Using LOFT Large Break Experiment L2-5*, NUREG/IA-0032, April.

Bang, Y. S., H. J. Kim, and S. H. Kim, 1992, *Assessment of RELAP5/MOD2 Cycle 36.04 with LOFT Large Break Experiment L2-3*, NUREG/IA-0070, April.

Bestion, D., 1985, *Interfacial Friction Determination for the 1D-6 Equations Two Fluid Model Used in the CATHARE Code*, European Two-Phase Flow Group Meeting, Southhampton, England.

Birchley, J. C., 1992, *RELAP5/MOD2 Analysis of LOFT Experiment L9-3*, NUREG/IA-0058, April.

Brain, C. R., 1992, *Assessment of the Subcooled Boiling Model Used in RELAP5/MOD2 (Cycle 36.05, Version E03) Against Experimental Data*, NUREG/IA-0056, March.

Brodie, P., and P. C. Hall, 1992, *Analysis of Semiscale Test S-LH-2 Using RELAP5/MOD2*, NUREG/IA-0065, April.

Carlson, K. E., R. A. Riemke, S. Z. Rouhani, R. W. Shumway, and W. L. Weaver, 1990, *RELAP5/MOD3 Code Manual, Volumes 1 to 4*, NUREG/CR-5535, June.

Castrillo, F., A. Gomez, and I. Gallego, 1991, *Assessment of the "Turbine Trip Transient" in Cofrentes NPP with TRAC BF1*, ICSP-CO-TTRIP-T, June.

Castrillo, F., A. G. Navarro, I. Gallego, 1991, *Assessment of the "One Feedwater Pump Trip Transient" in Cofrentes NPP with TRAC-BF1*, ICSP-CO-TURFW-T, February.

Cho, S., N. Arne, B. D. Chung, and H. J. Kim, 1993, *Assessment of CCFL Model of RELAP5/MOD3 Against Simple Vertical Tubes and Rod Bundle Tests*, NUREG/IA-00100, June.

Choi, C. J., C. H. Ban, and S. Y. Lee, 1992, *ICAP Assessment of RELAP5/MOD3 Cycle 5M5 with LOFT Large Break Experiment L2-5*, Korea Atomic Energy Research Institute.

Christensen, H., 1961, *Power-to-Void Transfer Function*, ANL-6385.

Chung, B. D., H. J. Kim, and Y. J. Lee, 1990, *Assessment of RELAP5/MOD2 Code Using Loss of Offsite Power Transient Data of KNU No. 1 Plant*, NUREG/IA-0030, April.

Collier, J. G., 1972, *Convection Boiling and Condensation*, McGraw-Hill Book Co.

Committee on the Safety of Nuclear Installations, 1987, *CSNI Code Validation Matrix of Thermo-Hydraulic Codes for LWR LOCA and Transients*, CSNI Report 132, March.

Crespo, J. L., R. A. Fernandez, 1991, *Assessment of MSIV Full Closure for Santa Maria de Garona NPP Using TRAC BF1 (G1J1)*, ICSP-GA-MSIV-T, June.

Croxford, M. G., and P. C. Hall, 1989, *Analysis of the THETIS Boildown Experiments Using RELAP5/MOD2*, NUREG/IA-0014, July.

Croxford, M. G., C. Harwood, and P. C. Hall, 1992, *RELAP5/MOD2 Calculation of OECD LOFT Test LP-FW-01*, NUREG/IA-0063, April.

Curca-Tivig, F., 1991, *Assessment of RELAP5/MOD3/V5M5 Against the UPTF Test No. 11 (Countercurrent Flow in PWR Hot Leg)*, KWU E412/91/E1002, March.

De Vlaminck, M., P. Deschutter, and L. Vanhoenacker, 1992, *Assessment Study of RELAP5/MOD2 Cycle 36.05 Based on the Doel-4 Reactor Trip of November 22nd, 1985*, NUREG/IA-0051, March.

Driskell, W. E., and R. G. Hanson, 1989, "Summary of ICAP Assessments of RELAP5/MOD2," *Nuclear Safety*, Vol. 30, No. 3, July-September.

Egen, R., D. A. Dingee, and J. W. Chastain, 1973, *Vapor Formation and Behavior in Boiling Heat Transfer*, BMI-1163.

EPRI Valve Test Program Staff, 1982, *EPRI PWR Safety and Relief Valve Test Program Safety and Relief Valve Test Report*, NP-2628-SR, December.

