SAND92-1273
Unlimited Release
3rd Draft, April 1994

Survey of MELCOR Assessment and Selected Applications

L. N. Kmetyk
Thermal/Hydraulic Analysis
Department 6418, MS-0745
Sandia National Laboratories
Albuquerque, NM USA 87185-0745

Note

This draft report is compiled and distributed by me as an information exchange with other MELCOR users. This activity is not funded by the NRC under MELCOR development and assessment at Sandia, and is not funded under or considered part of the MCAP program run by LANL.

The purpose of this draft report is to keep MELCOR users informed of what other users have done or are doing with MELCOR. Both published information and draft information are included, as available, with extensive references.

You are getting this because you have contributed material in the past, because you have expressed interest in the past, and/or because you are a new MELCOR user.

This draft report is updated and distributed on no fixed schedule, but whenever I have accumulated enough material. To date, it has been reissued about once a year. I will update and reissue again if and when I receive enough new material.

If you have anything you would be willing to share with other users, please send it to me. I will not copy or redistribute any draft, proprietary or confidential reports; I will simply summarize them so other users interested would know what has been done and whom to contact for more information.

Abstract

MELCOR is a fully-integrated, engineering-level computer code that models the progression of severe accidents in light water reactor nuclear power plants, being developed at Sandia National Laboratories for the U. S. Nuclear Regulatory Commission and the U. S. Department of Energy. The entire spectrum of severe accident phenomena, including reactor coolant system and containment thermal/hydraulic response, core heatup, degradation and relocation, and fission product release and transport, is treated in MELCOR in a unified framework for both BWRs and PWRs. The MELCOR computer code has been developed to the point that it is now being successfully applied in severe accident analyses, particularly in PRA studies.

This report presents a review of MELCOR verification, validation and assessment to date, both completed and underway. This review reveals that most of the severe accident phenomena modelled by MELCOR have received or are receiving some evaluation, primarily through assessment against experimental data. However, in many of these areas, the assessment to date does not cover all phenomena of interest, or is based on a limited number of experiments and analyses which may be insufficient to cover the scales of interest and which may be insufficient to allow identification of experiment-specific problems *vs* generic code problems and deficiencies. Furthermore, there has been no assessment yet at all of MELCOR for some phenomena, as identified here.

There is no experiment (not even the TMI accident) which represents all features of a severe accident, and only the TMI accident is at full, plant scale. It is therefore necessary for severe accident codes to supplement standard assessment against experiment (and against simple problems with analytic or otherwise obvious solutions) with plant calculations that cannot be fully verified, but that can be judged against expert opinion for reasonableness and internal self-consistency (particularly using sensitivity studies) and also can be compared to other code calculations for consistency. A number of plant analyses have been done with MELCOR, with sensitivity studies and/or code-to-code comparisons.

Contents

1 Introduction

MELCOR [1] is a fully-integrated, engineering-level computer code that models the progression of severe accidents in light water reactor nuclear power plants, being developed at Sandia National Laboratories for the U. S. Nuclear Regulatory Commission (USNRC) and the U. S. Department of Energy (USDOE). The entire spectrum of severe accident phenomena, including reactor coolant system and containment thermal/hydraulic response, core heatup, degradation and relocation, and fission product release and transport, is treated in MELCOR in a unified framework for both boiling water reactors and pressurized water reactors. The MELCOR computer code has been developed to the point that it is now being successfully applied in severe accident analyses.

Some limited technical assessment activities have been performed to date and a number of assessment calculations now are being done for the NRC. The available MELCOR assessment was surveyed in early 1990, as part of the MELCOR Peer Review process [2] and as part of developing a comprehensive multi-year, multi-facility assessment plan [3]. Both the MELCOR peer review and the NRC recognized the need to undertake a more comprehensive and more systematic program of MELCOR assessment.

However, there is now also a need to again review and evaluate the available MELCOR assessment done to date, updating and expanding those previous surveys to include recent and non-NRC analyses. This survey of existing assessment will be combined with a review of NPR *vs* LWR severe accident phenomena, and with a review of available NPR-specific experimental data and other calculations, to develop a comprehensive plan for verification and validation of the MELCOR/NPR code being developed for the U. S. Department of Energy (USDOE) for analyzing the New Production Reactor (NPR).

A MELCOR verification and validation ("V&V") program was funded at Sandia in 1985-1986 [4]. That limited effort primarily involved containment phenomena, because of the data available in that area and because of the synergism with the CONTAIN code development effort at Sandia. Results from MELCOR 1.0, 1.5.0 and 1.6.0 were compared with experimental data, with more mechanistic codes and with analytical solutions. Problems analyzed during the 1986 V&V program are described in Section 2.

A separate MELCOR assessment program also has been underway at Brookhaven over the past few years, and is currently ongoing. Unlike the Sandia or UKAEA programs, the BNL effort concentrates on assessment of in-core damage phenomena using tests PBF SFD 1-1 [5], PBF SFD 1-4 [6], NRU FLHT-2 [7, 8], NRU FLHT-4 [9] and NRU FLHT-5 [10]. It also includes plant analyses for the Peach Bottom BWR [11, 12], Zion, a 4-loop Westinghouse PWR [13], Oconee, a B&W PWR plant [14, 15, 16], and Calvert Cliffs, a CE PWR plant [17], including comparison to other code calculations. Problems analyzed by BNL are discussed in Section 3.

MELCOR has been used by Sandia to participate in the TMI-2 [18] plant accident standard problem, and HDR T31.5 (ISP-23) [19] hydrogen mixing and PHEBUS B9+ (ISP-28) [20, 21] core damage standard problem exercises. MELCOR calculations are currently being submitted for the CORA 13 (ISP-31) [22, 23, 24] core damage standard

problem exercise. However, the TMI-2 plant transient and its available data are incomplete and open to various interpretation, while some individual features of the PHEBUS and CORA test facilities could not be modelled with the baseline MELCOR code. The Sandia standard-problem analyses are described in Section 4.

The control-volume method for calculating containment thermal/hydraulics during severe accidents has been assessed by the United Kingdom Atomic Energy Agency (UKAEA) by comparing results obtained from the MELCOR code [25, 26, 27] against the BMC-F2 and HDR E11.2 tests, two experiments performed in large-scale, multi-compartmented facilities, as summarized in Section 5. These calculations were done as part of international benchmark exercises organized by the Commission of European Communities (CEC) and the Organization for Economic Cooperation and Development Committee on the Safety of Nuclear Installations (OECD/CSNI), respectively, with HDR E11.2 being ISP-29. AEA Technology at Winfrith Technology Centre are assessing MELCOR. A major part of this assessment was examining the performance of the code in plant calculations, in particular for the TMLB' sequence with and without surge line failure [28], as described in Section 6.

Section 7 summarizes MELCOR calculations done for the DEMONA F2 aerosol transport experiment [29], the PHEBUS B9+ core damage international standard problem [30], and the CEC thermal-hydraulic benchmark exercise for the Battelle Model Containment (BMC) FIPLOC verification experiment F2 [31], all from the Catedra de Tecnologia Nuclear, Universidad Politecnica de Madrid; MELCOR calculations have also been done for the FALCON fission product release and transport international standard problem ISP-34 [32], and for the core heatup and degradation phase of the first Phebus-FP fission product release and transport test, FPT-0 [33]. In addition, three accident sequences (AB, V, and SGTR) have been done for the Ascó II plant, a 3-loop Westinghouse PWR [34, 35, 36, 37]; two station blackout sequences in the Garoña plant, a GE BWR/3 with a Mark I containment, have also been done [38].

Writeups on two MELCOR assessment exercises, forming Section 8, have been contributed by Edo Velema of the Netherlands Energy Research Foundation, Energieonderzoek Centrum Nederland (ECN). MELCOR has been used by ECN mainly to analyze severe accidents for the General Electric ABWR and SBWR designs. As part of this effort, MELCOR 1.8 calculations were done to validate the MELCOR steam condensation models, in the presence of noncondensable gases. [39] The experiment used was a small-scale experiment performed at the University of California at Berkeley. A comparison of the temperature distribution in irradiated capsules calculated with MELCOR *versus* an analytical model was performed, also.

MELCOR is being used in the Nuclear Power Engineering Center of the Japan Institute of Nuclear Safety (NUPEC/JINS) as a second generation code for once-through analysis of light water reactor severe accidents, used to improve the accuracy of containment event tree analysis and source term analysis in level 2 PSAs for Japanese light water reactors. Calculations for both experimental analysis and plant analysis have been performed, as summarized in Section 9. Preliminary calculations for experimental analysis and plant analysis have been performed using MELCOR 1.8.0, including core degradation

calculations for the Phebus-FPT0 experiment [40], and calculations of two Peach Pottom BWR plant severe accident sequences [40]. More recently, a number of calculations have been done at NUPEC with MELCOR 1.8.1 [41], including numeric studies on machine dependencies and time step effects [42] (repeated with MELCOR 1.8.2 for direct comparison [43]), analysis of NUPEC's hydrogen mixing and distribution tests M-4-3 [44] and M-7-1 (ISP-35) [45, ?, 46], containment calculations for Phebus-FP test FPT-1 [47, 33], and a number of PWR [48] and BWR [49] plant sequence analyses in support of PSA studies.

MELCOR 1.8.2 calculations for NUPEC's hydrogen mixing and distribution test M-7-1 (ISP-35) have also been performed by Tractebel Energy Engineering (TEE) [50, 51], as described in Section 10.

A Level-3 probabilistic risk assessment (PRA) was done for N Reactor, a USDOE production reactor, with phenomenological supporting calculations performed with HECTR and MELCOR [52]. In order to ensure that the codes and the input adequately modelled N Reactor, a number of benchmarking calculations (discussed in Section 11) were performed. The purpose of the benchmarking exercises was to demonstrate that MELCOR could perform acceptable source term calculations for N Reactor accident sequences. Each of the benchmark calculations performed was intended to exercise a particular model or section of the code, and these separate effects calculations helped develop confidence that the models work as intended; with the processes represented by these calculations "proven", it could then be assumed that integral calculations would be essentially correct.

MELCOR was used to perform independent safety calculations for two proposed SP-100 space reactors designs [53], as described in Section 12. It proved possible to model and analyze simple pressure and temperature excursions for litium coolant with the existing code. This successful application to space reactors helps demonstrate the code's worth as a flexible analysis tool.

Section 13 summarizes the results of several simple, well-characterized problems done by Dennis Liles as part of the MELCOR Peer Review [2]. Demonstration calculations for station blackout scenarios in a typical PWR and BWR were also done and presented by Sandia staff as part of the Peer Review process, also discussed in Section 13.

A number of assessment calculations have been done at Sandia since the Peer Review as part of a quality control and technical assessment program, including some repeats of analyses done in the earlier assessment effort [54]. These are summarized in Section 14. That program at Sandia concentrated on PWR primary system response, analyzing the FLECHT SEASET natural circulation tests [55] and the OECD LOFT integral severe accident experiment LP-FP-2 [56], and on fission product and aerosol release and deposition, analyzing the LACE LA4 containment-geometry aerosol deposition test [57] and the ACRR ST-1/ST-2 in-pile source term experiments done at Sandia [58].

The MELCOR 1.8.1 code has been used at the Atomic Energy Institute in Hungary to simulate the PMK bleed-and-feed experiments done in a scale-model VVER-440 test facility [59], with comparison to results from corresponding RELAP5/MOD2 calculations, as summarized in Section 15.

MELCOR has been used at Sandia in a number of PRA applications, as described briefly in Section 16. In the NUREG-1150 study [60] reassessing risk at five plants, MELCOR was used to perform containment response calculations [61]. In the phenomenology and risk uncertainty evaluation program (PRUEP), MELCOR calculations were performed as part of an integrated risk assessment for the LaSalle plant [62]. MELCOR calculations have been done updating the source term for three accident sequences (AG, S2D and S3D) in the Surry plant [63]. MELCOR also has been used extensively in a program assessing risk during low power and shutdown modes of operation at the Grand Gulf plant [64] (with Brookhaven performing a parallel study for a PWR [65]).

SCDAP/RELAP5 calculations of natural circulation in the Surry TMLB' accident scenario [66] were independently reviewed and assessed by Sandia [134]. A number of identified uncertainties were examined by building a corresponding MELCOR model of the Surry plant and performing sensitivity studies with MELCOR on several modelling parameters, as described in Section 17.

MELCOR has been used as a severe accident analysis tool for several Oak Ridge programs. MELCOR has been validated by ORNL as part of the High Flux Isotope Reactor (HFIR) Safety Analysis Report (SAR) quality assurance program, before using MELCOR as the primary analysis tool for their Chapter-15 design-basis accident analyses. Problems analyzed during the ORNL V&V effort [68] are discussed in Section 18.1. As part of a focused severe accident study for the Advanced Neutron Source (ANS) Conceptual Safety Analysis Report (CSAR), MELCOR is being used at Oak Ridge to predict the transport of fission product nuclides and their release from containment [69], as summarized in Section 18.2. ORNL has also completed a MELCOR analysis characterizing the severe accident source term for a low-pressure, short-term station blackout sequence in a BWR-4 [70], as described in Section 18.3. A detailed assessment of the MELCOR Radionuclide (RN) Package's fuel fission product release models has been performed at ORNL via simulation of ORNL's VI-3, VI-5, and VI-6 fuel fission product release tests, and comparison of MELCOR's predicted fission product release behavior with that observed in the tests, as summarized in Section 18.4. Section 18.5 describes work on a projects to prepare a fully qualified, best-estimate MELCOR deck for the Grand Gulf facility; duplicate a short-term station blackout sequence with the deck used for NUREG-1150, and the QAed deck; and to compare the results of the two analyses.

A MAAP/MELCOR comparison for the Zion plant has been completed, but we have not been able to obtain the final report or any other information on the results of this study, so it is noted as existing but not included in this summary.

Mr. Hidaka of JAERI has very kindly sent us a conference paper and a more detailed, supporting internal memorandum [71, 72] on a comparative study of source terms in a BWR severe accident as predicted by THALES-2, the Source Term Code Package (STCP), and MELCOR. A summary of this conference paper is presented in Section 19.

MELCOR calculations have been done for two plant scenarios in the Teollisuuden Voima Oy (TVO Power Company) nuclear power plant, including a MAAP/MELCOR comparison study with the MAAP runs done by TVO and the MELCOR runs done by

Valtion Teknillinen Tutkimuskeskus (VTT), the Technical Research Centre of Finland. These analyses began using MELCOR 1.8.0 [73] and continued using MELCOR 1.8.1 [74], for the thermal/hydraulic aspects of the accidents; more recently, MELCOR 1.8.2 has been used to expand the TVO plant analyses to include fission product behavior in two accident scenarios [75]. In addition, an initial station blackout with a 10% break in the main steam line with recovery of power and reflooding of the overheated reactor core with auxiliary feedwater system has been analyzed for the TVO plant using the MAAP, MELCOR and SCDAP/RELAP5/MOD3 computer codes [76]. The results are described briefly in Section 20.

Msrs. Schmocker and Isaak prepared a summary paper of MELCOR experience at HSK (Hauptabteilung für die Sicherheit der Kernanlagen, the Swiss Federal Nuclear Safety Inspectorate) especially for this survey report [77]. The extensive set of plant analyses done include a number of sensitivity studies and a MELCOR/MAAP comparison. Their contribution is given almost verbatim in Section 21.

Section 22 gives results of a MAAP/MELCOR comparison study for the Point Beach plant just completed as a master's thesis at the University of Wisconsin [78].

Some additional MELCOR assessment calculations are being done currently for the NRC under the MELCOR development project. Completed analyses include Marviken-V aerosol transport tests ATT-2b and ATT-4 [79], PNL ice condenser tests 11-6 and 16-11 [80], the SNL and ANL IET direct containment heating (DCH) tests [81], ACRR early-phase core damaged fuel test DF-4 [82] and ACRR late-phase core melt progression tests MP-1 and MP-2 [83], a TMLB' station blackout analysis for the Surry plant, comparing results from MELCOR 1.8.2 with results from MELCOR 1.8.1 for the same transient [84], and the GE large vessel blowdown and level swell tests [85]. Ongoing calculations include the SURC-2 core-concrete interaction test and the CSE containment spray experiments. These will be summarized in Section 23. A number of MELCOR benchmark problems are being collected, updated, rerun, and documented [86, 87]; these will be summarized in Section 24.

Several sets of MELCOR calculations [88] have been completed studying the effects of air ingression on the consequences of various severe accident scenarios, as described in Section 25. One set of calculations analyzed a station blackout with surge line failure prior to vessel breach, starting from nominal operating conditions; the other set of calculations analyzed a station blackout occurring during shutdown (refueling) conditions. Both sets of analyses were for the Surry plant, a three-loop Westinghouse PWR. For both accident scenarios, a basecase calculation was done, and then repeated with various amounts of air ingression from containment into the core region following core degradation and vessel failure. In addition to the two sets of analyses done for the Surry PWR, a similar air-ingression sensitivity study was done as part of a low-power/shutdown PRA; that PRA study also analyzed a station blackout occurring during shutdown (refueling) conditions, but for the Grand Gulf plant, a BWR/6 with Mark III containment.

Section 26 discusses a number of MELCOR calculations which have been done for severe accident sequences in the ABWR and the results compared with MAAP calculations for the same sequences [89].

MELCOR is being used by a number of groups to model VVER nuclear power plants, as already noted in Section 15, even though the code models are not all readily applicable to the VVER design and even though there has been no development of MELCOR for VVER phenomenology. MELCOR is being used in Russia to model a VVER-440/213 reactor and plant [258], described in Section 27.

NRC has funded several MELCOR assessment activities at Energy Research, Inc., including a review of the existing heat and mass transfer correlations in MELCOR including identification of potential heat and mass transfer correlations for inclusion in the MELCOR code [90], sensitivity studies varying heat and mass transfer correlations in plant calculations for selected accident scenarios (a station blackout and a LBLOCA in the Surry plant) [91], and calculations for FIST BWR thermal/hydraulic experiments 6SB2C and T1QUV [92]. Some results of this work are summarized in Section 28.

The results of this review of MELCOR verification, validation and assessment to date are summarized in Section 30.

2 1986 V&V Program

A MELCOR verification and validation ("V&V") program was funded at Sandia in 1985-1986. That limited effort primarily involved containment phenomena, because of the data available in that area and because of the synergism with the CONTAIN [93] and HECTR [94] code development efforts at Sandia. Results from MELCOR 1.0, 1.5.0 and 1.6.0 were compared with experimental data, with more mechanistic codes and with analytical solutions [4]. The comparisons that were made with experimental data all were done by simply converting CONTAIN and/or HECTR input models into MELCOR models, rather than by developing MELCOR models directly from facility information. Because modelling conventions and guidelines appropriate for those other codes cannot guarantee always to produce the best possible MELCOR model, those analyses may not have adequately evaluated the modelling potentials of MELCOR. Furthermore, all those calculations were done with old versions of MELCOR, and it is not known whether any of the results would change using more current code versions.

2.1 Adiabatic Expansion of Hydrogen, Two-Cell Flow

MELCOR 1.6 calculations for the adiabatic flow of hydrogen between two control volumes were performed, and the results compared to an exact analytic solution for an ideal gas. Six cases were considered, varying the initial conditions, control volumes sizes and flow path parameters over a wide range. The MELCOR results differ only slightly from the analytic solution. The differences are caused by the use of a temperature-dependent heat capacity in MELCOR, which introduces some deviation from the ideal gas assumptions.

2.2 Saturated Liquid Depressurization Test

The analysis of severe accidents involves predicting the depressurization of the reactor vessel into its containment; for some accident sequences, the reactor vessel contains significant quantities of high-pressure, high-temperature water which will undergo rapid flashing during depressurization. MELCOR's ability to predict this depressurization is tested using a simple model and comparing to an analytic solution obtained from mass and energy balances. The results show good agreement between MELCOR predictions and the analytical solution. The calculations were done with MELCOR 1.6 on a VAX.

2.3 Cooling of Structures in a Fluid

MELCOR 1.0 calculations were performed for the cooling of two uniform heat structures (one rectangular and one cylindrical) with constant thermal properties and heat transfer coefficients. The temperatures as a function of time were compared to an exact analytic solution and to SCDAP results [108]. The good agreement of the MELCOR

results with the SCDAP results and the exact analytic solution show that the finite-difference methods used in the HS package produce accurate results for internal heat conduction without internal or surface power sources.

2.4 Radial Conduction in Annular Structures

MELCOR 1.0 predictions of the steady-state temperature distributions resulting from radial heat conduction in annular structures were compared to an exact analytic solution for two sets of boundary conditions and two cylinder sizes; in addition, the self-initialized steady-state temperature distributions were compared to the results of a transient calculation in which a structure with an initially-uniform temperature profile and the appropriate, fixed boundary conditions is allowed to reach a steady-state temperature profile. The agreement between the MELCOR results and the analytic result is excellent in all four cases studied.

2.5 Transient Conduction in a Semi-Infinite Solid Heat Structure

MELCOR predictions have been compared to exact analytic solutions for transient heat flow in a semi-infinite solid with convective boundary conditions. This problem simulates the conduction heat transfer in thick walls, such as the concrete containment walls of a nuclear power plant during a severe accident. Comparisons were made for steel and concrete, various thermal conductivities, ambient atmospheric temperatures, noding resolutions and time steps. MELCOR results appeared to be more accurate for cases involving materials with low thermal conductivities (like concrete) rather than high thermal conductivities (like steel), although in either case the accuracy of the MELCOR results is quite acceptable, within <1% of the analytic solution for the integrated heat flux. The calculations reported were run with MELCOR 1.1; a few cases were later rerun with MELCOR 1.6 with no significant differences in results.

2.6 HDR Containment Experiment V44

MELCOR 1.6 was used to simulate the HDR steam blowdown experiment V44 [95], a reactor-scale containment test conducted by Kernforschungszentrum Karlsruhe (KfK) at the decommissioned HDR reactor facility near Frankfurt in Germany. The peak containment pressure predicted by MELCOR was ~24% higher than measured, but the longer-term calculated pressures are in good agreement with observation. The temperature in the main compartment predicted by MELCOR peaked ~20K higher than observed, but again good long-term agreement was obtained. The agreement found between MELCOR calculations and experimental results was similar to that using the CONTAIN code.

2.7 Battelle-Frankfurt Gas Mixing Tests

The Battelle-Frankfurt mixing tests [96, 97] were a series of experiments in which hydrogen-nitrogen mixtures were injected into a model containment at the Battelle Institute e. V. Frankfurt; the containment model was a concrete structure with cylindrical central regions which could be isolated from the upper and asymmetric outer compartments. MELCOR calculations were done for tests BF-2 and BF-6, where only the inner regions of the containment were used, and for tests BF-10 and BF-19, in which the inner regions could communicate with the outer compartments. These four tests had been simulated also with the RALOC [98] and HECTR [94, 99] codes, and the MELCOR results compared both to test data and to results from these other codes.

MELCOR produced generally good agreement with test data, especially for those hydrogen-mixing tests where initial temperatures were assumed uniform and very near the injected gas temperature (*i.e.*, BF-2 and BF-10). Relatively large flows were calculated for what was a zero-flow steady state, which could be eliminated by more careful selection of initial pressures, by eliminating elevation discontinuities in the model, and by using a large computer word length. A fairly large number of iterations were required to obtain good agreement between MELCOR and experiment in those cases (*i.e.*, BF-6 and BF-19) in which the initial temperatures were not uniform, as also found in the RALOC and HECTR analyses.

2.8 ABCOVE Experiments AB5, AB6 and AB7

The Aerosol Behavior Code Validation and Evaluation (ABCOVE) program was a cooperative effort between the USDOE and USNRC to validate aerosol behavior codes under the conditions found in an LMFBR containment during a severe accident. A series of validation experiments were conducted at the Containment Systems Test Facility (CSTF) at Hanford, in which single- and double-component aerosols were injected into a closed vessel. MELCOR 1.5 was used to simulate ABCOVE aerosol experiments AB5, AB6 and AB7, comparing both to test data [100, 101, 102] and to results [103] from the CONTAIN [93] code.

MELCOR results were nearly identical to the CONTAIN results. The code predictions for the suspended mass of aerosol(s) track the experimental data to the end of the experiment to within a factor of two or three (over many orders of magnitude). Results for the masses deposited by settling agree within an 11% error with the data for all three tests. In AB5, code predictions for the material deposited by plating agree with data within 12%. For the other two tests, the codes did not give good accurate results for the amounts of material deposited on the walls at the end of the test; these errors were considered probably related to turbulence in the vessel which could cause inertial impaction, which was not modelled in either code.

2.9 Browns Ferry Reactor Building Burns

MELCOR 1.6 and HECTR [94] calculations were done for hydrogen burns that could occur in the reactor building following drywell failure in postulated severe accidents at Browns Ferry. When using the same flame speed, the two codes predict similar pressure responses, although the magnitudes of the pressure increases differ because the preburn conditions are slightly different, due to different treatments of the control volume gravity head and heat transfer and condensation. Some improvements needed in MELCOR were identified based upon HECTR calculations with models absent in MELCOR, such as including radiative heat from atmosphere to surfaces and enhancing the existing spray model. (The comparison between MELCOR and HECTR also identified some errors and limitations in the HECTR code.)

3 Brookhaven Program

BNL has a program with the NRC to provide independent assessment of MELCOR as a severe-accident/source-term analysis tool. The scope of this program is to perform quality control verification on all released versions of MELCOR, to benchmark MELCOR against more mechanistic codes and experimental data from severe fuel damage tests, and to evaluate the ability of MELCOR to simulate long-term severe accident transients in commercial LWRs, by applying the code to model both BWRs and PWRs.

Over the past few years, all released versions of MELCOR have been installed and maintained on BNL's VAX mainframe. Version 1.8.1 was installed in FY92. Whereas BNL's main emphasis has been on VAX, the IBM 3090 mainframe has also been used, and currently has MELCOR version 1.8DN operational on it. BNL also intends to get into the workstation environment in the near future. As part of verification, BNL has submitted 36 defect investigation reports (DIRs) to Sandia thus far; these have served to identify code errors and deficiencies, and recommend code improvements.

In accordance with a 1988 study on experimental data alternatives for benchmarking MELCOR [104], benchmarking analyses have been carried out for the following integral severe fuel damage tests:

1. Power Burst Facility (PBF) Severe Fuel Damage (SFD) test 1-1,
2. PBF SFD test 1-4,
3. National Research Universal (NRU) Full-Length High-Temperature (FLHT) test 2,
4. NRU FLHT test 4, and
5. NRU FLHT test 5.

MELCOR has been and is being used to simulate dominant severe accidents in the following commercial LWR plants:

1. Peach Bottom (GE BWR/4, Mark-I containment, 3293Mwt),
2. Zion (Westinghouse 4-loop PWR, large dry containment, 3250Mwt),
3. Oconee (B&W 2-loop PWR, large dry containment, 2584Mwt), and
4. Calvert Cliffs (CE 2-loop PWR, large dry containment, 2570Mwt).

On NRC request, support was provided to the NRC-sponsored Peer Review Committee [2].

3.1 PBF SFD 1-1 Core Damage

MELCOR 1.7.1 calculations were done for the Power Burst Facility (PBF) Severe Fuel Damage (SFD) test 1-1 [105] performed at the Idaho National Engineering Laboratory (INEL). The SFD 1-1 test was designed to simulate the heatup and resulting fuel damage in the upper half of a PWR core at ~2-3hr after initiation of a small break accident, when the core would be approximately 75% uncovered. Results [5] analyzed included the transient two-phase interface level in the core, fuel and clad temperatures at various elevations in the fuel bundle, clad oxidation, hydrogen generation, fission product release and heat transfer to surrounding structures. These results were compared to experimental data and to predictions from STCP [106, 107] and SCDAP [108, 105].

There were a number of uncertainties due to the performance of the test, including the bundle nuclear power used for evaporating of condensed steam; failure of thermocouples above 2000K; effects on shroud thermal conductivity due to failure of the shroud inner liner leading to steam penetration into the low-density zirconia insulation; and measurement uncertainties in hydrogen generation due to oxidation of cladding. The use of a constant inlet water flow rate in the MELCOR simulation introduced a further discrepancy into the analysis. Despite this, the calculated results showed good overall agreement with test data and with SCDAP results. The simplistic clad rupture model in MELCOR predicted failure times in the neighborhood of experimentally observed values, in no worse agreement with data than predicted times from SCDAP and STCP. Fission product releases predicted using both CORSOR and CORSOR-M models in MELCOR were an order of magnitude higher than either experimental data or SCDAP analysis using the FASTGRASS model [109], possibly because the models used in MELCOR are not intended for trace-irradiated fuel. Hydrogen production predicted by MELCOR was in very good agreement with measurement.

3.2 PBF SFD 1-4 Core Damage

MELCOR 1.8 calculations were done for PBF SFD test 1-4 [110], performed at the INEL. The test consisted of a 1.3hr-long nuclear transient simulating a small-break loss-of-coolant accident without energy core coolant (S_2D) in a commercial PWR. Results [6, 111] analyzed included the transient liquid level in the test bundle, clad temperatures and shroud temperatures, clad oxidation and hydrogen generation, bundle geometry changes, fission product release and heat transfer to the bypass flow. These results were compared to experimental data and to predictions from SCDAP/RELAP5 calculations [112].

There were many sources of uncertainties in the performance of the test, such as failure of shroud inner liner and thermocouple failures, as well as measurement uncertainties in hydrogen generation, liquid level in the bundle, fission product release, inlet water flow rate and power transferred to the bypass flow. There were also several model uncertainties and simplifications in MELCOR. Despite this, in general, MELCOR calculations represented the bundle behavior during the test reasonably well, showing the same trends as SCDAP/RELAP5 calc tions and the measured data.

3.3 NRU FLHT-2 Core Damage

MELCOR 1.8DN calculations were done for the Full-Length High-Temperature (FLHT) test 2 [113], performed by Pacific Northwest Laboratory (PNL) at the National Research Universal (NRU) Reactor at Chalk River, Canada. The objectives of the test were to simulate heatup and resulting fuel damage of full-length fuel rods during a hypothetical small-break loss-of-coolant accident in a commercial PWR. Results [7, 8] analyzed included the transient liquid level in the fuel bundle, heat transfer to the bypass flow, cladding temperatures, shroud temperatures and hydrogen generation. These results were compared to experimental data and to SCDAP results [108, 114]. Several sensitivity calculations were done, varying user-input modelling and time step control parameters.

There were some measurement uncertainties in the test, causing uncertainties in fission power, heat transfer to the bypass, hydrogen generation, liquid level in the bundle and inlet water flow rate; there were also several model uncertainties and simplifications in the MELCOR analyses. However, the MELCOR calculations generally represented the bundle behavior during the test reasonably well, showing similar trends to measured data.

Both MELCOR and SCDAP appeared to underpredict the sharp temperature rise due to accelerated zircaloy oxidation. The calculated temperature peak was delayed also, but the delay (compared to test data) was much greater in the SCDAP calculation. That better agreement between MELCOR clad temperatures and test data was reflected in the total hydrogen production, also. The discrepancies with data of the MELCOR results were attributed partially to the lack of a clad ballooning model, the absence of oxidation on the inside of the clad, and the treatment of the shroud zircaloy inner liner as a heat structure which is assumed not to oxidize; however, while there is no explicit model for clad ballooning in MELCOR, the selection of a default clad failure temperature of 1173K appears justified due to its closeness to the experimentally detected value of 1200K. (As discussed in Section 4.3, the oxidation of a similar shroud zircaloy inner liner in the PHEBUS B9+ ISP-28 analysis done with MELCOR by SNL was represented through a code modification and, as discussed in Section 14.5, the oxidation of a similar shroud zircaloy inner liner in the LOFT LP-FP-2 assessment analysis done with MELCOR by SNL was studied *via* simple, input-defined bounding calculations.)

The MELCOR calculations did not predict any noticeable flow blockages anywhere in the bundle region; this agrees well with post-irradiation examination of the FLHT-2 bundle, which revealed very small area reductions due to blockage. This is in contrast to the PBF tests, with a shorter-length fuel bundle, where large, cohesive blockages were observed to form in the lowest regions of the bundle.

Sensitivity calculations showed that bundle axial nodalization affected the predicted results, with a finer nodalization giving better agreement with test data. For this full-length test, a nodalization with 20 axial segments gave better results than using 5 or 10 segments, as would be expected; the results from 20 and 30 axial levels showed very little difference.

Both the MELCOR and SCDAP calculations used a constant fission power, without considering the increase in local power as water was replaced by a steam-hydrogen mixture during the boilaway transient, and both codes underpredicted the observed test temperature behavior. Sensitivity calculations showed that using higher fission power gave higher clad temperatures and reduced the delay in the peak clad temperature. The view factor for radiation radially outward from the core cell boundary also was shown to be an important parameter.

A very large heat transfer coefficient was assumed for the heat transfer to the outside boundary because of the high mass flow rate of the bypass coolant. Sensitivity calculations on convective heat transfer coefficients between shroud and bypass flow showed that variation of this coefficient did not affect the result as long as it was high enough, because much higher heat transfer resistance existed in the insulating shroud layers.

Probably because this experiment did not involve competing and threshold phenomena, reduction of the allowed Δt_{MAX} resulted in a converged solution. Other parameters varied in sensitivity studies were power deposited to the shroud, its radial distribution among the different layers, and the radiative exchange factors for radiation axially upward from a core cell boundary and for radiation from the liquid pool to the core. Changes in these parameters had little impact on the calculated results.

3.4 NRU FLHT-4 Core Damage

MELCOR 1.8.1 calculations also were done for FLHT-4 [115], performed by PNL at the NRU reactor. The objectives of the test series were to simulate heatup and resulting fuel damage of full-length fuel rods during hypothetical loss-of-coolant accidents in commercial PWRs. Unlike the FLHT-2 test, the period of high temperature and severe damage in FLHT-4 was prolonged to assess the continuation of hydrogen production after clad melting occurred. Results [9] analyzed included the transient liquid level in the test bundle, cladding temperatures, shroud temperatures, hydrogen generation, fission product release and material relocation. These results were compared to experimental data and to SCDAP results [108, 114]. Several sensitivity calculations were done also, studying the effects of variations in maximum allowable time step size for the calculation, in critical minimum thickness of unoxidized zircaloy in cladding and steel, and in fuel release models.

In general, MELCOR calculated the bundle behavior during the test reasonably well. The results showed similar trends to the measured data and were in better agreement with data than those calculated by SCDAP. The heatup portion of the transient was predicted well.

However, significant differences in predicted and measured results were noted in the total hydrogen production, in the cladding temperature escalation time, and in material relocation. The MELCOR calculations showed severe material relocation and noticeable blockage in the lowest regions of the bundle; in contrast, post-irradiation examination of

the FLHT-4 bundle revealed very small area reductions due to blockage. (The prediction of less hydrogen production than observed was also found in the simulation of the FLHT-2 test, where there was no noticeable blockage anywhere in the bundle region, but the difference was smaller than for FLHT-4.) These differences were attributed mainly to deficiencies in the material relocation model, to the lack of an oxidation model for the shroud zircaloy inner liner, and to the lack of a clad ballooning model. However, there were some measurement uncertainties in the test which caused uncertainties in hydrogen generation, such as liquid level in the bundle, inlet water flow rate, *etc.* (As discussed in Section 4.3, the oxidation of a similar shroud zircaloy inner liner in the PHEBUS B9+ ISP-28 analysis done with MELCOR by SNL was represented through a code modification while the oxidation of a similar shroud zircaloy inner liner in the LOFT LP-FP-2 assessment analysis done with MELCOR by SNL was studied *via* simple, input-defined bounding calculations, as discussed in Section 14.5.) SCDAP also underpredicted the sharp temperature rise due to accelerated zircaloy oxidation; furthermore, the calculated temperature peak was also delayed, much more so in the SCDAP analyses than in the MELCOR results.

Progressive reductions in Δt_{MAX} led to a converged solution, in the absence of material relocation, because the rest of the experiment does not involve threshold phenomena. The critical minimum thickness of unoxidized zircaloy in cladding and steel, a user-input parameter, also had a significant effect on the calculated behavior. Reducing this parameter by a factor of 20 from its default value resulted in the zircaloy mass staying longer in the hot bundle region before relocating, thus producing more hydrogen due to reaction with steam.

Of the eleven rods used in the FLHT-4 test, ten were fresh and only one rod was three-cycle irradiated. How this was represented in the MELCOR model is not documented. The results given, for xenon and krypton release fractions, showed MELCOR overpredicting the release (using the CORSOR option) and SCDAP underpredicting the release. A sensitivity study using the CORSOR-M option showed no significant impact on final release results except for tellurium where CORSOR-M predicted a much higher release than CORSOR. (This analysis was performed and documented before SNL identified and corrected an error in the tellurium release rate oxidation adjustment, during the ACRR ST-1/ST-2 source term assessment described in Section 14.4.)

3.5 NRU FLHT-5 Core Damage

MELCOR version 1.8.2 has been used suvvessfully to simulate the FLHT-5 experiment [10]. The FLHT-5 test [116] was conducted under more severe conditions than FLHT-2 or FLHT-4 and fuel degradation occurred over a longer period of time. Post-test analyses of the test data also have been performed with the SCDAP code [117].

MELCOR-calculated results are presented for the transient liquid level in the test bundle, cladding temperatures, shroud temperatures, hydrogen generation, fission product release and material relocation. Comparisons are made with experimental data and with SCDAP calculations.

The test train was modelled as a BWR geometry, which allowed the mass of zircaloy in the shroud inner liner, carriers and clad of one unfueled rod to be modelled as a canister component, and hence participate in oxidation with steam, as in the experiment. This was a modelling change from earlier simulations which treated the test train as a PWR geometry, in which the liner, being treated as a heat structure, could not participate in oxidation. The impact of this modelling change was to increase predicted cumulative hydrogen production by about 55-60%.

MELCOR predicted the heatup and temperature escalation of the clad very well, slightly better than SCDAP. There was also an improvement over earlier MELCOR calculations of FLHT-4 (discussed above in Section 3.5). There were, however, significant differences between measured and calculated saddle temperatures. These discrepancies can be partially attributed to uncertainties in estimating the effective thermal conductivity of the shroud during the transient.

Both MELCOR and SCDAP predict early termination of autocatalytic zircaloy oxidation. This is primarily due to overprediction of zircaloy relocation to cooler regions of the bundle, where oxidation is suppressed. This results in lower predicted cumulative hydrogen produced, compared to the experiment.

There is also a period of about 250s during which MELCOR predicts virtually no hydrogen production, and this corresponds to complete blockage of the fuel channel by massive relocation of core material calculated by MELCOR. In comparison, post-test visual examination of the fuel bundle revealed rundown of molten cladding, but no massive relocation from the high temperature region to the cooler regions above the coolant pool, and no flow blockage. Hence, oxidation and hydrogen generation continued unabated in the test.

The relocated material calculated by MELCOR included the liner (modelled as canister). The experiment showed substantial oxidation but almost no relocation of the shroud liner. If oxidation of heat structures were modelled in MELCOR, the shroud liner could have been modelled as a heat structure, which is not allowed to relocate; that would have resulted in more zircaloy oxidation, less overall relocation of core material and more hydrogen production. Hence, it is strongly recommended in the conclusions of [10] that the heat structure package in MELCOR be upgraded to allow oxidation of heat structures, as is the case with SCDAP.

The results of several sensitivity calculations with MELCOR also are presented, which explore the impact on the predicted behavior of varying user-input modelling options and timestep control parameters. This latest release version of MELCOR has several new or improved models, and has corrections to mitigate numerical sensitivities; the impact of these new models is also investigated.

All sensitivity calculations used one radial ring and twenty axial segments in the active bundle region. Parameters varied include maximum allowable timestep size, material holdup parameters, refreezing heat transfer coefficient, core radiation view factors, and fission product release models. Some of the new MELCOR 1.8.2 models such as

eutectic interactions and the core boundary fluid temperature option were included in the reference calculation; these models were deactivated in sensitivity calculations.

Sensitivity calculations show a noticeable improvement in the numerical behavior of MELCOR. While there is no convergence in going to smaller values of user-specified maximum allowed time steps, there is less deviation in predicted results for different values of user-specified maximum allowed time steps than was observed with previous versions. Most other parameters have been shown to have small or negligible impact on the predicted results, the maximum deviation in the predicted total amount of hydrogen produced being, in most cases, less than $\pm 10\%$ from the base case.

An important input parameter is the core support flag, which can be used to control the predicted material relocation in MELCOR. Using a support flag set to 01 at every axial level results in less relocation, somewhat more (8%) hydrogen production and no flow blockage. This may provide some physical justification for specifying a support flag of 01 at various axial levels for benchmarking against experiments, since the predicted behavior more closely resembles the experimentally-observed relocation behavior. However, whether this justification is equally applicable to full-plant simulations with a multi-ring core model is not at all clear, and needs to be investigated.

Another possible approach to improving predicted relocation behavior and prevent the predicted formation of a complete blockage would be to model the fuel bundle with 2 or 3 rings, rather than 1 ring as done in these calculations. Experimental observation of the FLHT-5 test showed evidence of heterogeneous melting and relocation. Modelling with only 1 ring forces MELCOR to assume that all fuel rods behave the same, leading to homogeneous relocation.

3.6 Peach Bottom BWR Plant Calculation

BNL performed MELCOR calculations [11, 12] for a long-term station blackout accident sequence at Peach Bottom, a BWR-4 plant with a Mark I containment, and compared the results to Source Term Code Package (STCP) [106] calculations of the same sequence [118].

Most of the calculations were performed using MELCOR 1.8BC; however, results from more recent calculations using MELCOR 1.8CZ and 1.8DNX (DN with updates for mass inconsistencies in debris ejection to cavity) are also included in the documentation. The calculations were done on a VAX 6450 computer.

Several sensitivity studies were done also, which explored the impact of varying user-input modelling and timestep control parameters on the accident progression and release of source terms to the environment. The studies include variations in fuel release models (CORSOR and CORSOR-M, both with and without a surface/volume correction), refreezing heat transfer coefficients, debris ejection models (solid debris ejection *vs* only molten debris ejection), burn propagation parameters, and the maximum allowable timestep size (10s in the basecase, reduced to 5, 3, 2 and 1s).

Results from a number of calculations done with the release version of MELCOR 1.8.2 have been documented in a recently-added appendix. The impact of debris fall velocity in the new debris quench model added in MELCOR 1.8.2 was examined, and the high pressure station blackout sequence was calculated using ORNL's BH bottom head model, available as an option in MELCOR 1.8.2.

Most interesting is the study on the impact of varying the maximum allowable time step with the latest code version, MELCOR 1.8.2 (1.8NM). The same set of maximum allowed time step sizes was used as before. While there was no convergence of the solution for reduced time steps, there was very close agreement in the timing of key events, from gap release of fission products, to core collapse lower plenum dryout, vessel failure, drywell failure, onset of deflagrations in the reactor building and debris ejection to the cavity. In most cases, deviations in timing were limited to a few hundred seconds; earlier calculations using MELCOR 1.8DNX for the same plant transient showed much larger deviations, many as high as 10,000s. This is certainly evidence of improved numerical behavior in MELCOR 1.8.2.