Eriksson, J., 1986, *Assessment of RELAP5/MOD2, Cycle 36, Against FIX-II Split Break Experiment No. 3027*, NUREG/IA-0005, September.

Eriksson, J., 1989a, *Assessment of RELAP5/MOD2, Cycle 36.04 Against FIX-II Guillotine Break Experiment No. 5061*, NUREG/IA-0016, July.

Eriksson, J., 1989b, *Assessment of RELAP5/MOD2, Cycle 36.04 Against FIX-II Split Break Experiment No. 3051*, NUREG/IA-0029, October.

Eriksson, J., 1990, *Assessment of RELAP5/MOD2, Cycle 36.04 Against LOFT Small Break Experiment L3-6*, NUREG/IA-0033, July.

Eriksson, J., 1992, *Assessment of RELAP5/MOD2, Cycle 36.04 Against LOFT Small Break Experiment L3-5*, NUREG/IA-0037, March.

Fletcher, C. D., and R. R. Schultz, 1992, *RELAP5/MOD3 Code Manual Volume 5 - User's Guidelines*, NUREG/CR-5535, January.

Gerth, G., 1992, *Assessment Study of RELAP5/MOD2 Cycle 36.04 Based on the Commissioning Test Reactor Trip at Full Load at the Phillipsburg 2 Nuclear Power Plant*, NUREG/IA-0057, April.

Giles, M. M., et al., 1992, *TRAC-BF1/MOD1: An Advanced Best-Estimate Computer Program for BWR Accident Analysis - User's Guide, Volume 2*, NUREG/CR-4356, June.

Grush, W. H., M. Tanaka, and P. Marsili, 1984, *Best Estimate Prediction for the OECD LOFT Project Small Cold Leg Break Experiment LP-SB-3*, OECD LOFT-T-3603, February.

Guntay, S., et al., 1985, *Best Estimate Prediction for OECD LOFT Project Fission Product Experiment LP-FP-2*, OECD LOFT-T-3803, June.

Guntay, G., 1990, *RELAP5/MOD2 Assessment: OECD/LOFT Small Break Experiment LP-SB-3*, NUREG/IA-0018, April.

Hall, P. C., 1990, *RELAP5/MOD2 Calculations of OECD LOFT Test LP-SB-02*, NUREG/IA-0021, April.

Hall, P. C., and G. Brown, 1990, *RELAP5/MOD2 Calculations of OECD LOFT Test LP-SB-01*, NUREG/IA-0012, January.

Hall, P. C., and D. R. Bull, 1992, *Analysis of Semiscale Test S-LH-1 Using RELAP5/MOD2*, NUREG/IA-0064, April.

Harwood, C., and G. Brown, 1990, *RELAP5/MOD2 Calculation of OECD LOFT Test LP-SB-03*, NUREG/IA-0013, January.

Hyvarinen, J., and T. Kervinen, 1992, *Assessment of RELAP5/MOD2 Against Natural Circulation Experiments Performed with the REWET-III Facility*, NUREG/IA-0059, April.

Kao, L., K. S. Liang, J. L. Chiou, L. Y. Liao, S. F. Wang, and Y. B. Chen, 1992, *Assessment of RELAP5/MOD2 Using LOCE Large Break Loss-of-Coolant Experiment L2-5*, NUREG/IA-0045, April.

Keevill, M. B., 1992, *RELAP5/MOD2 Analysis of LOFT Experiment L9-4*, NUREG/IA-0066, April.

Kukita, Y., Y. Anoda, H. Nakamura, and K. Tasaka, 1987, "Assessment and Improvement of RELAP5/MOD2 Code's Interphase Drag Models," *AIChE Symposium Series, Heat Transfer-Pittsburgh 1987*, No. 257, Volume 83.

Langerman, M. A., 1982, *Application of RELAP5/MOD1 for Calculation of Safety and Relief Valve Discharge Piping Hydrodynamic Loads*, EPRI NP-2479-LD, July.

Lee, E. J., B. D. Chung, and H. J. Kim, 1990, *ICAP Assessment of RELAP5/MOD2, Cycle 36.05 Against LOFT Small Break Experiment L3-7*, NUREG/IA-0031, April.

Lee, E. J., B. D. Chung, and H. J. Kim, 1992, *ICAP Assessment of RELAP5/MOD2 Cycle 36.04 Using LOFT Intermediate Break Experiment L5-1*, NUREG/IA-0069, April.