3.7 Zion PWR Plant Calculation

As part of an NRC-sponsored review of the MAAP 3.0B code [13], calculations were performed for two severe accident sequences using the MAAP and MELCOR codes. The two accidents analyzed were a loss of all electric power in the Peach Bottom BWR and a small break LOCA in the Zion PWR.

The MELCOR calculations were made by BNL staff using version 1.8.0, while the MAAP calculations were carried out by Fauske and Associates (FAI) and the results forwarded to BNL for the MAAP-MELCOR comparisons.

The MELCOR calculation for the loss-of-power sequence in the Peach Bottom BWR was basically similar to the MELCOR calculations for a long-term station blackout accident sequence at Peach Bottom already discussed in Section 3.6, with a few minor changes in the model to better match the corresponding MAAP calculation. The MELCOR calculation for the small break LOCA in the Zion PWR was done specifically as a MAAP-MELCOR comparison for this MAAP review, and is not documented elsewhere.

3.8 Oconee B&W PWR Plant Calculation

MELCOR calculations have been done for two severe accident sequences (LOCA and TMLB') in the Oconee-3 nuclear power station, a B&W PWR [14, 15, 16]. Results are presented for timing of key events, thermal/hydraulic response in the reactor coolant system and containment, and environmental releases of fission products, and include comparisons with STCP calculations performed at Battelle of the same scenarios [119]. MELCOR version 1.8DNY was used for these calculations. Sensitivity studies were done varying user-input modelling parameters such as concrete type, vessel failure temperature and break location.

3.9 Calvert Cliffs CE PWR Plant Calculation

MELCOR calculations have been done for a station blackout (TMLB) in the Calvert Cliffs nuclear power station, a CE PWR with a large dry containment[17]. Results include predicted timing of key events, thermal/hydraulic response in the reactor coolant system and containment, and environmental releases of fission products. This analysis is the first done with MELCOR for a Combustion Engineering (CE) PWR.

MELCOR version 1.8.1 (released in August 1991) was used, on BNL's VAX 6450 computer.

4 Standard Problems (SNL)

Outside the formal assessment efforts, MELCOR has been used by Sandia to participate in the TMI-2 [18] plant accident standard problem exercise, and the HDR T31.5 (ISP-23) [19] hydrogen mixing and PHEBUS B9+ (ISP-28) [20, 21] core damage standard problem exercises. MELCOR calculations are currently being submitted for the CORA 13 (ISP-31) [22, 23] core damage standard problem exercise. However, the TMI-2 plant transient and its available data are incomplete and open to various interpretation, while some individual features of the PHEBUS and CORA test facilities could not be modelled with the baseline MELCOR code.

4.1 TMI Standard Problem

The first four phases of the TMI-2 standard problem [120] were analyzed with MELCOR 1.8.0 on a VAX 8700 computer [18]. The two purposes of this analysis were to perform the first MELCOR PWR calculation (with MELCOR having been used extensively for BWR plants) and to identify any PWR-specific features that were needed within MELCOR; and to allow predictions of the MELCOR models to be compared to full-scale plant data, and to the results of more mechanistic analyses, over a significant spectrum of severe accident phenomena.

The TMI-2 accident is partitioned into four distinct phases. Phase 1 covers the period from accident initiation (0min) to shutdown of the last RCS coolant pump (100min). Phase 2 (100 to 174min) begins with a core boildown, leading to core uncovery, heatup and early degradation. Phase 3 (174 to 200min) was initiated by an RCS pump transient which injected coolant into the core, followed by a continued heating of core debris already in an uncoolable geometry. Phase 4 (200-300min) was initiated by restoration of full HPI flow, leading to a recovering of the core. In phase 4, a relocation of molten core debris from the core region to the lower plenum occurred at ~225min; through this redistribution of core debris, a coolable configuration was reached and the accident progression terminated.

In Phase 1, the MELCOR predictions were in reasonable agreement with the data. The key trends in the pressure response and the inventory loss were predicted well. The lack of better quantitative agreement was attributed to simplistic treatment of the primary-to-secondary heat transfer.

In Phase 2, the MELCOR analysis was quite good, compared to available data. While the timing of some events was slightly incorrect, the general trends were predicted very well. Hydrogen production and the state of the core at the end of Phase 2 were in reasonable agreement with the estimates given in the standard problem package. Those results indicate that the core degradation modelling in MELCOR is applicable to severe accident analysis.

The Phase 3 and 4 calculations demonstrated that MELCOR is capable of handling some recovered core sequences, even if in a limited manner: more sophisticated core debris

and relocation models would have been required to correctly represent all the events in the TMI-2 accident.

One conclusion made in the TMI-2 MELCOR analysis was that the ability of a computer code such as MELCOR for prediction of severe accident progression is best early in the accident and becomes progressively less certain later in the accident, due both to the accumulation of uncertainty in calculation and through the addition of severe accident phenomena with their associated uncertainty to the calculation. The TMI-2 analyses demonstrate this principle: The Phase 1 results were predicted fairly easily, although there was some uncertainty as to what the RCS inventory would be as a function of time. The Phase 2 calculations evinced an ability to generate divergent results, due to the addition of highly nonlinear processes such as core oxidation and countercurrent limited flow in the pressurizer drain line; without the known "correct answer" of plant data from the accident to benchmark the calculations, it would be easy to generate different consequences ranging from minimal to a highly damaged core.

This analysis indicated that the ability to simulate an accident sequence is highly dependent on the code user, who must select the appropriate nodalization and provide the appropriate models for phenomena important in the accident sequence (assuming they are available in the code). The user must also decide whether to impose possible operator actions as timed events or key them off of system variables. Finally, to fully understand the possible ramifications of a severe accident, it is necessary to try to identify, explain and follow possible divergent paths in the calculation(s).

4.2 HDR T31.5 Containment Blowdown and Hydrogen Mixing – International Standard Problem 23

A series of experiments have been conducted by Kernforschungszentrum Karlsruhe (KfK) in the decommissioned Heissdampfreaktor (HDR) containment building in West Germany, to obtain data to increase the understanding of the thermal/hydraulic behavior in a large-scale multi-compartment facility resulting from severe accident design basis accident scenarios. MELCOR 1.7.1 and later 1.8 was used to predict the thermal/hydraulic conditions in the HDR facility for one of these tests. [19] In that test, T31.5, designated as International Standard Problem 23 (ISP-23), a steam source was injected into one of the HDR containment compartments to simulate a large-diameter pipe rupture or loss-of-coolant accident. The short-term containment pressurization and temperature buildup during the blowdown as well as the long-term cooling and natural convection within the containment were parameters of particular interest for this exercise. The second phase of the experiment consisted of an injection of a light gas mixture of hydrogen and helium gas to investigate hydrogen transport and mixing in a large multi-compartment containment.

Generally, the MELCOR blind calculation compared favorably with the experimental results. The pressures and temperatures were in reasonable agreement with the data and in the range of predictive capability of the variety of codes which participated in the standard problem exercise.

Open, post-test recalculations identified some areas where input modelling could be improved. Sensitivity studies showed that improvements in comparisons with data could be obtained by adjusting flow loss coefficients and convective velocities used in the heat transfer correlations. In addition, by assessing the MELCOR calculations against data and other containment analysis codes, areas where code modelling improvements may be needed were noted.

The ISP-23 calculation was run on both VAX 8700 and Cray XMP-416 computers with practically identical results.

4.3 PHEBUS B9+ Core Damage – International Standard Problem 28

MELCOR 1.8EA was used to calculate the core degradation phenomena of the PHEBUS severe fuel damage experiment B9+, which was selected as International Standard Problem 28 (ISP-28). [20, 21]

It was necessary to make special code modifications to model the PHEBUS fuel bundle configuration, because its experimental geometry is not typical of the LWR reactor core configurations that MELCOR is intended to model. The major code change required involved the heat transfer from the Zircaloy liner to the porous zirconia insulating shroud. In the PHEBUS test bundle MELCOR model, the Zircaloy liner, which is modelled as a MELCOR core structure to treat oxidation and degradation, is able to transfer heat to the surrounding insulating shroud structure by radiation only; in reality, however, the conduction losses from the liner to the highly cooled insulator are substantial. Failure to model this heat loss resulted in very high calculated bundle temperatures very early in the experiment, inconsistent with the thermal measurements. Therefore, to correctly simulate the bundle heat loss in MELCOR, an additional conduction energy heat flux was added to the radiation flux to represent the net energy transfer from the test bundle to the insulator.

(A similar situation exists in the PBF SFD and NRU FLHT core damage tests analyzed by BNL, as summarized in Section 3. BNL chose to model the zircaloy inner liner and porous zirconia insulation as a MELCOR heat structure, which correctly represents the heat transfer, but cannot oxidize, melt or relocate. As discussed in Section 14.5, the oxidation of a similar shroud zircaloy inner liner in the LOFT LP-FP-2 assessment analysis done with MELCOR by SNL was studied *via* simple, input-defined bounding calculations.)

A number of other, minor code modifications were also used, mostly involving either very small masses in the experiment relative to the reactor-scale numbers expected or involving inconsistencies in mixture material properties. (Most of these have since been implemented in the production code.)

Comparisons of the thermal behavior of the bundle during high fission power heating and oxidation phases show good agreement with the data. Sensitivity studies were done

on the effects of varying the steam injection flow rate and the bundle nuclear power within the experimental uncertainties, as well as on the insulation thermal conductivity, the radiation view factors, and the convective heat transfer coefficients. To correct for the assumption of negligible crossflow between radial rings in the core package, the fluid flow areas were repartitioned among the three core rings used to better simulate the mixing between fluid channels in the test.

Other sensitivity studies were done on parameters affecting material degradation and relocation (rather than heatup), including the minimum oxide shell needed to hold up material, the failure temperature of the clad and liner, the refreezing heat transfer coefficients, and the amounts of UO_2 and ZrO_2 carried along with candling molten clad.

4.4 CORA 13 Core Damage – International Standard Problem 31

The MELCOR 1.8.1 code was used by SNL [22, 23, 24] to simulate one of the core degradation experiments conducted in the CORA out-of-pile test facility at Gesellschaft für Reaktorsicherheit (GRS) in Germany. This test, CORA-13, was selected to be OECD ISP-31.

The experiment setup consisted of a small core bundle of PWR fuel elements that was electrically heated to temperatures > 2800K. There were three phases in the experiment: a 3000s gas preheat phase, a 1870s transient phase, and a 180s water quench phase. In this blind calculation, only initial and boundary conditions were provided.

Four subroutines were added to the standard MELCOR code to model electrical heating in the core, and the standard COR package also was modified to communicate with these additional routines. This capability has since been added to the standard MELCOR code beginning with version 1.8JD, and will be available in the new MELCOR 1.8.2 release version.

MELCOR predictions have been compared both to the experimental data and to eight other ISP-31 submittals. Temperatures in various components, energy balance, zircaloy oxidation and core blockage were all examined. In general, the MELCOR calculation compared very well to the other submittals.

Up to the point where oxidation was significant. MELCOR temperatures agreed very well with the experiment (usually to within 50K). MELCOR predicted oxidation to occur about 100s earlier and at a faster rate than observed in the experiment. Because of the more rapid oxidation calculated, the MELCOR temperatures did not agree as well with test data later in the transient as they did in the pre-oxidation time period. MELCOR also predicted a hgher temperature gradient radially than observed in the experiment.

The large oxidation spike that occurred during quench was not predicted. However, the experiment produced 210g of hydrogen while MELCOR predicted 184g, which was one of the closest predictions of the nine submittals. None of the codes did well in terms of predicting oxidation and hydrogen generation; all of the codes overpredicted hydrogen

production in the early phase and underpredicted it in the later phase, and none of the codes predicted the intensive hydrogen production during quench.

Core blockage was of the right magnitude, slightly on the high side, but material collected on the lower grid spacer at an axial location of 450mm in the experiment while in MELCOR the material collected at the 50 to 150mm location (since MELCOR does not model a grid spacer as a physical impediment to melt and debris relocation).

5 Culcheth (UK)

The control-volume method for calculating containment thermal/hydraulics during severe accidents has been assessed by the United Kingdom Atomic Energy Agency (UKAEA) by comparing results obtained from the MELCOR code against two experiments performed in large-scale, multi-compartmented facilities. [25, 26] These calculations were run with MELCOR 1.8BC on a SUN Sparc1, and were done as part of international benchmark exercises organized by the Commission of European Communities (CEC) and the Organization for Economic Cooperation and Development Committee on the Safety of Nuclear Installations (OECD/CSNI), respectively. These experiments were chosen because they are among the few relevant and well-instrumented experiments performed in large, multi-compartment facilities.

In general, the results show that there are important uncertainties associated with the accurate prediction of containment thermal/hydraulics in complex geometries by control-volume models. These include leakage rates (especially after containment failure), modelling of bidirectional and/or strongly stratified flows, resolution of sump pool thermal gradients, and flows near dead-end rooms. In particular, an appreciation of the flow conditions to be expected is required to choose an appropriate nodalization scheme and hence obtain meaningful results, without excessive detail and resulting costs.

5.1 BMC-F2 Containment Thermal/Hydraulics

The Battelle Model Containment (BMC) is a $640m^3$ containment with internal structures which subdivide the containment into rooms connected by flow paths which can be opened or closed; for the BMC-F2 experiment [121, 31], the flow paths were arranged so that the containment was divided into nine rooms. The experiment consisted of several phases. The object of the heatup phase (Phase 1), which lasted 48hr, was to establish well-defined boundary thermal/hydraulic conditions in the containment for the subsequent phases. During this phase, the containment pressure was increased by steam injection; the steam was observed to accumulate in the upper dome and then gradually to enter the lower compartments of the facility as more steam was added, maintaining a distinct, strongly-stratified air/steam interface. During Phases 2 to 4 (48hr to 75hr), measurements were taken of the convective flows resulting from air, steam and dry heat injection into various rooms of the containment.

Early calculations with MELCOR for Phase 1 of the BMC-F2 test showed that if the calculational time step was too large then incorrect results were predicted, with the flow solutions showing rapid and severe oscillatory behavior. As a consequence, stratification of the upper and lower atmosphere regions was not predicted. The problem was overcome by choosing a small enough time step, but this illustrates that the numerical solution scheme in MELCOR is not robust in some cases; in particular, the coupling of the flow solutions with the heat and mass transfer to structures was identified as requiring more detailed evaluation. (All subsequent calculations were checked to ensure the time step was small enough.)

The calculated results for the BMC-F2 test were found to be sensitive to the containment leakage, which was very uncertain. The exercise coordinators had recommended that the leakage be modelled by flow paths connecting the containment atmosphere to the external environment, specifying the location and area of the leakage flow paths. However, the results indicated that the specified areas were too large, since the containment pressure was underpredicted, and hence reduced areas had to be used for a best-estimate calculation. During Phase 1, when the flows were strongly stratified, the results for dead-end rooms were shown to be sensitive to the leakage from them (if the areas were too small, air was predicted to be trapped in them, which prevented steam from entering). During Phases 2-4, differences between the predictions and the experimental pressures, especially after air was injected into the containment, were attributed to an inadequate leakage model which meant that the air content of the containment atmosphere was not predicted to decrease rapidly enough.

The temperature stratification of the atmosphere during Phase 1 was reproduced well by MELCOR, except for the dead-end rooms as just discussed. There was no stratification of the containment atmosphere during Phases 2-4, except for the dead-end rooms, because the convective flows during these phases resulted in a well-mixed atmosphere, which was correctly predicted by MELCOR.

Despite the uncertainty over the leakage in the dead-end rooms and the effect this had on the predictions, it was found impossible to consistently predict the atmosphere composition and the depth of water that drained into the sumps of these rooms. It was concluded that this was due to an inherent shortcoming in the lumped-parameter/control-volume approach – convective flow loops can be set up within dead-end rooms and give rise to bidirectional flow at flow path junctions, which can not be predicted.

The results calculated for certain rooms in the lower regions of the containment were shown to be sensitive to the nodalization choice during Phase 1. During this phase a steam/air interface moved down the length of the inner rooms with time. One of these rooms was connected horizontally at different elevations to two central rooms. When this inner room was modelled by one control volume, the strongly stratified flow which resulted in steam flow through the higher horizontal flow path before the lower flow path was not predicted. Better results were obtained using a more refined nodalization of the inner room, subdividing it at an elevation between the two horizontal flow paths. In the subsequent phases, the flows were much larger, which resulted in the atmosphere being well-mixed, so this nodalization sensitivity was not observed.

Large temperature differences were measured in the water pools of the room sumps. These temperature differences were not predicted correctly by MELCOR, which predicted a single temperature for the water pool in each room. This caused the heat loss from the pool to underlying structures to be too large and the concrete basemat temperatures at shallow depths to be overpredicted. Better agreement with the observed pool temperature gradients might be possible with a more refined nodalization scheme.

Flow velocity predictions during Phases 2-4 were in reasonable agreement with experimental values. The predicted flow directions sometimes were correct and sometimes

not, but this made no difference to the thermal/hydraulics since the atmosphere was well-mixed at the later times.

5.2 HDR E11.2 Hydrogen Distribution – International Standard Problem 29

The HDR E11.2 experiment [122] was designed to examine the distribution of light gas throughout a containment under severe accident conditions. A small-break loss-of-coolant accident was simulated involving injection of steam and a hydrogen simulant. A significant temperature difference was observed between the upper dome and the lower compartments, and very little steam was measured in the lower regions of the containment until a later phase of the experiment during which more steam was injected at a lower location. The light gas was measured to accumulate in the upper dome, and very little was measured in the lower compartments.

Reasonable agreement with the containment pressure was obtained in blind post-test calculations. This was, however, contrary to the findings of other codes which significantly overpredicted the pressure. Subsequently, the MELCOR input deck was checked and an error was discovered in the steam enthalpy, which was about one third of the value specified by the exercise coordinators. The steam enthalpy was corrected for the ISP-29 calculations with the result that the pressure was overpredicted, much in line with the predictions of the other codes. Reasons for the pressure overprediction were investigated. Cooling of the measurement instruments was shown to have an important influence resulting in lower pressures, but not enough to explain the overprediction. Heat losses due to venting of the annular gap between the steel shell and the outer concrete containment was also investigated, but shown to have an insignificant effect on the containment pressure. It was therefore concluded that there was an error in the specified/measured boundary conditions of the experiment (and, indeed, a serious inconsistency in the specified injection steam enthalpy and experimental measurements was later discovered). The ISP-29 exercise then was suspended pending further investigation by the organisers.

As noted above, the German experimenters subsequently identified the problem and revised the specified steam enthalpies accordingly. The MELCOR calculations were repeated [27] using the same input model of the containment [25] but incorporating the revised steam enthalpy boundary condition; the sensitivity of the results to the location of instrument cooling and flow path characteristics was investigated further, also. It was concluded that:

1. The instrument cooling in the facility is an important feature which must be modelled; otherwise, the containment pressure is grossly overpredicted.
2. Inadequate data is available on how best to model the location of the cooling throughout the facility. Weighting the cooling of certain compartments by the fraction of cooling pipe therein caused the calculation to crash as it froze those compartments where very little heatup is measured. The greatest heat losses will

have been from those compartments with cooling pipes and with highest atmospheric temperatures. This is a time-dependent problem and modelling it correctly is nontrivial.

3. The location of the instrument cooling in the facility had some effect on the short-term results, especially for the penetration of the light gas into the lower cells of the model. However, these differences were eroded over a 4hr period with similar concentrations throughout the facility predicted at the end of the calculations.

4. The overprediction of the temperatures in the lower cells of the model resulted initially from an overprediction in the amount of steam ingress, and this is an inherent limitation of the lumped-parameter method. The temperature overpredictions following the late blowdown injection were most likely due to the fact that no cooling was modelled in this region.

5. The nodalization of the containment facility was inadequate in certain respects. In particular, the results for one cell representing six rooms of the facility at about the 10m level were quite different from others at a lower level and to which it was connected. A more refined nodalization of these rooms should be explored.

6. Increasing the flow loss coefficients in those flow paths from the lower cells of the model to those higher up restricted steam and light gas ingress into the lower regions. This gave better agreement with the experimental values in the lower cells but resulted in overpredictions of the pressures, steam concentrations and consequently temperatures in the upper dome cells.

7. The overpressures which resulted from the increased flow loss coefficient approach indicate that this is not an adequate model refinement for better prediction of the containment thermal/hydraulic behavior.

8. The analysts concluded that the consistent overprediction of the containment pressures throughout all the sensitivities examined was due to a problem in the mass/energy balance in the MELCOR code which results in pressures being overpredicted for steam injection into a containment.

9. MELCOR predicted in all cases that there is no stratification of the atmosphere in the upper dome and that it is continuously well-mixed. This was in contrast to the experiment where significant differences in the steam and light gas concentrations were observed.

10. MELCOR did not predict the light gas distribution in the containment correctly. In the experiment, all the light gas rose into the upper dome and stayed there; in contrast, MELCOR initially predicted flow up into the upper dome but then distributed the light gas evenly throughout the containment.

11. The analysts concluded that the lumped-parameter/control-volume approach is inadequate for accurate prediction of hydrogen distribution in a containment under severe accident conditions.

6 Winfrith (UK)

AEA Technology at Winfrith Technology Centre are assessing MELCOR, funded by the UK Health and Safety Executive. A major part of this assessment was examining the performance of the MELCOR 1.8.1 code in plant calculations, in particular for the TMLB' sequence with and without surge line failure [28].

The analyses were performed using version 1.8.1 installed on the network of SUN workstations at Winfrith Technology Centre. The calculations for the intact circuit case were based upon an input deck for the Surry plant prepared by Sandia in support of the MELCOR peer review (as described in Section 13.5). The case with surge line break failure did not calculate whether or when surge line failure occurs, but simply used a valved flow path between the hot leg and the pressurizer cubicle which was specified to open to an area equal to the surge line flow area at a time of 10,000s; that time was selected based on SCDAP/RELAP5 calculation results.

The results of these analyses have been critically compared with those from corresponding calculations with the detailed, best-estimate code SCDAP/RELAP5 and CONTAIN for a typical large 4-loop PWR with a dry containment. As the plant designs are similar, useful information can be gained from a fairly brief comparison of results; however, there are sufficient differences in plant design and code modelling capabilities to prevent a detailed comparison from being worthwhile.

In general, MELCOR was found to be robust and easy to use. Though the surge line failure calculation failed to run to completion, the error occurred in the CORCON code, which is also used in the CONTAIN code and gave similar problems in the CONTAIN calculations.

In general, the results obtained using MELCOR were similar to those from the detailed calculations, apart from differences mainly attributable to known deficiencies in MELCOR 1.8.1, *e.g.*, to the absence of models for high-pressure melt ejection, direct containment heating and the solubility of aerosols. In most instances, however, the comparisons lend credence to the MELCOR predictions. One area where the codes disagreed in both scenarios was in the rate at which water was boiled off in the primary system, with MELCOR taking noticeably longer than SCDAP/RELAP5. In addition, the containment atmosphere tended to be more superheated in the MELCOR calculations than predicted by CONTAIN. The reasons for these apparent discrepancies are not known.

This study has usefully extended the assessment of MELCOR in general, and provides a good basis for assessment of future releases of the code.

(These calculations have also been done by Sandia using MELCOR 1.8.2. The station blackout calculation without surge line break was used as a MELCOR 1.8.1 *vs* MELCOR 1.8.2 study [84], and is described in Section 23.5, while the station blackout calculation with surge line break has been done as part of a set of MELCOR calculations [88] studying the effects of air ingression on the consequences of various severe accident scenarios, summarized in Section 25.)

7 Universidad Polytecnica de Madrid

The main activity of the Chair of Nuclear Technology, Faculty of Industrial Engineering, at the Polytechnical University of Madrid, is education on nuclear engineering to undergraduate and doctoral students, but it also performs research within several fields of specialization, such as nuclear safety and, more specifically, on severe accident phenomenology in light water reactors.

As part of these activities, MELCOR calculations have been done for the DEMONA F2 containment benchmark experiment [29], the PHEBUS B9+ core damage international standard problem ISP-28 [30], and the CEC thermal-hydraulic benchmark exercise for the BMC FIPLOC verification experiment F2 [31].

MELCOR calculations have also been done for the FALCON fission product release and transport international standard problem ISP-34 [32], and for the core heatup and degradation phase of the first Phebus-FP fission product release and transport test, FPT-0 [33].

Three accident sequences (AB, V, and SGTR) have been analyzed for the Ascó II plant, a 3-loop Westinghouse PWR [34, 35, 36, 37]; two station blackout sequences in the Garoña plant, a GE BWR/3 with a Mark I containment, have also been done [38].

7.1 DEMONA F2

Thermal/hydraulic conditions in the containment for late containment failure scenarios, are the most interesting, since the other scenarios (early failure, bypass) are of relatively much lower probability. It is necessary to determine the split between the energy transferred to containment *vs* that transferred to concrete during the course of such interactions. Similarly, the larger energy sink is water condensation on walls, floors and other surfaces of the containment building.

Calculations were performed with MELCOR version 1.8.0 run on a VAX Station 3100-M38. As a result of the analysis, estimates of the code's capability for simulating important phenomena related to thermal/hydraulic behavior of a multi-compartment containment were obtained.

Participation in this benchmark exercise [29] allowed verification in MELCOR of

- the adequacy of the models which describe interactions between phases and structures,
- the adequacy of the set of conservation equations solved in the code, and
- the important of nonhomogeneous effects on condensation rates (*i.e.*, the condensation rate is very sensitive to the presence of air in the compartment).
- Natural circulation is influenced in MELCOR by the momentum equation, due to the difference between the hydrostatic pressure term and the gravitational term.

- Clearly, this kind of fluid-dynamic problem includes several sources of inaccuracies and uncertainties, and requires a considerable experience in using the code.

7.2 PHEBUS B9+ Core Damage – International Standard Problem 29

MELCOR version 1.8.0 was used to model the Phebus B9+ core damage experiment as part of the ISP-28 exercise. (MELCOR calculations for this problem were submitted also by SNL, as discussed in Section 4.3, and by Taiwan.)

In this analysis, the default oxidation rate constants were modified, as were the mass relocation criteria (temperature and oxide shell thickness), and the zircaloy melt temperature. The following conclusions were reached in view of the results obtained [30]:

1. The thermohydraulic behavior of the code is satisfactory, as the trends of the calculated temperatures keep very good agreement with the experimental ones, although there is a displacement at the beginning of the calculation.

2. The temperature values at different elevations indicate insufficient heat transfer to fit correctly with specifications. Parameters like the thermal conductivity of porous zirconia and flow rates (steam and helium) have a great influence on the thermohydraulic behavior.

3. The results for core degradation phenomena are deeply influenced by the thermohydraulic behavior.

4. The MELCOR code, despite its simple core degradation model, allows the user to change the parameters which govern degradation. This gives a great deal of freedom to the user in studying such phenomena.

5. The modelling of the shroud was very difficult, due to its special geometry and composition.

6. MELCOR probably had problems with the dimensions of the PHEBUS-CSD facility, as it was designed to model larger reactor cores.

(Recall that Section 4.3 mentioned that Sandia used a number of minor code modifications for this standard-problem analysis, some involving very small masses in the experiment relative to the reactor-scale numbers expected, which have since been implemented in the production code.)

7.3 BMC-F2 Containment Thermal/Hydraulics

The objective of Experiment F2 was to investigate the thermal/hydraulic, long-term phenomena which may occur in a multi-compartment containment under severe accident conditions and to provide a data base for code improvement and validation; in this experiment special emphasis was placed on the study of natural convection phenomena in a loop-type multi-compartment geometry affected by variations of steam and air injection as well as of heat supply into various compartments.

Calculations were performed with MELCOR version 1.8.0 run on a VAX Station 3100-M38. MELCOR calculations for this problem also were submitted by the UK SRD/AEA, as discussed in Section 5.1.

Comparison results between computations and data were reported [31] on all important quantities relevant for containment analyses during long-term transients: pressure, steam and air content, velocities and their directions, heat transfer coefficients and saturation ratios; these quantities primarily define and specify the prevailing conditions and states inside the containment which are responsible for gas and aerosol transport and depletion.

7.4 FALCON Fission Product Transport and Deposition – International Standard Problem 34

Calculations have been completed for the FALCON internationsl standard problem ISP-34, but not submitted in time to be included in the code comparison study.

7.5 Phebus FPT-0 Benchmark Calculations

There have been several rounds of benchmark core degradation calculations for Phebus-FP, all concerned with the first test FPT-0; MELCOR was used by the Universidad Polytecnica de Madrid, while organizations used ICARE, KESS and SCDAP/RELAP5 [33].

There was general agreement about the temperature distribution predicted within the test bundle. All the codes showed that the oxidation of zircaloy accompanied by a rapid temperature excursion, after which all the zircaloy in the central region of the bundle will be oxidized. Most of the codes, including MELCOR, predicted that the cladding will be fully oxidized before it has a chance to melt and dissolve some of the fuel in a eutectic. All the codes coped adequately with the special boundary conditions imposed by a bundle experiment. All the codes had difficulty with the late phase of the transient when the fuel melting temperature was exceeded. One SCDAP calculation stopped completely, while the other calculations all predicted a partial blockage near the bottom of the vessel although the extent, composition and position of the blockage varied.

7.6 PWR Plant Calculations

The MELCOR code has been used to analyze three accident sequences in a 3-loop Westinghouse PWR, the 900MWe Ascó II plant [34, 35, 36]. The study includes:

1. an AB sequence initiated by the 200% rupture, in the loop including the pressurizer, of the cold leg close to the vessel,
2. a V sequence initiated by the rupture, within the auxiliary building, of a low-pressure injection pipe connected to the hot leg, and
3. a SGTR sequence initiated by the simultaneous rupture of ten inlet tubes in the steam generator close to the plate.

For each sequence, plots and tables describe the thermal/hydraulic results, core degradation results, fission product behavior, and attack to the concrete cavity for the base case analysis and for a variation study; radionuclide release, transport and deposition are documented in substantive detail in every case.

In the base AB sequence, none of the emergency systems are functional, except the accumulators, because of the loss of all AC electric power. The variation done on the AB LBLOCA sequence involved the addition of two systems, the containment sprays and keeping a constant cooling capacity in the steam generator secondary side. The main feature of this sensitivity case is the radionuclide retention in the containment pool.

The V sequence originates an intermediate LOCA in the hot leg; the reactor coolant water is driven outside the containment, bypassing it, to one of the RHR pump rooms. The V-sequence variation studied pool scrubbing effects. The variation case takes into account that the RWST is emptied out through the pipe connecting it with the pump room at the same time that the accident is initiated. Such a large amount of water coming from the tank would fill the pump room well above the level of the broken pipe for more than an hour. In this case, vapor as well as fission products would bubble through the pool, a fraction of them being retained.

In the base SGTR sequence, it was assumed that the ECCS does not operate, and that there is no auxiliary feedwater to the steam generators and no steam dump to the condensers; again, only the accumulators participate in the sequence because they are passive elements. The sensitivity study done for the SGTR sequence studied the effect of auxiliary feedwater being available from the start of the accident.

A separate report [37] discusses the behavior of hydrogen in a dry PWR containment for an AB sequence, studying two mitigation procedures for avoiding air/hydrogen/steam mixture flammability. The mitigation procedures are containment hydrogen purge and atmosphere inertization by injecting nitrogen. The influence of the containment spray system was also studied.

These PWR calculations were done using MELCOR 1.8.0 and MELCOR 1.8.1.

7.7 BWR Plant Calculations

The objective of this project was to analyze the role of the Phebus-FP experimental program, as seen from severe accident analyses in BWRs, considered with reference to BWR accident phenomenology and fission product behavior from this plant analysis.

Two station blackout sequences in the Garoña plant, a GE BWR/3 with a Mark I containment, have been analyzed. These two sequences are a long-term station blackout with SRVs opening and reclosing properly, and a long-term station blackout with a SRV stuck open. In both cases, two variations were done studying the effect of operating the HPCI system, for a total of four calculations.

The two selected sequences, and their variations, are being analyzed with both the MAAP and MELCOR codes. The MELCOR calculations are documented in [38] (with the MAAP calculations in a separate report). For all four cases, the thermohydraulic aspects, as well as the core, vessel and cavity degradation processes are analyzed, together with the timing and mode of vessel failure. The behavior of fission products is also carefully considered. The chronology of key events describing the accident sequences has been clearly stated so that comparisons are facilitated.

The results from these analyses have been used to help reconsider the role of the Phebus-FP project. The main features in BWRs which may influence the behavior of fission products in sta :on blackout sequences are the water-separator and steam-dryer assemblies, the downcomer with the jet pump and the pressure suppression pools. These devices act as effective sinks for fission product vapors and aerosols. The simulation of such devices in some of the Phebus-FP experiments should be considered; moreover, similar characteristics may also be present in certain accident sequences for PWRs.

8 Netherlands Energy Research Foundation (ECN) MELCOR Assessment Analyses

The Netherlands Energy Research Foundation, Energieonderzoek Centrum Nederland (ECN, received MELCOR in 1989 and implemented the code on a CONVEX C-220 mini supercomputer. Recently, the code was also installed on an IBM RISC-6000 workstation.

MELCOR has been used by ECN mainly to analyze severe accidents for the General Electric ABWR and SBWR designs. Some assessment of the MELCOR steam condensation models in the presence of noncondensable gases has been performed at ECN, as described below; experimental data from the University of California at Berkeley, obtained in the framework of the SBWR project, was compared with MELCOR calculational results. In addition, the heat conduction and heat transfer (for free convection) models in MELCOR were validated against an analytical model for small capsules with an internal heat source due to irradiation in a research reactor.

Future ECN assessment of MELCOR will involve the comparison of MELCOR results with experimental data for international standard problem ISP-34; this ISP provides data on the deposition and transport of fission products in the primary system as well as in the containment. Two experiments are being performed for this exercise in the FALCON facility at Winfrith (UK), one involving low humidity and high particle concentration in the containment, the other with high humidity and a low particle concentration.

One of the phenomena ECN is particularly interested in is the issue of core-concrete interactions, especially with regard to the debris coolability. In connection with future activities, ECN is interested in the assessment of MELCOR for East-European reactors, especially the VVER 440/230.

8.1 Validation of the MELCOR Steam Condensation Models

MELCOR 1.8 calculations were done to validate the MELCOR steam condensation models, in the presence of noncondensable gases. [39] The experiment which was used was a small-scale experiment performed at the University of California at Berkeley. In this experiment the heat transfer degradation due to the presence of noncondensables (air in this case) was measured. The test facility consisted of a condenser tube which was placed in a natural circulation loop. The condenser tube was surrounded by an annular cooling jacket through which the coolant was forced. Steam was injected into the natural circulation loop by a boiler which operated at different power levels. The condensate was collected and drained from the system.

Three separate MELCOR analyses have been performed: nodalization sensitivity analyses, secondary-side heat transfer analyses, and primary-side pipe friction sensitivity analyses.

The condenser tube was divided into 1, 2, 3, and 4 axial nodes to study the influence of the nodalization. Each of these nodalizations gave the same condensate flow rate

at the outlet of the condenser tube. The heat removed from the loop by this steam condensation equals the boiler power. The local condensation mass flux improved with increasing number of nodes, but the difference between 3 and 4 nodes was very small. Therefore, a nodalization with 3 axial nodes was used for the rest of the calculations.

The condenser tube wall temperatures, as calculated by MELCOR, were too high compared with the experiment. The MELCOR heat transfer correlations, applied on the secondary side, calculate heat transfer coefficients much lower than in the experiment. This is probably due to increased turbulence (due to flanges, thermocouple wires, *etc.*) in the experimental facility. Therefore, the wall temperatures were fixed at the experimental value for the remaining analyses.

Due to the small size of the test facility, the steady state system pressure was very sensitive to frictional pressure losses. Since the experimental pressure losses were not measured, it was very difficult to correctly predict the system pressure with MELCOR. Small variations in hydraulic diameters or form loss coefficients led to great variations in the calculated pressure. Also, the interfacial shear between the condensate and the steam/air mixture is only applicable for an annular flow regime, which may not always be the case.

8.2 Temperature Distribution inside a Capsule – MELCOR *vs* Analytic Model

In the High Flux Reactor the influence of radiation on material properties is investigated, with capsules filled with different kinds of materials irradiated. Due to radiation, heat is produced inside the capsule material. For safety purposes, it is necessary to know the maximum temperature of an irradiated capsule. To calculate the temperature profile, an analytic model was developed for heat conduction and heat production. The boundary conditions are obtained from heat transfer correlations found in the "VDI Wärmeatlas" [123]. For validation purposes, the calculation is also performed with MELCOR.

The calculated temperature from the analytical model, implemented on a PC, shows good agreement with the MELCOR results.

8.3 ABWR and SBWR Analyses

MELCOR has been used by ECN mainly to analyze severe accidents for the General Electric ABWR and SBWR designs. The accidents analyzed for the ABWR involved a station blackout with emergency cooling for 8 hours and a loss-of-all-core-cooling accident. In both scenarios the reactor pressure vessel is depressurized successfully, resulting in vessel failure at low pressure. To study the influence on the source term, the loss-of-all-core-cooling scenario also was analyzed with the assumption of unfiltered venting from the wetwell. The SBWR scenarios concern a low-pressure core melt, a bottom drain line break and a main steam line break. In the latter scenario all passive safety systems were

assumed to function, resulting in no core damage. The other two scenarios lead to vessel failure at low pressure. No source terms were calculated for these scenarios since the failure mode of the SBWR is still unknown.

9 NUPEC Experiment Analysis and Plant Analysis

MELCOR's role in the Nuclear Power Engineering Center of the Japan Institute of Nuclear Safety (NUPEC/JINS) is seen as that of a second generation code for once-through analysis of light water reactor severe accidents, used to improve the accuracy of containment event tree analysis and source term analysis in level 2 PSAs for Japanese light water reactors.

Preliminary calculations for experimental analysis and plant analysis have been performed using MELCOR 1.8.0. These analyses include core degradation calculations for the Phebus-FPT0 experiment [40], and calculations of two Peach Bottom BWR plant severe accident sequences [40].

A number of calculations have been done at NUPEC with MELCOR 1.8.1 and MELCOR 1.8.2 [41], including numeric studies with MELCOR 1.8.1 on machine dependencies and time step effects [42] (repeated with MELCOR 1.8.2 for direct comparison [43]), analysis of NUPEC's hydrogen mixing and distribution tests M-4-3 [44] and M-7-1 (ISP-35) [45, 46], containment thermal/hydraulic calculations for Phebus-FP test FPT-1 [47, 33], and a number of PWR [48] and BWR [49] plant sequence analyses in support of PSA studies.

9.1 Preliminary Plant Analysis Calculations

Preliminary calculations of a BWR plant severe accident were done to examine code characteristics and input data preparation, and to accumulate code application experience [40]. Plant data were taken from the Peach Bottom FSAR so as to be able to compare with other US calculations for checking purposes. Two sequences, failure of ECCS and safety relief valves after transient (TQUX) and a large LOCA (AE), were selected for calculation. Results given include the predicted timings of key events and the cesium iodide distribution fractions in the plant, for an accident progression of 19hr for the TQUX sequence and 14hr for the AE sequence.

9.2 Phebus-FP FPT-0 Core Degradation Analyses

The Phebus-FP experiment is the integral, in-pile test which is being prepared by the CEA and the CEC to study fission product transport behavior in light water reactor severe accidents. Preliminary calculations on the Phebus-FP experiment have been performed [40] to obtain information on MELCOR's capabilities and limitations in experimental analyses, in order to use the MELCOR code for thermal/hydraulic and fuel behavior analysis of the Phebus-FP experiments.

MELCOR input data was prepared from the physical dimensions and thermal/hydraulic boundary conditions given by the CEA and the CEC. The first Phebus-FP test planned, FPT-0, is a scoping test. Pretest analysis of FPT-0 has been carried out to examine the

fuel and control rod temperature, cladding oxidation, fission gas release, and relocation behavior of fuel and control rod, and to check the code capabilities.

Two calculation cases were selected. One used default values for core degradation parameters. In this case, the fuel relocates at the same time as the cladding melts, which is different from the expected behavior based upon experiments such as the Phebus/SFD tests. The fraction of cladding oxidation indicated by hydrogen production was ~25%. Iodine release from the fuel was ~22% at 7500s, and the same fraction (22%) of rare gases and cesium were released; 20% of the tellurium was released from the fuel.

The other calculation used specially-prepared input data to simulate the in-pile experiment, which shows that fuel does not form debris instantaneously when the cladding reaches the melting point. The cladding oxidized fraction based upon predicted hydrogen generation in this case was ~29%. All of the iodine, rare gases and cesium were released from the fuel into the bundle section, and ~89% of the tellurium was predicted to be released by 7500s.

A lack of information on cadmium aerosol in the printed output was noted, as was a need for a new control flag to simulate the delayed formation of fuel debris after clad melt.

9.3 Containment Thermal/Hydraulic Analyses of Phebus-FP

In preparation for the Phebus-FP tests, a series of thermal/hydraulic tests have been performed in which steam is injected into the containment vessel in a series of steady states; before these tests, a series of calculations were performed by various teams using a number of reactor codes [33]. NUPEC participated in these analyses using MELCOR 1.8.1 [47].

The test protocol defines a target humidity of 50% in the first test and near saturation in FPT-1 during the injection phase of the test. The conditions in the containment are defined by the injection flow rate, the sump temperature, the vessel wall temperature and the condenser temperature. The objective of the Phebus-FP containment thermal/hydraulic calculations and separate-effects experiments was to arrive at a consensus view on test conditions and procedure, to use MELCOR as one of the analysis tools, to study how saturated conditions could be obtained and to check the possibility of no wall condensation except on the condenser wall.

MELCOR calculations were done by NUPEC for a base case and for eight parametric variations:

1. increasing the pool surface area to cover the whole of the containment vessel floor,
2. twofold increase of the inlet steam flow and the condenser power,
3. hydrogen injection during the steam injection,
4. increasing the condenser power by 20%,

5. decreasing the condenser surface area by 50%,

6. zero condenser power during the non-injection phase,

7. 2bar initial pressure in the containment, and

8. vessel wall temperature at 80C.

These MELCOR calculations indicated that the humidity in the containment depends very strongly on the sump surface area and that condensation onto the vessel wall was not avoidable in all cases. MELCOR also predicted that enough liquid condensate film would accumulate in some cases that post-test analysis could be disturbed due to transfer of material condensed onto the wall to the floor and sump. (Note, however, that the LACE LA4 MELCOR assessment analysis [57] showed that the default aerosol washoff model in MELCOR significantly overpredicted removal of aerosols from walls by condensate film.)

Double blind test calculations were done by MELCOR and by other codes (CONTAIN, CONTEMPT, CONT and JERICHO) for the thermal/hydraulic tests done in the containment vessel, and a comparison to data presented in [33]. The results of the calculations revealed that the codes agreed quite well about the mass transfer rates; the experimental results surprisingly showed that the measured vapor pressure was outside the range of the calculated values implying that the codes' mass transfer coefficients were too high. Different treatments for sensible heat transfer in the various codes resulted in rather larger differences in the calculated atmosphere temperature than was the case with the vapor pressure; the results were nonetheless all higher than the measured values. The asymptotic trends in the codes' predictions were reasonable; changing the surface temperatures or the steam injection rate changed the atmosphere temperature and steam content in the right direction. But quantitatively the results were rather poor and confirmed that close agreement between code predictions does not mean that there will be close agreement with experiment. The errors in temperature and steam concentration both led in the same direction – an underprediction of humidity. In a test with fission products this could mean underprediction of the likelihood of steam condensation onto aerosol particles.

9.4 Numeric Studies

A number of calculations have been done at NUPEC with MELCOR 1.8.1 studying numeric effects due to machine dependencies and/or time step effects [42]. Those calculations have now been repeated with MELCOR 1.8.2, for direct comparison [43].

Calculations were performed with MELCOR 1.8.1 with both single and fully double precision for all real variables on engineering workstations. Calculations were done using the "DEMO" test problem included in the standard MELCOR distribution package, on HP9000/730, Sun SPARC 2, and IBM6000/560 workstations, with various compiler optimization levels. Three cases were selected as time step control schemes. The numerical results with single precision depend on compiler options and machines utilized,

confirming that MELCOR 1.8.1 produced different results with different compiler options on different machines. For fully double precision calculations, the final cycle, final time and final temperatures were equal with any optimization level on any machine, if the time step scheme was kept the same; the fully double precision calculations did, however, show anomalous behavior and encountered abnormal termination after about 700s of the transient.

Calculations for the "DEMO" problem also have been preformed on various workstations with various compiler optimization levels using MELCOR 1.8.2, and those results compared to those from MELCOR 1.8.2 [43]. Calculations were done on HP9000/730, DEC3000/500 AXP and IBM6000/560 workstations. The dependence of calculation results on computer environments was observed to be reduced from MELCOR 1.8.1 to MELCOR 1.8.2. Through calculations by MELCOR 1.8.2 with fully double precision, the dependence of calculation results on computer environments was found to be reduced; furthermore, the tendency of convergence of solutions was observed with decreasing time step.

9.5 NUPEC Hydrogen Mixing Tests M-4-3 and M-7-1 (ISP-35)

A number of calculations have been done at NUPEC with MELCOR 1.8.1 and 1.8.2, including analyses of NUPEC's hydrogen mixing and distribution tests M-4-3 [44] and M-7-1 [45, 46]. Test M-7-1 has been selected as international standard problem ISP-35.

The Hydrogen Mixing and Distribution Tests are part of the Ministry of International Trade and Industry (MITI) sponsored project entitled "Proving Test on the Reliability for Reactor Containment Vessel", and are part of NUPEC's ongoing severe accident safety analysis program. The aim of these tests is to investigate hydrogen distribution behavior within a model containment and at the same time provide a set of experimental data useful for validation of severe accident analysis codes.

The test vessel is a 1/4-scale large dry PWR containment with a total volume of $1300m^3$; it has a diameter of about 10m and a height of 17m, with three floors. The containment was divided into 25 volumes connected by 66 openings.

The first test, M-4-3, was characterized by a helium and steam gas mixture injection into the containment, initially maintained at room temperature. The MELCOR model for the containment test facility consists of 25 control volumes and 66 flow paths. The inner structures were modelled as two-sided heat conductors, while the outer walls and floor were modelled with insulated boundary conditions on the heat structure outer surfaces; the thermal insulators at the outer wall were not modelled. Calculations were done both with the original, standard MELCOR heat transfer coefficients, evaluated in the natural convection regime, and with Uchida's correlation added and evaluated as a condensation regime. Using the original heat transfer model, MELCOR overestimated the pressure and atmospheric temperatures measured, apparently due to low heat transfer rates from

atmosphere gases to heat structures. Using Uchida's correlation, the calculation results agree with the experimental results.

For the M-7-1 test, the containment was preheated to about 70°C and, in addition to the gas mixture, an inner spray was active during the test. The MELCOR model for the containment test facility consists of 32 control volumes, 74 flow paths and 122 heat structures. For heat structures modelling walls directly cooled by the spray, the Kirkbridge and Badger correlation for filmwise condensation was used while the spray was active; on the opposite wall and after the spray stopped, Uchida's correlation was used, considered as condensation regime. For other walls, the MELCOR original, default model was used. Sensitivity studies were done evaluating the influence of the spray droplet diameter, of dividing the dome nodes, of heat transfer modelling including a liquid film model, and of the insulator.

In the M-7-1 experiment good mixing, enhanced by spray water, was observed and the MELCOR calculation showed good agreement with experimental results when a proper spray model and system noding was selected. The size of spray droplets and their distribution did not affect the thermal/hydraulic state in the containment because sufficient equilibration between the droplets and the atmosphere was obtained in the early phase of the droplet flow. However, modelling the spray water after it reached the bottom of the dome or structure walls was important in predicting the experimental results. Some of the dead-end volumes with only one opening were correctly predicted by subdividing the volumes to simulate the countercurrent flow through the opening.

9.6 PWR PRA Calculations

Severe accident analyses have been done at NUPEC with MELCOR 1.8.2 for a reference PWR [48] in support of PSA studies The reference PWR is a 3411Mw(th) PWR with four loops and a large dry containment, similar to the Zion plant.

Two loops are modelled: a single loop which connects with the pressurizer and a second loop with three plant loops lumped together. Each loop is modelled with 5 control volumes, as is the reactor vessel. The containment is modelled with three control volumes. The core and lower plenum are modelled in detail with seven radial rings and 16 axial levels; the upper 10 levels are fueled while the lower 6 represent support structure and the lower plenum.

Thirteen accident scenarios have been analyzed:

1. S_1D, a medium break (6in) LOCA in the hot leg, accompanied by failures in the operation of the HPI and LPI systems;

2. S_1DC, a medium break (6in) LOCA in the hot leg, accompanied by failures in the operation of the HPI, LPI and containment spray systems;

3. S_1H, a medium break (6in) LOCA in the hot leg, accompanied by failures in the recirculated operation o' the HPI and LPI systems;

4. S_1HF, a medium break (6in) LOCA in the hot leg, accompanied by failures in the recirculated operation of the HPI, LPI and containment spray systems;

5. S_2D, a small break (2in) LOCA in the hot leg, accompanied by failures in the operation of the HPI and LPI systems;

6. S_2DC, a small break (2in) LOCA in the hot leg, accompanied by failures in the operation of the HPI, LPI and containment spray systems;

7. S_2H, a small break (2in) LOCA in the hot leg, accompanied by failures in the recirculated operation of the HPI and LPI systems;

8. S_2HF, a small break (2in) LOCA in the hot leg, accompanied by failures in the recirculated operation of the HPI, LPI and containment spray systems;

9. TML, a station blackout (loss of offsite power and failure of emergency diesel generator with recovery of AC power) with failure of turbine-driven auxiliary feedwater pump,

10. TMLB', a station blackout (loss of offsite power and failure of emergency diesel generator without recovery of AC power) with failure of turbine-driven auxiliary feedwater pump,

11. S_2F, a small break (2in) LOCA in the hot leg, accompanied by failures in the recirculated operation of the containment spray system;

12. SGTR, a steam generator tube rupture, with failure to isolate the broken steam generator and failure in the operation of the LPI system, and

13. V, a large break LOCA in a pipe in the Residual Heat Removal (RHR) system, with failures in the operation of the HPI and LPI systems.

Sensitivity studies have been done on core-concrete interaction in the S_2DC sequence, and on debris coolability for the S_2HF sequence. The basecase S_2DC calculation showed basemat melt-through due to core-concrete interaction; if the characteristics of the heat transfer from the core debris to the reactor cavity concrete and the heat transfer between layers inside the debris pool are changed, the concrete erosion could change and the timing of containment failure be affected. The basecase S_2HF calculation showed containment failure due to overpressurization by steam production during debris cooling in the reactor cavity; if the characteristics of the heat transfer from the core debris to the water pool and the heat transfer between layers inside the debris pool are changed, the behavior of debris cooling and steam production could change and the timing of containment failure be affected.

In the sensitivity analyses concerned with core-concrete interaction, the effects of decay heat and of the heat transfer correlation of the debris pool were investigated. Calculations for the S_2DC sequence were done assuming a loss of secondary cooling,

since in that case accident progression and reactor vessel failure occurs faster than in the basecase and the decay heat level in the debris becomes higher. Another analysis for the S_2DC sequence was performed using a modified heat transfer model as the debris-concrete heat transfer model (instead of the CORCON-Mod2 heat transfer model); the modified heat transfer model consists of D. R. Bradley's modification on the bottom of the debris pool and the Kutateladze model in other places.

In the sensitivity analyses concerned with debris coolability, the effects of the heat transfer coefficient for the surface of the debris pool and of the heat transfer correlation of the debris pool were investigated. Calculations for the S_2HF sequence were done using 1200w/m^2-K as the heat transfer coefficient from the debris pool surface to the water, instead of the basecase value of 1000w/m^2-K. Another analysis for the S_2HF sequence was performed using the same modified heat transfer model as the debris-concrete heat transfer model as used in the S_2DC core-concrete interaction sensitivity study (instead of the CORCON-Mod2 heat transfer model).

The summary and conclusions of the results of selected accident sequence analyses are:

1. In the cases with failure of the operation of HPI and LPI systems, the containment remains intact due to the operation of the containment spray system.
2. In the cases with failure of the operation of HPI, LPI and containment spray systems, the containment fails by basemat melt-through.
3. In the cases with failur › of the recirculated operation of HPI, LPI and containment spray systems, the containment fails by overpressure.
4. From the results of sensitivity analyses, changing the parameters concerned with core-concrete interaction influences accident progression greatly, but changing parameters concerned with debris coolability has little influence on the accident progression.

9.7 BWR PRA Calculations

Severe accident calculations for a reference BWR plant have been done by NUPEC with MELCOR 1.8.1 in support of PSA studies [49]. The reference BWR plant is a 3293Mw(th) BWR-5 with Mark II containment, similar to the LaSalle plant. A number of accident scenarios have been analyzed, including 6 transients (TQUV, TQUX, TB, TBU, TW and TC sequences), one large break LOCA (AE sequence) and one interfacing-systems LOCA (V sequence).

Sensitivity studies also have been done. For the TQUV sequence, cases investigated include varying thermal loading on the core support plate due to slumping (by widening the slumping area) and varying the corium spreading within the wetwell (from the space velow the lower pedestal to the entire wetwell area). For the TB sequence, the effect of

drywell pressure increase upon reactor pressure vessel failure was studied by increasing and decreasing the vessel breach area from that of an instumentation guide tuve opening. All sequences except the V sequence were run with different containment failure criteria.

10 Tractebel Analysis of NUPEC M-7-1 Hydrogen Mixing and Distribution Test – International Problem 35

MELCOR 1.8.2 calculations for NUPEC's hydrogen mixing and distribution test M-7-1 (ISP-35) have been performed by Tractebel Energy Engineering (TEE) [50, 51], as part of the OECD International Standard Problem 35.

The Hydrogen Mixing and Distribution Tests are part of the Ministry of International Trade and Industry (MITI) sponsored project entitled "Proving Test on the Reliability for Reactor Containment Vessel", and are part of NUPEC's ongoing severe accident safety analysis program. The aim of these tests is to investigate hydrogen distribution behavior within a model containment and at the same time provide a set of experimental data useful for validation of severe accident analysis codes.

The test vessel is a 1/4-scale large dry PWR containment with a total volume of 1312m^3; it has a diameter of 10.8m and a total height of 19.4m, with three floors. The containment was divided into 25 volumes connected by 56 openings. Experiment M-7-1 was selected as ISP-35 by the OECD to simulate the impact of the containment spray system on the helium distribution in a steam-rich atmosphere. The M-7-1 test consists of injecting steam and helium in a steel containment when a spray system operates. The test begins after a preconditionng phase during which the containment is heated up by steam injection.

The conclusions drawn by Tractebel from participation in this standard problem exercise are summarized below:

- Test M-7-1 has confirmed the homogenization effect of the containment spray system on the atmosphere composition.
- ISP-35 was an excellent basis to exercise the different codes, but no attempt can be made to associate the operating conditions in the NUPEC facility with severe accident conditions.
- The pre-conditioning phase could not be predicted by the codes; the majority of participants chose to begin the test with the initial conditions specified.
- Test M-7-1 confirmed the effect of user experience on on the results as can be seen from the four CONTAIN and two RALOC calculations, which provded different results and demonstrate the user effect. Moreover, the nodalization adopted has a strong impact on the results; the subdivision of a compartment can create artificial convection loops without any experimental confirmation.
- An important code limitation for test M-7-1 is the lack of heat transfer modelling between the spray droplets and the heat structures. This could explain the difficulties the codes had reproducing the behavior in some of the compartments and the large discrepancies in the predictions of wall temperatures inside the containment.

- The results obtained with MELCOR 1.8.2 demonstrate the capability of the code to handle complex flow situations including the actuation of containment safeguards systems.

  The global energy balance of the atmosphere is correctly computed as shown by the good agreement between the theoretical and experimental values of the total pressure. Local discrepancies observed in the gas temperatures could be explained by:

  1. assumptions related to the distribution of the spray mass flow in the different compartments,
  2. code limitations for the distribution of the spray mass flow in the different compartments,
  3. reversal of flow direction in some junctions.

- The temperature of the external walls is very sensitive to the outer boundary condition. The imposed adiabatic condition does not allow a fitting of the calculation results with the experimental data. A much better agreement was obtained by modelling convective heat transfer along the outer face.

- The validation of computer codes requires well defined and well documented experimental data. In this exercise, TRACTEBEL experienced many difficulties mainly related to:

  1. inconsistences and successive corrections of geometric data;
  2. uncertainties about the thermodynamic data of the injected steam and the actuation time of the spray system;
  3. location of some measuring points (*e.g.*, helium concentration measurement in a flooded compartment);
  4. not steady state initial conditions of the pre-conditioning phase.

11 MELCOR Benchmark Calculations for N Reactor PRA

A Level-3 PRA has been performed for N Reactor, a USDOE production reactor, with phenomenological supporting calculations performed with HECTR and MELCOR [52]. The differences between the N Reactor core and a commercial LWR core (for which MELCOR had been developed) required a completely new core package in MELCOR. In order to ensure that the codes and the input adequately modelled N Reactor, a number of benchmarking calculations were performed.

The purpose of the benchmarking exercises was to demonstrate that MELCOR could perform acceptable source term calculations for N Reactor accident sequences. Each of the benchmark calculations was intended to exercise a particular model or section of the code, and these separate effects calculations should develop confidence that the models work as intended; with the processes represented by these calculations "proven", it was then assumed that integral calculations would be essentially correct.

11.1 Hydrogen Mitigation Design Basis Accident

The goal of the hydrogen mitigation design basis accident (HMDBA) benchmark analysis was to reproduce with MELCOR the TRUMP-BD calculations [124] done by Westinghouse at Hanford. The objective of those calculations was to determine whether or not the design basis of the hydrogen mitigation system would be challenged under a postulated design basis severe accident scenario where a large break loss-of-coolant accident (LBLOCA) occurs, the emergency core cooling system (ECCS) fails completely on demand, but the graphite and shield cooling system (GSCS) remains functional throughout the (10-hr) accident.

Six cases were run with MELCOR for use in the benchmarking study. Case 1, a "black box" case using all MELCOR defaults, corresponds to what might be done as a blind MELCOR scoping analysis. Cases 2 through 6 correspond to the cases and sensitivities performed using TRUMP, and were done with MELCOR input intentionally designed to mimic the TRUMP input, for comparison purposes.

The MELCOR HMDBA benchmark calculations agree well with the Westinghouse Hanford TRUMP calculations, as to hydrogen production, peak temperatures and temperature curve shapes; while there was a notable difference in the temperature turnover times between the TRUMP and MELCOR calculations, these differences were generally attributable to the very large node sizes used in the MELCOR model.

11.2 Cold Leg Manifold Break with CV-2R Failure

MELCOR results were compared to TRUMP-BD results [125] for a cold leg manifold break with failure of a CV-2R valve. For this case, most of the core would be cooled by

ECC and only the region with the failed valve would be expected to heat up. Lateral conduction between the two regions therefore could be important for this case. Because the TRUMP-BD calculations used a heavily-noded grid for conduction within the core, they provided a good basis for assessing MELCOR's ability to adequately model lateral conduction in the N Reactor core.

Two MELCOR models were used for this comparison. In the first, two core regions were modelled, with a single axial node in each; this model corresponded to a TRUMP-BD calculation reported for a segment in the central core region cooled by the GSCS, and allowed a direct comparison of predicted temperatures. In the second model, the full core was modelled (1/16 "affected", the remainder "unaffected") using the axial power distribution used for the full-plant calculations, allowing a rough comparison of the hydrogen generation rates using MELCOR and TRUMP-BD.

The results indicated that the MELCOR models for lateral conduction in the CRN (N Reactor core) package were adequate. The MELCOR results showed the same trends as the TRUMP-BD results for both the single-segment and full-core calculations, with MELCOR's coarser noding yielding lower peak temperatures as well as a slower rate of cooling after the peak. These differences were not large enough, however, to significantly affect conclusions that would be drawn from the calculations concerning the amount of core damage or hydrogen generation.

11.3 Fission Product Release from N Reactor Fuel

The purpose of this benchmark exercise was to verify that the MELCOR radionuclide release model for metallic fuels was indeed implemented as intended. The MELCOR calculation was based on the HMDBA calculation described in Section 11.1. Two radionuclide release calculations were run for this benchmark exercise. In the first, the non-oxidation release models were disabled by user input, so only fuel-failure and oxidation-based releases were calculated; in the second calculation, all release models were used in order to verify the non-oxidation release model.

The results of the first calculation showed exact agreement between hand calculation and MELCOR for all radionuclide classes except the noble gases; further investigation revealed an interaction error between the oxidation-release model and the fuel-failure release model which controls the initial release of a large fraction of the noble gas inventory. In the second calculation, an exact comparison with hand calculation was not possible; however, examination of the plots indicated that the expected qualitative trends were followed in all cases, and the non-oxidation release model was also verified on a stepwise basis using MELCOR under the VAX/VMS debugger.

11.4 Confinement Response

A comparison was made between MELCOR and HECTR calculations to assess MELCOR's ability to model confinement thermal/hydraulic phenomena. Steam and hydrogen

sources [126] for both HECTR and MELCOR were for a cold leg inlet manifold with failure of ECC; fog sprays were included in the HECTR and MELCOR calculations. The MELCOR deck was constructed to match as closely as possible the HECTR analysis [127]. The results of the MELCOR and HECTR calculations agreed extremely well in all areas (pressure, temperature, mole fractions, and timing of key events) both during the initial blowdown phase and during later periods with hydrogen injection.

11.5 Fission Product Transport

The purpose of the fission product transport benchmark calculation was to ensure that MELCOR and the N Reactor plant model adequately modelled fission product transport processes. The results of the MELCOR benchmark calculation were compared to the results of CONTAIN calculations [128]. Only the confinement and confinement systems were modelled for this calculation because the water, hydrogen and fission product sources were the same as those used in the CONTAIN calculation.

The results of this code-to-code comparison exercise demonstrated that MELCOR adequately modelled the transport of radionuclides in the N Reactor confinement. The transport of noble gases was very well predicted; although MELCOR predicted a higher release of molecular iodine, the amount of mass released in both calculations was so small that the difference was insignificant.

11.6 Steady-State

The steady-state benchmark calculation was performed to ensure that MELCOR and the N Reactor input deck adequately modelled the N Reactor normal operating state. Several of the plant operational parameters [126] were compared with the MELCOR calculation results. The conclusion of this study was that MELCOR predicted the steady-state conditions of the N Reactor plant extremely well. This benchmark calculation instills confidence that the results of any transients or loss of coolant accidents initiated from this steady state would not be affected by numerical instabilities previous to the accident initiation.

11.7 Scram Transient

The scram transient benchmark calculation was run to ensure that MELCOR and the N Reactor input deck adequately predicted the thermal/hydraulic response to a scram transient. Two calculations were performed for this benchmark: in one calculation, the final pressurizer pressure setpoint following scram was 9.25MPa ([129]) and in the other calculation the final setpoint was 8.46MPa (from Westinghouse Hanford personnel).

MELCOR adequately predicted the trends in pressurizer level, pressurizer pressure and HPI mass flow during a scram transient. Although a simplified model of the HPI was

used, the error introduced would not be significant since the severe accident scenarios that MELCOR was used to analyze either did not have HPI available or other cooling mechanisms were available such that HPI was unneeded.

11.8 Hot Dump Test

The hot dump test benchmark calculation was run to ensure that MELCOR and the N Reactor input deck adequately predicted the response of N Reactor to a transient in which the V-4 valves open, depressurizing the system. This calculation was chosen because accurate prediction of the dump line behavior can have a significant effect on the timing and progression of accidents and, ultimately, on the release and retention of fission products.

This test was performed as part of the N Reactor startup. The initial conditions for this MELCOR calculation were based on those used in a RELAP5 hot leg dump calculation [129]. The MELCOR N Reactor model calculated the events that occurred in the hot leg dump very well. Although the timing was slightly off, the critical parameter for this benchmark calculation was the mass flow rate through the V-4 valves since the prediction of radionuclide transport late in time is of primary importance, and MELCOR predicted this extremely well.

11.9 Cold Leg Manifold Break with Failed CV-2R Valve

A fully integrated MELCOR 1.8.0 calculation was done for a double-ended rupture in the cold leg manifold in which a CV-2R valve fails to open and blocks the ECC flow from getting into one inlet riser. A comparison was made with the same calculation [130] done with the RELAP5/MOD2 computer code. Because the RELAP5 code was designed and developed over many years specifically to calculate the hydrodynamics of a reactor primary system, its hydrodynamics results therefore form a good data base for benchmarking the performance of MELCOR's hydrodynamics.

The MELCOR hydrodynamic results showed reasonable agreement with the RELAP5 results during both the rapid system depressurization and later after the ECC becomes fully established. There were some intermediate differences between the results of the two calculations due to MELCOR's coarser nodalization, the level of detail in the treatment of two-phase flow, and underpredicted inertial and interfacial forces. The blocked riser core volume nodalization was too coarse to correctly predict the temperatures in the fuel; more vertical volumes would allow the upper fuel to dry out and heat up faster.

The N Reactor LBLOCA was a complex hydrodynamic calculation which is really RELAP5's domain and, in fact, gave RELAP5 some difficulty. The reason for using MELCOR in this application was that it could do the entire integrated calculation including radionuclide transport from the fuel through the primary system and confinement and release to the environment. This calculation was one of the more difficult N Reactor calculations done with MELCOR and was only attempted twice. Experience gained in

these attempts should have been applied to a subsequent run because MELCOR is considered capable of doing a better job of predicting the correct results, especially the fuel failure (assuming that the RELAP5 results are correct), but, since the calculation was relatively expensive, the results of the second, later attempt were deemed adequate.

12 SP-100 Space Power Reactor

MELCOR has been used to perform independent safety calculations for two proposed SP-100 space reactors designs [53]. It proved possible to model and analyze simple pressure and temperature excursions for litium coolant with the existing code. This successful application to space reactors proves the code's worth as a flexible analysis tool.

Calculations were done for both a LANL reactor design and a GE reactor design. For both reactor designs, both loss-of-circulation and loss-of-coolant accident scenarios were studied, and strengths and weaknesses of each design were identified.

Both designs use liquid lithium as the working fluid. Since modelling actual lithium coolant with phase change allowed would require extensive internal code modifications as well as a much more extensive definition of lithium material properties than were available, the NCG package in MELCOR was used to model the liquid lithium. The evaluation of the pressure/temperature response surface of the lithium and the generation of the necessary material properties is described in the report, together with the lithium boiling curve needed for some of the analyses.

13 MELCOR Peer Review

As part of the MELCOR Peer Review process [2], the CVH/FL package was used to simulate several simple and well-characterized problems to develop a better appreciation for the CVH/FL hydrodynamics models and their numerical implementation into the MELCOR architecture. These were done by Dennis Liles using a version of MELCOR 1.8.0.

In addition to the problems done by Dennis Liles as part of the MELCOR Peer Review, demonstration calculations for station blackout scenarios in a typical PWR and BWR were also done and presented by Sandia staff as part of the Peer Review process.

13.1 GE Vessel Blowdown

A series of medium-scale blowdown tests were performed in the early 1980s at General Electric (GE) [131]. Blowdown tests were conducted with a blowdown line connected to either the top or bottom of the vessel. Initially saturated water partially filled the vessel; saturated steam filled the remainder of the tank.

The first test analyzed was GE Test 5801-13, a top-blowdown experiment. The pressure results showed good agreement with the test data, but the code computed slowly. A bottom-blowdown test (GE Test 5803-1) was also simulated with MELCOR. A little poorer comparison of the MELCOR results with data was observed in this case, particularly in the mid-range of the test when a distinct flow quality change in the entrained fluid shows up in the data. The calculations revealed no particular time-step sensitivity for this problem.

The stated conclusion was that MELCOR seems adequate in predicted most blowdown scenarios; for the intended purpose extreme accuracy is unnecessary and the results are generally satisfactory.

13.2 Condensation

To assess the condensation model in MELCOR, a simple vertical test problem was set up. A stack of six control volumes was capped with a much bigger volume on the top which provided a large source of steam, with the lowest volume was run as a constant source of subcooled liquid. Two cases simulated the thermodynamic equilibrium and nonequilibrium options, respectively.

Both cases showed the classic difficulties associated with low-order finite-difference schemes in an Eulerian formulation, including pressure spikes when the void fraction of a mesh cell approaches zero and stair steps rather than more realistic continuous behavior; however, a certain amount of stair-stepping would also be evident in TRAC or RELAP for this type of problem.

Ignoring these difficulties, the equilibrium case appeared to function as intended. For the nonequilibrium case almost no condensation was taking place because of the use of a very small interfacial area as the default value, and there is very little liquid entrainment. This simple test points out a basic problem with MELCOR: for vertical volumes, the flow path opening heights selected partially determine the void distribution. (This problem has since been addressed by the MELCOR code developers.)

The conclusion derived was tha MELCOR 1.8.0 probably cannot handle ECC injection problems accurately with the default interfacial area value; although MELCOR is not really a reflood code, it could be expected that too much liquid and too much subcooling would enter the lower plenum unless the interfacial rates were increased by a factor of >10. Because condensation is a flow-regime-specific phenomenon, such a single value of the augmentation factor is inappropriate for all cases. A separate problem associated with the MELCOR condensation model is the potential of overprediction in condensation rates for conditions when a steam atmosphere overlies a large quiescent water pool.

13.3 Air-Water Closed Loop

In another gedanken problem, ten vertical volumes, each 1m high, were stacked with a much larger tank on the top. The diameter of the volume stack, 0.5m, was chosen as the flow path opening height, needed to connect vertical mesh cells. A flow path containing a fan (*i.e.*, a momentum source) connected the top tank volume with the lowest pipe cell to form a closed loop. The initial condition had half the pipe filled with water before the fan was activated, while the upper half and the tank were filled with air.

The results showed each control volume settling out to a uniform void fraction of 0.5, entirely due to the opening heights being equal to the mesh cell height divided by 2. The opening heights uniquely determine the void fraction distribution for this problem.

The conclusion drawn in the Peer Review report was that this problem can only be addressed by designating the full vertical height as an opening height, using one rather than multiple flow paths as connectors, and developing a more detailed flow map and interfacial drag package and incorporating them into the code. (The code developers have addressed this issue recently, as discussed in Section 14.5 for the LOFT LP-FP-2 analysis, in a different fashion.)

13.4 MELCOR BWR Demonstration Calculation

Results of a MELCOR calculation of a postulated BWR short-term station blackout accident sequence, done by SNL, were provided to the Peer Review committee on two occasions. The calculations represented the LaSalle County Station, a BWR/5 with a Mark-II containment, and addressed the full scope of severe accident behavior, *i.e.*,

in- and ex-vessel aspects of core melt progression, the accompanying containment thermal/hydraulic response, and attendant fission product release and transport to the environment.

The first calculation was performed with MELCOR 1.8.0, and presented very early in the review process. The second calculation addressed the same BWR accident scenario and was performed at a later time with a preliminary version of MELCOR 1.8.1. The combination of these two calculations illustrated the strengths and weaknesses of the current code models, and underscored the developmental status of MELCOR. Noteworthy findings or observations from the committee's review are:

1. Substantial differences in important calculated results from the MELCOR version 1.8.0 and preliminary 1.8.1 calculations were observed. Some of these differences indicate improvements in code models or their implementation: for example, large energy errors in the COR package and mass balance deficiencies in the RN package observed in the MELCOR 1.8.0 calculation appear to have been eliminated or reduced in the MELCOR 1.8.1 calculation. Other differences, however, clearly illustrate the lack of maturity and continuing development of some MELCOR models: for example, in-vessel hydrogen differed by 17% in the two calculations, time to containment failure changes from 48,863s for MELCOR 1.8.0 to 24,631s for MELCOR 1.8.1, and radionuclide release to the environment decreased by a factor of 2 to 10, depending on species.

2. In both calculations, the reactor vessel failed via a penetration failure ~2min after molten debris began to relocate to the lower plenum; this occurred in spite of the relatively small mass of molten UO_2 entering a large water pool in the lower plenum and reflects the lack of an in-vessel molten debris-coolant interaction model.

3. Some details of the calculated in-vessel core melt progression (in particular, results related to material relocation) were surprising and warrant further investigation.

4. Large temporal variations in the airborne mass and size distribution of aerosols throughout the problem were calculated with no apparent physical explanation; in particular, the aerosol masses in virtually all sections (size bins) changed in a near-oscillatory fashion during a period of the accident when relatively little else was occurring.

5. Finally, this problem (as well as the experiences of two committee members in MELCOR application) emphasizes the need for the code user to "adjust" input parameters to obtain a plausible sequence of in-vessel events. Melting, relocation and refreezing of BWR fuel canister and control blade materials were observed to be strongly dependent on, among other things, the user's selection of criteria for oxide shell failure, debris and lower core support plate porosity, and selected melt-structure heat-transfer coefficients, highlighting the need for more extensive user guidelines and possibly improvements in default values.

13.5 MELCOR PWR Demonstration Calculation

SNL also performed a calculation of a Surry station blackout (TMLB') accident with MELCOR. This was the first fully-integrated PWR severe accident calculations performed with the code (since the TMI analysis only included in-vessel phenomena).

In general, the committee was favorably impressed with the overall performance of MELCOR and the results of the Surry calculation; based on station blackout predictions made with other codes, the results appeared reasonable. An exhaustive review was not performed on the Surry results, primarily because the committee did not allocate sufficient resources and time to this effort. Some of the surprising or noteworthy results, from a brief review, are:

1. After the steam generators dried out, the pressurizer level rose and remained near the top of the unit until vessel failure. RELAP5 results [132] and hand calculations both indicate that entrainment of water when the power-operated relief valves (PORVs) lift should lower the water level further after the surge line uncovers. Countercurrent flow of water and steam in the surge line, modelled in CVH by using a flooding correlation, should also reduce the water level in the long term. Neither effect was very significant in the MELCOR calculation.

2. Primary system gas temperatures were quite low, especially in the coolant loops. Part of this can be explained by the fact that core/upper plenum and upper plenum/steam generator natural circulation are not currently represented in MELCOR. In addition, the loop seals in the reactor coolant pump suction piping did not clear, even long after vessel failure, so that natural circulation around the entire primary system was precluded.

3. In-vessel hydrogen production was quite low; this may be due, in part, to steam starvation caused by the failure to model natural circulation between the core and the upper plenum.

4. The reactor vessel failed very soon after debris relocated to the lower plenum. This reflected a modelling change in MELCOR 1.8.1. Still, the code does not model the breakup of molten debris as it enters the lower plenum, so there is apparently no way to mechanistically avoid prompt vessel failure in this sequence.

5. One of the advantages of a unified modelling approach was evident in the steam generator results. After the steam generators dried out, a noticeable natural convection flow was calculated between the downcomer and tube bundle. Given the heat load from fission products deposited on the primary side of the tubes this is to be expected, but other severe accident codes that use a coarse modelling could not explicitly predict this effect. Such a capability would be useful for making estimates of peak tube temperatures in studies evaluating induced steam generators tube ruptures [133].

6. It was thought that the containment results would be greatly affected if debris dispersal from the reactor cavity was taken into account. This could not then be modelled. (The results after such a model was included are discussed in Section 23.5.) The same is true for induced hot-leg or surge-line rupture. (Note that MELCOR could model the system response to an assumed hot-leg or surge-line failure, but cannot calculate the occurrence of such a failure without a number of very specific and non-standard input-model changes very similar to those used with RELAP5 to calculate such failure [134].)

7. Cavity and upper containment water mass fluctuated in an erratic fashion after vessel failure; the reason for this is not known.

This Surry calculation was the basis of the more extensive PWR assessment demonstration calculations discussed below in Section 23.5.

14 SNL QCTA Program

A number of assessment calculations have been done at Sandia as part of a quality control and technical assessment program, including some repeats of analyses done in the earlier 1986 V&V assessment effort (summarized in Section 2). In contrast to that 1986 V&V effort, this program at Sandia concentrated on PWR primary systems and on fission product and aerosol release and deposition.

14.1 HDR Containment Experiment V44

The reactor-scale steam blowdown experiments conducted at the HDR facility near Frankfurt, West Germany, by KfK in 1982 contribute to the understanding of the physical processes taking place within the containment after a loss-of-coolant accident and expand the data base of energy and mass transfer within a large and complex containment building. One of the more common uses of MELCOR is to predict the containment response to accident scenarios in which primary coolant inventory is vented or otherwise lost to containment. The HDR series of tests is the largest-scale data available for assessing the MELCOR code's accuracy and reliability in predicting such response.

Experiment V44 [95] was one of a series of six water and steam blowdown experiments conducted in the HDR facility to simulate full-scale loss-of-coolant accidents; this test was initiated from saturated steam conditions, and had the highest reactor pressure vessel liquid level with the vessel nearly full. A MELCOR 1.6.0 calculation was performed for this experiment as part of a previous verification and validation program [4]. That MELCOR calculation was rerun with version 1.8.0 of the code, using assorted versions from 1.8.0DN to 1.8.0EC for the various calculations done, with results documented in a letter report [54]. Those results were compared to the previous MELCOR results and experimental data, and to a CONTAIN calculation for HDR V44 [135].

Rerunning a past assessment calculation, as done here for HDR V44, checks whether accumulated major and minor code modifications in the intervening years have significantly changed the predicted behavior. Furthermore, the later MELCOR analysis includes sensitivity studies on time step control, on noding detail, and on heat structure modelling and heat transfer coefficient correlations, most of which were not done for the original HDR V44 assessment study.

Earlier-time (<1hr) pressures and temperatures were higher in the later-code calculation than the values obtained with the older code version; the late-time pressures and temperatures for both MELCOR versions were in very good agreement (within 1%) with test data and with each other. The major modification in the MELCOR code, from version 1.6.0 to version 1.8.0, believed responsible for these different results, is a change in the heat transfer coefficient correlations calculated in the HS package (between versions 1.6 and 1.7, in late 1987).

The time step throughout most of the MELCOR basecase calculation was limited to a maximum of 1 second. When the code was allowed to use whatever time step the

internal code logic dictated, the overall behavior predicted was almost unchanged; the main difference is that, with the unrestricted time step, the heat transfer during a time step is occasionally greater than required to completely vaporize a thin pool of liquid water on the bottom of the control volume and on the surface of the heat structure modelling the floor, resulting in occasional temperature overshoots. The decrease in run time (a factor of 4-5) and the increase in average time step (a factor of ~10) were substantial.

The HDR V44 transient was analyzed using three different input decks, with varying degrees of modelling detail. The results show that including more detail in a MELCOR model does not unconditionally guarantee more accuracy. While each model yielded results agreeing better with some facet of the test data than did the others, none of the three gave obviously superior results for all aspects of the problem. In particular, the finer-node model gave better results only for the measured peak pressures, which occur for a very brief period very early in the transient, while the two coarser models gave better agreement with the observed pressures, and temperatures and temperature gradients for most of the problem period thereafter. There was a difference of two orders of magnitude in the run time required for the coarsest and finest nodings used. The comparisons with test data for this particular problem suggest that, for overall system response, the results to be expected using a finer input model for MELCOR often may not justify the increased costs.

The results suggest that the turbulent, rather than laminar, heat transfer coefficient correlations should be examined more carefully, to determine their impact on the overprediction of early-time peak pressures and temperatures. During most of the first minute (the steam blowdown period), the turbulent natural convection heat transfer correlation is used for pool heat transfer while heat transfer to atmosphere uses turbulent forced, mixed and natural convection correlations. The heat transfer around the system then switches slowly to mixed laminar/turbulent natural convection and pure laminar natural convection conditions later in the problem, when the predicted results are in significantly better agreement with test data.

When we reviewed the MELCOR 1.6.0 input model, the user-specified characteristic lengths input for the heat structures (used in evaluating the heat transfer coefficient correlations) seemed unexpectedly large, so we did a few studies in which these lengths were reduced. The pressure predicted using the smallest characteristic lengths agrees very well with test data, both in the magnitude of the peak and during the early blowdown period in general, but then underpredicts the late-time pressurization (which the large-characteristic-length "old basecase" analyses match well). These results imply that a single, constant, user-specified characteristic heat transfer length may not be adequate to represent a wide range of fluid conditions and heat transfer processes.

14.2 LACE LA4 Aerosol Transport and Deposition

The LWR Aerosol Containment Experiments (LACE) program [136] was a cooperative effort to investigate inherent aerosol behavior for postulated accident situations for

which high consequences are presently calculated in risk assessment studies because either the containment is bypassed altogether, the containment function is impaired early in the accident, or delayed containment failure occurs simultaneously with a large fission-product release. A series of six large-scale experiments has been conducted at the Containment Systems Test Facility (CSTF) at Hanford Engineering Development Laboratory (HEDL).

The MELCOR code has been used to simulate LACE experiment LA4 [57], an integral aerosol behavior test simulating late containment failure with overlapping aerosol injection periods [137, 138, 139]. In this test, the behavior of single- and double-component, hygroscopic and nonhygroscopic, aerosols in a condensing environment was monitored. MELCOR results were compared to experimental data, and to CONTAIN [140] calculations for LACE LA4 [141]. The reason for the difference in predicted suspended aerosol masses in the two codes is the larger aerosol particles calculated by MELCOR; the reason for the difference in aerosol particle sizes is primarily the different agglomeration shape factors used.

MELCOR calculated the thermal/hydraulic and aerosol response phenomena observed during the LACE LA4 experiment. The lack of any hygroscopic effects in the MELCOR aerosol treatment is visible mostly as the lack of any calculated difference in the behavior of the hygroscopic CsOH and the nonhygroscopic MnO aerosols. MELCOR predicted aerosol particles generally larger than measured, which then settled faster than observed, and consequently less suspended aerosols were leaked and/or plated in the calculation than in the experiment.

The MELCOR LA4 analysis included sensitivity studies on time step effects, wall and pool condensation, radiation heat transfer, number of aerosol components and sections, impact of non-default values of shape factors and diameter limits in the aerosol input, and the degree to which plated aerosols adhere to the walls or are washed off by draining liquid condensate films. The results showed that water should be modelled as a separate aerosol component in this problem, and that more sections (size bins) than the MELCOR default should be used. Including atmosphere-structure radiative heat transfer, even at the relatively low temperatures (300-400K) characteristic of this test, produced better agreement with data, as did using a detailed volume-altitude table reflecting the differences in sump pool liquid surface area with elevation in the elliptical lower head. There was a strong effect of whether plated aerosol mass was allowed to wash off heat structures with condensate films draining down into the pool. The suspended aerosol results depended most strongly on the value used for the agglomeration shape factor, with a much weaker (but still visible) dependence upon the dynamic shape factor.

Although there has been a lot of discussion recently on numeric effects seen in other MELCOR calculations, no machine dependencies were seen in this problem, and smooth convergence in results with reduced time steps was demonstrated.

14.3 FLECHT SEASET Natural Circulation

The Full-Length Emergency Cooling Heat Transfer Separate Effects and Systems Effects Test (FLECHT SEASET) program [142, 143] was a cooperative NRC/EPRI/Westinghouse effort to investigate heat transfer and hydraulic phenomena in a Westinghouse PWR primary system. One part of this program [144, 145] consisted of a series of natural circulation tests in a 1:307-(volume-)scale facility, with prototypic full lengths and full heights. The FLECHT SEASET test series was selected for assessment of MELCOR's ability to correctly model early-time natural circulation both because it is done in a larger-scale facility than the equivalent 1:1705-scale Semiscale natural circulation tests more commonly used for code assessment [146, 147, 148, 149], and because it covers a wider range of primary-system-inventory conditions than the equivalent 1:134-scale PKL natural circulation tests [150, 151, 152].

Steady-state single-phase liquid, two-phase and reflux condensation modes of natural circulation cooling were established, and flow and heat transfer characteristics in the different cooling modes were identified. In addition, other tests studied the variation of single-phase liquid natural circulation with changing core power or with different secondary side heat removal capabilities, and the effect of noncondensables on two-phase natural circulation flows.

MELCOR version 1.8HN was used for all the calculations in the report [55]. That report gives results of MELCOR calculations for single-phase liquid and two-phase natural circulation conditions, including comparisons to experimental data, sensitivity studies on time step effects and machine dependencies, and on noding variations and code modelling options. The MELCOR code developers provided a discussion of the importance and generality of some of the code problems encountered during this assessment analysis, and of possible fixes.

MELCOR correctly calculated the thermal/hydraulic phenomena observed during steady-state, single-phase liquid natural circulation. MELCOR predicted the correct total flow rate and the flow split between two unequal loops without any *ad hoc* adjustment of the input. The code could reproduce the major thermal/hydraulic response characteristics in two-phase natural circulation, after a number of nonstandard input modelling modifications; MELCOR could not reproduce the requisite physical phenomena with "normal" input models.

One major input model change consisted of subdividing the steam generator U-tubes into stacks of multiple control volumes. The top elevations of the control volumes containing the U-tubes were adjusted to lie above the top of the connecting horizontal flow path opening heights, and small incremental volumes were added in the volume-altitude tables in those control volumes; this is an input trick to ensure that a minimal atmosphere is always present and the nonequilibrium physics model always used in the control volume. Other required input changes included enabling the nondefault bubble rise model to account for interactions of bubbles with the pool, and increasing the junction opening heights between vertically-stacked volumes.

With these various input modifications, the correct dependence of mass flow on system mass inventory was obtained; the pressure and temperatures were then calculated to be in good agreement with test data. However, even in this case, the two-phase flow was overpredicted by ~30%, possibly because of incorrect two-phase interface and/or wall friction code models. As in the single-phase liquid natural circulation calculations, the two-phase simulations experienced a lot of subcycling and repeated advancement attempts, and very oscillatory time steps.

Although there has been a lot of discussion recently on numeric effects seen in other MELCOR calculations, no significant machine dependencies were seen in sensitivity studies for this problem; however, much smoother two-phase mass flow rates were calculated with a substantially reduced time step.