Lee, E. J., B. D. Chung, and H. J. Kim, *RELAP5 Assessment Using Semiscale SBLOCA Test S-NH-1*, Korea Institute of Nuclear Safety (copies can be obtained from Ms. Nancy Larson, INEL).

Lee, S., B. D. Chung, and H. J. Kim, 1993, *RELAP5 Assessment Using LSTF Test Data SB-CL-18*, NUREG/IA-00095, May.

Lee, S., and H. J. Kim, 1992, *RELAP5 Assessment on Direct-Contact Condensation in Horizontal Cocurrent Stratified Flow*, NUREG/IA-0077, April.

Liang, K. S., L. Kao, J. L. Chiou, L. Y. Liao, S. F. Wang, and Y. B. Chen, 1992, *Assessment of RELAP5/MOD2 Using Semiscale Large Break Loss-of-Coolant Experiment S-06-3*, NUREG/IA-0046, April.

Loomis, G. G., and J. E. Streit, 1985, *Result of Semiscale Mod-2C Small Break (5%) LOCA Experiments S-LH-1 and S-LH-2*, NUREG/CR-4438, November.

Lubbesmeyer, D., 1992, *Post-Test Analysis and Nodalization Studies of OECD LOFT Experiment LP-02-6 with RELAP5/MOD2 Cycle 36.02*, NUREG/IA-0088, August.

Lubbesmeyer, D., *Post-Test Analysis and Nodalization Studies of OECD LOFT Experiment LP-LB-1 with RELAP5/MOD2 Cycle 36.02*, NUREG/IA-0089,(copies can be obtained from Ms. Nancy Larson, INEL).

Maciaszek, T., and A. Menponteil, 1986, "Experimental Study on Phase Separation in a Tee Junction for Steam-Water Stratified Inlet Flow," Paper C2 at the European Two-Phase Flow Group Meeting, Munich, Federal Republic of Germany, June 10-13.

Moeyaert, P., and E. Stubbe, 1988, *Assessment Study of RELAP5/MOD2 Cycle 36.04 Based on Spray Startup Test for Doel-4*, NUREG/IA-0020, July.

Nalezny, C. L., 1983, *Summary of Nuclear Regulatory Commission's LOFT Program Experiments*, NUREG/CR-3214, July.

Nilsson, Lars, 1990, *ICAP Assessment of RELAP5/MOD3 Against Twenty-Five Post-Dryout Experiments Performed at the Royal Institute of Technology*, STUDSVIK/NS-90/93, October.

Pena, J. J., S. Enciso, and F. Reventos, 1992, *Thermal-Hydraulic Post-Test Analysis of OECD-LOFT LP-FP-2 Experiment*, NUREG/IA-0049, April.

Preece, R. J., and J. M. Putney, 1991, *Preliminary Assessment of PWR Steam Generator Modelling in RELAP5/MOD3*, TEC/L/0539/R91, PWR/THSG/T(91)38, November.

Putney, J. M., and R. J. Preece, 1991, *Assessment of PWR Steam Generator Modelling in RELAP5/MOD2*, TEC/L/0471/R91, September.

Ransom, V. H., et al., 1982, *RELAP5/MOD1 Code Manual; Volume 1: System Models and Numerical Methods*, NUREG/CR-1826, March.

Ransom, V. H., et al., 1985, *RELAP5/MOD2 Code Manual; Volume 1: Code Structure, Systems Models, and Solution Methods*, NUREG/CR-4312, August.

Ransom, V. H., et al., 1987a, *RELAP5/MOD2 Code Manual Volume 2: Users Guide and Input Requirements*, NUREG/CR-4312, March.

Ransom, V. H., R. J. Wagner, and G. W. Johnsen, 1987b, *RELAP5/MOD2 Code Manual Volume 3: Developmental Assessment Problems*. EGG-TFM-7952, December.

Rhee, G. S., letter to ICAP Members, 1989, "ICAP Code Assessment Report," April 26.

Richner, M., G. Th. Analytis, and S. N. Aksan, 1992, *Assessment of RELAP5/MOD2, Cycle 36.02, Using NEPTUN Reflooding Experimental Data*, NUREG/IA-0054, August.

Riemke, R. A., 1989, *Junction-Based Interphase Drag and Vertical Stratification Modifications for RELAP5/MOD3*, EGG-EAST-8580, June.

Riemke, R. A., 1990, "RELAP5/MOD3 New Models," 1990 Joint RELAP5 and TRAC-BWR International User's Seminar, Commonwealth Edison Co., Chicago, IL, USA, September 17-21.