14.4 ACRR ST-1/ST-2 Source Term Experiments

Calculation of the fission product release from degraded fuel in a light water reactor (LWR) core uncovery is the first step in determining the overall radiological source term. Currently, the basis for most in-vessel fission product release calculations in large codes is the CORSOR model [153]. The CORSOR model is a simple correlational relationship based on data from early out-of-pile experiments at atmospheric pressure in a steam/helium atmosphere, performed at Oak Ridge, and on Kernforschungszentrum Karlsruhe's SASCHA tests. [154] Release of volatiles is assumed to be limited by diffusion, and all volatiles share the same release parameters, obtained by averaging experimental results; release of nonvolatiles is assumed to be limited by vaporization, and vapor pressures are scaled for consistency with experimental observations. Other parameters possibly affecting release rates (such as pressure, atmospheric composition, fuel characteristics, chemistry, radiation environment, flow rates and the extent of fuel degradation) are not considered explicitly. The CORSOR code has just been updated, and this CORSOR-Booth model [155] has been added to MELCOR.

The ACRR Source Term (ST) test series was designed to investigate some of the parameters not modelled in CORSOR that may affect fission product release, in particular, the pressure, atmospheric composition and fuel degradation states. The ST experiments provide time-resolved fission product release data to help validate models and to identify important release mechanisms which may be neglected in current codes. ST-1 and ST-2 were performed with approximately the same temperature history, fuel characteristics, hardware configuration, sampling methods, and hydrogen partial pressure. The main difference in the experiment conditions was in the pressure and in the gas velocity through the fueled test section. A more significant difference for our assessment analyses, however, was that there is substantially more documentation available for the ST-1 experiment (a final data report [156] as well as a journal article [157], a conference paper [158] and at least one supporting report [159]) than for the ST-2 experiment (a few comments in the aforementioned conference paper).

The MELCOR 1.8.1 code has been used to simulate both the ST-1 and ST-2 experiments, with MELCOR versions 1.8IG, II and IM were used for various calculations

whose results are shown in the report [58]. Note that, unlike the recent MELCOR 1.8.0 and current 1.8.1 calculations for the PHEBUS B9+ core damage test [20, 21], which is somewhat similar in geometry and scale to the ACRR ST experiments, no special code modifications were used to represent any of the test geometry, although extensive use was made of both the control function and sensitivity coefficient features in MELCOR to represent various aspects of these tests and their geometry. The only special code version built for and used in the ST assessment analyses was one with the CORSOR-Booth option wired in, because of a programmatic requirement to have this assessment completed and documented before that new model was to be installed in MELCOR 1.8.1.

MELCOR analyses were done using the CORSOR, CORSOR-M and CORSOR-Booth release models. Both release rates and total releases calculated by MELCOR generally agreed well with test data. Both qualitative and quantitative differences between volatile and refractory species were correctly reproduced. The more volatile species (Xe, Cs, I and Kr) were released starting earlier and peaking earlier than the more refractory species (Ba, Sr and Te) and most of the volatiles' initial inventories were released, while only part of the initial masses present were released for the more refractory species. Very low release fractions were predicted for the most refractory species (U and Zr) which agreed well with the limited test data.

None of the release model options produced consistently better agreement with test data for all species considered. The new CORSOR-Booth model matched the europium test data best, while CORSOR and CORSOR-M significantly underpredicted Eu release. CORSOR-Booth predicted less release of all volatiles than the nearly complete release calculated using either the CORSOR or CORSOR-M options; none of the models predicted the different release fractions measured for the various volatiles. CORSOR-Booth and CORSOR-M underpredicted the releases of refractory species such as Ba/Sr and Zr/U, while the CORSOR results for those species appear in good agreement with test data.

The MELCOR results also were compared directly to the release rate correlations as functions of temperature, using control functions, and to ST-1/ST-2 results obtained by Battelle using their standalone CORSOR code, to verify that the models have been implemented correctly within MELCOR.

Because the release is a very strong function of temperature, it was important to match the experimental temperature distribution as well as possible. Sensitivity studies showed no significant temperature dependence on changes in power, pressure or gas flow (within the experimental uncertainties and variations), or on convective heat transfer coefficients; the temperatures calculated were sensitive to the insulation thermal conductivity and the view factors used in radiation heat transfer from the fuel to the shroud.

The fuel damage observed in the ST experiments could not be predicted by the current version of MELCOR, because it does not include the fuel/clad interaction postulated to have occurred in the tests' reducing environment [159]. Results using a control function model representing portions of the postulated interaction suggest that such an interaction would produce the observed fuel end state.

Sensitivity studies checking for time step and noding effects, and for machine dependencies, were done. The major problem identified was a machine dependency associated with exponentials and very small numbers; it resulted in significantly different releases being predicted on different machines for refractory species. Other problems associated with differences in roundoff of small numbers were also found. All these problems were corrected immediately, and no machine dependencies were found in our final calculations.

14.5 LOFT LP-FP-2

The MELCOR code (version 1.8KA) has been used to model experiment LP-FP-2 [160, 161, 162, 163, 164, 165, 166, 167, 168, 169, 170], thus simulating many of the primary system and core thermal/hydraulic conditions that would be expected during a PWR V-sequence. These conditions led to uncovering of the core and to severe fuel damage in the central fuel module (CFM) which contained the test fuel bundle. Temperatures exceeding the zircaloy melting point were maintained for $\sim$ 260s which resulted in the release of fission products and the generation of aerosols. The relatively large scale of the test, and the extensive instrumentation used to model the thermal/hydraulic response, the core behavior and the effluent release from the primary system, make the LP-FP-2 experiment an important integral source of data for qualifying severe accident code predictive capabilities. This assessment analysis [56] proved that MELCOR was, in fact, able to calculate most of the thermal/hydraulic, core damage, and source term response phenomena observed during the LP-FP-2 experiment.

Our MELCOR results can be put into perspective best, perhaps, by examining them in relation to the performance of other codes in predicting this very challenging experiment [170]. MELCOR does at least as well as other "best-estimate" (*i.e.*, SCDAP/RELAP5) or integral (*i.e.*, MAAP) codes in predicting the thermal/hydraulic and core responses in this experiment; in fact, MELCOR and MAAP appear to give the best agreement with data, especially for clad temperature histories. Further, MELCOR does at least as well as "best-estimate" fission product codes in predicting the source term (with a number of such codes having to be run in tandem and driven by test data or other "best-estimate" thermal/hydraulic and code damage codes to provide results equivalent to a single, integrated MELCOR calculation).

The predicted primary system pressure was generally lower than measured, while the predicted primary system mass inventory was generally higher than measured, but with a large uncertainty on the test data. The pressurizer was predicted to empty within 1min, in good agreement with test data, and the early-time intact-loop mass flow also was calculated in good agreement with measurement, despite the lack of a complicated pump coastdown model in MELCOR. Despite the differences in calculated and observed thermal/hydraulic response, the core uncovery, dryout and onset of clad heatup were calculated in excellent agreement with thermocouple data.

Sensitivity studies on parameters which directly affect the thermal/hydraulic response showed a significant dependence on several break flow modelling parameters, including

areas, discharge coefficients and loss coefficients used. Results showed little or no dependence on structural heat transfer, either on the magnitude of the convective heat transfer coefficients or on the correlation sets and characteristic lengths used, on the radiative heat transfer emissivity or path length used, or on the modelling of piping insulation, on bubble rise physics in flow paths, or on secondary system leakage. The sensitivity studies did find a strong dependence on the junction opening heights used in flow paths connecting vertical stacks of control volumes, particularly at the core inlet and outlet.

The core heatup predicted was in very good agreement with test data (even to the effect of enhanced core cooling and a partial rewet soon after core dryout and uncovery) until the onset of rapid metal-water reaction late in the transient. This behavior could not be predicted using the default models and parameters in MELCOR, but required changing the temperature switching from a low-temperature to a high-temperature set of zircaloy oxidation rate constants.

Post-irradiation examination (PIE) of the CFM [169] concluded that the material relocation and stratification in LP-FP-2 resulted in low-melting-point metallic melts near the bottom of the fuel bundle, a high-temperature $(U,Zr)O_2$ ceramic melt region above this, and a debris bed of fuel pellets near the top of the fuel bundle. The final material distribution in MELCOR is in reasonable qualitative agreement with the test results. A debris bed consisting mostly of solid UO_2 fragments overlies a central region where much of the oxidized and unoxidized zircaloy clad has refrozen, with the steel in the other structure refreezing at a somewhat lower average elevation and the control rod poison material flowing down to the lower core and core support plate before refreezing. The PIE identified a 79-86% blockage due to material relocation and stratification in LP-FP-2. There is no internal blockage model in MELCOR. With flow blockage approximated *via* input at $\geq$1400s, predicted clad temperatures are in better agreement with data; the agreement might be improved further if the blockage could be modelled as occurring at the "correct" (moving) core elevation, rather than simply at the CFM inlet.

The hydrogen generated in our MELCOR analyses is in good agreement with data. The reference MELCOR calculation, with the inner zircaloy liner of the insulating shroud assumed to oxidize at the same temperature and rate as the adjacent clad, showed 267g of hydrogen in the BST, while a sensitivity study in which oxidation of the shroud inner liner was neglected gave 218g of hydrogen in the BST. Two experimental data sets are available for comparison. Grab samples from the suppression pool indicating 205$\pm$11g reflect hydrogen generation during the transient because the tank was isolated just prior to reflood; the PIE indicated 63g and 118g of hydrogen, respectively, generated as a result of zircaloy oxidation in cladding shells and in relocated material in the lower bundle, for a total of $\sim$181g.

Modelling the CFM shroud proved important primarily because of its effect on preventing radiative heat transfer and coolant temperature equilibration in the two parallel, isolated core flow channels. Minor changes were noted varying zircaloy melt temperature or core axial noding resolution, eliminating a gaseous diffusion oxidation rate limit or axial conduction, or varying convective heat transfer in the core, refreezing heat transfer

coefficient values, minimum oxide shell thicknesses for material holdup or other structure composition (*i.e.*, steel or inconel).

A significant fraction of the most volatile species (Xe, Cs and I) were released using both the CORSOR and CORSOR-M expressions, with all three classes having nearly equal releases of ~7-11% (with the test data in the lower half of this range, with more I found released than Cs, Xe and Kr). Only the gap inventories were released for the most highly refractory species (*e.g.*, Ce, La and U) for all options, and also for Ru, Mo and Cd in the CORSOR-M version. CORSOR gave higher releases for several classes (Ba, Mo, Cd and Sn, and – to a lesser degree – Ru), while CORSOR-M produced significantly higher release of Te (with data indicating a Te source term somewhere inbetween). CORSOR-Booth predicted significantly lower releases (2-4%) for the most volatile species (Xe, Cs and I) than either of the older CORSOR options, in very good agreement with test data, while the releases of other species (Ba, Te, Cd and Sn) were intermediate between the CORSOR and CORSOR-M predictions. Calculations were done with both the low- and high-burnup CORSOR-Booth default constants, although the CFM fuel in the LP-FP-2 test would clearly lie on the low-burnup side of the expressions.

Different gap first-release times calculated with the different CORSOR and CORSOR-Booth options indicate that some differences existed in these calculations prior to clad failure. Analyses using CORSOR-M showed identical results up to the time of first clad failure and gap release, but this was not the case in preliminary calculations; a number of code problems had to be identified and corrected to obtain this expected result. We also thought that no differences should exist in calculations varying assorted MAEROS parameters prior to clad failure and subsequent aerosol release, but found that small differences were caused by the effect of the MAEROS input parameter changes on water droplets present in control volume atmospheres during the first portion of the transient (confirmed in a sensitivity study with specification of zero fog density through sensitivity coefficient input.)

Both machine-dependency and time-step studies, and evaluation of the new heat transfer model for partially covered core cells, indicate strongly that additional time step controls must be developed in the COR and/or CVH packages to avoid what appear to be unphysical, numerically-driven liquid level oscillations during core uncovery and dryout, and valve-setpoint over- and undershoots. The Cray, SUN, VAX and IBM gave very similar results, while the "same" analysis done on a PC gave visibly different results throughout most of the latter half of the transient, primarily due to the increase in both number and magnitude of liquid level oscillations during core uncovery. A compiler error was later found which caused these discrepancies, and more recent code versions now give nearly identical results on all platforms tested. Increasing the time steps used generally resulted in progressively larger and more numerous liquid level oscillations.

The results of both the reference analysis and the large number of sensitivity studies done suggest that more separate-effects assessment of MELCOR is needed, particularly for break flow in the early-time thermal/hydraulics and for rapid metal-water reaction during core damage. Numerical effects were significant in both the COR and HS packages for heat transfer under two-phase conditions, in the COR and CVH packages for liquid

level oscillations during core dryout, and in the CVH and FL packages for valve setpoint over- and undershoots. New time step control algorithms are now being developed to check and adjust for rapid liquid level changes in control volumes, and for valve setpoint over- and undershoots; preliminary results indicate that these will resolve many of the outstanding difficulties in these LOFT analyses.

This LOFT LP-FP-2 assessment analysis clearly demonstrates MELCOR's ability to fulfill a large portion of its primary intended use, the calculation of severe accidents from full-power steady-state initiation through primary-system thermal/hydraulic response and core damage to fission product release, transport and deposition. After a number of identified code errors were corrected, few nonstandard inputs and no code problem-specific modifications were needed to provide reasonable agreement with test data in all areas considered.

15 PMK Bleed-and-Feed

The MELCOR 1.8.1 code has been used at the Atomic Energy Institute in Hungary to simulate the PMK bleed-and-feed experiments done in a scale-model VVER-440 test facility [59], with comparison to results from corresponding RELAP5/MOD2 calculations. Nodalization studies and studies on several code modelling options were also done. Good agreement was found between calculations done by RELAP5/MOD2 and MELCOR 1.8.1 (JY version). The conclusion of this study was that the ability of the user to "match" the observed behavior through a small set of nonstandard input modelling changes allows MELCOR to be used in accident management and PRA studies for VVER-440 reactors in which such physics are expected to be encountered.

The PMK-2 integral experimental facility [172] is used to understand the effectiveness of bleed-and-feed manipulations in VVER-440 reactors. The facility design is a 1/2007-scale single-loop model of the 6-loop VVER-440/213 reactor in the Paks Nuclear Power Plant, with full heights preserved.

The calculations indicated that the key to correctly simulating the bleed-and-feed experiment is a detailed representation of the steam generator in the test facility. MELCOR was able to correctly represent the basic physical phenomena found in the RELAP calculations, using a detailed heat structure model in the steam generator. To produce reasonable results, the steam generator primary side was subdivided into three control columes, and in each volume 16 heat slabs, stacked vertically, were used. So many heat slabs were used because the steam generator heat sink was very sensitive to the collapsed liquid level in the steam generator, and because the added expense in computational time for heat structures in this problem was not so high as for increasing the number of control volumes.

During these calculations, some divergencies were found in the heat structure response with a heat slab node thickness less than 1mm and a large heat source. The calculation run with MELCOR 1.8.1 (HY version) run on a VAX terminated, but the JY version (run on a PC) continued running, albeit with unrealistic results. Similar difficulties were found in the case of horizontal heat slabs with similar thin nodes.

16 MELCOR Applications in PRAs at SNL

MELCOR has been used at Sandia in a number of PRA applications. In the NUREG-1150 study [60] reassessing risk at five plants, MELCOR was used to perform containment response calculations [61]. In the phenomenology and risk uncertainty evaluation program (PRUEP), MELCOR calculations were performed as part of an integrated risk assessment for the LaSalle plant [62]. MELCOR calculations have been done updating the source term for three accident sequences (AG, S2D and S3D) in the Surry plant [63]. MELCOR is currently being used in a program assessing risk during low power and shutdown modes of operation at the Grand Gulf plant (with Brookhaven performing a parallel study for a PWR).

16.1 NUREG-1150 Supporting Calculations

MELCOR was used to help address many phenomenological questions in the accident progression event trees and to provide guidance for the expert opinion studies for the NUREG-1150 probabilistic risk study [60]. The MELCOR analyses included integral calculations covering an entire accident sequence, as well as calculations addressing specific issues that could affect several accident sequences. Analyses were performed for both PWR and BWR plants.

Two integral MELCOR calculations were performed for a station blackout scenario at Grand Gulf. The basecase was a station blackout with nominal leakage between the drywell and outer containment, and no outer-containment burns. A variation was performed in which a large outer-containment burn was assumed to occur before vessel breach, creating a large hole in the drywell wall. Additional MELCOR calculations were performed using a simplified deck to examine the flammability in various regions of containment, as well as numerous calculations to characterize containment response to burns initiated over a wide range of conditions. MELCOR and HECTR calculations were done to examine the effect of spray injection into a steam-filled containment, and MELCOR calculations were done to investigate the potential for pushing water over the weir wall onto the drywell floor.

An analysis of the Peach Bottom containment response following vessel breach was performed using MELCOR, as was a very limited analysis to estimate the timing for boiling the Sequoyah reactor cavity dry with a coolable debris bed submerged under a large pool of water. The LaSalle reactor building response following wetwell venting or drywell failure was examined using MELCOR, and an integral calculation for a short-term station blackout was done.

16.2 LaSalle PRUEP Study

Phenomenological calculations have been done in support of the Level II/III portions of a PRA for the LaSalle County Unit 2 nuclear power plant [62], using detailed integrated

thermal-hydraulic calculations to evaluate baseline representations of the dominant accident progressions of the PRA, and investigating the uncertainties arising from model limitations, phenomenological uncertainties and uncertainties in the initial conditions, using sensitivity calculations and expert judgement.

The latest released version of the MELCOR code at the time, version 1.8.0, was used. The sequences analyzed included high- and low-pressure, short-term station blackouts; an intermediate-term station blackout; and a long-term station blackout. Twelve studies were performed for the high-pressure short-term station blackout sequence to investigate the sensitivity of the source term results to the size and location of the containment failure, and to combustion parameters; a sensitivity study was performed also on the low-pressure short-term station blackout investigating the effect of pedestal wall failure.

An early version of one of these short-term station blackout analyses was presented to the MELCOR Peer Review as the required BWR demonstration calculation (with results and conclusions described in Section 13.4).

16.3 Surry AG, S2D and S3D

This report [63] presents the results from three MELCOR calculations of nuclear power plant accident sequences and presents comparisons with Source Term Code Package (STCP) calculations for the same sequences. The three low-pressure sequences were analyzed to identify the materials which enter containment and are available for release to the environment (source terms), and to obtain timing of sequence events. The source terms include fission products and other materials such as those generated by core-concrete interactions. All three calculations, for both MELCOR and STCP, analyzed the Surry plant, a pressurized water reactor (PWR) with a subatmospheric containment design.

The AG sequence assumed the availability of both passive and active Emergency Core Cooling System (ECCS) safety systems for protection of the primary system. Containment protective systems available for use included the containment fan coolers and containment sprays. Since the containment recirculation spray system coolers were inoperable, there was no capability for containment heat removal as the accident progressed. The small break LOCA's, S2D and S3D, assumed the ECCS systems were unavailable, with the exception of the passive accumulators. For those two accident sequences, the containment spray systems were fully operable, including the capability for containment heat removal via the containment spray recirculation system coolers. Since each of the three accident sequences progressed through core melt, core slumping, reactor vessel failure, and ex-vessel core-concrete interaction, they provided a good test of the ability of MELCOR to simulate integrated accidents that progressed to the point of radionuclide release to the containment or environment.

There were no major differences in the behavior predicted for the AG large break LOCA sequence. Both MELCOR and STCP predicted a slow pressurization of containment as the ECCS water delivered to the core is boiled off removing the decay heat.

The containment was predicted to fail slightly later in time in the MELCOR calculation than in the STCP analysis, partly due to a slower pressurization rate and partly due to a higher failure pressure setpoint. After containment failure and associated loss of ECC, both codes predicted core damage, lower head failure, and debris ejection to the cavity. The core degradation process calculated by MELCOR was somewhat more gradual and extended than that predicted by STCP. Both codes predicted almost all the noble gases and alkali metal volatiles (CsOH) released, and most of the halogens (I). Significantly more alkali earth (Ba) release and significantly less chalcogen (Te) releases were calculated by MELCOR than by STCP. A small fraction ($\leq$5%) of the Mo, Cd and Sn were calculated to be released in the MELCOR analysis, with no STCP values for comparison. Both codes predicted only trace amounts of the refractories (Ru, Ce, La and U) to be released.

Fission product release results from STCP were not available for the S2D and S3D sequences for comparison to MELCOR predictions. Therefore, only timings of major events could be compared in these two cases. Neither code predicted containment failure in either case, primarily due to the continued availability for containment heat removal via the containment spray recirculation system coolers. Time to core uncovery, core damage and relocation, lower head failure and debris ejection to the cavity were not all that different. One major difference between results from from MELCOR and from STCP was the prediction of deflagrations occurring in both sequences in the MELCOR analyses, with associated containment pressure and temperature spikes; there were no deflagrations in the STCP analyses for either small break sequence.

The overall fission product source terms calculated by MELCOR for the S2D and S3D sequences, and for the AG sequence as well, showed some general similarities in predicted response. In all three cases almost all of the noble gases ($\leq$99%) and most ($\sim$85-95%) of the Cs and I volatiles were released; very little remained in the RCS and almost all were in the containment or (for the AG sequence) released to the environment. Intermediate amounts of Ba, Te, Sn, Cd and Sn (2-30%) were released, and only trace amounts ($\leq$1%) of the refractories Ru, Ce, La and U were predicted to be released.

This report adds three sequences (AG, S2D and S3D) in the Surry plant, a 3-loop PWR with subatmospheric containment, to the growing list of various accident scenarios analyzed using the MELCOR code. In addition to comparing the MELCOR results to those from previous analyses for these sequences performed using the STCP code, this report provides substantial documentation on the MELCOR calculations for primary system thermal/hydraulics, core degradation, containment response, core- concrete interaction, and fission product release and transport, in an attempt to provide reasonably complete documentation on the source term for future applications such as PRAs.

16.4 Grand Gulf Low-Power Shutdown PRA

The safety of commercial nuclear plants during full power operation has been previously assessed in many probabilistic safety assessment studies. Recent events at several

nuclear power generating stations, recent safety studies, and operational experience, however, have all highlighted the need to assess the safety of plants during low power and shutdown modes of operation. In contrast to full power operation, there is very little information on the safety of plants during low power and shutdown modes of operation. In the past, the assumption has been that power operation is the risk dominant mode of operation because the decay energy is greatest at the time of shutdown and then decays as a function of time. Thus, the rationale was that during shutdown modes of operation the decay heat would be sufficiently low that there would be plenty of time to respond to any abnormal event that may threaten the core cooling function. Furthermore, given the unlikely event that a release did occur, radioactive decay would lessen the radiological potential of the release. This argument's Achilles' heel is that the technical specifications allow for more equipment to be inoperable in off power conditions. Thus, while there may be more time to respond to an accident during shutdown, many of the systems that are relied on to mitigate an accident during power operation may not be available during shutdown.

To gain a better understanding of the risk significance of low power and shutdown modes of operation, the Office of Nuclear Regulatory Research at the NRC established programs to investigate the likelihood and severity of postulated accidents that could occur during low power and shutdown (LP&S) modes of operation at commercial nuclear power plants. To investigate the likelihood of severe core damage accidents during off power conditions, probabilistic risk assessments (PRAs) were performed for two nuclear plants: Unit 1 of the Grand Gulf Nuclear Station which is a BWR-6 Mark III boiling water reactor (BWR) and Unit 1 of the Surry Power Station which is three loop, subatmospheric, pressurized water reactor (PWR). These studies consist of the following five analysis components: accident frequency analysis, accident progression analysis, analysis of the release and transport of radioactive material (*i.e.*, source term analysis), consequence analysis, and a risk integration analysis. A principle product of such a Level 3 PRA is an expression for risk.

The analysis of the BWR was conducted at Sandia National Laboratories while the analysis of the PWR was performed at Brookhaven National Laboratory. A multi-volume report [64] presents and discusses the results of the BWR analysis. Volume 1 summarizes the overall results. Volumes 2-5 present the accident frequency analysis (*i.e.*, Level 1). Volume 6 presents the Level 2/3 analysis performed under FIN L1679. Part 1 of Volume 6 presents the accident progression, radionuclide release and transport, consequence and risk analyses. Part 2 of Volume 6 presents the deterministic code calculations performed with the MELCOR code that were used to support the development and quantification of the PRA models.

In that report, the background for the work is summarized, including how deterministic codes are used in PRAs, why the MELCOR code is used, what the capabilities and features of MELCOR are, and how the code has been used by others in the past. Brief descriptions of the Grand Gulf plant and the configurations and plant operating states (POS) during LP&S operation, and of the MELCOR input model developed for the Grand Gulf plant in its LP&S configuration are given. The results of MELCOR analyses

of various accident sequences for the POS 5 plant configuration are presented, for accidents initiated at several different times after scram and shutdown, including shortened thermal/hydraulic and core damage calculations done in support of the Level 1 analysis and full plant analyses, including containment response and source terms, supporting the Level 2 analysis. MELCOR calculations of various accident scenarios for POS 6 also are given; these include a reference calculation and sensitivity studies on both plant configuration assumed and on code input options used.

A series of MELCOR calculations were done to support the quantification of the Level 1 PRA models for POS 5. POS 5 is rigorously defined as: "Cold Shutdown (Operating Condition 4) and Refueling (Operating Condition 5) only to the point where the vessel head is off." For these calculations, the parameters of interest include the times to reach various pressure and/or level setpoints, the time to top-of-active-fuel (TAF) uncovery, the times to core heatup and clad failure and the time to vessel failure. Several general scenarios when the plant is in POS 5 have been considered:

1. open MSIVs,
2. low pressure boiloff,
3. high pressure boiloff with closed RPV head vent,
4. high pressure boiloff with open RPV head vent,
5. large break LOCA,
6. station blackout with failure to isolate SDC,
7. station blackout with firewater addition,
8. station blackout with 10 hr firewater addition followed by high pressure boiloff, and
9. station blackout with 10 hr firewater addition followed by failure to isolate SDC.

In all these Level 1 cases, the drywell personnel lock is open; the containment equipment hatch and both of the containment personnel locks are open.

Calculations were performed for several different times from shutdown for each of these accident scenarios: 7 hr, 24 hr, 59 hr, 12 days, and 40 days. The first two times correspond to the times used to determine the decay heats for the first and second time windows; the third time corresponds to the midpoint of the second time window; the last time corresponds to the time corresponding to the decay heat level in the third time window. Because the primary interest was in time to core damage, these Level 1 support calculations were run until any of the following: vessel failure, code abort or 24 hr of transient. If any sequence produced no significant core damage within 24 hr for a given decay heat level, no further calculations were done with longer shutdown time s (*i.e.*, lower decay heat levels).

Table 16.4.1. MELCOR Level 2 Support Calculations – Sequences and Relative Contribution of Plant Damage States to Core Damage Frequency

Plant Damage State	Time After Shutdown	Fraction Contributed	Sequence Description
PDS 3-1	40 day	0.338	LBLOCA with flooded containment
PDS 2-2	24 hr	0.242	SBO w/o firewater, break in SDC
PDS 2-1	24 hr	0.170	LBLOCA with flooded containment
PDS 2-4	24 hr	0.104	Low-P Boiloff with flooded containment
PDS 1-3	7 hr	0.032	SBO w/10 hr-firewater, High-P Boiloff
PDS 1-1	7 hr	0.019	LBLOCA with flooded containment
PDS 1-2	7 hr	0.015	SBO w/o firewater, break in SDC
PDS 1-5	7 hr	0.008	Low-P Boiloff with flooded containment
PDS 2-5	24 hr	0.007	High-P Boiloff with closed containment
PDS 2-6	24 hr	0.006	Open MSIVs with closed containment
PDS 2-3	24 hr	0.054	Same as PDS 2-2, but with potential to recover AC power
PDS 1-4	7 hr	0.005	Same as PDS 1-2, but with potential to recover AC power

Based partly on the results of the MELCOR calculations done in support of the POS 5 Level 1 analysis, a number of accident sequences were eliminated from consideration as not resulting in core damage within the first 24 hr from the start of the accident. The remaining sequences, those leading to core damage within 1 day and with a frequency greater than the Level 1 truncation frequency, were grouped into plant damage states or PDSs. The plant damage states are ranked by their relative contribution to core damage frequency as:

Complete MELCOR accident analyses have been done for these sequences in support of the Level 2 PRA, with results described in detail. (The last two sequences in the table are identical to other sequences in the table with regard to MELCOR calculations, but with different recovery assumptions in the Level 2 PRA.)

An abridged risk analysis was performed on the early portion of the refueling mode of operation. In the Level 1 coarse screening analysis this mode of operation is referred to as plant operating state 6 (POS 6). During a refueling outage, the plant will enter POS 6 prior to loading fresh fuel (*i.e.*, going down) and then following fuel transfer on the way back up to power conditions (*i.e.*, going up). In this POS 6 study, only the going-down phase is analyzed. POS 6 begins when the vessel head is detached and ends when the upper reactor cavity has been filled with water. Prior to this mode of operation,

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the containment equipment hatch and personnel locks have been opened, the drywell head has been removed and the drywell equipment hatch and personnel locks have been opened. Thus the suppression pool is effectively bypassed both from the vessel and from the drywell (*i.e.*, steam lines are plugged and the drywell is open).

All the MELCOR POS 6 calculations were done assuming that, at the start of the accident, shutdown cooling, suppression pool cooling and containment sprays are all unavailable and remain unavailable during the accident; coolant injection is not provided to the vessel during the accident, and suppression pool makeup is not dumped into the suppression pool. The MELCOR POS 6 calculations done included a number of variations on the exact plant configuration assumed. In addition, a few sensitivity studies were done on various code options and/or parameters.

In addition, three preliminary calculations were done [175] for a low decay power boiloff without any ECCS and all piping intact and for two LOCA accidents with a recirculation-loop double-ended pipe rupture, all with a simplified containment model and no auxiliary building model. The first LOCA calculation assumed only one LPCI pump was operated, while the other LOCA calculation assumed two pumps were available.

17 Independent Review of SCDAP/RELAP5 Natural Circulation Calculations

SCDAP/RELAP5 calculations of the Surry TMLB' accident scenario [66] showed that natural circulation (both in-vessel and hot leg countercurrent flow) transfers core energy to other regions of the primary coolant system. Furthermore, the SCDAP/RELAP5 code results predicted that either the hot leg pipe or the surge line pipe would fail and depressurize the system, precluding failure of the reactor vessel at high pressure.

The main objective of this exercise [134] was to review and assess the results and conclusions in [66]. Because the SCDAP/RELAP5 model relied heavily on results from Westinghouse experiments [176] and from supporting calculations [177] with the COMMIX code, those studies also were examined in detail, and use of these results was identified as a major source of uncertainty in the SCDAP/RELAP5 analyses.

Some of these uncertainties were examined by building a corresponding MELCOR model of the Surry plant and performing sensitivity studies with MELCOR on several modelling parameters. The MELCOR model developed for this problem used more empirical models than the SCDAP/RELAP5 model; in particular, the core was modelled explicitly in the SCDAP/RELAP5 model but was modelled simply as a volumetric heat source in the MELCOR model. Despite such differences, MELCOR results for the base-case TMLB' event were in good agreement with those of SCDAP/RELAP5. The MELCOR model used the same heat transfer and flow loss coefficients as the SCDAP/RELAP5 model as well as the same steam generator mixing fractions obtained from the COMMIX calculation and the Westinghouse test data.

The effect of the hot leg inlet vapor temperature history was examined by varying the decay heat and the oxidation energy in the reactor vessel core. The effects of various modelling parameters on the mass flow rate developed in the hot leg countercurrent flow loop was evaluated by varying flow loss coefficients, including radial heat conduction between the top and bottom portions of the split hot leg, and modelling heat and mass transfer between the split hot leg countercurrent flows.

Of all the parameters studied, variation in the decay heat most affected the results, with a 25% change in decay heat changing surge line failure times by as much as 25min. Variations of other parameters affecting hot leg countercurrent flow modelling assumptions altered the predicted failure time times by less than 10min.

A total-loop natural circulation calculation, for a pump seal leak scenario, was performed with this MELCOR model. The total-loop circulation flow rate was higher that the hot-leg/steam-generator circulation flow rate in the base calculation. The surge line was not heated as much in this scenario and was not vulnerable to failure. Instead, this calculation showed that the steam generator tubes are more vulnerable to failure in this case than found in the case that did not have total-loop natural circulation.

(In addition, to study the relationship between the steam temperatures in the inlet plenum of the steam generator and the steam generator circulation rate, an independent

computer model was developed, which revealed a deficiency in the SCDAP/RELAP5 natural circulation modelling. Sensitivity studies on various inlet plenum mixing parameters were performed with this model.)

18 ORNL Analyses

MELCOR has been used as a severe accident analysis tool for several Oak Ridge programs. MELCOR has been validated by ORNL as part of the High Flux Isotope Reactor (HFIR) Safety Analysis Report (SAR) quality assurance program, before using MELCOR as the primary analysis tool for their Chapter-15 design-basis accident analyses. Problems analyzed during the ORNL V&V effort [68] are discussed in Section 18.1. As part of a focused severe accident study for the Advanced Neutron Source (ANS) Conceptual Safety Analysis Report (CSAR), MELCOR is being used at Oak Ridge to predict the transport of fission product nuclides and their release from containment [69], as summarized in Section 18.2. ORNL has also completed a MELCOR analysis characterizing the severe accident source term for a low-pressure, short-term station blackout sequence in a BWR-4 [70], as described in Section 18.3. A detailed assessment of the MELCOR Radionuclide (RN) Package's fuel fission product release models has been performed at ORNL via simulation of ORNL's VI-3, VI-5, and VI-6 fuel fission product release tests, and comparison of MELCOR's predicted fission product release behavior with that observed in the tests, as summarized in Section 18.4. Section 18.5 describes work on a projects to prepare a fully qualified, best-estimate MELCOR deck for the Grand Gulf facility; duplicate a short-term station blackout sequence with the deck used for NUREG-1150, and the QAed deck; and to compare the results of the two analyses.

18.1 HFIR SAR MELCOR V&V

A series of calculations were done to validate and benchmark the HFIR MELCOR severe accident analysis model for application to loss-of-coolant severe accident scenarios. [68] The effort has focused primarily on validation of the reactor coolant system (RCS) portion of the model. All of the calculations described were performed on an IBM RISC-6000 Model 530 computer using MELCOR 1.8.1 (specifically, 1.8HN).

18.1.1 Null Transient

A null transient is a calculation in which no forcing functions are applied to the system and the model's predictions for steady state operational values (*e.g.*, pressures, temperatures, *etc.*) are compared with known operational data. Steady states were obtained both for full-power operation and for a post-scram state.

The calculated results agree favorably with nominal operating conditions for both full power and shutdown operation. The agreement between calculated and operating values ensures that the initial conditions calculated for the large break LOCA transient are reasonable.

18.1.2 Adiabatic Null Transient

Due to problems encountered in trying to provide completely consistent initial conditions for a complex model such as the integrated HFIR model, and because MELCOR predicted a significant pressure drop immediately following problem initialization in the absence of a "steady-state initialization" control volume acting as an auxiliary makeup system, a calculation was done in which all energy sources and sinks were removed from the model in order to demonstrate whether the RCS model was stable in the absence of nonequilibrium boundary conditions.

The results of this calculation indicate that the model is stable in the absence of external forcing functions. The results also indicate that the "RCS initialization volume" is required to provide makeup flow immediately after problem initialization to counter the coolant temperature decrease associated with heat transfer from the RCS coolant to the system piping and components; it may be possible in the future to eliminate the initialization volume by providing additional initial condition information for the RCS structures and confinement volumes.

18.1.3 CVH Energy Sources

This problem is an adiabatic heatup in which a known, time-dependent energy source is applied to the water in an otherwise adiabatic RCS, to demonstrate that the model can accurately predict the RCS loop-average temperature increase based on a known energy input.

The results of this calculation demonstrate that MELCOR does perform realistic energy balances on the RCS coolant inventory in the absence of RCS heat sinks.

18.1.4 "Spring Constant" Experiments

Because water-solid systems such as HFIR's are extremely sensitive to small changes in water temperature and makeup/letdown system flow, the HFIR's RCS volumetric expansion coefficient ("spring constant") was measured during a series of hydraulic tests [178]. Two calculations were made in which MELCOR's predictions for the RCS spring constant were compared to test data and to analytic calculations.

The calculated spring constants are significantly higher than measured values. Since part of the spring constant is due to expansion/contraction of the RCS loop piping, and part of the constant is due to the expansion/contraction of the water, it is clear that MELCOR should in fact overpredict the overall spring constant. An analysis of the HFIR RCS spring constant was done which evaluated the relative contributions of loop structural elasticity and water compressibility to the overall system spring constant. The spring constant results obtained from the MELCOR analysis compare extremely well with the theoretical predictions in which structural elasticity is ignored.

These two calculations demonstrate that MELCOR significantly overpredicts the RCS spring constant, due to the code's inability to model RCS loop structural elasticity. Thus, MELCOR's predictions for RCS pressure are significantly more sensitive to RCS mass and energy sources and sinks than is the actual system. This is believed to be of little importance for LOCA scenarios in which the system actually depressurizes very quickly; the significance for non-LOCA severe accident sequences will have to be evaluated on a case-by-case basis.

18.1.5 LBLOCA Comparison to RELAP5

Thermal/hydraulic performance parameters are especially important for severe LOCA sequences at HFIR because the fission product source term for these accidents is characterized by a tendency for the volatile fission products to remain dissolved in the (relatively) cool primary coolant water; thus, the total source term for these species is a direct function of the time-integrated leakage of the reactor coolant from the break. The reactor coolant break flow rate is determined by the pump performance in the broken and intact loops. Therefore, the MELCOR model must accurately predict the pump performance during the LOCA, but the simple pump model in MELCOR does not include detailed calculations of degraded pump performance such as are included in the RELAP5 code. RELAP5 has been extensively verified and validated against a wealth of LWR transient and experimental data, and is believed to provide the best available simulation of LOCA transient behavior. Agreement of the RELAP5 and MELCOR integrated leakage rates provides added confidence that the MELCOR model can be used for the prediction of the thermal/hydraulic and fission product source term for the HFIR.

A thermal/hydraulic computational model of the HFIR has been developed using the RELAP5 code [179]. That model includes a detailed representation of the reactor core and other vessel components, three heat-exchanger/pump cells, pressurizing pumps and letdown valves and secondary coolant system (with less detail than the primary system); limited validation has been performed [180] against plant data. This model is being used to simulate operational transients and LOCAs in support of the HFIR SAR.

The total integrated effluent mass from the primary system as calculated by MELCOR and RELAP5 differed by ~4% for the first hour following a large break LOCA for the HFIR, which is excellent agreement between two large codes designed for totally different objectives. However, the inability of the MELCOR model to predict the intact loop flows could significantly impact RCS fission product retention estimates for certain other transients such as a small break LOCA. The differences in cold leg header flows were insignificant compared to the total mass lost from the system.

This result is valid only for the large break LOCA for the HFIR, and caution should be used in extending the use of MELCOR to other accident types. Also, the RELAP5 calculation did not represent a realistic scenario (*i.e.*, it did not include fuel melt or decay heat); its use here was limited to predicting flow rates during a large break LOCA.

18.2 ANS Containment

As part of a focused severe accident study for the Advanced Neutron Source (ANS) Conceptual Safety Analysis Report (CSAR), MELCOR is being used at Oak Ridge to predict the transport of fission product nuclides and their release from containment [69], with the MELCOR Accident Consequence Code System (MACCS) [181] used to determine subsequent accident dispersion and radiation exposures.

The report describes the postulated severe accident scenarios, methodology for analysis, modelling assumptions, modelling of several severe accident phenomena, and evaluation of the resulting source term and radiological consequences.

Due to the early stage in severe accident technology development for the ANS, relevant tools have not been developed for evaluating core melt progression phenomena. Consequently, three different types of severe accident scenarios were postulated with a view of evaluating conservatively scoped source terms. To provide initial source term estimates for the high-consequence, low-probability end of the severe accident risk spectrum, early containment failure cases are also evaluated for the scenarios analyzed and reported. In addition, containment response for an intact containment configuration is also analyzed.

The first scenario evaluates maximum possible steaming loads and associated radionuclide transport. The core debris is this case is assumed to be confined within a water pool. At the beginning of the MELCOR calculations, it is assumed that a partitioning of fission products has occurred: all of the noble gases and 50% of the halogen inventory escape from the water and get sourced into the atmosphere of the primary containment high bay area volume, while the balance of the radionuclides stay behind and deposit their decay heat into the water, eventually causing steaming.

The next scenario is geared toward evaluating conservative containment loads from release of radionuclide vapors and aerosols with associated generation of combustible gases during molten core-concrete interaction. It is postulated, due to the very high power density of the ANS fuel debris, that during a core meltdown accident core debris could ablate penetration seals or other structures and relocate onto the concrete floor of the subpile room; thereafter, the core debris would spread and molten core-concrete interaction would begin. The containment will get challenged from the resulting loads arising from combustible gas deflagration and released radionuclides, in addition to other gases produced from molten core-concrete interaction and steaming (if flooding is employed). If flooding is employed, it is postulated that steam explosion loads, combined with aerosol suspension of nonvolatile fission products, will not occur. It is not apparent that a steam explosion in the subpile room or detonable quantities of combustible gases could directly threaten containment. From the standpoint of conservatism, the analysis of containment failure during molten core-concrete interaction was included. Several different containment configurations (including primary and/or secondary containment failure) are studied in combination with and without flooding during molten core-concrete interaction events.

The third scenario follows the prescriptions given by the 10 CFR 100 guidelines; it was included in the CSAR for demonstrating site-suitability characteristics of the ANS.

Various containment configurations are considered for the study of thermal-hydraulic and radiological behaviors of the ANS containment. These range from an intact primary and secondary containment (*i.e.*, containment isolation) to at least partial failure of both the primary and secondary containment. The worst containment failure mode (*viz.*, the failure of both primary and secondary containment) would occur in such a manner that a leakage path of some particular size would open to the environment. Severe accident mitigative design features such as the use of rupture disks were accounted for.

For all the intact containment configurations, including the 10 CFR 100 scenario, MELCOR predicted that only a negligible amount of radionuclides get released into the environment. The scenarios with the failure of the primary containment (with intact secondary containment) revealed that about 10% of the noble gas inventory and a few percent of volatile radionuclide inventories get released into the environment. For the cases with failure of both primary and secondary containment walls, however, the results show that about 10% to 20% of initial inventories of noble gases and volatile radionuclides are released into the environment. This source term information was used to drive MACCS for the evaluation of radiological consequences.

18.3 Peach Bottom Plant Analyses

ORNL has completed a MELCOR analysis whose purpose was to provide best-estimate source terms for two low-pressure, short-term station blackout sequences (with a dry cavity and with a flooded cavity) and a design-basis loss-of-coolant accident concurrent with complete loss of the ECCS in a BWR-4 [70]. The source terms include fission products and other materials generated by core/concrete interactions. The in-containment source terms generated by MELCOR are compared to those developed for NUREG-1150 [60] using STCP [106].

The plant analyzed, the Peach Bottom Atomic Power Station, is a BWR-4 with a Mark-I containment. The selected severe accident analyzed, a low-pressure, short-term station blackout, assumes that all power is lost except the DC power needed to actuate the automatic depressurization system (ADS) and the safety relief valves (SRVs).

Version 1.8.1 of the MELCOR code (specifically versions 1.8HN, IR and KH) and (in a few calculations) the MELCOR/CORBH package (of which a later version is included in MELCOR 1.8.2) were used to calculate the best-estimate timing of events and best-estimate source terms. The CORBH package [182, 183, 184] is a BWR lower plenum debris bed package developed by ORNL and interfaced with the MELCOR code, which has a more detailed model of the molten core/debris behavior in the lower plenum. (Note that then-unresolved problems with the CORBH package prevented execution to completion, and that the CORBH package was not interfaced with the fuel fission product release algorithms.)

Several different input models were used. The first model used a single control volume node for the drywell and a single cavity. The second model used a multi-node drywell model and two cavities, to provide the best-estimate source term results and the best-estimate containment failure time. The single-node drywell, single cavity model was used with the CORBH package to provide the best-estimate failure time. The calculations used the CORSOR release model with the surface-volume correction term, and used 16 RN classes in the source term calculations, including CsI as class 16. All of the released iodine (class 4) was assumed converted into CsI.

There are three possible containment failure modes in the Peach Bottom plant: by high drywell pressure, by high-temperature failure of the head flange seals coupled with moderate containment pressure, and by liner failure when molten debris contacts the liner. The third failure mode cannot be modelled due to CORCON cavity model limitations. The likelihood of the other two failure modes was studied for the short-term station blackout with a dry cavity through a number of calculations using different input models (a single control volume node for the drywell and a single cavity *vs* a multi-node drywell model and two cavities), and by varying the vessel failure time and ejection mass as noted below. All the calculations with a single drywell node predicted containment failure at the head flange seals by high temperature. Running a multi-node drywell model in which the debris remained in the first cavity for a long time predicted early failure of the drywell liner by high pressure; using a multi-node drywell model with the debris transferred from the first to the second cavity at a reasonable time predicted containment failure at the head flange seals by high temperature about 1hr later than the single-node drywell model.

Containment failure at the head flange seals resulted in small source terms released, with the containment remaining pressurized during containment failure. The source terms released into the environment were significantly larger for the cases with a large drywell breach area (liner melt=through or drywell liner ruptured by high drywell pressure) than when the drywell fails at the head flange seals by high temperature (which results in a very small leakage area).

For the station blackout with a flooded cavity, MELCOR predicted high pressure-induced containment failure in the wetwell more than 1hr later than predicted for the station blackout with a dry cavity. The flooded cavity kept the drywell volumes cooler than in the sequence with a dry cavity, and the head flange seals did not reach their failure temperature. Thus, one potential major impact of drywell flooding is to change the location of containment failure from the drywell to the wetwell.

In the LOCA sequence, the events occurred faster than in the station blackouts because of the large coolant loss during the initial blowdown. Vessel failure was predicted to occur about 1hr after transient initiation and containment failure by high temperature at the drywell head flange seals was predicted almost 4.5hr after transient initiation.

The in-containment source term for noble gases is 100% of the total inventory for all three sequences analyzed, in agreement with [185]. Source terms calculated by MELCOR for I, Cs, Sr and Ba exceed the values in [185], while MELCOR calculates smaller source

terms for Te, Ru and Ce. MELCOR also calculates smaller amounts of nonradioactive aerosols than the values in [185].

Of the three sequences, the LOCA has the highest in-containment source terms for I (as CsI) and Cs, because the LOCA has the lowest retention of these products in the reactor coolant system (about 21%). The station blackout with a dry cavity has the lowest releases of I (as CsI) and Cs of the three sequences and also the largest retention of these products in the reactor coolant system (about 40%). For the remaining fission products, the station blackout with a dry cavity has the largest source terms, the station blackout with a flooded cavity the smallest source terms for Te, La and Ce and the LOCA the smallest source terms for Ba, Sr and Ru.

The largest release of noble gases into the environment (89%) was predicted for the station blackout with a flooded cavity; in this sequence, the wetwell failed by high pressure and all the noble gases accumulated in the wetwell escaped. The LOCA has the largest releases of Cs and CsI into the environment; the LOCA also has the largest in-containment source terms for Cs and I (as CsI). The station blackout with the flooded cavity has the lowest releases for all the classes (except for the noble gases); the scrubbing effect of the water in the flooded cavity and in the wetwell retained most of the fission products.

The MELCOR results for the timing of significant events (vessel failure, containment failure, *etc.*) are compared in [70] to results from a calculation [186] for the same low-pressure short-term station blackout sequence at the Peach Bottom plant done using the BWRSAR [187, 188] and CONTAIN [189] codes. The environmental source terms calculated by MELCOR for the station blackout sequences were also compared to STCP results [60]; MELCOR calculated smaller releases than the STCP calculations.

Several sensitivity studies were done for the short-term station blackout sequence with a dry cavity as part of this analysis, with results summarized here. The uncertainties in these calculations are in five different areas: (1) containment failure mode and timing, (2) vessel failure timing, (3) source terms, (4) plant input data and (5) MELCOR input parameters.

One sensitivity study looked at the effect of the timing for the ADS actuation to depressurize the vessel and to "steam cool" the hot core, including a base calculation with no depressurization and four calculations with ADS actuated at four different times (when the water level is in the lower plenum or is exactly at bottom-of-fuel, and when the active fuel is either 1/3 or 2/3 covered with water).

Runs were done with two different core intact component porosities (0.99 and 0.53). The use of the larger porosity value for the intact components produces a large "packed" volume for the fuel and clad with little or no "free" volume left for relocation of debris. A porosity value of 0.53 (because only about 53% of the core cell volumes are occupied by solid components) allows free space for relocation of debris into the space between the fuel canisters. (The grid spacers are assumed to block debris relocation inside the fuel channels.)

An investigation of the effect of the amount of lower-plenum steel mass input was completed. The MELCOR input deck for Peach Bottom as transmitted from BNL had less steel inside the vessel than the actual amount of steel in the lower plenum, core plate and core, with the main source of the discrepancy in the masses of the control rod guide tubes and in the structural material in the lower plenum. The unexpected initial results of more steel inventory present resulting in less steel ejected from the vessel demonstrate a deficiency in the inability of the MELCOR COR package to transfer decay heat from the fuel debris to lower plenum structures when the lower plenum contains water (because the COR package only considers energy transfer from debris to structures by radiation when there is no water present).

Several attempts were made to force the fuel debris and lower plenum structures into contact to enhance the energy transfer. Increasing the porosity of the debris to 0.9 (which would cause the fuel debris to lodge between the lower plenum structures instead of falling directly onto the bottom head) significantly increased the amount of steel melted and subsequently ejected from the vessel. Other calculations varied the logical support flags, failure temperatures, structure surface areas and open flow areas in the various lower plenum levels. Supporting debris with a high failure temperature resulted in longer times to vessel failure containment failure, and more steel melted and ejected. Smaller core plate open flow areas resulted in less debris relocated into the plate and longer times to core plate failure and vessel failure. Smaller surface areas resulted in less heat transferred to lower plenum structures and less steel melted and ejected from the vessel.

The effect on reactor vessel failure timing of varying several lower-head penetration failure parameters was also investigated. The penetration failure temperature was increased from its default value to the melt temperature of steel. The heat transfer coefficient between the debris and the penetrations was reduced by factors of 1/10 and 1/100. Also, the masses of the penetrations were increased and that increased mass was removed from the lower plenum structural steel (to avoid duplication of mass). The time of vessel failure was found to be not very sensitive to input values for these various penetration parameters.

18.4 RN Package Assessment - VI Fission Product Release

This summary of work done at ORNL under FIN J6014 was provided for our survey report by the principal investigator, Sherrell R. Greene (615-574-0626).

The purpose of this project is to perform a detailed assessment of the MELCOR Radionuclide (RN) Package's fuel fission product release models. This comparison is being performed via simulation of ORNL's VI-3, VI-5, and VI-6 fuel fission product release tests, and comparison of MELCOR's predicted fission product release behavior with that observed in the tests.

The object of the VI-3 test was to investigate fission product release at two temperatures (2000 and 2700K) under strongly oxidizing conditions. The objective of the VI-5

test was to investigate fission product release at two temperatures (2000 and 2700K) under strongly reducing conditions (hydrogen-helium atmosphere) to provide a direct comparison with test VI-3 in steam. The objective of VI-6 test ws to obtain fission product release data for fuel heated at 2300K, first in hydrogen (to allow cladding melting and runoff), then insteam (to enable oxidation of the UO2 fuel pellets). Total test times ranged from one to approximately three hours.

The calculations conducted include runs with each of the six basic MELCOR fuel fission product release options (CORSOR, CORSOR with surface/volume ration correction, CORSOR-M, CORSOR-M with surface/volume ratio correction, CORSOR-Booth for low burnup fuel, and CORSOR-Booth for high burnup fuel) for each of the three tests. Time-dependent cumulative release fraction comparisons were conducted for twelve elements (Kr, Cs, Sr, Ba, I, Te, Ru, Mo, Ce, Eu, U, and Sb). In addition to the base calculations, several sets of parametric calculations were conducted prior to the VI-3 test. This measurement yielded an approximate particle radius of six microns (the default MELCOR value is 10 microns). The four sensitivity calculations conducted in this series employed particle radius values of one, three, six, and ten microns. The default activation energy value in the Booth model is 3.8E5 J/kg-mole. Six additional calculations were performed for the VI-3 test, in which the debris particle radius was fixed at six microns, but he activation energy parameter (Q) was varied between half of the default value and twice the default value.

The six most important radionuclides (in terms of health impacts) are I, Te, Cs, Sr, Ru, and Ba. None of the existing MELCOR fuel fission product release models provided final release estimates within the range of data uncertainty for all six of these elements in any of the three tests. The results of the comparisons indicate that there were only two cases in which a MELCOR fission product release model provided final release fraction estimates within the uncertainty range of the data for all three tests, (viz. CORSOR with the surface/volume ratio correction for Ba; and CORSOR-M and CORSOR-M with the surface/volume ratio correction for Te.) Thus, none of the existing models reliably provided final release fraction estimates within the rage of data uncertainty. (Comparisons of the time-dependent cumulative release fraction estimates from MELCOR to those observed in the tests were also conducted and will be documented in the final report.)

The last remaining VI-5 iodine samples were irradiated in March and the data should be available sometime in April. At this writing (early April 1994) completion of the final assessment report (an ORNL Letter Report) is pending receipt of the final VI-5 data and completion of the code prediction/data comparisons for the VI-5 experiment.

18.5 Grand Gulf Fully Qualified MELCOR Deck

This summary of work done at ORNL under FIN W6093 was provided for our survey report by the principal investigator, Juan J. Carbajo (615-574-5856).

Many of the MELCOR computer decks employed in NUREG-1150 were based on manipulations of previous input decks. For example, in the case of BWRs, the original

input deck was developed for Peach Bottom APS (BWR-4/Mark I) facility. It was then adjusted to represent the LaSalle facility (BWR-5, Mark II) and later adjusted again to represent the Grand Gulf (BWR-6/Mark III). Since the NUREG-1150 results are widely employed by researchers and regulators in many endeavors, the need exists to confirm the vailidity of the earlier input deck approximations and to verify that the NUREG-1150 results would not have been different if fully qualified (QED), best estimate computer models had been used in lieu of adjusting existing models.

The objective of this project is to prepare a fully qualified, best-estimate MELCOR deck for the Grand Gulf facility; duplicate a short-term station blackout sequence with the deck used for NUREG-1150, and the QAed deck; and to compare the results of the two analyses. The lack of significant differences will verify the NUrEG-1150 results. Significant differences will identify the need to prepare plant-specific QAed input decks and the need to re-evaluate the technique of adjusting the deck from one plant to make it "look like" another plant.

A secondary objective of this project is to identify source terms into containment when water is injected into the reactor vessel after core relocation has started but before vessel failure. Three recovery scenarios will be simulated by injecting water into the vessel at different times and with different flow rates.

The results of this effort will affect all regulatory efforts which are based on or rely on the results of NUREG-1150 by either supporting the conclusions or identify weaknesses in the reuslts. This includes the proposed updatted source terms (reported in NUREG-1465), revisions to 10 CFR Parts 50 and 100, and the value aspects of regulatory analyses. Work is currently ongoing and the final report for the project is expected to be avialable in August 1994.

19 THALES-2/STCP/MELCOR Source Terms in a BWR Severe Accident

A comparative study [71, 72] was performed by Japan Atomic Energy Research Institute (JAERI) for source terms in a severe accident at a BWR with Mark-I containment, using the THALES-2, STCP and MELCOR codes to identify phenomena in which uncertainties in analytical methods have a significant effect on source term evaluation.

The Browns Ferry nuclear power plant was selected as the reference plant. The accident sequence analyzed was S_2E, a small break LOCA with no ECCS or RCIC coolant injections; the break location was selected at the lowest elevation of a recirculation loop and the size of the break was set at 2in diameter.

THALES-2 is a coupled fission product transport and thermal hydraulic code developed at JAERI [190], while the Source Term Code Package (STCP) is a set of five codes [106] developed for the U. S. NRC which MELCOR is intended to supercede.

Results from the calculations were compared for timings of major events in the accident progression and for source terms. A number of differences in analytical models among the three codes were identified and their effects on source term were examined.

This study concluded that

1. the timings of major events in accident progression and source terms are primarily influenced by melt progression and revaporization, and
2. THALES-2 and MELCOR gave similar predictions for CsI release while STCP gave much lower CsI release (by two orders of magnitude).

The following models were found to have significant influence on the calculated source terms:

1. candling model for fuel rods,
2. models of core support plate failure and whole core collapse,
3. revaporization model, and
4. crust formation model

20 VTT Analyses of Plant Transients in TVO NPP

MELCOR calculations have been done for two plant scenarios in the Teollisuuden Voima Oy (TVO Power Company) nuclear power plant, including a MAAP/MELCOR comparison study with the MAAP runs done by TVO and the MELCOR runs done by Valtion Teknillinen Tutkimuskeskus (VTT), the Technical Research Centre of Finland. These analyses began using MELCOR 1.8.0 [73] and continued using MELCOR 1.8.1 [74], for the thermal/hydraulic aspects of the accidents. More recently, MELCOR 1.8.2 has been used to expand the TVO plant analyses to include fission product behavior in two accident sequences [75].

In addition, an initial station blackout with a 10% break in the main steam line with recovery of power and reflooding of the overheated reactor core with auxiliary feedwater system has been analyzed for the TVO plant using the MAAP, MELCOR and SCDAP/RELAP5/MOD3 computer codes [76].

20.1 Station Blackout and Main Steam Line Break Sequences

MELCOR 1.8DN and MAAP/BWR 3.0B/rev 6.05 and 7 were used on a Cray XMP/432 and an Acer 1100SX (386SX+387SX) PC, respectively, to calculate two accident sequences for the TVO power plant [73]. The sequences chosen were:

1. TB sequence – initial station blackout with manual depressurization of the primary system and simultaneous pedestal flooding at 1hr into the accident; wetwell venting was assumed to start when drywell pressure exceeded 6bar; and
2. MSL-Break sequence – initial station blackout with a large (200%) break in main steam line; ADS operates with normal logic and pedestal flooding valves were opened at 1800s.

In both accident cases, a leak of 5cm^2 area was assumed between drywell and wetwell.

The fission product models were not activated in these calculations, because of convergence error problems encountered after the core material relocated into the lower head. Further investigation of the problems with the fission product calculations was postponed until the new version (1.8.1) of MELCOR would be installed and tested. Also, the material relocation models used in the core and lower plenum regions seemed to give unrealistic predictions. The heatup of the core was very rapid in some core nodes, particularly in the main steam line break case, without any physically reasonable explanation.

The most significant differences between the results of the two codes were:

1. The debris was assumed to cool in the bottom head water pool in MELCOR, delaying the RPV failure by several thousands of seconds; in MAAP, the corium slumps through the bottom head within a minute after core plate failure.

2. The in-vessel hydrogen production in MELCOR was higher than in MAAP if the core blockage model was used in MAAP; if the core blockage model was turned off in MAAP, the in-vessel H_2 production in MAAP was higher than in the MELCOR prediction.

3. MELCOR assumes that corium is not coolable in the cavity; thus MELCOR predicted significant ex-vessel H_2 production and concrete ablation, whereas corium was assumed to be coolable in the pedestal, according to MAAP.

4. The pressurization of containment was faster according to MELCOR than according to MAAP; this could probably partly be explained with the fact that large amounts of CO_2 were released in the MELCOR calculation during the corium-concrete interaction.

The results obtained with MELCOR would need more variations run to check the sensitivity of various input parameters. Also, the control-volume nodalization could be more detailed, but this particular amount of calculational nodes was chosen to match roughly with the MAAP nodalization. One observation on the core behavior was that the heatup mode of MELCOR was very sensitive and the results could probably vary widely with different input assumptions. Also, different compiler options on the Cray seemed to give significantly different gas temperatures in the RCS, which could be a sign of bad numerics or device-dependent programming in MELCOR.

A new MELCOR code version, 1.8.1, was implemented on an HP Apollo 730 workstation and the same two accident scenarios were rerun [74]. (Installation and execution problems were noted installing MELCOR 1.8.1 on a Cray and, although installing without problems on a VAX, the TVO calculations ran very slowly and then eventually died in CORCON.) Fission product calculations were successfully included in these analyses. The main steam line break case was run up to the time of containment failure but the station blackout case run stopped at 5.1hr due to a CORCON error.

Some significant differences between the results obtained with MELCOR versions 1.8DN and 1.8.1 were noticed. The core material relocations and the behavior of corium in the lower plenum exhibited the largest differences. In the earlier versions, the corium was partly cooled in the lower plenum water pool before RPV failure, but in the newer code version the RPV failure follows the core plate failure very closely (within 1-2min in the TB case and 8-17min in the MSL-break case). This is due to code modifications in version 1.8.1 to limit the heat transfer between particulate debris and surrounding water pool by taking into account hydrodynamic phenomena.

Because of the different relocation histories, the in-vessel hydrogen production changed also. In the TB case, significantly less hydrogen was generated in-vessel using the new code version than in the old one (404kg to 150kg). In the MSL-break case, the in-vessel hydrogen production increased by 46kg to 319kg, using the new code version compared to the earlier version. In the containment behavior for the MSL-break case, the gas temperatures remained lower and the pressurization of the drywell was slower.

Core/concrete reaction results were judged to be unrealistic in both cases calculated; the heavy and light oxide layers kept flipping back and forth.

In the MELCOR 1.8.1 runs the fission product models were activated and no problems discovered. The default values for the numerous aerosol model input parameters were applied in these calculations, but the number of aerosol particle sections was 15. Iodine and cesium were modelled to form CsI through the input. However, these fission product results are considered still very tentative and a closer look is needed in defining the aerosol model parameters as well as some sensitivity runs to find out their real significance. Also, the orientation of aerosol deposition surfaces in the vessel may be occasionally inaccurate (because all heat structures are either horizontal or vertical).

Recently [75], MAAP 3.0B rev 9 and MELCOR 1.8.2 were used to calculate two different accident scenarios for the TVO I plant to predict the possible release of fission products from the containment. The accident scenarios were a station blackout (TB sequence) and a main steam line break with initial loss of all power (AB sequence). The containment venting location was either in the wetwell or in the drywell, and the starting pressure of the venting was varied to be 6 or 7 bar.

MAAP and MELCOR gave reasonably good agreement in timing of the important events in the course of the calculated severe accidents (*e.g.*, core plate failure, reactor pressure vessel failure, containment failure).

Several model parameter runs were carried out with MAAP. The varied sensitivity parameters were those controlling fission product release from the fuel, revaporization of volatile fission products, and the in-vessel or ex-vessel release of tellurium. MELCOR calculations were performed using default values for radionuclide package sensitivity coefficients. Ten aerosol size sections and one aerosol component per each class were used in the calculation. Both MAAP and MELCOR input had three control volumes in the containment: drywell, wetwell and pedestal.

The calculated source terms were higher in the TB cases than in the AB cases. In the TB cases the temperatures were higher in the primary system, causing revaporization of CsI and CsOH from the surfaces of the reactor pressure vessel internals. The revaporized aerosols coming from the vessel in the TB sequence maintained the airborne aerosol concentration in the drywell at higher levels than in the AB sequence. In the TB sequence, also, the retention of CsI in the primary circuit was higher than in the AB sequence. In the AB sequence most of the CsI was dislodged from the reactor pressure vessel within two hours after vessel failure, whereas in the TB sequence the removal of CsI from the primary circuit continued slowly throughout the calculation. Both MAAP and MELCOR predicted a similar trend in revaporization from the reactor coolant system.

The results from MAAP and MELCOR calculations showed that in general, with the default values of sensitivity coefficients and model parameters, MELCOR predicted higher source terms from the containment than MAAP. However, if the revaporization sensitivity coefficients in MAAP were changed in the range suggested by the code developer, the MAAP prediction of the source term varied within 2 to 4 orders of magnitude,

also giving release fractions from the containment that were higher than those calculated by MELCOR. With a particular choice of parameters in the MAAP calculation, the MAAP and MELCOR predictions of CsI and CsOH releases from the containment agreed quite well. However, because the MAAP 3.0B model was very sensitive to these model parameters, the code user should be cautious when evaluating the source term calculated by MAAP. On the other hand, the MELCOR radionuclide package lacks a model for particle growth due to hygroscopic steam condensation. The lack of this model makes MELCOR predictions of the source term conservative in cases where relative humidity is high (~100%) in the containment.

The tendency of MELCOR to overestimate the airborne aerosol concentration has been seen in preliminary comparisons of MELCOR and CONTAIN calculations and test data in the case of the AHMED Aerosol and Heat Transfer Experiments [191] at VTT; if the relative humidity in the test was low (<30%) the MELCOR calculation agreed well with the test data if a correct aerosol particle density was used. This conclusion is also based upon the MELCOR assessment results reported for primary system aerosol deposition in Marviken ATT-2b and ATT-4 [79] and on the MELCOR assessment results reported for containment aerosol deposition in LACE LA4 [57], summarized in Sections 23.1 and 14.2.

20.2 10% Main Steam Line Break with Reflooding

The computer codes MAAP 3.0B, MELCOR 1.8.1 and SCDAP/RELAP5/MOD3 were applied for the TVO nuclear power plant in the case of an initial station blackout with a intermediate size, 10% break in the main steam line, and with recovery of power and reflooding of the overheated reactor core with auxiliary feedwater system [76].

Four different variations were calculated to investigate the effect of water injection location on core coolability: core spray only, injection to downcomer only, both core spray and injection to the downcomer, and no core cooling or reflood.

The SCDAP/RELAP5 and MELCOR calculations were performed by VTT; the MAAP calculations were carried out by the TVO power company. The MAAP calculations were done on a 486-PC MicroMikko 5 CXe486/66; MELCOR and SCDAP/RELAP5 were run on a HP 9000/735 workstation. MAAP calculations were done with two different core nodalizations to facilitate direct comparison of results with those from MELCOR and SCDAP/RELAP5.

The core reflooding was started before any material relocation occurred, at the time when the maximum cladding temperature reached 1500K. Since MELCOR does not have a specific core spray model, the water addition above the top of the core was modelled by splitting the upper plenum control volume, adding a water source to the lower of the two upper plenum volumes, and letting that water flow through a normal flow path from the upper plenum into the core volume; however, a normal flow path does not model sparging of water into droplets that penetrate the core from the top.

All three codes predicted that some debris formation took place in the upper parts of the core after the water addition, due to heating by efficient zircaloy oxidation. The core was quenched by reflooding in all MAAP and MELCOR calculations. The major difference between the code predictions was in the amount of hydrogen produced, although the trends in hydrogen production for different coolant injection locations were similar. All the codes predicted that additional steam production during the quenching process accelerated oxidation in the unquenched parts of the core; this result is in accordance with several experimental observations.

In the MAAP calculations, the quenching front moved downward from the top to the bottom of the core when the core spray was active and from bottom upwards when the coolant was injected through the downcomer. MAAP predicted fuel relocation to occur to a small extent, and the largest amount of hydrogen was produced in the case with no restoration of core cooling.

According to MELCOR the normal operation of the auxiliary feedwater system gave the lowest in-vessel hydrogen generation, while injection into the downcomer and bottom reflood gave the largest in-vessel hydrogen production. In that case the water was vaporized when it reached to lower parts of the core, establishing an effective steam source for oxidation in the upper parts of the core. The quench front moved from the bottom upwards and some debris formation took place in the upper parts of the core. The debris formation and material relocation were induced by runaway zircaloy oxidation in the upper third of the core.

SCDAP/RELAP5 predicted in all cases cladding failure due to interaction with inconel grid spacers soon after a cladding temperature of 1500K was reached. The case with both core spray and injection into the downcomer gave the largest hydrogen production. The core was quenched although thermal shocking damaged the middle part of the core. In the other reflooding variations the lower parts of the core quenched but the steam production accelerated the metal oxidation in the middle and upper parts of the core, leading to the melting of ZrO_2 at 2950K and formation of cohesive debris in the other half of the core.

21 MELCOR Use at HSK

During 1989/1990 MELCOR Version 1.8.0 was installed on the VAX/CONVEX cluster at the Paul Scherrer Institute (PSI) in Villigen and on the CRAY computers at the Federal Institute for Technology (ETH) in Zürich and Lausanne. During 1991 MELCOR 1.8.0DN was installed, also on various workstations at HSK. At the end of 1991 and in the beginning of 1992 MELCOR 1.8.1 was installed on the VAX/CONVEX cluster at PSI, the CRAY YMP at ETH-Zürich and on the new IBM RISC 6000 workstations at HSK and ERI.

In 1989, the first MELCOR calculations were performed on a VAX 6350. At that time typical runs for a BWR simulation needed about 80 hours of cpu time. During the past years, computing power at PSI has increased by about a factor of 10. The VAX 9000, which was introduced about one and a half years ago, has roughly the same cpu power as a CRAY XMP (in a scalar mode). On the VAX 9000 typical MELCOR runs for a BWR simulation need about 8 hours of cpu Time, typical runs for a PWR simulation at present need about 3 times more. The fastest workstations available today reach the cpu power of the CRAY YMP. For instance on the IBM RISC 6000-550, typical BWR runs now need about 5 hours of cpu time.

MELCOR input models have been developed for the following four plant types:

1. Mühleberg: GE BWR/4 (Mark-I Double Torus Containment) 1097 MW(t)

2. Beznau: Westinghouse 2 Loop PWR (Large Dry Containment) 1130 MW(t)

3. Gösgen: Siemens-KWU 3 Loop PWR (Large Dry Containment) 3002 MW(t)

4. Leibstadt: GE BWR/6 (Mark-III Containment) 3138 MW(t).

To summarize the experience and recommendations, preparation of the plant input model requires about 3 man-months of effort. Debugging at present requires at least several months of effort. Performance of plant-specific calculations is painstaking labor intensive, because of the existing numerical, modelling, and FORTRAN problems discussed below.

Numerous software problems (in CVH, COR, CORCON) hinder calculations and should be addressed as soon as possible. A serious attempt should be made to analyze and correct the reason for time step and computer dependencies of the analyses. An attempt should be made to correct the numerical instabilities encountered in the thermal-hydraulic calculations in small control volumes. Editing capabilities should be improved, *e.g.* printout of containment failure time and pressures around the time of vessel breach. Allow for static parameters (failure pressure, valve setpoints, flow path controls) to be modified in COR input. If most of the core (more than 99%) is ejected from the vessel lower head, no core debris should be allowed to remain. This could cure some numerical instabilities associated with the lower plenum and the CVH package.

21.1 MELCOR Calculations for Mühleberg

In the Mühleberg plant a Containment Venting System (CVS) is installed. In over 80% of all severe accidents radioactivity is released through the CVS, resulting only in relatively mild consequences to the environment. Risk important drywell failure modes have been found to be (1) a massive drywell rupture resulting in a large blowdown of the primary containment into the large and isolated reactor building (secondary containment), (2) drywell head flange lifting, leading to a slow leakage and depressurization and (3) small and large break interfacing systems LOCAs (ISLOCAs) into the reactor building. The following accident sequences were simulated with MELCOR:

- Long-term station blackout sequence without ADS, assuming (1) massive drywell rupture at 0.725 MPa and (2) drywell flange leak at 0.525 MPa,
- Long-term station blackout sequence with ADS, assuming (1) massive drywell rupture at 0.725 MPa and (2) drywell flange leak at 0.525 MPa,
- Small break LOCA into the drywell, assuming (1) massive drywell rupture at 0.725 MPa and (2) drywell flange leak at 0.525 MPa, and
- Small and large break Interfacing Systems LOCA (ISLOCA) into the reactor building (V-Sequence).

All calculations were performed using the CORSOR and CORSOR-M models for in-vessel fission product release. The base calculations were performed on the VAX 9000 using MELCOR 1.8.0DN. The impact of computer environment on calculated results was investigated using MELCOR 1.8.0DN on CONVEX, CRAY and several workstations (SOLBOURNE Series 5/600, IBM RISC Series 300/500 and HP APOLLO 9000 Series 700). A limited number of calculations were also performed using MELCOR 1.8.1 on the VAX 9000.

Problems encountered with MELCOR 1.8.0DN during the analysis were presented at the CSARP Review Meeting in May 1991 [192]. The major problems are numerical instabilities throughout the code, particularly related to CORCON, and dependence of results on computer environment and minor time step variations. It is felt that the calculated results also depend very strongly on the quality of the FORTRAN compiler. In the Mühleberg analysis the problems related with CORCON were eventually bypassed with small changes in concrete composition and deliberate variation of maximum time steps.

Limited calculations performed with MELCOR 1.8.1 showed that the same major underlying numerical problems still exist. The only improvement noticed was in the treatment of revaporization from the RCS (no revaporization was included in MELCOR 1.8.0). However the full impact of this is still under investigation. In any case, the improved treatment leads to much larger releases of the more volatile species, including

tellurium. Another small improvement was noticed in some results in conjunction with the core relocation and blockage model.

Source term results have been published in a recent paper [193]. The calculated results for Mühleberg using MELCOR 1.8.0DN in general show good agreement with STCP calculations for Peach Bottom. The observed differences are partly due to modelling, scenario assumptions and other important design attributes. However, a major concern is the calculation of suppression pool decontamination factors, which appear to be much too low for MELCOR [193].

21.2 MELCOR Calculations for Beznau

The following MELCOR calculations have been performed for the Beznau plant:

- Long-term station blackout sequence (SBO), assuming containment venting at 0.5 MPa. In this calculation the ECCS was recovered after core damage and prior to vessel breach, resulting in a flooded cavity.
- Long-term station blackout sequence (SBO), assuming (1) forced early containment failure after vessel breach and (2) late containment failure at 0.75 MPa. No ECCS injection after core damage was simulated to represent a seismically initiated accident.
- Small break LOCA in the hot leg, assuming (1) forced early containment failure after vessel breach, (2) late containment failure at 0.75 MPa and (3) venting at 0.5 MPa,
- Intermediate break LOCA in the hot leg, assuming (1) forced early containment failure after vessel breach, (2) late containment failure at 0.75 MPa and (3) venting at 0.5 MPa,
- Large break LOCA in the hot leg, assuming (1) forced early containment failure after vessel breach, (2) late containment failure at 0.75 MPa and (3) venting at 0.5 MPa,
- An interfacing-systems LOCA (V-sequence), and
- Steam generator tube rupture (SGTR) with stuck open safety relief valve.

In addition to all the base cases, several sensitivity analyses were performed to resolve containment response issues, such as hydrogen generation, effect of operator actions (recovery of the safety systems), and power upgrade. All calculations were performed using the CORSOR model for in-vessel fission product release. Preliminary calculations were performed using MELCOR 1.8.0DN on VAX 9000 and IBM RISC 6000. Base calculations were performed on IBM RISC 6000 using MELCOR 1.8.1. The dependency of results on time steps and computer environment may be investigated in the future.

Both MELCOR 1.8.0DN and 1.8.1 show the same principal deficiencies as in the Mühleberg analysis. Some additional problems that were encountered in the Beznau analysis include numerical instabilities in thermal-hydraulic calculations in small control volumes, such as accumulators. An additional specific problem encountered is related to the CVH/COR coupling for convective heat transfer from fuel, clad and debris. A "TOO LARGE HEAT SINK" message followed by excessive time step reduction stopped some sensitivity analyses with increased power level. An arbitrary doubling of the convective heat transfer hydraulic diameters bypasses the excessive time step reduction error condition.

Sensitivity analyses to time steps could not be conducted seriously, because the results could be obtained essentially only after a painstaking effort of finding "good" combinations of time steps. Any other combination which was attempted usually led to premature failure of the analysis. Contrary to the Mühleberg analysis, the CORCON instability could not be bypassed by using different concrete composition or "good" combinations of time steps. In some cases, CORCON failure follows shortly after containment failure so that source terms may not be adequately estimated.

Also, sometimes the code allows a very small fraction of the core (less than 0.5%) to remain in the lower head. The code becomes unstable under these conditions, both in the core thermal-hydraulics (temperature and pressure in the lower plenum control volume oscillate violently), and apparently also in CORCON (the thermal-hydraulics of the lower plenum are boundary conditions for the cavity). Probably because of this, the CORCON problem could not be bypassed. Almost all runs failed sooner or later due to CORCON failure. Doubling the number of rays in the cavity from 35 to 70 prolongs the calculations. Any further increase in the number of rays, up to the maximum of 100, is disastrous.

The results of the base case and sensitivity analyses were compared with the utility MAAP 3.0B calculations. Full results of the comparisons will be reported in [194]. In summary, large differences were evident in the predicted vessel failure times for the station blackout sequence. This is partly due to differences in the lower plenum heat transfer models, and also as a result of differences in the lower head failure approach of the two computer codes. The results for containment bypass sequences compared extremely well. For all other sequences accident modelling differences were too large to allow for an exact comparison of the containment performance. Approximate comparisons, however, were favorable. In particular, the prediction of hydrogen generation and containment pressurization were almost the same in the two analyses. A comparison of the radiological source terms for some of the relevant sequences is given in Table 21.2.1. In general, the MELCOR and MAAP calculated releases of volatile species are within a factor of 2, while the calculated release for refractory groups are within one to two orders of magnitude.

21.3 MELCOR Calculations for Gösgen

Calculations for a large dry, German-design PWR are underway. The input model for this power plant includes more than 10 control volumes in the containment. This detailed

Table 21.2.1. Summary of MELCOR and MAAP Predictions of Radiological Source Terms for a PWR with Large Dry Containment

Radiological Group	Code	SBO Vent	SBO Early	SBO Late	SGTR	V
Xe	MELCOR	0.95	0.99	0.93	0.84	0.99
	MAAP	0.94	1.00	0.99	0.70	1.00
CsI	MELCOR	0.08	0.21	0.05	0.28	0.25
	MAAP	3E-3	0.11	0.03	0.23	0.56
CsOH	MELCOR	0.05	0.15	0.03	0.16	0.26
	MAAP	3E-3	0.10	0.07	0.23	0.56
Te	MELCOR	0.04	0.34	0.02	0.06	0.08
	MAAP	1E-4	0.14	0.01	0.10	0.17
Sr	MELCOR	4E-3	0.13	3E-3	3E-3	0.03
	MAAP	2E-4	7E-3	2E-3	0.03	0.10
Ru	MELCOR	2E-5	3E-4	9E-6	1E-4	1E-3
	MAAP	-	2E-5	5E-6	4E-7	7E-7
La	MELCOR	4E-5	3E-3	4E-5	1E-5	5E-4
	MAAP	5E-2	8E-3	2E-3	2E-3	3E-3
Ce	MELCOR	3E-6	1E-5	2E-6	2E-6	2E-5
	MAAP	5E-6	1E-3	1E-4	2E-3	3E-3

nodalization is necessary to properly account for phenomena of natural circulation and non-condensible gases mixing or stratification. The reactor cavity is also extremely small. The following accident sequence has been analyzed so far:

- Long-term station blackout, assuming (1) late containment failure at 1 MPa and (2) venting at 0.7 MPa.

Preliminary calculations using the CORSOR model for in-vessel fission product release were performed with MELCOR 1.8.1 on IBM Risc 6000-550.

Basically the same problems were encountered as in the Beznau analysis. However, probably due to the higher power, most of these problems, including the long-term CORCON failure could be circumvented in the case analyzed by finding "good" combinations of time steps and number of rays in the cavity.

22 MELCOR/MAAP Comparisons for Point Beach

A comparison study of MAAP 3B [195] and MELCOR 1.8.1 analyses of a station blackout accident scenario for the Point Beach Nuclear Power Plant Unit 1, a two-loop PWR, has been completed as a master's thesis at the University of Wisconsin [78].

In the course of this analysis, they have

- briefly described the current understanding of the progression of a PWR station blackout sequence (TMLB),
- given a brief explanation of the operation of the MAAP and MELCOR accident analysis codes,
- provided a description of the specific plant models developed for use with the MAAP and MELCOR codes,
- gained a qualitative and quantitative appreciation of the underlying phenomena that affect the TMLB accident sequence at Point Beach,
- evaluated the effects of two containment nodalizations on the MELCOR analyses of the progression of a station blackout sequence at Point Beach, and
- made a comparison of the station blackout responses calculated by MAAP and MELCOR for Point Beach, using a similarly noded containment.

The results of this study indicated that

- the results calculated by MELCOR were extremely sensitive to the choice of maximum time step;
- the choice of containment nodalization was important in the prediction of gas combustion;
- a good agreement existed between the simulated responses from MAAP and MELCOR prior to reactor vessel failure with, however, further investigation of the early stages of the transient and the core melt ejection modelling necessary;
- the very different post-vessel-failure scenarios predicted by MAAP and MELCOR were the result of the assumption of molten core debris coolability in MAAP, while in MELCOR the core debris in the cavity was assumed not to be coolable.
- The modelling of heat transfer between the molten debris and an overlying coolant pool is extremely important to the simulated progression of a station blackout and led to the following observations:

  - substantially more MCCI predicted by MELCOR than by MAAP,

- steam inerting of the containment in the MAAP calculations,
- a significantly lower "end of the day" containment pressure predicted by MELCOR (70psi) relative to MAAP (120psi).

23 Assessment within SNL MELCOR Development

23.1 Marviken-V ATT-2b/ATT-4 Primary System Aerosol Transport and Deposition

A series of five aerosol transport test (ATT) experiments were done in the large-scale Marviken facility investigating the behavior of vapors and aerosols under typical LWR primary system accident conditions. The main objectives of these large-scale experiments was the creation of an extensive database on the transport and attenuation of aerosols and volatile fission products within typical LWR primary coolant systems under conditions simulating severe fuel damage. MELCOR results [79] have been compared to experimental data, primarily to the deposited masses recovered from various identified portions of the system, and to TRAP-MELT2 [198, 199, 200, 202, 203], RAFT [205, 206] and VICTORIA [199, 207] code calculations for these experiments [208]. A large number of sensitivity studies were done investigating the effects of various MELCOR modelling parameters, defaults and assumptions.

The MELCOR code has been used to analyze two of the Marviken-V aerosol transport tests, ATT-2b [196] and ATT-4 [197]. In test 2b, the system geometry consisted of a pressurizer and four pipe sections prior to a relief tank, which was used to scrub materials which would otherwise escape the system; fissium aerosol was injected horizontally, near the bottom of the pressurizer. In test 4, the aerosol was injected into a simulated reactor vessel containing internal structures, whose top was connected by piping to the pressurizer volume and the remainder of the fissium transport system.

MELCOR was able to match most of the vessel, pressurizer and piping gas and wall temperature histories for both tests, despite the obvious existence of substantial recirculation and localized temperature gradients, particularly in the pressurizer in test ATT-2b and in the reactor vessel in test ATT-4. In both test analyses, the "net plasma input" taken directly from the test report energy balance estimates was found to produce good overall agreement in predicted and measured gas and wall temperature histories. That "net plasma input" accounts for heat losses in components not included in the MELCOR model, such as electrode and cable cooling and vaporization chamber wall losses, but does not include any heat losses in the reactor vessel, piping to pressurizer, pressurizer, piping to relief tank and relief tank, which are explicitly represented in our MELCOR model. The good agreement obtained on overall temperature histories without any input adjustment required on system heating or on system heat losses provides a validation of the MELCOR thermal/hydraulic model (both input and coding).

The initial (*i.e.*, injected) aerosol particle size distributions were not well known experimentally. As done by others analyzing these tests, we did a sensitivity study for both ATT-2b and ATT-4 in which the assumed AMMD of the injected aerosol particles was varied. A value of $5\mu m$ gave the best overall agreement with the measured aerosol distribution in test ATT-2b, particularly in the pressurizer, and was therefore used in our reference calculation and in the rest of our analyses. In general, the retention nearest the

injection point increases as the particle size is increased; this increase is most dramatic for aerosol species, but a similar effect is also seen for vapors in ATT-4 as those species condense onto the aerosols and settle out with them. For aerosol species, the fraction of material reaching the relief tank continually declines as the injection particle size is increased. For vapor species, the fraction of material reaching the tank is minimized for intermediate values of aerosol particle injection sizes, because the corium particles provide condensation sites for the fissium species: very small (corium) particles do not grow large enough through either condensation or agglomeration in the primary piping components for significant retention, while very large (corium) particles settle out so quickly in the vessel (where the temperatures are too high for significant condensation of the fissium species onto the corium aerosols) that few large particles are available in the cooler downstream components to interact with the condensing fissium species.

There are a number of compensating effects visible in the final material distributions for both test ATT-2b and ATT-4. Gravitational settling onto floors appears to be overpredicted while deposition onto vertical surfaces such as walls is underpredicted. The lack of a bend impaction deposition model is balanced by enhanced deposition in horizontal piping. However, the results generally show overall good agreement with retention in major components (*i.e.*, vessel, piping, pressurizer, tank).

The predicted retention results for the three fission product simulant species (CsOH, CsI and Te) for the low-temperature test ATT-2b are very similar, as are the experimental data. The predicted retention results for the two corium aerosol simulant species (Ag and Mn) for test ATT-4 also are very similar, as are those experimental data. These two species remained aerosols throughout the transient. MELCOR correctly predicts different final distribution patterns for the corium species (which remain aerosols throughout) and the fissium species (which exist in vapor forms at the elevated temperatures found in test ATT-4). In particular, MELCOR correctly calculates reduced retention in the vessel for the fissium species. Deposition of the fissium species onto the upper vessel walls and internals is underpredicted; the difference may be due in part to the neglect of vapor chemisorption, onto either aerosols or structures, since the chemisorption of CsOH onto stainless steel surfaces is expected to be significant at these temperatures. Instead, there is much more deposition by gravitational settling of Cs and Te onto the centreplate than was measured experimentally.