Rogers, J. M., 1992, *An Analysis of Semiscale MOD-2C S-FS-1 Steam Line Break Test Using RELAP5/MOD2*, NUREG/IA-0052, March.

Rosdahl, O., and D. Caraher, 1986a, *Assessment of RELAP5/MOD2 Against Marviken Jet Impingement Test 11 Level Swell*, NUREG/IA-0006, September

Rosdahl, O., and D. Caraher, 1986b, *Assessment of RELAP5/MOD2 Against Critical Flow Data from Marviken Tests JIT 11 and CFT 21*, NUREG/IA-0007, September.

Roth, P. A., and R. R. Schultz, 1991, *Analysis of Reduced Primary and Secondary Coolant Level Experiments in the BETHSY Facility Using RELAP5/MOD3*, EGG-EAST-9251, July.

Roth, P. A., C. J. Choi, and R. R. Schultz, 1992, *Analysis of Two Small Break Loss-of-Coolant Experiments in the BETHSY Facility Using RELAP5/MOD3*, EGG-NE-10353, July.

Rouel, G. P., and E. J. Stubbe, 1992, *Assessment Study of RELAP5/MOD2 Cycle 36.05 Based on the Tihange-2 Reactor Trip of January 11, 1983*, NUREG/IA-0044, March.

Schneider, K., 1991, *Assessment of RELAP5/MOD3/V5M5 Against Upper Plenum Test Facility Test No. 6, Run 131, 132, 133, 135, and 136 Downcomer Countercurrent Flow*, April.

Schultz, R. R., and L. Ericson, 1981, "The Marviken Critical Flow Test Program," *Nuclear Safety*, Vol. 22, No. 6, November-December.

Schultz, R. R., 1990, "TRAC-BWR Updates - Fiscal Year 1990," Presentation at the 1990 ICAP Management Meeting, October.

Schultz, R. R., et al., 1991, *An Investigation of Core Liquid Level Depression in Small Break Loss-of-Coolant Accidents*, NUREG/CR-4063, EGG-2636, August.

Scriven, A. H., 1992a, *Analysis of LOBI Test BL02 (Three Percent Cold Leg Break) with RELAP5 Code*, NUREG/IA-0036, March.

Scriven, A. H., 1992b, *Application of the RELAP5/MOD2 Code to the LOFT Tests L3-5 and L3-6*, NUREG/IA-0060, April.

Scriven, A. H., 1992c, *Pre- and Post- Test Analysis of LOBI MOD2 Test ST-02 (BT-00) with RELAP5/MOD1 and MOD2 (Loss of Feedwater)*, NUREG/IA-0061, April.

Shrock, V. E., S. T. Revankar, R. Mannheiner, and C. H. Wang, 1986, *Small Break Critical Discharge - the Roles of Vapor and Liquid Entrainment in a Stratified Two-Phase Region Upstream of the Break*, NUREG/CR-4761, December.

Shumway, R. W., et al., 1984, *TRAC-BD1/MOD1: An Advanced Best Estimate Computer Program for Boiling Water Reactor Loss-of-Coolant Accident Analysis, Volume 2: Users Guide*, NUREG/CR-3633, April.

Shumway, R. W., et al., 1985, *TRAC-BD1/MOD1: An Advanced Best Estimate Computer Program for Boiling Water Reactor Loss-of-Coolant Accident Analysis, Volume 4: Development Assessment*, NUREG/CR-3633, August.

Singer, G. L., et al., 1984, *TRAC-BD1/MOD1: An Advanced Best Estimate Computer Program for Boiling Water Reactor Loss-of-Coolant Accident Analysis, Volume 3: Code Structure and Programming Information*, NUREG/CR-3633, April.

Sjoberg, A., and D. Caraher, 1986, *Assessment of RELAP5/MOD2 Against 25 Dryout Experiments Conducted at the Royal Institute of Technology*, NUREG/IA-0009, October.

Slater, C. E., 1992, letter to D. E. Solberg, "Resolution of NRC Analyses," April 28.

Smoglie, C., 1984, *Two-Phase Flow Through Small Branches in Horizontal Pipe with Stratified Flow*, KfK Report KfK 3861, December.

Stubbe, E. J., 1986, *Assessment Study of RELAP5/MOD2 Cycle 36.01: Based on the Doel-2 Steam Generator Tube Rupture Incident of June, 1979*, NUREG/IA-0008, October.