In the majority of our MELCOR ATT-2b analyses, only aerosol particles were considered, given the temperatures observed in the ATT-2b test. Sensitivity study results for ATT-2b showed very similar retention patterns whenever aerosols were injected, regardless of whether zero or non-zero vapor pressures were used. However, a large difference in deposition distribution was found when CsOH, CsI and Te were specified to be sourced in as vapors. The system was not hot enough to maintain vapor conditions and the injected fissium simulant materials quickly condensed into solid aerosol particles. However, because MELCOR automatically places newly created aerosol particles into the smallest MAEROS size bin available (in this case 0.1-0.15μm), the resulting aerosol particles were much smaller than the 5μm-AMMD specified for aerosol injection. In fact, the final aerosol distribution predicted for ATT-2b assuming injected vapors is quite similar to

that calculated assuming injection of aerosol particles with initial sizes of 0.1-0.5μm.

In most of our MELCOR ATT-4 analyses, MELCOR's built-in properties for the vapor pressures of CsOH, CsI and Te were used. This was considered necessary, given the temperatures in the ATT-4 test. (The silver and manganese have zero vapor pressure, so that only aerosol particles are considered for the corium simulants.) In most of our ATT-4 analyses, the fissium species were sourced in as vapors and the corium species as aerosols. As for ATT-2b, a sensitivity study was done evaluating the importance of condensation/evaporation effects for the fissium species. When injected as aerosols with zero vapor pressures, the retention patterns of the three fissium species closely resemble that calculated for the corium simulant species, as would be expected. Thus, neglecting the vapor pressures of the fission product simulants substantially overestimates the retention in the vessel of these three species because, at the temperatures attained in the Marviken test vessel, these materials can and do exist as vapors; overestimating the retention in the vessel by treating the CsOH, CsI and Te as always aerosols then results in underestimates for retention further downstream in the system, as might be expected. The final distribution patterns are very similar whether the three fissium species are injected as either vapors or as aerosols, as long as the MELCOR vapor-pressure values are used, because the temperatures in the vessel are high enough to vaporize these species if injected as aerosols. However, injecting all species (*i.e.*, Ag and Mn as well as CsOH, CsI and Te) as vapors produces results similar to those found in the corresponding sensitivity study calculation in the ATT-2b analysis – the retention is significantly reduced in the vessel, piping and pressurizer, and most of the material ends up in the relief tank. Also as found for our ATT-2b analysis, the results resemble those for sensitivity study calculations in which much smaller aerosol particle injection sizes are assumed, because the corium materials injected as vapors that have zero vapor pressure immediately condense, and MELCOR assumes that the condensed particles are created at the smallest size represented in the MAEROS size distribution.

No significant effects were found in test ATT-2b when the number of MAEROS aerosol components was varied from 1 (the default) to 2 (a separate one for fog) to 4 (a separate component for each class present). For test ATT-4, there was no change in results increasing the number of components from 1 to 2, because there was no fog present in the mostly superheated control-volume atmospheres. When 6 components (one per class) were used, somewhat different answers were obtained. AMMD plots best show the source of the different results calculated using six components. The two "always-aerosol" corium species (Ag and Mn) have AMMDs throughout most of the transient slightly larger than their injection AMMDs of 5μm. The condensed fissium species, especially CsOH and Te, have much smaller average particle sizes ($\leq 1\mu$m during most of the injection period), as these vapor-injected classes initially condense into the smallest MAEROS size bin available, and then further condense and agglomerate. (Different components can have different maxima in their size distributions.)

The predicted results vary smoothly as the number of MAEROS sections (*i.e.*, the resolution detail in the aerosol size distribution) is varied, with 10 sections appearing to offer a good compromise between accuracy and run time. The MELCOR results also

varied smoothly with the aerosol density assumed.

There was little effect found on final material distributions and retention factors when varying the wall radiation heat transfer emissivity between 0.3 and 0.9, for either test ATT-2b or test ATT-4. Using an emissivity of 0 (*i.e.*, no radiation heat transfer) did cause a significant change in results, especially for test ATT-4, by increasing volume atmosphere temperatures while reducing adjacent heat structure wall temperatures, which should significantly change deposition due to diffusiophoresis and thermophoresis.

The control-volume noding used in these MELCOR ATT assessment analyses is significantly more detailed than would be used in modelling corresponding components in plant analyses. This is common in assessment against experimental data, where the noding is often driven by instrumentation location and resolution. However, to determine the effect of coarser CVH noding more typical of plant models, control volumes were progressively combined into fewer, larger volumes and flow paths were combined and eliminated accordingly. The number of heat structures was left unchanged, but the one-on-one relationship generally prevailing between control volumes and heat structures in the reference, finer-node input models was lost, with multiple heat structures seeing a single average boundary volume. Combining control volumes in this way obviously affects the atmosphere and wall temperatures being calculated. The point of interest is how much losing detailed resolution of thermal/hydraulic conditions affects predictions of aerosol transport and deposition.

Simplifying the control-volume noding used for test ATT-2b causes the distribution patterns to fall into three main groups. Using two or three control volumes in the pressurizer gives similar, ~40% retention in the pressurizer, probably because even a 2-node representation can resolve the temperature gradient existing because of the energy source at the injection point in the bottom of the pressurizer. The retention in the pressurizer then decreases to ≥30% when a single control volume is used for the pressurizer (with a single exception). The retention in the piping does not vary greatly, and the varying amounts reaching the relief tank pool are a direct result of the pressurizer and piping deposition calculated using these different control-volume models. In one input model variation, the retention in the pressurizer was only about 15%. The important difference in this case was that an uninsulated wall heat structure was included in the same control volume as other, well-insulated pressurizer and piping heat structures, which significantly increased the heat losses from that volume. The better approach is to include the all the uninsulated piping volumes in the same control volume, allowing the insulated portions of the facility (the pressurizer and the LO4 pipe) to see one average temperature and the uninsulated portions of the facility (the LO5 and LO6 pipes and downstream) to see a different, lower average temperature.

A similar noding study was done for test ATT-4. The biggest change observed is a substantial reduction in vessel retention for all species going from a 2-volume model, with its ~500K temperature gradient from the lower to the upper vessel, to a 1-volume model with a single average temperature. This modelling simplification affects the aerosol results slightly, reducing the floor and lower wall deposition, but has no effect on the fissium species (for which no floor or lower wall deposition is predicted). The increase

in upper-vessel atmosphere and structure temperatures eliminates any settling onto the center plate for Cs and Te (there was none for I in any case) and also eliminates deposition of Cs on the upper wall and internals; the deposition of Te on these upper internals and wall is unchanged, and there was no deposition of I in any case. Since the temperatures of the upper wall, internals and center plate heat structures are very similar in each calculation, the change in temperature obviously affects gravitational settling (through changes in carrier gas density and flow velocity) to a different degree than it affects plating due to thermophoresis and diffusiophoresis, and affects condensation of vapors onto vertical structures differently than condensation onto horizontal structures.

The retention predicted for test ATT-4 in the piping connecting the vessel and pressurizer does not change very much as the control-volume modelling is simplified, for most of the species. There is a total variation of <1% for the Ag and Mn, <3% for Te and ≤5% for Cs. There is a strong correlation between the iodine deposition and the local heat structure surface temperatures. The highest temperatures are seen for those models which exhibit the lowest iodine deposition. The behavior in the pressurizer and in the piping from the pressurizer to the relief tank is generally similar to the results seen for test ATT-2b.

These results indicate that more detailed control-volume modelling may be a benefit in calculating radionuclide retention factors in regions with both high temperatures and significant temperature gradients between adjacent regions. There appears to be an overall net decrease in retention in the primary system as coarser control-volume nodings are used.

There has been a lot of discussion recently on numeric effects seen in various MELCOR calculations, producing either differences in results for the same input on different machines or differences in results when the time step used is varied. Identical calculations for both tests were run on a Cray, SUN and IBM workstations, VAX and 486 PC, and otherwise identical calculations were run using the code-selected time step and with the user-input maximum allowed time step progressively reduced by factors of 2, 5 and 10, to identify whether any such effects existed in these assessment analyses. No machine dependencies or time step effects were seen.

As part of this Marviken-V ATT MELCOR assessment effort, results from other computer code calculations have been reviewed. The Marviken ATT experiments have been analyzed with several versions of TRAP-MELT2 (by Battelle [202], by the UKAEA [198, 199, 203] and by ENEL [200, 204]), with RAFT [205, 206] and with VICTORIA [201, 207]. This review indicated that all the codes are predicting essentially similar behavior. All the codes predicted that the deposition patterns for Cs, I and Te under the conditions found in test ATT-2b are virtually identical. Many of the analyses done for test ATT-4 considered only a subset of the test system so that detailed, quantitative comparison is quite difficult but, in general, all the codes appeared able to predict some difference in retention of aerosols *vs* volatile species in test ATT-4.

The Marviken aerosol transport tests have been analyzed at AEE Winfrith using the UK enhanced version of the TRAPMELT-2 computer code. Their results are qualitatively very similar to many of our MELCOR results, and include underprediction of wall

deposition, and the lack of a bend impaction deposition model being partly offset in the code by exaggerated sedimentation rates. Total cesium retention in the reactor vessel in test ATT-4 was predicted to be about half the experimental value (as in our MELCOR results); the difference was explained as due in part to the neglect of vapor chemisorption, onto either aerosols or structures.

Comparisons of code predictions with selected Marviken ATT project experimental results (and with other results pertinent to turbulent deposition of particles) also have been done by Battelle Columbus for TRAP-MELT2.2. The results found were generally similar to the results from our MELCOR analyses and/or the UK TRAPMELT-2UK analyses: good agreement for those system components dominated by settling, much greater deposition predicted on the centreplate in the vessel than observed and, in general, wall deposition greatly underpredicted by the code, while gravitational removal is somewhat overpredicted.

The review of other code analyses for the Marviken aerosol transport tests show MELCOR generally producing similar behavior to the results from "best-estimate" aerosol transport codes, with the additional advantages of also predicting self-consistent, interdependent thermal/hydraulic and aerosol response in a simple, single integral calculation rather than the multi-stage process often required by the more detailed, best-estimate codes.

23.2 PNL Ice Condenser Tests 11-6 and 16-11

MELCOR has been used to simulate ice condenser tests 11-6 and 16-11, two of a series of large-scale experiments conducted at the High Bay Test Facility (HBTF) at Pacific Northwest Laboratories (PNL) to investigate the extent to which an ice condenser may capture and retain air-borne particles [209]. Experiment 11-6 was a low-flow test with some natural recirculation, while experiment 16-11 was a relatively high-flow test with no recirculation; in both tests, ZnS was used as the aerosol and temperatures and particle retention were monitored.

MELCOR results [80] have been compared to experimental data, and also to the results of CONTAIN calculations [210, 211] for these two tests. MELCOR version 1.8LF was used for the final calculations.

Agreement was very good between MELCOR predictions and PNL experimental data. MELCOR particle retention results agreed qualitatively with the data in that the value began at one and decreased quickly, levelled out during the time that the ice was melting, and then finally began decreasing again late in the experiment when the ice supply had been exhausted. Quantitative agreement with the experimental results was also excellent, based on the few values given for the experimental particle retention. Agreement with temperature data was also excellent, with MELCOR results usually falling within the low-temperature/high-temperature experimental data envelope given at three axial locations; the time at which all of the ice in a region melted also was well-predicted by MELCOR.

The MELCOR results were in better agreement with experimental data for particle retention than the CONTAIN results. On average, MELCOR and CONTAIN results were quite similar for the diffuser inlet and outlet temperatures, although differences in nodalization complicate the comparison. Unfortunately, there was no CONTAIN data published or available for temperatures in the ice-condenser region, the region of most interest.

A number of sensitivity studies were performed for each experiment simulation, also. The results of a time step study showed a small time step dependency with the results clearly converging with reduced time steps. No machine dependencies were observed when running the same problems on a Cray-XMP/24, SUN Sparc2, IBM RISC-6000 Model 550, VAX 8650 and 486 PC.

Thermal/hydraulic sensitivity studies examined the effects of varying flow loss coefficients, equilibrium *vs* nonequilibrium thermodynamics, and the possibility of including SPARC bubble rise physics. Parameters associated with the aerosol input examined through sensitivity studies included number of aerosol components, number of aerosol sections, aerosol particle density and aerosol particle size range. The last set of studies done studied the impact of varying input parameters associated with the ice condenser model directly, and included varying the energy capacity of the ice, the ice heat transfer coefficient multiplier, the ice heat structure characteristic length, the number of nodes in the ice condenser heat structure, and radiation heat transfer for the ice condenser heat structure.

Separating different aerosols into different components was found to be desirable in the MAEROS-components study. The gain in accuracy and the more accurate physical representation is usually worth the additional computer time required. This conclusion was also reached in other assessment studies involving the RN package [57, 79]. In the MAEROS-sections study, using five sections (the MELCOR default) was adequate in this case; using 10 sections improved the results but incurred an additional cost, while using 20 sections did not change results significantly but doubled the computer time. The aerosol results were also quite sensitive to the value input for aerosol density.

The parameters that set the energy capacity of the ice affected the time to complete ice melt much more than they affected the calculated temperatures or aerosol results. The ice heat transfer coefficient multiplier affected both the ice melt rate and the temperatures. The study on varying the ice heat structure characteristic length found that this parameter affected the results the most – temperatures, ice melt rate and particle retention were all sensitive to this parameter. A length representative of a single ice cube in the condenser seemed to be the most reasonable and most accurate value. In the heat structure noding study, results showed that using two nodes gave the most predictable and accurate behavior, because the MELCOR ice condenser model does not remove ice in a cell as in a moving-boundary model but rather, when all ice in a cell is melted, replaces the ice with another heat structure material, affecting both the ice melt rate of cells further in and the energy exchange between the ice heat structure and its adjacent control volume.

23.3 Direct Containment Heating Tests IET-1 and IET-6

The MELCOR computer code has been used [81] to analyze several of the IET direct containment heating experiments done at 1:10 linear scale at Sandia [213, 214, 215, 216, 217, 218] at at 1:40 linear scale [219, 220, 221, 222] at Argonne National Laboratory.

Note that these MELCOR calculations were done as an open post-test study, with both the experimental data and CONTAIN results [223, 224, 225, 226, 227, 228, 229] available to guide the selection of code input. Most individual parameters in our MELCOR input models were not separately adjusted in each of our MELCOR IET experiment analyses to best match data for each individual experiment. Instead, the basic control-volume/flow-path/heat-structure model was kept the same for all SNL/IET experiments analyzed, and a single set of debris source, distribution and interaction time parameters was used for all the SNL/IET experiments analyzed. The only test-specific changes made were to set the initial pressures, temperatures, gas composition, and liquid pool heights to match individual experiment initial conditions. A similar approach was taken for the ANL/IET analyses.

The processes modelled in the MELCOR FDI/HPME/DCH model include oxidation of the metallic debris components in both steam and oxygen, surface deposition of the airborne debris by trapping or settling, and heat transfer to the atmosphere; first-order rate equations with user-specified time constants for oxidation, heat transfer and settling are used to determine the rate of each process.

A single set of characteristic interaction times was specified for all the seven of the 1:10-scale tests analyzed (SNL/IET-1 through SNL/IET-7). The characteristic times for settling of debris in the control volume atmospheres onto floor heat structures were based upon free-fall times for the various volume heights, and therefore proportional to volume heights and constant in the various tests; there could be some test-to-test variations in turbulent flow circulation patterns, thermal buoyancy effects, *etc.*, but these were assumed negligible. The characteristic oxidation and heat transfer times were assumed to depend primarily on parameters such as average airborne or deposited particle concentrations, which in a given geometry should be approximately constant for identical melt debris and blowdown steam sources such as used in the tests analyzed.

The characteristic times for oxidation and heat transfer of debris in the control volume atmospheres, as well as a characteristic time for oxidation of debris deposited on heat structures, were selected after a number of iterations in sensitivity studies as giving reasonable agreement with a subset of test data (in particular, vessel pressure, subcompartment temperature and hydrogen production and combustion) in the SNL/IET experiments simulated. Note that there is no reason to assume that the debris source and interaction input parameter set used in our reference analyses is unique (*i.e.*, the only set to provide reasonable agreement with the selected test data). It is also not guaranteed that the iterative procedure followed results in an input parameter set that yields the best agreement with data, or agreement with data for the "correct" reasons (*i.e.*, representing the actual behavior). For example, freezing some of the parameter values early in this iterative process undoubtedly affected the values assumed for other parameters. Further,

experiment ambiguities may have led to incorrect modelling assumptions that would also affect the values chosen for various parameters (such as the characteristic oxidation time, as discussed below).

The results of the MELCOR reference calculations for the Surtsey 1:10-scale tests correctly reproduce the subdivision of the pressure response into two major families, caused by the effect of hydrogen combustion, as seen in the test data, with a peak pressure rise of ~100kPa due to HPME and an additional pressure rise of ~150kPa due to hydrogen combustion. The results also correctly reproduce the lack of any significant effects of presence *vs* absence of pre-existing hydrogen or presence *vs* absence of basement condensate water.

The hydrogen production and combustion calculated by MELCOR is generally in reasonable agreement with test data (after careful adjustment of the BUR package input, as described below). However, it is difficult to quantitatively compare the measured and calculated hydrogen production and combustion because of the basic assumption made by the experimenters that all oxygen depletion was due to reaction with hydrogen. The experimenters assumed in their data analysis that debris reacted only with steam, not with free oxygen, whereas MELCOR assumes that oxidation of metals with free oxygen occurs preferentially to oxidation with steam. Therefore, throughout this report, pairs of values are given for the hydrogen production and combustion calculated by MELCOR, presenting both the actual amounts of hydrogen calculated to be produced by HPME steam/metal reactions and burned, and the amounts of hydrogen produced and burned that would be calculated using the initial and final oxygen and hydrogen moles from the MELCOR analyses in the same formulae as in the experiment data analysis.

The two sets of MELCOR values differ by twice the number of moles of O_2 consumed by direct metal/oxygen reactions. There is little difference found in the hydrogen production evaluated using the experimental procedure and actually calculated by MELCOR in the tests with little or no free oxygen present (*i.e.*, SNL/IET-1 and SNL/IET-1R); however, note that, for those two tests and for SNL/IET-5, assuming all oxygen depletion was due to combustion reaction with hydrogen does result in a small mass of hydrogen calculated to be burned, similar to the experimental results. The actual moles of hydrogen produced and burned in these MELCOR analyses appear generally less than measured values, especially in the experiments with hydrogen combustion, while the hydrogen production and combustion calculated using the experimental procedure on the MELCOR results are generally greater than measured. Also, the actual amount of hydrogen calculated to be produced by MELCOR is lower in the tests with oxygen initially present (SNL/IET-3 through SNL/IET-7) than in the experiments with no significant oxygen initially present (SNL/IET-1 and SNL/IET-1R). However, deriving the amounts of hydrogen produced and burned in the MELCOR calculations by assuming that all oxygen depletion is the result of hydrogen burning yields greater hydrogen production in SNL/IET-3 through SNL/IET-7 than in SNL/IET-1 and SNL/IET-1R, reproducing the trend seen in the tabulated experimental data.

Overall, the "correct" answers are likely to lie somewhere between the two limiting assumptions. It is unlikely that there is no oxidation of metal with free oxygen at all (as

assumed in the experimental analysis protocol). However, MELCOR would be expected to exaggerate the relative degree to which metal oxidizes with free oxygen *vs* with steam, because of the hierarchical assumptions in the MELCOR FDI/HPME/DCH model and because, in the experiment, the debris transport probably lags the steam/hydrogen mixture flow, so that not much of the debris gets to see much oxygen, while in the MELCOR model the debris is immediately transported to its ultimate distribution (within a user-specified time period, in this case 1s) while the steam blowdown is modelled "normally" as a transient process taking several seconds.

The quantification of hydrogen production and combustion in the SNL/IET experiments assuming that all oxygen depletion was due to reaction with hydrogen had an unforeseen effect on our MELCOR analyses. In particular, the choice of a very short time constant for airborne debris oxidation (0.025s in the reference analyses) was driven by trying to explicitly match the reported hydrogen production and combustion data; sensitivity study results show very little difference in calculated pressures or temperatures for $0.01\text{s} \leq \tau_{ox} \leq 0.1\text{s}$, because the oxidation rate is essentially limited by availability of steam and/or oxygen at the shorter characteristic interaction times, and the hydrogen production and combustion results later derived from a molar balance assuming only steam/metal reactions (as in the test data analysis) is in better agreement with test data for calculations using longer characteristic airborne-debris oxidation times ($\tau_{ox} \geq 0.1\text{s}$), which seems a more reasonable value based on physical grounds. (This does not affect any of the comparative conclusions drawn from the various other sensitivity studies.)

The hydrogen combustion observed in these tests could not be calculated using the default burn package input, because the default ignition criteria are never satisfied in these experiments. Instead, in the majority of our IET analysis calculations, the hydrogen mole fraction ignition criterion in the absence of igniters was set to 0.0, which (in the absence of CO) also gives a combustion completeness correlation value of 0.0; in addition, burn was suppressed in all control volumes except the vessel dome. This particular combination of input was found to produce reasonable agreement with test data in all cases. The combustion completeness being set to 0 prevents the burning of any pre-existing hydrogen, but allows burning of any additional hydrogen generated during the HPME. Suppressing burn except in the dome mimicked the experimental behavior of a jet flame burning at the outlet from the subcompartments to the dome; because little or no hydrogen was generated by debris oxidation in the dome in our analyses, only hydrogen advected into the dome from the subcompartments burned, and only on the time scale over which it was advected into the dome.

(While these non-standard combustion criteria could be specified with the standard BUR package input for these experiment analyses, the same input modification could not be made in plant analyses, because the non-standard input would affect the results calculated both before and after the HPME period. This problem was addressed by providing new, optional input parameters in the BUR package, essentially allowing the user to specify one set of input parameters to be used during periods of HPME and another set of input parameters to be used during the remaining times.)

Most of our calculations were run with control volume flow areas reduced by factors

of $\geq$10 from their default values, to enhance convective heat transfer from the control volume atmospheres to the heat structure surfaces. (The control volume flow areas are used only to obtain volume velocities for use in the calculation of convective heat transfer coefficients; changing control volume flow areas does not affect flow path calculations at all.) The convective heat transfer was enhanced for two reasons:

First, our preliminary calculations showed that the flow through the system in these calculations was primarily that associated with the steam blowdown only, flowing from the steam accumulator through the ca‥ty and chute volumes to the subcompartments and then to the dome. The MELCO‥ ‥HPME/DCH model does not model transport of debris between and through volu‥ ‥t instead deposits the debris directly at its ultimate destination, using the same time-dependent deposition in all volumes regardless of their distance from the debris source. Thus, instead of debris being transported into an "upstream" volume with the blowdown steam and the resultant additional heating adding to the driving force pushing flow further "dov stream", the MELCOR logic does not represent this additional flow driving force a‥ ‥ontrast has debris appearing "upstream" and heating the atmosphere in upstre‥ ‥umes, if anything contributing a retarding force to the expected flow. This results in lower velocities, and is more benign than the transient HPME blowdown actually occurring in the experiments, with transport of hot debris together with the steam blowdown. Decreasing volume flow areas resulted in increased volume velocities more characteristic of the turbulent conditions that might be expected during HPME, and the associated turbulent forced convection heat transfer to structures.

In addition, the MELCOR FDI/HPME/DCH ‥del does not account for any radiation directly from airborne debris to surrounding structures (or from deposited debris directly to atmosphere). Although radiation heat transfer was included in the MELCOR input model, there is little or no calculated atmosphere-structure radiation heat transfer early in these transients (except in IET-5), because MELCOR only considers radiation heat transfer for steam and/or CO_2 in atmospheres. In IET-5, some atmosphere-structure radiation heat transfer is calculated because of the large amount of CO_2 used to inert the system; however, in most of the experiment simulations there is very little steam present early in the transient, because any blowdown steam is consumed in debris oxidation soon after arrival, and very little CO_2 present at all. The lack of steam and/or CO_2 in the atmosphere would if anything enhance radiation heat transfer from airborne debris to structures because there would be little absorption in the intervening atmosphere. Hand calculations indicate that this could be a significant heat transfer mechanism, early in the transient. Because there is no way in MELCOR to model this effect, too much energy may be deposited in the atmosphere by the airborne debris; because there is no convenient way to enhance atmosphere-structure radiation heat transfer in general, we relied on increasing convective heat transfer instead to help remove that energy.

(Again, while this could be done with the standard CVH package input for these experiment analyses, the same input modification could not be made in plant analyses, because the non-standard input would affect the results calculated both before and after the HPME period. This problem also was addressed through new input capabilities,

adding a sensitivity coefficient to the CVH package that optionally multiplies the volume velocities in any given control volume during the HPME period only.)

The MELCOR FDI/HPME/DCH model does not model transient transport of debris into and through the system, but instead immediately places the debris at its ultimate destination. MELCOR uses a single function for the time-dependence of the melt injection in all control volumes and heat structures; in reality, the melt reaches the subcompartments later than the cavity, and the dome later than the subcompartments. The time period over which melt injection was specified to occur was varied in sensitivity study analyses, and the time-dependence of the melt addition in the MELCOR input was adjusted to match the rate of pressure and temperature increase in the vessel. Based upon results results for vessel pressure, hydrogen generation and subcompartment temperatures, our analyses were run with a melt injection period of 1s, with most of the injection occurring during the second half of that period. This $\leq$1s melt injection period is in reasonable agreement with test observations indicating molten brass, steel and thermite entering the cavity between 0 and $\sim$0.3s, and debris entrainment from the cavity into the subcompartments between about 0.4s and 0.8s.

The total debris mass collected in these experiments was usually greater than the initial thermite charge due to melting of the inner wall of the crucible, vaporization of the fusible brass plug, ablation of concrete in the cavity and structures, and oxidation of metallic debris. Thus, despite the careful duplication of the initial thermite charge, the different amounts of debris collected from the melt generator and from the vessel result in some uncertainty in the actual amount and composition of melt injected into the vessel. The majority of our MELCOR analyses simply specified the original thermite charge mass, neglecting both the retention of any debris in the melt generator and the addition of any debris due to melting, vaporization, ablation, and/or oxidation. To determine the effect of the injection mass source uncertainty, calculations were done varying the melt mass. As would be expected, the vessel pressurization increases slightly as more melt mass is injected during the HPME; there is also a small increase in both hydrogen production and combustion with increasing melt injection amount, and a small increase in subcompartment temperature.

Sensitivity studies varying the debris temperature showed, as would be expected, that increasing debris temperature increases the vessel pressures calculated, but has very little effect on either the amounts of hydrogen generated or burned. The debris temperature variation has the strongest effect on the subcompartment temperatures predicted, with a 1000K increase in debris temperature producing a $\sim$500K increase in subcompartment peak temperatures.

The MELCOR FDI/HPME/DCH model does not model transient transport of debris into and through the system, but instead immediately places the debris at its ultimate destination. The debris fractions placed in each control volume and on each heat structure are controlled solely by user input. In these IET analyses, the debris injected was all placed in various control volume atmospheres and then allowed to settle out onto floor heat structures; no debris was specified to be deposited directly onto any heat structures.

The debris distribution was kept the same in our MELCOR input for all tests analyzed, because there were only small differences in the test data debris distributions.

However, in most plant analyses, there will be no equivalent data set providing guidance on HPME melt distribution. To evaluate the effect of the debris distribution assumed on the overall DCH behavior calculated, calculations were done in which the experimental debris distribution for each test was used, and in which most of the debris was placed either in the cavity and chute or in the dome. The major difference is seen for the calculation with most of the debris specified to go into the vessel dome, a volume with a longer characteristic settling time (proportional to the volume height), which allows more time for oxidation and especially for heat transfer from airborne debris to the atmosphere; assuming most of the debris goes to the dome results in much less hydrogen production calculated for most of the tests because then most of the debris is oxidized by the relatively large amount of free oxygen available in the dome.

The effects of varying the characteristic debris interaction times were also investigated. With a very long characteristic airborne debris oxidation time, the overall pressurization, and both the hydrogen production and combustion, are all underpredicted. Using shorter characteristic airborne debris oxidation times ($\leq$0.1s) gives generally similar results because in all these cases the oxidation is mostly limited by the availability of oxygen and/or steam. As would be expected, control volume temperatures are affected most by varying the airborne debris characteristic heat transfer time; the control volume atmosphere temperatures increase as the airborne debris characteristic heat transfer time is shortened; the vessel pressurization also increases, and there is decreasing hydrogen production and combustion. The effect of increasing the airborne debris characteristic settling time is to increase the vessel pressures and temperatures calculated, as well as the amount of hydrogen both produced and burned, because this increases the available time for both oxidation and heat transfer to occur.

After work on these analyses had been in progress for some time, we became concerned about the interaction of debris with heat structures. In the original HPME model added to MELCOR, any debris immediately deposited onto a heat structure or later settled onto a heat structure essentially left the problem; there was no subsequent interaction of any kind for that debris, except for decay heating of the structure surface. This was identified as a major potential problem area, especially given MELCOR's emphasis on mass and energy conservation. For example, the lack of any thermal interaction of debris with structures could adversely affect the ability to correctly predict late-time revaporization of volatile fission products. Also, the lack of any oxidation of deposited debris meant that the total amount of hydrogen producable during HPME was very highly dependent on the user-specified initial debris distribution and on the characteristic settling time constants - any debris deposited or settled could not continue to generate hydrogen through further oxidation, regardless of oxygen and/or steam availability or debris temperature and/or amount. Therefore two effects were added to the original HPME model: heat transfer to the structure surface from deposited hot debris, and the continued oxidation of the deposited debris. The heating of the structure surface by deposited hot debris is controlled by a heat transfer coefficient adjustable through sensitivity coefficient in-

put, and the continued oxidation of the deposited debris is controlled by a user-input structure oxidation characteristic time (distinct from and usually much longer than the characteristic time input for oxidation of airborne debris). With these input and coding modifications, HPME debris deposited on structures now can continue to affect the overall system response through several potential interactions.

In these IET experiment analyses, there was generally little effect on either the peak or the long-term pressurization as the deposited debris characteristic oxidation time was varied. The total amount of hydrogen generated increased as the characteristic oxidation time for deposited debris decreased, as would be expected, because more hydrogen accumulates late in the transient as the debris settled and/or deposited onto structures continues to oxidize. There was little effect seen on the amount of hydrogen burned, however, because the hydrogen combustion primarily occurs early in the transient, on a time scale of a few seconds or less, as the airborne debris provides an ignition source during the high-pressure melt ejection.

Much of this "lack of effect" of deposited debris oxidation is probably due to the fact that the values of most of the other input parameters used in the MELCOR input were set in earlier sensitivity study calculations, before this effect was included in the MELCOR FDI/HPME/DCH model. A longer characteristic interaction time constant for oxidation of airborne debris (which would probably be more reasonable on physical grounds) would have left more debris unoxidized during the first few seconds of the transient and thus would allow more oxidation of deposited debris later in the transient. Oxidizing less airborne debris within the first few seconds and more deposited debris later in the transient could also potentially allow both high enough hydrogen generation and combustion and low enough vessel pressurization and subcompartment temperatures to match the measured test data, without requiring as large an increase in heat transfer to structures early in the transient.

Several counterpart tests to the IET direct containment heating experiments done at Sandia in the 1:10 linear scale Surtsey facility were performed at ANL in the 1:40 linear scale COREXIT facility, in an experimental program to investigate the effects of scale on DCH phenomena. The results of the 1:40-scale IET experiment MELCOR simulations were generally inconclusive. The vessel pressures predicted in our SNL and ANL counterpart-test calculations were quite similar when both the geometry and the characteristic interaction times in the FDI HPME input were scaled, but the test data showed a number of non-scaled effects. In particular, the results of both our limited review of the facility and data scalability and of our ANL test simulations suggest that, in the experiments, the DCH energy-transfer efficiency is greater at smaller scale, that there is less pressurization due to hydrogen combustion at smaller scale, and that there appears to be a greater effect of pre-existing hydrogen in the ANL 1:40-scale tests than in the counterpart SNL 1:10-scale tests. These scale-dependent differences are not reproduced in the corresponding MELCOR analyses. Other sensitivity studies indicated that some of the greater pressurization due to DCH at small scale observed in the experiments but "missing" in our MELCOR calculations may be due to differences in heat transfer to structures at smaller compared to larger scale, that the pressure dropoff rate in the ANL

data clearly would be matched better by assuming a recirculation flow area greater than the 10% value assumed in the SNL experiment analyses, and that the data for those tests in which hydrogen combustion occurred could not be matched by using the same, non-default hydrogen burn package input that gave good agreement with test data for the 1:10-scale test simulations.

The reference MELCOR calculations for the 1:10 linear scale IET experiments done in the Surtsey vessel have been compared to similar calculations done with the CONTAIN code, when available. The CONTAIN DCH model is quite different from the MELCOR FDI/HPME DCH model, being a more detailed, more mechanistic treatment rather than a more parametric approach. Despite these differences, the results obtained with the two code models are generally quite similar: in particular, a pressure rise of $\leq$100kPa was calculated by both for tests with no significant hydrogen combustion, and a larger pressure rise of ~200-250kPa for cases with substantial hydrogen burn.

Several calculations have been done to identify whether any numeric effects exist in our DCH IET assessment analyses, producing either differences in results on different machines or differences in results when the time step used is varied. The SNL/IET reference calculations were run, using the same code version, on an IBM RISC-6000 Model 550 workstation, on an HP 755 workstation, on a SUN Sparc2 workstation, on a CRAY Y-MP8/864, and on a 50MHz 486 PC. There is generally excellent agreement among results generated on these various hardware platforms. The SUN and PC were always slowest in run time required; the IBM, HP and Cray were all significantly faster with the HP the fastest for these analyses. In addition, otherwise identical MELCOR SNL/IET calculations were run on a SUN Sparc2 workstation with both the user-input maximum allowed time step and the initial time step size for HPME initiation simultaneously reduced by factors of 2, 10, 20 and 100 from the basecase values. The results showed about half of the SNL/IET experiment analyses fully converged for all these time steps, with the other half demonstrating convergence with reduced time steps.

23.4 ACRR DF-4 In-Pile Core Damage and Relocation

MELCOR has been used to model the ACRR DF-4 damaged fuel experiment [82]. The DF-4 test [230] provided data for early-phase melt progression in BWR fuel assemblies, particularly for phenomena associated with eutectic interactions in the BWR control blade and zircaloy oxidation in the canister and cladding.

The MELCOR basecase input model for the DF-4 experiment consisted of 4 control volumes, 4 flow paths, and 15 heat structures; 14 core cells were modelled in a single ring, with 9 cells in the active fuel region. Of the non-default MELCOR input parameters used in the basecase input model, the most important were the activation of the new eutectics model, those that changed the zircaloy melt temperature and the transition temperature for zircaloy oxidation rate, and enabling a new code option for calculating heat transfer between core radial boundary heat structures and the core control volume atmosphere.

In addition to comparison with test data, the results of the basecase MELCOR calculation were compared to results of DF-4 analyses performed using 4 more mechanistic

codes (APRIL.MOD3 [231], BWRSAR/DF4 [232], MELPROG-PWR/MOD1 [233] and SCDAP/RELAP5/MOD2 [234]).

The basecase MELCOR model underpredicted control blade temperatures in the early parts of the experiment by almost 200K but, in later stages of the experiment when all the core damage was taking place, calculated control blade temperatures corresponded almost exactly to measured values. Control blade failure times in most of the test bundle were predicted almost exactly compared to experimental data. Cladding temperatures were predicted almost exactly compared to experimental data at all times and at all levels except for the uppermost axial level; MELCOR overpredicted temperatures in the uppermost axial levels by close to the same amount (~250K) as other codes did during the middle of the experiment, leading us to believe that the power coupling relationship did not predict power coupling well in this part of the core. Fuel failure times calculated by MELCOR corresponded almost exactly to experimental data. Calculated canister temperatures were also very close to experimental data, after correcting this data for the time and temperature lags associated with the slow-response thermocouples used for the canister.

Material distribution plots for the melting and relocation portions of the experiment very clearly show the effect of the B_4C-stainless steel eutectic interaction in the control blade. This reaction resulted in the first control blade failure around 7450s, which was within 10s of the first observed failure in the experiment. Eutectic dissolution of the canister wall was also evident and was responsible for the calculated failure of lower portions of the canister. Evidence of canister failure was seen in the postirradiation examination (PIE) of the DF-4 test bundle.

The material distribution plots also showed clearly that, in the MELCOR DF-4 calculations, core materials relocated by axial level and not by component. That is, all components at a single axial level (fuel, clad, canister and control blade) melted and relocated before significant component relocation at other levels. This behavior could be significantly affected by code input parameters. For example, the default candling heat transfer coefficients resulted in the control blade material refreezing quite close to the axial location from which it melted. Behavior would be quite different if the control blade materials were allowed to candle to the bottom of the test bundle, as they did during the DF-4 test. These results are important when considering the possibility of reactivity excursions due to control poison relocation without accompanying relocation of fuel material.

The amount of hydrogen production calculated by MELCOR was 36.4gm, which was within the amount derived from the PIE (38.0±4.0gm). MELCOR calculated the autocatalytic oxidation reaction to begin sooner than was measured, and predicted 5gm of hydrogen produced before the autocatalytic stage, compared to no hydrogen production measured in the experiment during that time; other codes predicted early hydrogen production and early transition to the autocatalytic stage as well.

A large number of sensitivity studies were performed on MELCOR input parameters, most of which were in the COR package but also some in the HS and CVH packages.

A study which deactivated the eutectics model showed clearly the benefits of using this new model, as deactivating it predicted much different behavior of the B_4C and did not show any canister dissolution. Hydrogen production without the eutectics model was well below both the measured and the MELCOR basecase values. A sensitivity study which varied the eutectic temperature of the B_4C-stainless steel reaction by $\pm 50K$ showed little variation of results. A study which used the default heat structure boundary fluid temperature option (which uses bulk atmosphere temperature instead of local dT/dz temperatures for calculating heat transfer between the core and its boundary heat structures) resulted in much earlier component failure and poorer temperature agreement withe experimental data; this study showed the usefulness of the new HS boundary fluid temperature option. Finally, a study on minimum oxide shell thickness and two other core material relocation parameters in the COR package showed no variation in results until the critical minimum thicknesses for intact zircaloy and stainless steel were set to zero; after these parameters were changed, the final core material configuration showed the fuel pellet stacking observed in the PIE, but did not relocate any of the ZrO_2 that resulted from cladding oxidation. Other studies showed sensitivities to zircaloy properties, COR component view factors, allocation of canister mass to either the canister or canister-b component, candling heat transfer coefficient, COR and CVH nodalization, and slight sensitivity to COR and overall time steps. No sensitivities were found to minimum component mass, B_4C oxidation modelling, HS outer boundary temperature, and the machine used to run the problem.

This assessment analysis resulted in improvements to the COR dT/dz model, in particular with the addition of the HS boundary fluid temperature option. Several other code errors were uncovered and corrected during this analysis.

23.5 Surry TMLB' with and without DCH

As part of the MELCOR Peer Review process [2], Sandia performed and presented a demonstration calculation of a Surry station blackout (TMLB') accident with MELCOR. This was the first fully-integrated PWR severe accident calculation performed with the code (since the earlier TMI analysis only included in-vessel phenomena). That calculation was done using the release version of MELCOR 1.8.1. The calculation has been rerun with the release version of MELCOR 1.8.2 [84], allowing direct comparison of predicted results for the same problem. That analysis also has been used as a standard test problem to investigate problems identified by the Peer Review (*e.g.*, lack of pressurizer draining prior to vessel breach) and to evaluate the impact on the results of model improvements and extensions (for example, adding the CORSOR-Booth fission product release model) and of new models (such as radial debris relocation, material eutectics interactions, and direct containment heating due to high pressure melt ejection).

No input changes were required between running with the release versions of MELCOR 1.8.1 and 1.8.2. Input changes made in the basecase model to take advantage of new models and/or upgraded models included using step functions in valve area-*vs*-time tables, and enabling the new eutectics model (not used as the default); the new debris

radial relocation model is enabled by default. Other input changes for various sensitivity studies included specifying high-pressure melt ejection debris distribution and interactions, varying the fission product release model option, varying the interfacial momentum exchange length in some flow paths, and changing in-vessel falling debris heat transfer parameters.

The results of the same transient run with MELCOR 1.8.1 and 1.8.2 show generally very similar early-time behavior, for the steam generator secondary inventory boiloff, for the pressurizer filling and venting through the PORV, and for the core uncovery and initial clad failure and gap release. The vessel was calculated to fail ~1hr earlier by MELCOR 1.8.2 than by 1.8.1; of that difference, ≥0.5hr was due to correcting the "levitating water" problem diagnosed and corrected during our LOFT LP-LP-2 MELCOR assessment [56], while ≤0.5hr was due to incorrect failure of the blocked core plate in the MELCOR 1.8.1 analysis (corrected in 1.8.2). More hydrogen was generated in-vessel in the MELCOR 1.8.2 analysis than in the MELCOR 1.8.1 analysis, but the total hydrogen generated (adding together in-vessel and in-cavity production) by the two code versions was within 5%. There was very little change in calculated containment response, with a pressure spike at vessel breach shifted in time due to the different vessel failure times, but the same long-term pressure and temperature response predicted by both MELCOR 1.8.1 and 1.8.2. (Note that this direct comparison did not use the new direct containment heating model added in MELCOR 1.8.2, but even with that model enabled there was simply an increase in the containment pressure spike at vessel failure, and no other significant long-term differences in predicted system response.)

During the MELCOR peer review [2], questions were raised concerning the failure of the pressurizer to drain until the time of vessel failure and subsequent primary system depressurization in the MELCOR 1.8.1 Surry TMLB' demonstration calculation; there was general agreement that this appeared to violate physical intuition, and might reflect a code problem. In particular, concern was expressed by members of the peer review committee that the failure of the pressurizer to drain was a result of the inadequacy of the momentum exchange model in MELCOR, leading to an incorrect two-phase countercurrent flow limit (CCFL). In response to this problem (and to other concerns), a number of modifications were made to the code including treating the momentum exchange length as a separate variable from the inertial length, defaulted to the buoyancy force characteristic dimension; user input can be used to override the default if desired. As part of evaluating the current momentum exchange model, the Surry TMLB' analysis which originally highlighted the pressurizer drainage problem was rerun with input appropriate to the new interfacial momentum exchange model in MELCOR, in a number of sensitivity study calculations. The results of this sensitivity study indicate that the ability of the user to change the interfacial momentum exchange length through input added in MELCOR 1.8.2 obviously allows wide variation in countercurrent flow limits and associated pressurizer drainage rates, but the question of the "correct" value to use remains open.

Another code model added in MELCOR 1.8.2 is a debris radial relocation model. Previous versions of MELCOR would predict each radial ring in the core package model

responding independently, with artificial "stacking" of debris columns often observed. This new model was added to relocate molten and/or particulate debris between rings (and axial levels), based upon hydrostatic head equilibration. Sensitivity study results for the Surry TMLB' sequence show more coherent behavior among rings when the debris radial relocation model is enabled. There is no effect on early core heatup or initial clad failure and gap release, but a slightly faster core damage progression and earlier lower head penetration failure (at 11,219s with the debris radial relocation model, *vs* 12,531s with that model disabled).