Stubbe, E. J., and L. Vanhoenacker, 1990, *Assessment Study of RELAP5/MOD2 Cycle 36.04 Based on Pressurizer Safety and Relief Valve Tests*, NUREG/IA-0034, July.

Stubbe, E. J., L. Vanhoenacker, and R. Otero, 1990, *RELAP5/MOD3 Assessment for Calculation of Safety and Relief Valve Discharge Piping Hydrodynamic Loads*, Tractebel, October.

Stubbe, E. J., and P. Deschutter, 1992, *Assessment Study of RELAP5/MOD2 Cycle 36.04 Based on the Doel-4 Manual Loss of Load Test of November 23rd 1985*, NUREG/IA-0043, March.

Taitel, Y., and A. E. Dukler, 1976, *AIChE Journal*, 22(1), 47.

Task Group on the Status and Assessment of Codes for Transients and ECCS, 1987, *CSNI Code Validation Matrix of Thermo-Hydraulic Codes for LWR LOCA and Transients*, CSNI Report 132, March.

Taylor, D. D., et al., 1984, *TRAC-BD1/MOD1: An Advanced Best Estimate Computer Program for Boiling Water Reactor Transient Analysis, Volume 1: Model Description*, NUREG/CR-3633, April.

Weaver, W. L., R. W. Shumway, G. L. Singer, and S. Z. Rouhani, 1986, *TRAC-BF1 Manual: Extensions to TRAC-BD1/MOD1*, NUREG/CR-4391, August.

Weaver, W. L., R. A. Riemke, R. J. Wagner, and G. W. Johnsen, 1989, "The RELAP5/MOD3 Code for PWR Safety Analysis," *Proceedings of the Fourth International Topical Meeting on Nuclear Reactor Thermal-Hydraulics*, Karlsruhe, Federal Republic of Germany, October.

Wheatley, P. D., M. A. Bolander, R. Chambers, J. M. Cozzuol, C. B. Davis, J. L. Steiner, 1985, *RELAP5/MOD2 Code Assessment at the Idaho National Engineering Laboratory*, NUREG/CR-1454, November.

U. S. Regulatory Commission, 1987, *Guidelines and Procedures for the International Code Assessment and Applications Program*, NUREG-1271, April.

Yrjola, V., 1992, *Assessment of RELAP5/MOD2 Cycle 36.04 Against the Loviisa-2 Stuck-Open Turbine Bypass Valve Transient on September 1, 1981*, NUREG/IA-0047, March.

NRC FORM 335
(2-89)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/IA-0128
EGG-EAST-8719

2. TITLE AND SUBTITLE

International Code Assessment and Applications Program: Summary of Code Assessment Studies Concerning RELAP5/MOD2, RELAP5/MOD3, and TRAC-B

3. DATE REPORT PUBLISHED

MONTH	YEAR
December	1993

4. FIN OR GRANT NUMBER
L2245

5. AUTHOR(S)
R. R. Schultz

6. TYPE OF REPORT

7. PERIOD COVERED *(Inclusive Dates)*
Oct 1989 to Sept 1992

8. PERFORMING ORGANIZATION – NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Idaho National Engineering Laboratory
EG&G Idaho, Inc.
Idaho Falls, ID 83415

9. SPONSORING ORGANIZATION – NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

Members of the International Code Assessment Program (ICAP) have assessed the U. S. Nuclear Regulatory Commission (USNRC) advanced thermal-hydraulic codes over the past few years in a concerted effort to identify deficiencies, to define user guidelines, and to determine the state of each code. The results of sixty-two code assessment reviews, conducted at INEL, are summarized. Code deficiencies are discussed and user recommended nodalizations investigated during the course of conducting the assessment studies and reviews are listed. All the work that is summarized was done using the RELAP5/MOD2, RELAP5/MOD3, and TRAC-B codes.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

ICAP, assessment, code, RELAP5/MOD2, RELAP5/MOD3, TRAC-B

13. AVAILABILITY STATEMENT
Unlimited

14. SECURITY CLASSIFICATION
(This Page)
Unclassified
(This Report)
Unclassified

15. NUMBER OF PAGES

16. PRICE

NRC FORM 335 (2-89)

Printed on recycled paper

Federal Recycling Program

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

SPECIAL FOURTH CLASS RATE
POSTAGE AND FEES PAID
USNRC
PERMIT NO. G-67

120555139531 1 1AN1CI1R4
US NRC-OADM
DIV FOIA & PUBLICATIONS SVCS
TPS-PDR-NUREG
P-211
WASHINGTON DC 20555