The core state at vessel failure is also greatly affected by the new debris radial relocation model. With the debris radial relocation model disabled, there is much less debris in the lower plenum at the time a lower head penetration first fails; in particular, the amount of debris in the lower plenum corresponds quite well to the mass of material initially present in the active fuel region in the ring whose core plate failed just previously (*i.e.*, the first, inner, high-powered ring). In the reference calculation with the debris radial relocation model enabled, the mass of debris in the lower plenum at the time a lower head penetration first fails is much greater, about half the total mass initially present in the active fuel region. Also, in the reference calculation with the debris radial relocation model enabled, most of the material remaining in the active fuel region is "intact" (either still in its initial location or refrozen onto intact components). However, in the sensitivity-study calculation with the debris radial relocation model disabled, almost all of the material still in the active fuel region (*i.e.*, above the core support plate) is predicted to be particulate debris. This is the old problem of "stacking" of debris in separate columns, seen in MELCOR 1.8.1 calculations; without the debris radial relocation model, debris in the outer two rings cannot move sideways to the empty inner ring and move down to fall through the failed core plate in that innermost ring.

The capability to model a variety of material eutectics interactions (such as inconel and zircaloy, zircaloy and stainless steel, B_4C and stainless steel, zircaloy and Ag-In-Cd, UO_2 and ZrO_2, and B_4C and zircaloy) was also added to the core package modelling in MELCOR 1.8.2. Earlier versions of MELCOR treated each material melting as a separate process, although there was coding for a specified fraction of solid material to be relocated by molten Zr or steel, to represent dissolution of UO_2 and/or ZrO_2 in melts; the new model has a better treatment of the dissolution of solid material by eutectics melts, based on phase equilibrium and dissolution rate limits, proceeding sequentially as determined by a solid dissolution material hierarchy.

Using the new eutectic materials interaction model generally had only a small effect on the results for the Surry TMLB' station blackout sequence. Both earlier core support plate failure (11,178s *vs* 11,675s) and earlier vessel lower head penetration failure (11,219s *vs* 11,685s) were calculated when the model was enabled, but the difference is quite small ($\leq$500s). The biggest difference found was in the lower plenum structural response. Without the eutectics interactions modelled, most ($\sim$80%) of the steel structure in the lower plenum melted and fell into the cavity; the behavior predicted by MELCOR 1.8.2 with the eutectics interactions not modelled was very similar to the results previously obtained using MELCOR 1.8.1. With the eutectics interaction model enabled, Zr and

stainless steel debris in the lower plenum melted at lower temperatures and flowed to the cavity somewhat sooner, with less heating of the lower plenum steel structure due to the lower melt temperature and shorter residence time of the debris; thus, most (~70%) of the lower plenum structure remained in the vessel throughout the entire transient period analyzed. The larger amount of stainless steel transferred to the cavity in the case without the eutectics interactions modelled resulted in a thicker metallic layer in CORCON existing for a longer time period, and the increased concrete ablation then resulted in slightly higher (≤5%) containment pressures at late times.

A set of MELCOR Surry TMLB' assessment analyses were run with different fission product release model options enabled in MELCOR, as a sensitivity study on fission product source term. These include the CORSOR and CORSOR-M models, each with and without a surface-volume correction term, and the new CORSOR-Booth model with low- and high-burnup coefficient sets, for a total of six possible variations (although obviously only the high-burnup version of the CORSOR-Booth model should apply to most plant analyses). In-vessel, the CORSOR and CORSOR-M options result in similar releases of the Xe, Cs and I volatiles. The CORSOR expression and constants give higher releases for many classes (Ba, Ru, Mo, Ce, La, Cd and Sn), while the CORSOR-M expression and constants produce significantly higher release of Te, with no release at all of Mo, La or Cd. The new CORSOR-Booth model predicts lower releases for the most volatile species (Xe, Cs and I), as well as for Ba, Te and U, than either of the older CORSOR options, while the releases of some other species are intermediate between the higher CORSOR and lower CORSOR-M predictions. The effects of using various CORSOR options are less evident in the total-release comparisons, because the later ex-vessel release can somewhat compensate for in-vessel differences.

In two cases in this source-term sensitivity study (using CORSOR without the S/V term and using the low-burnup form of CORSOR-Booth), there was no high-pressure melt ejection of debris immediately following lower head penetration failure, but instead debris falling into the lower plenum water pool was sufficiently quenched that it remained in the lower plenum for ~2,000-3,000s before reheating sufficiently (to melt) that it could fall into the cavity. The delay in debris ejection in these two cases affects the releases in the lower plenum, because there is more debris in the lower plenum for a longer period of time to contribute to the released source term. The delay in debris ejection in these two cases also affects the melting and ejection of the structural steel mass in the lower plenum, because there is more debris in the lower plenum for a longer period of time to heat the structural material there. The increased retention of steel mass in the lower plenum in the other four calculations resulted in a smaller, thinner metallic layer in the cavity, which was completely oxidized by the end of the transient. The VANESA code, which is used to calculate ex-vessel releases in MELCOR, has no provision for a disappearing metallic layer; therefore, as the metallic layer in the cavity goes to zero, the releases of radionuclide species associated with that layer (*i.e.*, Te, Ru, Cd, Sn, and unoxidized Zr and Fe) can begin growing exponentially.

Similar problems with a vanishing metallic layer in the cavity and associated exponential releases of some radionuclides were seen in several of our other MELCOR 1.8.2

sensitivity study calculations. This problem is inherent in the VANESA formulation itself, not in MELCOR, but is more likely to be encountered with MELCOR 1.8.2 than with MELCOR 1.8.1 because of the increased likelihood of more retention of lower plenum structural steel in-vessel with the new eutectics model enabled. That increased retention of lower plenum structural steel (together with the increased robustness of MELCOR 1.8.2, which makes it easier to run long transients to completion without code failure) results in an increased likelihood of oxidizing the entire cavity metallic layer before the end of the transient period of interest.

The new direct containment heating model added in MELCOR 1.8.2, which models high pressure melt ejection from the vessel into containment, also has been used in these PWR TMLB' analyses. These Surry TMLB' DCH analyses relied heavily on modelling insights and code improvements from the earlier MELCOR DCH assessment analyses of the IET experiments [81].

Initial calculations showed a rapid, brief pressure and temperature spike in containment immediately upon high-pressure melt ejection and direct containment heating. The effect was not extremely pronounced, because only ~15% of the available core material was predicted to be ejected during the high-pressure melt ejection phase in our reference Surry TMLB' calculation.

The amount of melt in the lower plenum at failure is a concatenation of early-time core damage, core plate failure criteria, falling debris heat transfer and possible quench in the lower plenum, and lower head penetration heat transfer and failure criteria. The core plate and bottom head penetration failure temperatures, and the falling debris and lower head penetration heat transfer coefficients were all set to their default values in the MELCOR reference calculation. Sensitivity studies were done varying some of these parameters, but there is little data available for these phenomena, either for evaluation of the MELCOR models' adequacy or for guidance on the values to use for the various input parameters controlling predicted response. In addition, calculations were done in which the peaking factors used were adjusted until ~60% of the available core material was predicted to be ejected during the high-pressure melt ejection phase; this was not to represent "correct" values for core power peaking, but simply to allow a comparison of DCH behavior in otherwise similar calculations with different amounts of high-pressure melt ejection.

Sensitivity studies also have been done varying the relative amounts of melt deposited directly in the cavity, in the various containment volume atmospheres, and on various heat structures in the cavity, basement and containment dome. As would be expected, depositing more debris directly into the cavity or onto heat structures reduces the magnitude of the pressure/temperature excursion, while increasing the amount of debris deposited in the containment atmosphere increases the magnitude of the pressure/temperature excursion. In addition, varying the relative amounts of debris deposited into various containment control volume atmospheres changes the relative magnitude of the pressure/temperature excursion predicted: specifying more debris into the cavity atmosphere (a relatively small volume) results in a very large pressure and temperature spike in that local volume, but much smaller pressure/temperature excursions throughout the rest of

containment, while specifying more debris into the containment dome atmosphere (a relatively large volume) results in a significantly smaller pressure and temperature spike more uniformly throughout the containment.

Including DCH in the Surry TMLB' analysis also affects the amount of material in the cavity (because some debris settled onto heat structures outside the cavity) and hence the amount of concrete ablated, and affects the source term because release of fission products from air-borne debris and from debris settled onto heat structures (instead of into the cavity) is neglected in the MELCOR model. This may or may not be a reasonable assumption. Debris dispersed throughout containment is quickly cooled and quenched, and fission product release is a strong function of temperature. However, the dispersal of debris into relatively small fragments during the HPME/DCH process, fragments which then undergo rapid oxidation, could conceivably facilitate fission product release from the greatly increased debris surface area.

In response to concerns raised on numeric effects seen in various MELCOR calculations, producing either differences in results for the same input on different machines or differences in results when the time step used is varied, several calculations have been done to identify whether any such effects exist in our Surry PWR TMLB' assessment analyses, and to evaluate their impact on the accident sequence prediction. The reference analysis has been run on a Cray, SUN Sparc2, HP Model 755 and IBM RISC-6000 Model 550 workstations, and on a 50MHz 486 PC, and with the code-selected time step and then the maximum allowable time step set by user input to 5, 2.5 and 1s. Similar, minor differences were found in both numeric studies, including: 1) accumulating offsets in both steam generator secondary and pressurizer relief valve cycling early in the transient; 2) timing shifts in clad failure and gap release, and core support plate and lower head penetration failure; 3) variations in amounts of radionuclides released; 4) magnitude and timing offsets in cavity and containment response; and 5) variations in hydrogen burn frequency and duration. However, despite the number of small differences observable, no significant branching into different response modes was found in the time-step or machine-dependency studies.

The differences seen in timing of key events such as clad failure, core plate failure, lower head penetration failure, *etc.*, in these machine-dependency and time-step studies vary by much smaller times (on the order of 10-100s) than the timestep-variation results observed by BNL for their Peach Bottom station blackout analysis with MELCOR 1.8.1 (which often varied by 1,000-10,000s) [12]. The fraction of core materials relocated and the amount of debris in the lower plenum at vessel failure vary in otherwise-identical calculations run on different hardware platforms and with different time steps, but the range found $\pm$6-7% of the total core mass in the active fuel region, not a large variation; the debris temperature in the lower plenum also varies somewhat, over a $\leq$200K range. The fraction of zircaloy oxidized by the time of vessel breach varies from $\geq$20% to $\leq$40%, with most of the numeric-effects sensitivity study calculations predicting $\leq$30%; the fraction of steel oxidized by the time of vessel breach also varies in these analyses, from 0.2% to 0.4%, with most of these calculations predicting $\leq$0.3%. A large part of this reduction in numeric sensitivity represents the significant efforts of the code developers

since the Peer Review in identifying and eliminating numeric sensitivities in MELCOR. BNL has seen similar significant reduction in time step sensitivity rerunning their Peach Bottom station blackout analysis with MELCOR 1.8.2 [12].

In both the machine-dependency and time-step studies, differences were noted early in the transient in the number of times that the steam generator secondary relief valve and, later, the pressurizer PORV cycled. Those differences were traced to differences in over- and undershooting the valve controller setpoint pressures with different time steps and/or different machine accuracies. The tabular function logic was modified to allow step function input, to minimize valves getting caught in a part-open state interpolating between table entries. A time-step controller has been developed to limit the time step whenever a valve pressure setpoint is being approached, through control function input. Based on prototype testing, this addition to the code's time-step control algorithm will decrease the numeric sensitivity significantly, but some other contributing effect still remain to be identified.

Another numeric effect recently identified in these Surry TMLB' demonstration analyses (in our machine-dependency and time-step sensitivity studies) are differences in the time that hydrogen burns occur in containment, and in the amount of hydrogen burned, which in turn can significantly impact containment failure times and releases to environment. A set of sensitivity study calculations were done in which the default overshoots allowed in the combustion ignition mole fractions were both reduced by an order of magnitude, with no visible improvement in the scatter of results calculated. This numerical sensitivity severely hampered and essentially prevented any substantive analysis of the effects of enhanced hydrogen ignition during HPME/DCH; because the numerical sensitivities in the burn coding can be large enough to dominate and cover up the actual physical effect we want to study.

The results from the MELCOR TMLB' analysis have been compared to results from similar analyses by other codes. The early-time behavior of the Surry PWR TMLB' accident has been calculated by several best-estimate codes, notably by SCDAP/RELAP5 [66], MELPROG/TRAC [235] and MELPROG-PWR/MOD1 [236]. The containment response of the Surry PWR to a TMLB' accident has been calculated by the best-estimate containment thermal/hydraulic code CONTAIN, both for the early-time containment response at vessel failure including direct containment heating effects [237, 238, 239], and recently for the long-term containment response [240] with no direct containment heating. The overall transient behavior of the Surry PWR TMLB' accident has been calculated several times by STCP, by various users [241, 242, 243, 244]; at the time they were done, these were best-estimate source term calculations, using a linked set of codes to analyze the entire accident sequence.

The results of comparisons for primary system response and core damage both with detailed, best-estimate, state-of-the-art codes such as SCDAP/RELAP5, MELPROG and MELPROG/TRAC, and with older, engineering-level integrated codes such as STCP, highlight the importance of continued assessment of MELCOR's ability to calculate the early-time thermal/hydraulics in the severe accident precursor. This portion of MELCOR (*i.e.*, the CVH/FL packages) is significantly different than the corresponding RELAP5

and/or TRAC modelling approach (and also significantly different than the corresponding MARCH modelling approach), and the biggest differences found in the results were in the predicted times to core uncovery, which then propagated throughout the remainder of the accident sequence. The maximum and average core heatup rates in the various calculations were generally similar, if best-estimate calculations without in-vessel natural circulation were used as the comparison values; including in-vessel natural circulation tends to slow the core heatup and degradation process somewhat.

The MELCOR calculations generally showed core damage and relocation at lower temperatures than the MELPROG, MELPROG/TRAC or STCP analyses using default failure temperature and other failure criteria, but the various failure criteria are adjustable through input. Because of this, MELCOR also generally seemed to have less debris in the lower plenum at the time of vessel failure (although there is some question of the exact definition of the quantities being compared), but since MELCOR can continue to lose debris from the vessel to the cavity throughout an integral transient calculation, this difference may not be as significant as at first glance. The greater core damage predicted before vessel failure in the STCP calculations resulted in significantly higher in-vessel releases of most fission products than in the reference MELCOR 1.8.2 calculation (although similar to the results from some sensitivity study calculations, notably calculations with the debris radial relocation model disabled). However, the added ex-vessel release of the volatiles in MELCOR produced similar total releases of noble gases, Cs, I and Te; the release fractions for more refractory species such as Ba, Ru and La varied greatly in the various STCP analyses with the MELCOR result somewhere in the range found.

The atmospheric pressures calculated in the containment by MELCOR and by STCP have also been compared. Differences in vessel failure timing are reflected as timing shifts in the early-phase containment responses predicted. Two of the STCP analyses available show qualitatively similar containment response to each other and to MELCOR. An initial pressurization due to PORV outflow followed by a much greater pressurization upon vessel failure; the containment pressure then drops somewhat until a slow pressurization is resumed, to eventual failure. Even the change from rapid pressurization while boiling off cavity water to slower pressurization during core-concrete interaction can be seen in one of the STCP calculations. Quantitatively, the containment pressure in the MELCOR analysis is generally lower than the STCP results, especially during the first portion of the TMLB' transient analyzed, until after ~16hr when MELCOR begins significant core-concrete interaction.

Comparisons with CONTAIN are complicated by the fact that the vessel failure timing and debris ejection, as well as the steam and hydrogen outflow and the containment conditions at vessel failure, may not be the same in the integral (internally-calculated) MELCOR analysis as in the (externally-defined) source(s) assumed to begin a CONTAIN calculation. However, given this uncertainty, the peak pressure rise predicted during DCH by CONTAIN and by MELCOR agreed well in a comparison with no additional pressurization from enhanced hydrogen burn during DCH; the numeric problems found in the burn logic in MELCOR precluded quantitative comparison with a calculation including

enhanced hydrogen burn during DCH. Also, CONTAIN calculations have also been done studying the late containment pressurization resulting from a station blackout in Surry, with no DCH modelled. Those results when compared to the long-term containment pressure response from the MELCOR reference calculation demonstrate that the long-term pressurization rate predicted by CONTAIN is very similar to the corresponding behavior predicted by MELCOR, neglecting the offset due to boiling off cavity water in the MELCOR analysis.

In summary, the effects of new models added in MELCOR 1.8.2 have been investigated, both individually and collectively, for a TMLB' transient scenario in the Surry plant. Results obtained are considered reasonable, based upon comparison to other codes. Significant reduction in numeric sensitivity and significant improvement in code robustness was found, compared to MELCOR 1.8.1. Some numeric effects still remain, in valve cycling, in core material damage and relocation, and in hydrogen combustion, but no significant branching into different response modes was found in any of our numerous sensitivity studies.

23.6 ACRR MP-1 In-Pile Late-Time Melt Progression

As part of the SNL assessment program, MELCOR has been used to model the MP-1 and MP-2 melt progression experiments [83]. MP-1 [245] and MP-2 [246] provided data on late phase melt progression in PWR geometries, particularly in the area of melting and relocation of metallic blockage and rubblized debris beds.

The purpose of the Melt Progression, or MP, series of experiments was to investigate late phase core melt progression and to obtain data for the benchmarking of severe accident codes for these types of phenomena. The MP-1 experiment began with an initial configuration representing a degraded core. The fueled portions of the test bundle consisted of three axial regions: a fuel rod region at the bottom, consisting of 32 PWR-type fuel rods; a crust region, containing the 32 fuel rods surrounded by a Zr-ZrO_2-UO_2 crust; and a debris region at the top, consisting of a ZrO_2-UO_2 rubblized debris bed. The debris and crust regions were fully blocked, and the test section was a closed system filled with helium gas at 69kPa (10psi). The MP-1 experiment used nuclear heating and progressed to partial melting and settling of the debris bed region; no material was melted in the crust region during the experiment. The MP-2 experiment was quite similar, but with a longer test section in the stub region and a longer heating period and more insulation to insure significant melt and relocation occurred.

The basecase input model for MP-1 consists of three radial rings and 13 axial levels (six debris levels, three crust levels, three stub levels, and one lower plenum level); the MP-2 basecase input model is similar but with 5 axial levels in the longer stub region. For both tests, the debris region initially contains particulate debris composed of ZrO_2 and UO_2 with a particle diameter of 2mm. The crust region is composed of zircaloy-clad fuel rods, modeled as intact components in MELCOR, and conglomerate debris, which resides on the cladding component and which is made up of Zr, ZrO_2 and UO_2. The crust region

is initialized in a fully blocked configuration, which is supposed to prevent downward relocation of the particulate debris. (A new option to initialize core components in a degraded state has been added to the MELCOR code as part of this task. With this new option, the user is allowed to initialize core materials in any state allowed by the MELCOR COR package. In particular, this allows an initial core configuration which contains particulate and conglomerate debris. This new input is available beginning with production version 1.8OD, and thus is not included in the version of MELCOR 1.8.2 (1.8NM) released in mid-1993.)

In general, MELCOR did a fairly good job predicting temperature and relocation phenomena in the MP-1 and MP-2 experiments, considering the relatively coarse modelling in the code. For MP-1, temperatures in some parts of the debris region were underpredicted by 250-500K during part of the experiment, but peak temperatures were predicted within 250K at all debris region levels. Crust region temperatures were overpredicted by no more than 200K at all levels. Stub region temperatures were substantially overpredicted at all levels, due to downward axial heat conduction in the cladding component and the lack of radial heat losses in the grid spacer. The poor agreement with data in the stub region was not considered important, because this test arrangement was not representative of a typical severe accident geometry. No melting or relocation was calculated for MP-1, compared to a small amount of material melting observed in the debris region in the experiment. This was probably due to skewing of power peaking factors into the crust and stub regions, and would explain the under- and overprediction of temperatures in the debris and crust regions, respectively.

In MP-2, debris region temperatures were also underpredicted, by as much as 500K at some levels. The calculated axial temperature gradient in the centerline of the bundle was a maximum of 500K, compared to a measured gradient of ~900K. Temperatures in the crust region were predicted within 100K in the outer two rings and within 200K in the inner ring, while stub region temperatures were high, again because of axial conduction within core components.

Material relocation from the crust to the stub region and within the debris region was close to the observed final material state in MP-2. MELCOR predicted a melt pool surface in the debris region at 21.75cm, compared to the observed surface at 21cm. The calculated penetration of debris region materials into the crust region was within 0.3cm of measured penetration (compared to a COR cell height of 1.17cm in that region). MELCOR predicted the relocation of lower melt point crust materials to the bottom of the stub region, and the refreezing of material onto the rod stubs, in appropriate amounts, but only after the candling model in MELCOR was modified to be more representative of the MP experiment geometry and behavior.

Fifteen sensitivity studies were performed on various COR, CVH and HS package input parameters. Sensitivities to minimum component mass indicated the need to use more appropriate values for small-scale experiments, since the defaults were chosen to be appropriate for full-scale plant analyses. Turning off axial heat conduction in the COR package resulted in less heat transfer from the crust region downward, raising crust and debris region temperatures substantially. As noted above, the use of a new candling

model option resulted in better material relocation agreement with test data. Enabling the eutectics model made agreement worse for material relocation, due to the retention of lower melt point materials in the crust. Inserting helium-filled gaps in the insulation heat structures, which was more representative of the experiment configuration, resulted in higher core temperatures, as expected. Using non-default core component radiation heat exchange view factors also affected results somewhat, due to radiation being the dominant heat transfer mechanism between core components in these tests. In general, no unexpected sensitivities were found in the MP-1 and MP-2 input models.

Little or no sensitivity was found to particulate debris diameter, candling model refreezing heat transfer coefficients, convective heat transfer coefficients, gamma heating, time step or computer platform. Making the nodalization finer resulted in little change in temperature profiles, while reducing the number of core cells and control volumes smoothed out temperatures between core regions.

The results of the MELCOR MP-1 and MP-2 basecase calculations were compared to corresponding results obtained by the DEBRIS [247] and TAC2D [248] codes, which were run by the same group performing the experiments. These other codes predicted temperatures more accurately than MELCOR, particularly in the stub region, where heat transfer downward and radially outward through the grid spacer to the cooling jacket was important. Axial heat transfer down through one of the radial insulating layers and into the grid spacer was also modelled better by the other two codes. This was to be expected, due to the full two-dimensional modelling in those codes. Material relocation was not predicted much better by the DEBRIS code, while TAC2D performs only heat transfer calculations.

23.7 GE Large Vessel Blowdown and Level Swell

The MELCOR computer code has been used to analyze a series of blowdown tests performed in the early 1980s at General Electric (GE) [85]. The GE large vessel blowdown and level swell experiments [249] are a set of primary system thermal/hydraulic separate effects tests studying the level swell phenomenon for BWR transients and LOCAs. This experiment series includes both top blowdown tests with vapor blowdown, characteristic of accidents such as steam line breaks, and bottom blowdown tests with liquid and two-phase blowdown, more characteristic of recirculation line breaks. Assessment against this data allows an evaluation of the ability of MELCOR to predict the inventory loss, and hence time to core uncovery and heatup, in the early stages of transients and accidents in BWRs. Also, an implicit bubble separation algorithm has been implemented recently in the CVH package in MELCOR, since the release of MELCOR 1.8.2 in mid-1993. Analysis of the GE tests was intended to validate this algorithm for general use.

The calculated pressure transients generally agree well with the measurement. In the top blowdown tests, there is a relatively fast depressurization for the first few seconds, with progressively slower depressurization later in the transient. Qualitatively, the MELCOR calculations correctly reproduce the increase in vessel depressurization rate as the

nozzle throat diameter and area increase, in the top blowdown experiment set. Quantitatively, there is progressively more difference between the calculated and measured vessel pressures as the nozzle throat diameter and area increases and the depressurization rate increases. This difference is due partly to the fact that the single value of form loss and discharge coefficients used in all these basecase calculations may not be optimum for all test conditions (as indicated by sensitivity studies), and partly due to increased discrepancies between measured and predicted level swelling as the nozzle throat diameter and area, and hence the depressurization rate, is increased.

The test data from the top blowdown tests show the two-phase mixture levels increasing more rapidly early in the transient as the nozzle throat diameter and area, and hence the depressurization rate, is increased, and also shows the two-phase mixture level reaching progressively greater maximum heights before beginning to drop off; for the test with the largest blowdown nozzle dimensions, the observed two-phase liquid level reaches above the top of the dip tube. The two-phase mixture, or swollen, levels calculated by MELCOR correctly reproduce the observed initial swelling, and the predicted two-phase levels initially increase at about the rate determined from measurement in each test; MELCOR correctly reproduces the qualitative trend seen in the test data that the measured two-phase liquid levels peak progressively earlier in the transient as the nozzle throat diameter and area, and hence the depressurization rate, is increased. However, the vessel swollen levels calculated by MELCOR for the different nozzle dimensions all reach a similar maximum value which is significantly below the maximum two-phase levels in the test data, and the two-phase levels begin decreasing earlier in the calculations than observed in the test. The swollen liquid levels in the calculation later decrease less rapidly than observed for the measured two-phase liquid levels, for all these top blowdown tests. After the swollen levels begin to drop, the MELCOR calculations show progressively lower swollen levels at any particular time as the nozzle throat diameter and area, and hence the depressurization rate, is increased; the test data in contrast show the two-phase mixture levels in tests with larger blowdown nozzle diameters remaining above two-phase mixture levels in tests with smaller nozzle diameters throughout the entire period when test data are available.

The discrepancies found in measured *vs* calculated two-phase mixture levels in the basecase calculations for the top blowdown tests are generally all attributable to the limiting in the MELCOR CVH package of the maximum allowed pool bubble fraction to 40%. The maximum swollen levels in each of the four MELCOR top blowdown test analyses correspond to the bubble fraction in the pool reaching a value of $\leq$0.40. As the blowdown nozzle dimensions and hence the vessel depressurization rates increase, the swollen vessel level is predicted to reach that limiting value earlier in the transient and the swollen liquid level of the pool in the vessel then drops more rapidly as the vessel loses inventory more rapidly drops, due to continued inventory loss out the blowdown line, to maintain that pool bubble fraction of $\sim$0.40.

The calculated pressure transients generally agree very well with measurement for the bottom blowdown tests. There is a relatively slow depressurization for the first $\leq$20s seconds, followed by a more rapid depressurization beginning to slow again late in

the transient. The relatively slow depressurization in the first phase of the transients corresponds to the time period where the two-phase mixture level is above the entrance to the blowdown line, so that liquid is being lost directly out the blowdown line. The subsequent more rapid depressurization begins when the mixture level drops below the blowdown line elevation, so that vapor blowdown can occur. As with the vessel pressure histories, the calculated mixture level transients for the bottom blowdown tests generally agree very well with measurement, both during the earlier liquid blowdown and the later vapor blowdown periods. The agreement of predicted level swell with test data is much better in this bottom blowdown test analysis than in any of the top blowdown test analyses because the pool bubble fraction is not being controlled within MELCOR by the maximum allowed value of 40%. There is significantly less level swell in this bottom blowdown test than in any of the top blowdown tests, and the pool bubble fraction is not affected by the maximum allowed value of 40% until very late in the transient, when little pool is left.

Sensitivity studies show that the blowdown flow and vessel depressurization are strongly dependent on the break discharge coefficient used, and weakly dependent on the form loss coefficient used in the blowdown line. The basecase calculations used a discharge coefficient of 0.85 for the top blowdown test analyses and 0.75 for the bottom blowdown test analyses, with a form loss coefficient of 1.5 applied to the nozzle throat area. Other sensitivity studies indicate that the nonequilibrium thermodynamics model must be enabled to calculate any level swell, and that the magnitude and timing of the level swell is dependent on the values used for the maximum allowed pool bubble fraction and for the bubble rise velocity assumed in the bubble separation model (not user input, but variable through sensitivity coefficient input). Comparison to test data suggests that the default maximum allowed pool bubble fraction of 40% is too low for the top blowdown tests, but that the default bubble rise velocity of 0.3m/s produces generally good agreement with data for both top and bottom blowdown tests. The underprediction of level swell in the basecase calculations for the top blowdown tests does not appear to have any significant adverse effect on the code's ability to correctly calculate overall break flow and vessel depressurization.

The results proved insensitive to enabling the optional SPARC bubble rise physics model, which accounts for finite transit time through and interaction with any intervening water pool in the downstream volume. This bubble rise model does not contribute to the behavior response being predicted by MELCOR for these blowdown and level swell experiment analyses, even though two-phase conditions exist for significant periods in the test vessel, because the model only affects vapor flowing out of a flow path into a two-phase pool region; in the GE large vessel blowdown and level swell experiments, the two-phase conditions are on the upstream, inlet side of the flow path and the downstream sink volume consists of only atmosphere.

The basecase MELCOR input model for these GE large vessel blowdown and level swell experiments used a single control volume for the test vessel. This is standard modelling in MELCOR, where multiple control volumes are used to subdivide regions only if there is some obvious change in geometry or flow pattern. Unlike best-estimate

codes such as TRAC or RELAP, MELCOR does not necessarily give better results if components or volumes are subdivided; most MELCOR models assume large, lumped component volumes. A sensitivity study was done in which the single vessel control volume was subdivided into a stack of multiple control volumes, with vertical flow paths added as needed to connect the stacked volumes. The heat structure modelling the vessel cylinder was subdivided correspondingly, also. This is a noding more typical of TRAC and/or RELAP than for MELCOR analyses. Since there is no obvious geometrically "correct" value for junction opening heights in flow paths connecting such a stack of volumes in MELCOR, both large (1ft) and small (1cm) junction opening heights were tried.

Subdividing the blowdown vessel into a stack of multiple control volumes has no significant effect on the vessel depressurization in the top blowdown test analyses. The results for two-phase level calculated using the single-volume basecase noding are in better quantitative agreement with test data in all of these top blowdown experiment analyses than the swollen levels calculated using a subdivided, stacked, multiple control volume model, even though the exact degree of level swelling is underpredicted with the basecase model. For bottom blowdown tests, using a subdivided noding yields small differences in depressurization history, a smoother break flow, and little or no difference in overall vessel level swell compared to test data or to basecase results when large junction opening heights are used. For both the top and bottom blowdown test analyses, using large junction opening heights (equal to the volume heights) in the flow paths connecting the subdivided, stacked control volumes in the finer noding produced much better agreement with both test data and with the 1-volume basecase results than did using small junction opening heights. However, the results of this sensitivity study demonstrate no significant improvement in agreement with test data using a subdivided, stacked, multiple control volume vessel model rather than a single large volume. The results with the subdivided finer noding show more level swell at the bottom of the stack than further up, which seems counterintuitive. There appear to be no benefits and significant drawbacks found in subdividing the vessel into multiple, stacked control volumes, especially given the increased run times required.

Several calculations have been done to search for differences in results for the same input on different machines or differences in results when the time step used is varied in our GE large vessel blowdown and level swell assessment analyses. We also compared results obtained with a recent code version (1.8OI) which includes a new implicit bubble separation algorithm with results obtained using the release version of MELCOR 1.8.2, 1.8NM (in which the bubble rise calculation is explicitly coupled to the rest of the thermal/hydraulics analysis).

The GE large vessel reference calculations were rerun, using the same code version and input models, on an IBM RISC-6000 Model 550 workstation, on an HP 755 workstation, on a SUN Sparc2 workstation, on a CRAY Y-MP8/864, and on a 50MHz 486PC (IBM clone). There is very little or no difference found in the results obtained on any of these hardware platforms. The SUN and PC are always slowest in run time required; the IBM, HP and Cray are all significantly faster with the Cray the fastest by a small fraction for

these analyses. There is also generally little or no difference found in the results obtained as the user-specified maximum allowed time step is progressively reduced and, as would be expected, reducing the time step and thus increasing the number of cycles required correspondingly increases the run times required.

An implicit bubble separation algorithm has been implemented recently [?] in the CVH package in MELCOR. Prior to the implementation of this algorithm, MELCOR was experiencing problems with natural circulation phenomena in the COR package; it is expected that the problems with calculating natural circulation will be eliminated with the implementation of the implicit bubble separation algorithm. A sensitivity study has been done on the effect of this implicit bubble separation algorithm comparing results from MELCOR version 1.8OI to results from the release version of MELCOR 1.8.2, which was MELCOR 1.8NM. The results of that study indicate that there are no major differences in vessel blowdown and/or level swell calculated by either the release version of MELCOR 1.8.2 (1.8NM) or by MELCOR version 1.8OI after an implicit bubble separation algorithm has been added. The results and conclusions of this assessment study should apply equally well to either the release version of 1.8.2 or to later versions.

One noticeable difference is that with the release code the vessel pool bubble fraction always increases to the maximum allowed value, albeit more slowly for larger bubble rise velocities, while with the new implicit bubble separation algorithm the vessel pool bubble fraction equilibrates at lower values for the larger bubble rise velocities. With no difference in vessel depressurization, blowdown flow or collapsed liquid level, this results in higher swollen liquid levels calculated with the release code version than with the new implicit bubble separation algorithm for bubble rise velocities increased above the code default of 0.3m/s. (There is not much difference in swollen liquid levels calculated with the release code version and with the new implicit bubble separation algorithm for the default bubble rise velocity of 0.3m/s.)

The GE large vessel blowdown and level swell tests have been used to validate both best-estimate thermal/hydraulics codes such as TRAC-B [250, 251, 252] and other engineering integrated, engineering-level severe accident analysis computer codes such as MAAP [253]. The results obtained with MELCOR have been compared to available results obtained using those other codes. The MAAP and MELCOR results for these GE large vessel blowdown and level swell tests are generally similar. Both codes underpredict the level swell observed at certain periods in the tests, but with little overall effect on the ability to calculate vessel depressurization during BWR accident scenarios.

TRAC-B correctly reproduces the observed two-phase level behavior, with initial swelling of the level up to the break, due to flashing, followed by a drop in level due to inventory loss. TRAC-B thus calculates a relatively faster depressurization rate in the first few seconds corresponding to steam blowdown, slower depressurization as the mixture level swells up to the blowdown tube inlet which results in two-phase carryover, and finally sustained depressurization corresponding to high quality steam blowdown as the mixture level in the vessel drops back below the blowdown pipe inlet. As already noted, the two-phase mixture levels calculated by MELCOR correctly reproduce the observed initial swelling in each of the top blowdown tests; however, the vessel swollen

levels calculated by MELCOR for the different nozzle dimensions all reach a similar maximum value which is significantly below the maximum two-phase levels in the test data, and the two-phase levels begin decreasing earlier in the calculations than observed in the test. (This discrepancy in measured *vs* calculated two-phase mixture levels in the MELCOR code is due to the limiting in the MELCOR CVH package of the maximum allowed pool bubble fraction to 40%.) MELCOR thus predicts only sustained steam blowdown, since the dip tube elevation remains uncovered throughout the calculation. The best-estimate code, TRAC-B, clearly does a better job of predicting the observed level swell behavior in this test. However, the depressurization histories predicted by both codes are generally similar, despite the differences in calculated two-phase levels and total outflows.

The overall results for these GE large vessel blowdown and level swell test assessment analyses show that MELCOR does reasonably well calculating break flow and vessel depressurization for typical BWR accident conditions. While the level swell is underpredicted at certain periods in the tests, this discrepancy appears to have little effect on the code's overall ability to calculate vessel blowdown during BWR accident scenarios.

23.8 SURC-2 Core/Concrete Interaction

23.9 CSE Containment Spray Experiments

24 Benchmark Problems

Small demonstration problems used to verify the most basic MELCOR models are being developed, collected and documented. Some of these problems are repeats of problems from the 1986 MELCOR V&V program at Sandia [4] summarized in Section 2, updated to the latest code version and platforms used and with expanded sensitivity studies; others were collected from the code developers' test problems and documented; still others were developed for this exercise. A series of reports is being generated in which a number of simple gedanken problems are presented, with results compared to analytical solutions.

Problems described in the first volume [86] include a saturated liquid depressurization, the adiabatic flow of hydrogen, transient heat flow in a semi-infinite solid with convective boundary conditions, cooling of rectangular and annular heat structures in a fluid, the self-initialization of steady-state radial temperature distributions in annular structures, and establishment of flow in a pipe. Problems described in the second volume [87] include manometer oscillation, control volume mass and energy sources, natural convection, flooding (*i.e.*, countercurrent flow limit), compressible pipe flow and emptying wine bottles. The third volume, also in progress, will consist of problems related to the BUR and ESF packages (hydrogen burn and sprays and fans).

Each of these problems has been run on a Cray XMP/24, IBM RISC-6000 Model 550, SUN Sparc2, VAX 8650 or 8700, and a 386 PC to check for machine dependencies. Time step studies, nodalization studies and studies on code modelling options were also done when appropriate. All code problems identified during these analyses have been corrected. Input listings for the various problems are given as appendices.

24.1 Saturated Liquid Depressurization Test

This problem tests the CVH/FL/CVT packages and the HS package. It was originally run and documented as part of the 1986 MELCOR V&V effort [4]. Those results were for MELCOR 1.6.0; the final results given here are for MELCOR 1.8.1 (version 1.8JG).

The results show good agreement between MELCOR predictions and analytical solution, demonstrating MELCOR's ability to predict the depressurization of a reactor vessel into its containment with the associated involvement of very rapid flashing of saturated water within the vessel.

No major machine dependencies were found. A problem with the heat structure package time step control due to roundoff problems on some machines was noted. This affected the run efficiency, but did not significantly change any results calculated. The problem has been reported to, and solved by, the MELCOR code developers, and corrected in version 1.8JG. In the meantime, a minor input modification bypassed the problem.

24.2 Adiabatic Expansion of Hydrogen, Two-Cell Flow

MELCOR calculations for the adiabatic flow of hydrogen between two control volumes have been run, and are compared to a closed-form analytical solution. This problem tests the CVH/FL/CVT packages and the NCG package. It was originally run and documented as part of the 1986 MELCOR V&V effort [4]. Those results were for MELCOR 1.6.0; the results given here are for MELCOR 1.8.1 (version 1.8IM).

These results show good agreement between MELCOR predictions and analytical solution, demonstrating MELCOR's ability to predict the adiabatic expansion of a noncondensable gas. The slight differences sometimes visible between the MELCOR predictions and the analytic solution are in part due to using temperature-dependent heat capacities in MELCOR, which introduces some minor deviations from the ideal gas assumption in the analytical solution, and partly due to the time step selection.

24.3 Transient Conduction in a Semi-Infinite Solid Heat Structure

Predictions of the heat conduction models in the MELCOR heat structures package have been compared to exact analytical solutions for transient heat flow in a semi-infinite solid with convective boundary conditions. The accuracy of the heat conduction models is demonstrated and the effects of various node spacings and time steps are investigated. The ability of MELCOR to predict the exact solution depends on the fineness of the node spacing and the time step used, and the precision of the computer.

This problem primarily tests the HS package, and was originally run and documented as part of the 1986 V&V effort [4]. Those results were for the MELCOR 1.1 code; two of the cases were later run with MELCOR 1.6 with no significant differences from the MELCOR 1.1 results presented in [4]. The results given here are for MELCOR 1.8.1 (version 1.8IM).

Results obtained with MELCOR 1.8.1 appeared more accurate, in general, than the earlier, MELCOR 1.1 analyses of this problem. Errors increased as noding detail was decreased, in both MELCOR 1.1 and MELCOR 1.8.1. However, errors remained nearly constant with MELCOR 1.8.1 as the time step was increased over a very wide range, a substantial improvement over MELCOR 1.1 where the error had increased as the time step was increased.

24.4 Cooling of Structures in a Fluid

MELCOR calculations for the cooling of two heat structures in a fluid have been compared the results to an exact, analytic solution. Both rectangular and cylindrical geometries were tested. This problem primarily tests the implementation of the internal heat conduction methodology in the absence of internal or surface power sources in the

HS package, and was originally run and documented as part of the 1986 MELCOR V&V effort [4]. Those results were for MELCOR 1.6.0; the results given here are for MELCOR 1.8.1 (version 1.8IU). The good agreement between the MELCOR results and the exact analytic solution show that the finite-difference methods used in the MELCOR HS package produce accurate results.

24.5 Radial Conduction in Annular Structures

MELCOR predictions of the steady-state temperature distributions resulting from radial heat conduction in annular structures were compared to results obtained from exact analytic solutions. Two sets of boundary conditions and two cylinder sizes were considered. In addition, a transient calculation was done with an initially uniform temperature profile to test whether MELCOR can achieve the correct steady-state temperature profile.

This problem primarily tests the HS package, and was originally run and documented as part of the 1986 MELCOR V&V effort [4]. Those results were for MELCOR 1.6.0; the results given here are for MELCOR 1.8.1 (version 1.8IV). The agreement between MELCOR results and the analytic solution is excellent in all cases.

A coding error was found in MELGEN which caused code aborts on a VAX when no control volumes were present in the input model. This problem was corrected in MELCOR version 1.8IQ.

24.6 Flow Establishment

MELCOR 1.8.1 predictions for the establishment of flow in a pipe connected to a large resevoir after a valve is suddenly opened have been compared to results obtained from exact analytic solutions, for both the final, asymptotic velocity attained and for the time required to establish this flow. Several variations in controlling parameters were considered.

This problem primarily tests the CVH/FL package. The results given here are for MELCOR 1.8.1 (version 1.8IR). The results of this problem show that the flow solution algorithm in MELCOR can correctly calculate both the flow startup and subsequent steady-state flow in a pipe fed from a liquid resevoir.

During this analysis, an error was found and corrected in the time-independent control-volume logic. Even after this error was corrected, there was still a noding sensitivity to the values used for the junction opening heights, but it affected the results by $\leq$1%.

24.7 Simple Manometer

This problem compares MELCOR calculations with analytic results for a simple undamped manometer. It exercises the CVH and FL packages, and tests the implementation

of the inertial and head terms in the hydrodynamic equations. It also investigates the performance of the code when a rising liquid level fills a pool with water, forcing the pool surface to move into the next control volume and the out-flow to change from gas to liquid (a frequent source of computational difficulties in hydrodynamic codes, often referred to as "water packing"). The results given here are for MELCOR 1.8.1 (version 1.8JK).

MELCOR calculations for the oscillation of a simple manometer were in excellent agreement with the analytic solution, even when the the liquid level must cross from one control volume to another during the motion. Some time-step dependent numerical damping of the amplitude of the motion was observed.

The MELCOR calculations reproduced the analytically-predicted period of oscillation, demonstrating that the inertia and head terms are correctly coded in the flow equation (at least for pool flow); in particular, it confirms that terms arising from motion of the pressure reference point in a control volume to follow the pool surface are correctly formulated.

There is close agreement (in the simplest cases) between the observed damping and that predicted by analysis of the numerical algorithm for implicit treatment of head terms in the liquid. This confirms both that the major source of damping has been identified and that the implicit treatment is correctly coded.

The relative absence of calculational difficulties when the liquid level crosses from one cell to the next gives evidence of the effectiveness of the numerical algorithm in avoiding "water packing" problems.

24.8 Mass/Energy Sources

This problem compares MELCOR results to hand calculations for injection of a steam/water mixture into a control volume using mass and enthalpy sources. It tests the CVH/CVT packages as well as the NCG and TF packages. This problem was derived from a CONTAIN test problem [212]. The results given here are for MELCOR 1.8.2 (version 1.8NE).

The results of this test problem confirm

1. correct implementation of sources, including mass/energy conservation,
2. consistent implementation of the mixed-material equation of state, and
3. qualitative behavior of mass/energy transfer at a pool surface.

This test problem suggests a need for more user-friendly source options to partition two-phase water correctly. Part of the problem is that (unlike for noncondensable gas mass/energy sources) specifying a temperature is not sufficient to fully define a two-phase state since the state also depends on pressure. (This is the reason that the code

requires enthalpy to be input in the source definition rather than temperature.) There is currently no direct association of an enthalpy source with particular H_2O mass sources in the user input (again, unlike for noncondensable gas mass/energy sources). One option would be to "bind" an enthalpy source to a water mass source, and use the pressure in the target volume to determine the saturation state. A more general option is to allow a flow path "connected to" a source; this would allow interaction of gases with pool if introduced below the pool surface, and could be extended to allow interaction of liquid with atmosphere if introduced above the pool surface.

24.9 Flooding

This problem compares MELCOR calculations with analytic results for a simple countercurrent flow problem. It exercises the CVH and FL packages, and tests the form and implementation of the two-phase momentum exchange, as well as relative gravitational terms between pool and atmosphere flows. The results given here are for MELCOR 1.8.2 (version 1.8NE).

This calculation tests the form and implementation of the momentum exchange and relative gravitational terms between pool and atmosphere flows, and the results confirm that the code model is now properly implemented, at least for vertical flow paths. No significant machine dependencies or time step effects were found. The results for MELCOR 1.8.1, also included, show substantially less countercurrent flow; in fact, the state of the system jumped from simple upflow of air to downflow of liquid so quickly that the flooding curve was not well defined, and might actually lie below the plotted line. (Old versions of the MELCOR code, such as 1.8.1, contained both a different default momentum-exchange length and a coding error which introduced an additional factor of that length.)

A sensitivity study was done evaluating the effect on the predicted flooding curve of varying the junction opening heights. The default flow path opening heights were used in the basecase analysis; for a vertical flow path, this corresponds to the radius of a circle whose area is equal to that of the flow path (in this case 0.127m). Values used in this sensitivity study ranged from much larger (10m) to much smaller (0.01m). The two largest junction opening heights considered, 10m and 5m, respectively are greater than and equal to the adjacent control volume heights and therefore truncate to the adjacent control volume heights. These two cases give different results because these opening heights are large enough to include the liquid accumulating in the bottom of the test section in their average junction void fraction. Opening heights of 1m and 0.1m give results very similar to those obtained for the basecase, default model, and also give results in good agreement with the theoretical Wallis flooding curve for this problem. The smallest junction opening height tried, 0.01m, shows some deviation from the expected result, possibly due to relatively large reductions in the buoyancy and momentum exchange terms compared to inertial terms in the momentum equation.

The recommended flow path inertial length was used in the basecase analysis; for a vertical flow path connecting vertically-stacked control volumes, this corresponds to the

distance from the midpoint of one volume to the midpoint of the other volume, in this case a length of 5m. The inertial-length values used in our sensitivity study ranged from much larger (25m) to much smaller (0.01m). There is very little effect on the flooding behavior predicted over a wide range of flow path inertial lengths in MELCOR 1.8.2. However, here, again, some deviation i seen when the inertial term is made extremely large.

The default value for the momentum exchange length was used in the basecase analysis; for a vertical flow path, this corresponds to the distance between the lowest point in the flow path and the highest point, as defined by the junction elevations and the opening heights), in our problem 0.1128m for a vertical flow path with junction opening heights of 0.0564m. As would be expected, in general changing the momentum exchange length directly affects the amount of countercurrent flow possible and thus shifts the predicted flooding curve. Shorter momentum exchange lengths allowed much greater countercurrent flow.

24.10 Natural Convection

This problem compares MELCOR results to known correlations for free convection flow in a volume between hot and cold walls. It tests the CVH/FL/HS packages as well as the NCG and TF packages. This problem was developed by Randy Cole. The results given here are for MELCOR 1.8.2 (version 1.8NM).

The results of this test problem evaluate the internal consistency of MELCOR formulations and correlations, and its ability to reproduce known results for free convection. The simple case here may be analyzed directly by any undergraduate heat transfer student, but most cases of real interest are not so simple, and no relevant correlations exist. In practice, results for the more complicated problems are obtained by constructing a multi-volume, multi-flowpath nodalization and solving the fluid equations. The calculated behavior will involve a balance of buoyant and frictional forces and a balance of advective and boundary layer heat transfer. The ability (or inability) of MELCOR to reproduce expected results for simple cases should enhance confidence in results calculated for more complicated cases.

The differences found in the different cases studied are generally moderate or minor, not bad for a temperature-driven problem. The relatively good agreement between the results of calculated convection and direct application of a free convection correlation for this simple case gives some confidence that MELCOR calculations will not be in serious error for more complicated cases of free convection. (Most cases of real interest are driven by heat sources, *i.e.*, by heat fluxes, and any reasonable heat transfer mechanism will move the correct amount of heat.)

24.11 Compressible Pipe Flow

This problem compares MELCOR results to analytic solution for steady flow of a compressible fluid such as a γ-law gas through a uniform pipe. It tests the CVH/FL packages as well as the NCG and CF packages. This problem was developed by Randy Cole. The results given here are for MELCOR 1.8.2 (version 1.8NM).

The hydrodynamics equations in MELCOR 1.8.2 do not include the v^2 terms describing kinetic energy ($\frac{1}{2}\rho v^2$) and momentum flux $\rho v \cdot \nabla v$. These terms are a well-known source of problems in control volume codes, because they cannot be properly defined without a detailed description of problem geometry which, by the very nature of the control volume approach, is not available (at least in the general case. However, it is part of hydrodynamic folklore that such terms do not become important until the Mach number becomes greater than about 0.3.

This exercise confirms that element of the folklore for a simple geometry. It also tests the proper implementation of the hydrodynamic equations, including friction terms, and of the equation of state for noncondensible gases.

24.12 Bottle Emptying

The object of this set of calculations is to test MELCOR's ability to handle simple flooding cases less artificial than the benchtop flooding problem analyzed in Section 4. This problem compares MELCOR calculations with real data [254] for the familiar case of emptying an inverted bottle filled with water. It exercises the CVH and FL packages, and tests the form and implementation of the two-phase momentum exchange, as well as relative gravitational terms between pool and atmosphere flows. The results given here are for MELCOR 1.8.2 (version 1.8NE).

25 Air Ingression Calculations for Selected Plant Transients

Two sets of MELCOR calculations [88] have been completed studying the effects of air ingression on the consequences of various severe accident scenarios. One set of calculations analyzed a station blackout with surge line failure prior to vessel breach, starting from nominal operating conditions; the other set of calculations analyzed a station blackout occurring during shutdown (refueling) conditions. Both sets of analyses were for the Surry plant, a three-loop Westinghouse PWR. For both accident scenarios, a basecase calculation was done, and then repeated with air ingression from containment into the core region following core degradation and vessel failure.

In addition to the two sets of analyses done for this program, a similar air-ingression sensitivity study was done as part of a low-power/shutdown PRA, with results summarized here; that PRA study also analyzed a station blackout occurring during shutdown (refueling) conditions, but for the Grand Gulf plant, a BWR/6 with Mark III containment.

These calculations have been of limited scope, to assess the magnitude of air ingression that would be necessary to produce any significant alteration of core degradation or radionuclide release. These calculations constitute initial steps toward the definition of experimental conditions that might be employed in the planned fifth test of the Phebus-FP program [255] which is to involve air ingression [33]. Because so little is known about air oxidation during severe reactor accidents, only limited modifications to the MELCOR code could be made to treat the effects of air ingression. The changes in reaction kinetics of air and Zircaloy, and the enhanced heat of Zircaloy-air reaction, were available in the standard MELCOR 1.8.2 code version. Modifications made for these calculations treated only the enhanced release of ruthenium from fuel in air [257]; effects air could have on the release and transport characteristics of other radionuclides were not modelled for these initial calculations.

All three studies lead to the same conclusions. For the two major phenomena dependent on air ingression:

1. There can be a significant increase in ruthenium release in-vessel, to ~50-80% of initial inventory, assuming moderate air ingression rates of ~10-100mole/s; without any air ingression, only trace amounts of ruthenium are released.
2. There is some increase in clad oxidation degree and energy, but only ~10-20%; most of the oxygen sourced into the core region escapes before it is consumed.

The enhanced ruthenium release with air ingression was expected. The relatively small changes in core temperatures, hydrogen production and steam consumption, and oxidation energy were not expected. The greater oxidation energy due to reaction of Zircaloy with oxygen does cause core temperatures to rise more quickly than for oxidation only

with steam, but those higher temperatures then cause the remainder of the core material to melt, relocate and be lost to the cavity sooner than predicted with no air ingression into the core. Oxidation of Zircaloy with air rather than with steam for the relatively short period of time that the clad remains in-vessel does not significantly affect the overall steam consumption and hydrogen production, and the total oxidation energy, because these are dominated by the long-term oxidation of structural stainless steel in the core and especially in the lower plenum.

The assumed air ingression does not significantly affect most of the accident scenario. There are some small effects on fission product releases in general:

1. There is very little change in the release of the volatile species, *i.e.*, noble gases, Cs, I and Te, which are released at lower temperatures ($T \leq 2000$K); most of their initial inventory has been released before vessel breach and air ingression.

2. In most cases, there is a small increase in the releases of those species, *i.e.*, Ba and Sn, requiring somewhat higher temperatures ($2000\text{K} \leq T \leq 2500\text{K}$) for release, probably reflecting the increased oxidation energies and temperatures from Zircaloy reacting with air.

3. There is usually a decrease in the release of refractory species, *i.e.*, Ce and U, which are released only at very high temperatures ($T > 2500$-3000K), possibly due to the cooling effect of sourcing relatively cool containment air into the core region.

These predicted changes in radionuclide release reflect only the effects of air ingression changing the calculated core temperatures and relocation history. There may be other, larger changes in fission product release with air ingression, if other species are also sensitive to the oxygen potential as is ruthenium; however, any such additional effects air could have on the release and transport characteristics of other radionuclides were not modelled for these initial calculations.

These studies help quantify the amount of air that would have to enter the core region to have a significant impact on the severe accident scenario. These calculations demonstrate the potential of air ingression to substantially enhance ruthenium release. These analyses indicate no substantive increases in maximum core temperatures, albeit with modest acceleration of the core degradation process, due to the increased heat of reaction of Zircaloy oxidized by air rather than by steam.

26 MELCOR ABWR Analyses

A number of MELCOR calculations have been done for severe accident sequences in the ABWR and the results compared with MAAP calculations for the same sequences [89]. The program task was to run the MELCOR program for two low-pressure and three high-pressure sequences to identify the materials which enter containment and are available for release to the environment (source terms), to study the potential effects of core-concrete interaction, and to obtain event timings during each sequence. The source terms include fission products and other materials such as those generated by core-concrete interactions. All calculations, with both MELCOR and MAAP, analyzed loss-of-cooling accidents in the advanced boiling water reactor (ABWR) plant.

All calculations, with both MELCOR and MAAP, analyzed loss-of-cooling accidents in the ABWR plant. The LCLP-PF-R-N and LCLP-FS-R-N sequences are accidents starting with a loss of all core cooling and with vessel failure occurring at low pressure; the LCHP-PF-P-M, LCHP-FS-R-N and LCHP-PS-R-N sequences also are accidents starting with a loss of all core cooling but with vessel failure occurring at high pressure. In all these sequences, the passive flooder automatically floods the lower drywell. The containment depressurizes as planned through a relief rupture disk, except in the LCHP-PF-P-M sequence where containment failure is through penetration leakage. In the LCLP-FS-R-N and LCHP-FS-R-N sequences, the firewater spray provides additional containment cooling; in the LCHP-PS-R-N sequence, the drywell spray provides additional containment cooling.

MELCOR generally reproduced the event sequences predicted by MAAP, albeit usually with timing shifts. The major differences found were in core degradation and vessel failure time, and in core-concrete interaction and containment depressurization time.

In all cases, the core was predicted to uncover slightly later by MELCOR than by MAAP, at 27min for MELCOR compared to 18min for MAAP. The core degradation process also was slower in the MELCOR analyses than for MAAP – MELCOR calculated vessel failure to occur later than in the MAAP analyses, at 3.3hr *vs* 1.8hr for the LCLP sequences and at 4.5hr *vs* 2.0hr for the LCHP sequences. However, both MELCOR and MAAP predict vessel failure to occur earlier in sequences with ADS depressurizing the primary system than in scenarios where the vessel fails at pressure.

The core debris in the cavity is quenched by the passive flooder in the MAAP analyses, so little or no core-concrete interaction occurs. MELCOR does not have an ex-vessel debris quench model, so the core debris in the cavity remains unquenched and hot in the MELCOR calculations, resulting in significant and continued core-concrete interaction predicted. This in turn results in faster containment pressurization and earlier rupture disk opening predicted by MELCOR, due both to more generation of noncondensables in core-concrete interactions and to continued boiling of the cavity water pool by contact with the hot, unquenched debris.

Both MELCOR and MAAP predict release of almost all the noble gas initial inventory and small releases of all other fission products in the sequences failing through the

rupture disk, although the releases calculated by MELCOR are slightly higher than those predicted by MAAP for the volatiles CsOH and CsI (although <1% in all cases). Some portion of the higher releases predicted by MELCOR is due to continued release from the unquenched, hot core debris in the cavity.

Both MAAP and MELCOR predict much greater releases for most fission products (but lower for the noble gases) if the containment fails through drywell penetration seal leakage than if the containment fails as intended through the rupture disk in the wetwell vapor space, verifying the benefit of suppression pool scrubbing on reducing the source term to the environment. The releases to the environment calculated by MELCOR for the volatiles CsOH and CsI in this case (10-15%) are in good agreement with the values predicted by MAAP (8-10%), as are the releases of the noble gases (about 50% with both codes).

Sensitivity studies were done on the impact of assuming limestone rather than basaltic concrete, and on the effect of quenching core debris in the cavity compared to having hot, unquenched debris present. Assuming limestone concrete in the cavity resulted in faster containment pressurization and earlier rupture disk opening due to more generation of noncondensables in core-concrete interactions. Having "quenched" debris in the cavity in the MELCOR calculations resulted in slower containment pressurization and later rupture disk opening, in better agreement with the MAAP results. Varying the concrete type or the debris temperature had no major effect on the fission product release calculated by MELCOR.

27 VVER Analyses in Russia

MELCOR is being used by a number of groups to model VVER nuclear power plants, as already noted in Section 15, even though the code models are not all readily applicable to the VVER design and even though there has been no development of MELCOR for VVER phenomenology. MELCOR is being used in Russia to model a VVER-440/213 reactor and plant [258].

MELCOR 1.8.0 has been used for VVER analyses by several organizations in Russia; MELCOR 1.8.2 has been received.

Input decks are being prepared for the VVER-440 and VVER-1000 nuclear power plants. To check the validity of the control and safety system logic, a small (10cm diameter) cold leg LOCA has been calculated, assuming the full set of control and safety systems in the input deck is available duirng the accident; the MELCOR results have been compared to results from RELAP5/MOD2.

The design basis accident, a small (10cm diameter) cold leg LOCA assuming the full set of control and safety systems in the MELCOR input deck is available, has been calculated. The control and safety systems available in the MELCOR input are

1. main coolant pump model,
2. pressurizer heaters, pressurizer 'spray' lines,
3. hydroaccumulators,
4. make-up system,
5. high pressure injection system (HPIS),
6. low pressure injection system (LPIS),
7. steam generator auxiliary feedwater system (AFW),
8. steam generator emergency feedwater system (EFW),
9. BRU-A, BRU-K, steam generator safety valves,
10. reactor scram control,
11. containment active spray system,
12. containment passive spray system,
13. pressurizer PORVs, and
14. long term residual system.

The results of MELCOR 1.8.0 calculations for the small break LOCA in the primary circuit have been checked against results from the RELAP5/MOD2 code, and for most points they are compatible; the MELCOR containment results were compared with results from STCP-VVER, and the results of the comparison are satisfactory.

Checking the input deck validity and models describing the core degradation process, coupled with VVER-440/213 containment behavior, is completed. The full severe accident calculation using MELCOR 1.8.0 showed several nonphysical results and numerical solution problems. Input modifications were made to try to avoid some of these problems, keeping in mind the physical sense of the results. Detailed descriptions of the problems encountered and resolved during the course of the calculation are given in [258].

Several of the problems were connected with simulation of material relocation and blockage processes in the active core and lower plenum regions. The lack of an in-vessel debris heat transfer model in MELCOR 1.8.0 accounting for debris falling into and through a water pool in the lower plenum was noted as an unresolved problem; such a model has been provided in MELCOR 1.8.2. Still-unresolved problems identified in [258] are lack of conduction heat transfer for particulate debris in adjacent core cells, lack of a reflood model, and lack of melting in core steel structures which were modelled as heat structures and calculated to be heated above the melting point.

The description of the bubbler tower is identified as the most difficult part of the VVER-440/213 containment thermal/hydraulic analysis. Explicit description of the steam flowing through the water layer in the bubbler trays leads to a nonphysical high pressure difference between the bubbler tower primary side and the air volume of the trays. This problem was resolved through input modifications.

Validation of the COR model against results of a CORA-VVER experiment is underway. Due to the lack of a model in MELCOR 1.8.0 to simulate the power distribution in the experimental assebly during the course of the experiment, the results of the MELCOR 1.8.0 calculation are valid only for the first stage of the experiment. Investigation of this subject will continue after receiving MELCOR 1.8.2 (which has an electric heater rod modelling capability [24], as noted in Section 4.4).

28 ERI MELCOR Assessment

NRC has funded several MELCOR assessment activities at Energy Research, Inc., including a review of the existing heat and mass transfer correlations in MELCOR including identification of potential heat and mass transfer correlations for inclusion in the MELCOR code [90], sensitivity studies varying heat and mass transfer correlations in plant calculations for selected accident scenarios (a station blackout and a LBLOCA in the Surry plant) [91], and calculations for FIST BWR thermal/hydraulic experiments 6SB2C and T1QUV [92].

28.1 Sensitivity Studies on Heat and Mass Transfer Correlations

The impact of selected heat and mass transfer correlations on results of key accident signatures calculated by MELCOR 1.8.2 has been assessed by performing a number of sensitivity calculations for two severe accident scenarios (*i.e.*, a station blackout and a large break LOCA) in the Surry plant. [91]

The input model used is that developed by Sandia for the station blackout analyses summarized in Section 23.5 and documented in [84].

ERI recommended a number of heat transfer correlations in [90] as candidates for sensitivity studies. In this study, only a selected subset of the recommended correlations were implemented into test code for use in sensitivity studies. These include correlations for natural and forced convection adjacent to heat structures, boiling heat transfer in rod bundles, natural and forced convection in debris beds, and condensation on heat structures. The limited scope of this study dictated the extent of the sensitivity studies done; therefore, the impact of other potentially important heat transfer processes, such as natural convection and radiation in the COR package and natural convection for internal flows, were not studied.

The numerical sensitivity of MELCOR 1.8.2 was assessed by varying the user-input maximum time step from 0.5s to 10s. Some differences were noted in the timing of key events during severe accident progression; however, these differences were found to be much smaller than those observed using earlier code versions.

Sensitivity studies performed using the modified correlations for natural convection on external surfaces (without changing the correlations for other phenomena such as condensation) revealed little sensitivity of key severe accident signatures, including containment pressure as a function of time, time of containment failure and radiological source terms. The sensitivity of the calculated results was more pronounced for the modifications involving the use of forced convection heat transfer correlations on external surfaces; in this case, small differences in containment pressure as a function of time were noted.

Replacement of the existing boiling heat transfer correlation in the MELCOR COR package showed little impact on reactor vessel pressure, fuel rod temperature and fluid temperature.

Two simplified correlations for natural and forced convection in debris beds were tested for the large break LOCA scenario; results indicated little or no sensitivity of primary system pressure and debris temperature to the modified correlations. The most pronounced sensitivity was observed for in-vessel hydrogen generation, for which the sensitivity study showed 16% more hydrogen generated as compared to that based on the existing MELCOR correlation.

Test implementation of two replacements for the existing condensation models in MELCOR, consisting of either a heat and mass transfer analogy, with the inclusion of the film resistance, or a diffusion model that includes a convection correlation based upon experiments, showed differences in the calculated containment pressure and the containment failure time for both the station blackout and the LBLOCA scenarios. Lower containment pressure trends were observed for the sensitivity studies, due to the higher condensation heat fluxes calculated by the test correlations. For the two cases examined, MELCOR results appeared to be most sensitive to the tested condensation correlations.

Even though the complexity of the thermal/hydraulic integrations and the possibility of compensating errors appear to reduce the influence of heat and mass transfer correlations on the global accident signatures, based upon the limited cases investigated in this study, the authors conclude by strongly recommending upgrading the existing heat and mass transfer correlation set in MELCOR, on the grounds that condition might be encountered for other accident sequences or plant types where larger sensitivities could be envisioned.

28.2 FIST BWR Thermal/Hydraulics Tests 6SB2C and T1QUV

The MELCOR code has been used, successfully, to simulate the 6SB2C and the T1QUV experiments performed in the Full-Integral Scale Test (FIST) facility [92]. The release version of MELCOR 1.8.2 (1.8NM) was used.

The FIST program [259, 260, 261] investigated heat transfer and thermal/hydraulic phenomena during the early stages of a reactor transient and/or accident in a simulated GE BWR facility. More than twelve different tests were conducted in the test facility, which is a full-height, 1/642-(volume)-scale model of a BWR/6-218. The BWR core was simulated using electrically heated rods.

The 6SB2C test was a simulation of a small break LOCA in the recirculation piping of a BWR/6-218, with ADS and low-pressure ECCS assumed to be operational. MELCOR correctly simulated the gross features of the test. With an input model developed only from the available facility descriptions and drawings, MELCOR predicted the correct system pressure behavior and provided a qualitative prediction of system water inventory. However, there are differences between the code calculations and the test data insofar as rod and peak cladding temperatures are concerned. The differences between the code predictions and the test data for the rod temperatures can be attrib.ted to the absence of models in MELCOR for the spray cooling by LPCS and the top reflood heat transfer from the heater rods.

The system pressure behavior predicted by MELCOR is judged to be in excellent agreement with the test data; associated with the system pressure, the break flows predicted by MELCOR are also in good agreement with test data.

A reasonable agreement was observed in the MELCOR prediction of the liquid inventory in various control volumes as a function of time. The qualitative behavior of the MELCOR predictions are comparable to the test data, and all the major trends and phenomena are predicted. However, quantitative differences exist between the code predictions and the test data for the liquid inventory.

The rod temperatures at the lower elevations of the heater rods were predicted well, at least qualitatively, by MELCOR. The MELCOR predictions were no worse than the TRAC calculations, at least as far as the temperatures in the lower elevations are concerned. However, the MELCOR code was unable to predict the spray cooling and the top reflood process as observed in the experiment. Hence, the rod temperatures calculated by MELCOR for the top 24in of the heater rods were in minimal agreement with the test data.

Of more interest to the prediction of the rod temperatures is the reflood phenomena. The fuel rods in the test were cooled partially by top reflood, by the LPCS, and by bottom reflood from the bypass by LPCI. MELCOR does not have a core spray model. The coolant added by core sprays and LPCI always filled the core channels "bottom-up". As a result, the top reflood behavior observed in the test (that caused the upper elevations of the fuel rods to cool down before the middle elevation of the rods) was not predicted by MELCOR. Hence, MELCOR showed that the higher elevations of the fuel rods to be hotter than the middle elevations, with the result that the peak clad temperature occurs at a higher elevation than that observed from the measured data.

The T1QUV test is a simulation of a transient with a failure to maintain water level. Inventory loss occurs through the SRV at high pressure. A reasonable agreement was obtained between the MELCOR predictions and the data for system pressure. Only a minimal agreement was obtained between the MELCOR predictions and the test data for water levels and rod temperatures. However, differences were noted between the decay power input to the code and the test data. A simulation was performed by reducing the decay power input to the FIST heater rods by 20%, and a better comparison was obtained between the predicted rod temperatures and the test data.

Several time step and machine dependency sensitivity studies were also performed. A small effect of the time step on the results was noted for the 6SB2C test. The calculations were repeated on two different computers

29 LANL MELCOR Assessment for MIST Thermal/Hydraulic Tests 3109AA and 3404AA

As part of the MCAP program, LANL has completed assessment of MELCOR 1.8.2 using experiments 3109AA and 3404AA from the Multiloop Integral System Test (MIST) experiment series. A report has been sent to the NRC; no further information was available for inclusion in this report.

The MIST facility [262] is a full-height, 1/815-volume-scale representation of a 2×4loop B&W plant primary system. The facility includes a 49-rod core with 45 heated rods supplying 330kw representing 10generator (OTSG) simulators and hot legs, and 4 cold legs and coolant circulation pumps, and 4 vent-valve simulators.

30 Summary and Conclusions

This review of MELCOR verification, validation and assessment to date reveals that most of the severe accident phenomena modelled by MELCOR have received or are receiving some evaluation. Figures 30.1, 30.2 and 30.3 summarize the available MELCOR assessment against experimental test data for primary system thermal/hydraulics, in-vessel core damage and fission product release and transport, and ex-vessel and containment phenomenology, respectively. Only analyses that are completed or already underway are included; analyses scheduled but not yet begun are not included.

30.1 Primary System Thermal/Hydraulics

Primary system loop natural circulation flows are the focus of the FLECHT-SEASET natural circulation test analyses. MELCOR results showed excellent agreement with multi-loop, single-phase liquid natural circulation data; however, significant code problems were encountered in the two-phase flow portions of the transient. While it was possible through a number of nonstandard input changes to force the expected behavior, and while some of the code problems found have since been resolved, the ability of MELCOR to predict two-phase natural circulation remains very questionable.

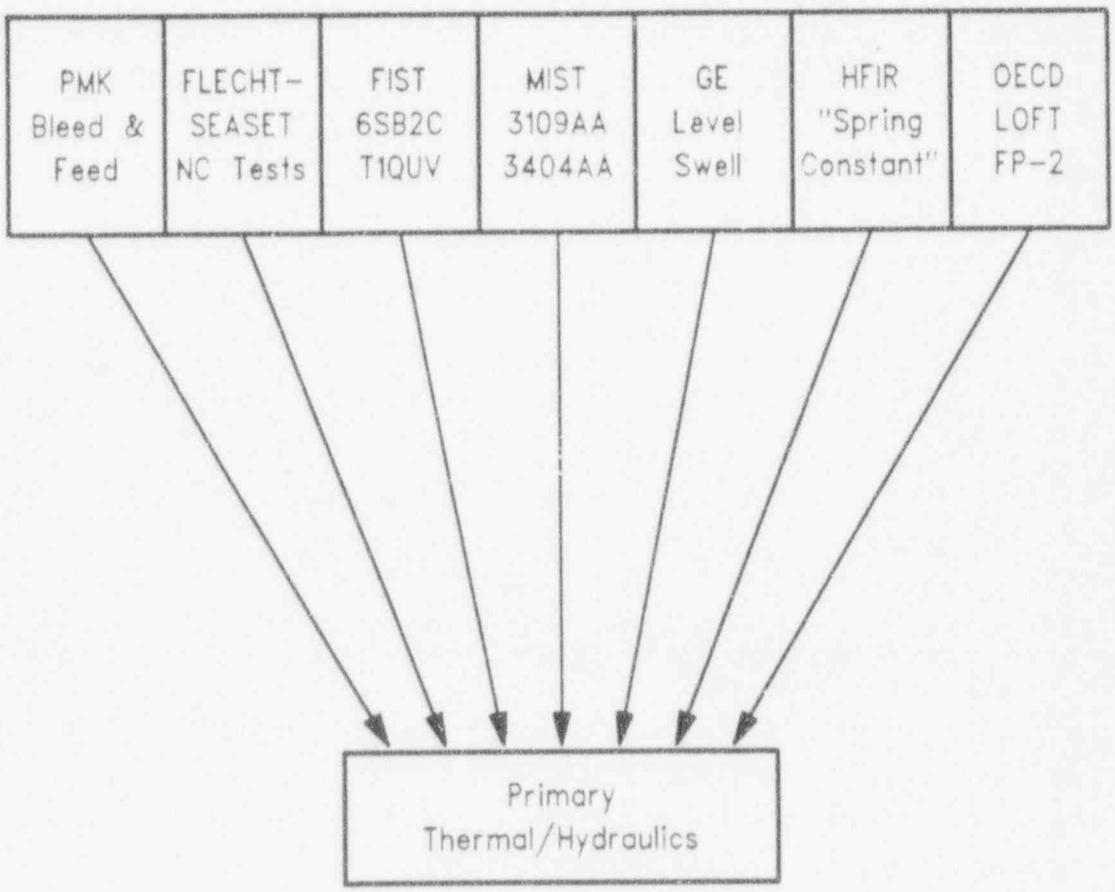
ORNL has validated MELCOR against RELAP5 results for several LBLOCA scenarios as part of their HFIR licensing analyses, with very good agreement between the two codes' results in most cases. The LOFT LP-FP-2 assessment analysis done by SNL also evaluated MELCOR's ability to calculate break flow, albeit indirectly (i.e., no break flow measurements were available for comparison); however, the generally good agreement found for primary system depressuriztion, inventory loss and core uncovery could not have been achieved without reasonable prediction of the multiple break flows in this test. The FIST assessment analyses done by ERI provided information on MELCOR's ability to model thermal/hydraulic response in the early stages of BWR severe accidents; the MIST assessment analyses done by LANL will do the same for the early stages of severe accidents in B&W PWRs. Early-time thermal/hydraulics in a BWR geometry also are evaluated in the GE level swell separate effects test modelled during the MELCOR Peer Review, and more recently using MELCOR 1.8.2, and the conclusion was that MELCOR seemed adequate in predicting most blowdown scenarios, with generally satisfactory results.

The LOFT LP-FP-2 analysis, the FLECHT SEASET simulation, the GE level swell test simulations and a gedanken test problem done for the Peer Review all showed a strong modelling sensitivity to values input for flow path opening heights in junctions connecting vertically-stacked control volumes. The results showed no conclusive pattern on the "correct" values to be used, and this remains an area of concern.

Heat transfer in heat exchangers connecting primary and secondary systems was studied in the ECN analyses of UCB tests on heat transfer degradation and steam condensation in the presence of noncondensables, and in calculations for the PMK bleed-and-feed

MELCOR
ASSESSMENT AGAINST EXPERIMENTS

EXPERIMENTS (IN-VESSEL THERMAL/HYDRAULICS)



MELCOR (IN-VESSEL THERMAL/HYDRAULICS)

Figure 30.1. MELCOR Primary System Thermal/Hydraulic Phenomena Assessment to Date

MELCOR
ASSESSMENT AGAINST EXPERIMENTS

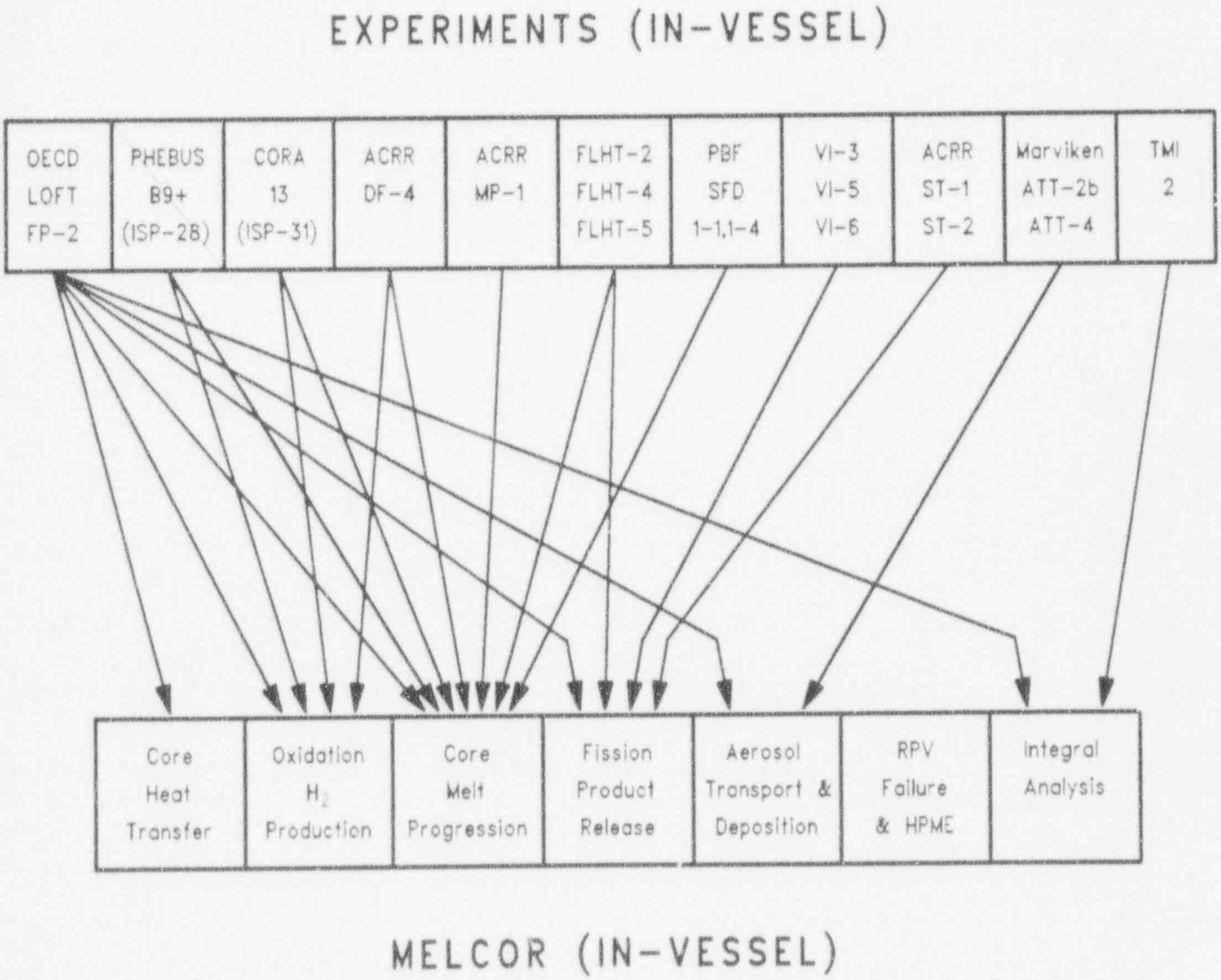
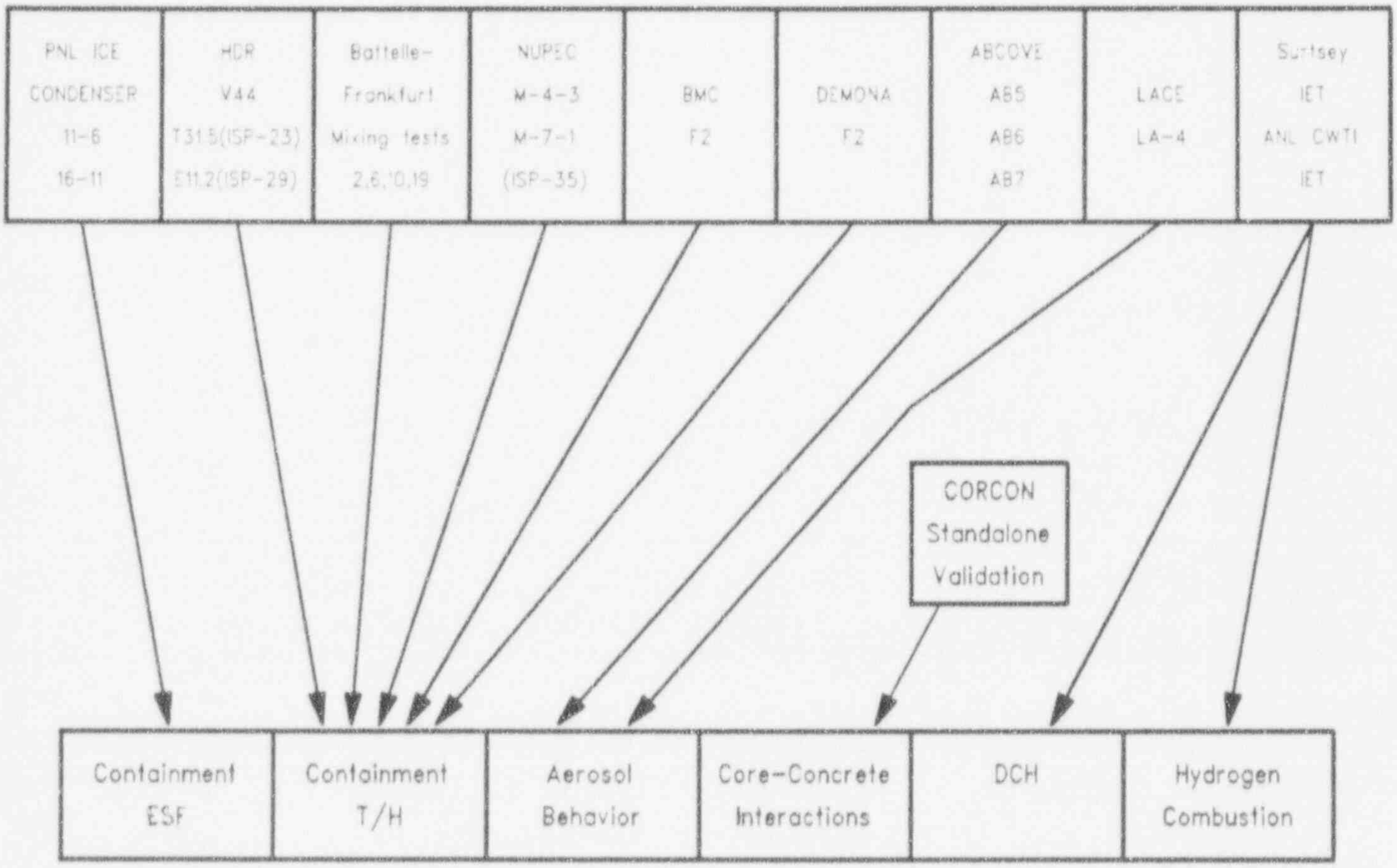


Figure 30.2. MELCOR In-Vessel Severe Accident Phenomena Assessment to Date

MELCOR
ASSESSMENT AGAINST EXPERIMENTS

EXPERIMENTS (EX-VESSEL)



MELCOR (EX-VESSEL)

Figure 30.3. MELCOR Ex-Vessel and Containment Phenomena Assessment to Date

experiments. Both these analyses included noding studies, and both demonstrated convergence to test data as the control-volume and/or heat-structure modelling detail was progressively refined (as was found in the FLECHT SEASET natural circulation analyses, also). The good agreement found when comparing to RELAP5/MOD2 results for the PMK bleed-and-feed tests and to test data demonstrate MELCOR's ability to correctly model the required thermal/hydraulic phenomena.

30.2 In-Vessel Core Damage

Core heatup, degradation and relocation mechanisms dominate the PBF SFD 1-1 and 1-4 tests and the NRU FLHT-2, FLHT-4 and FLHT-5 tests analyzed by BNL; the ACRR DF-4 early-phase core damage and MP-1/MP-2 late-phase melt progression experiments used for code assessment by Sandia; and the PHEBUS B9+ and CORA 13 core damage experiment simulations done by SNL as international standard problem submittals, for ISP-28 and ISP-31,respectively; Spain and Taiwan also submitted MELCOR calculations for the PHEBUS B9+ experiment (ISP-28). The results of all these calculations (and of the integral LOFT LP-FP-2 analysis) showed MELCOR representing the test behavior quite well in most cases, with results generally similar to those predicted by SCDAP, SCDAP/RELAP5, and other best-estimate core damage codes. Noding and time-step studies showed converging results. The major lacks noted in several of these analyses were the lack of a ballooning and/or blockage model in MELCOR, leading to mis-prediction of hydrogen generation rates and amounts. Several of these analyses found the onset of rapid metal-water reaction calculated to occur later than measured (aslo seen in many of the corresponding best-estimate code results).

30.3 Fission Product Source Term

PBF SFD 1-4 and the NRU FLHT-2 and FLHT-4 tests also study fission product release, which is the main emphasis of the ACRR ST-1/ST-2 test series analyzed by SNL and the VI tests analyzed by ORNL. The ACRR ST-1/ST-2 assessment analyses showed the new CORSOR-Booth release model producing generally less release of the volatiles than either CORSOR or CORSOR-M, and releases of more refractory species often intermediate between the releases predicted by the other two models. A major machine dependency affecting release rates calculated was identified and corrected during these analyses. Both the ACRR ST calculations and the LOFT LP-FP-2 integral analysis showed generally good agreement with data, with the standalone CORSOR code, and with best-estimate codes such as FASTGRASS or VICTORIA.

30.4 Fission Product Transport and Deposition

Aerosol transport and deposition in containment geometries were investigated in the old ABCOVE AB5, AB6 and AB7 test simulations (now being rerun with the most recent

code version as a graduate research project at University of New Mexico), and the recent LACE LA-4 experiment analysis done by SNL; fission product and aerosol transport and deposition in primary system components (upper plenum, hot leg, pressurizer and PORV piping) have been investigated in the Marviken-V ATT-2b and ATT-4 simulations done by SNL. The results show generally very good agreement with data for deposition and retention patterns; in particular, the differential retention of volatile fission products (such as Cs, I and Te) and aerosols such as Ag and Mn in the various primary system components in the ATT-4 test was calculated. The results also showed convergence with refined MAEROS aerosol size distribution resolution, and with reduced time steps. The only problems found primarily involved inconvenient input/output processing.

30.5 Containment Response

Containment thermal/hydraulic response is the emphasis of the HDR V44 steam blowdown experiment and of the four Battelle-Frankfurt hydrogen mixing tests analyzed by SNL, and the BMC-F2 experiment simulated by the UKAEA and by the Polytechnical University of Madrid; the Polytechnical University of Madrid also used MELCOR for the DEMONA F2 containment problem. Containment thermal/hydraulic response also was assessed in the HDR T31.5 steam blowdown and hydrogen mixing experiment analyzed by SNL in the ISP-23 exercise, in the HDR E11.2 steam blowdown and hydrogen mixing experiment analyzed by the UKAEA in the ISP-29 exercise, in the M-4-3 NUPEC hydrogen mixing and distribution test analyzed by NUPEC and in the M-7-1 NUPEC hydrogen mixing and distribution test analyzed by NUPEC and by TRACTEBEL in the ISP-35 exercise. The results generally indicated difficulties in accurately predicting localized, detailed thermal/hydraulic response in complex geometries, although overall, long-term behavior was generally in good agreement with data; this was identified as a general problem for any control-volume code, rather than a problem specific to MELCOR alone.

Assessment analyses have been completed for two of the PNL ice condenser tests, evaluating both temperature and ice melting predictions, and aerosol particle retention, with excellent agreement with test data and with CONTAIN calculations. This assessment produced a number of user guidelines for this new MELCOR model, and resulted in a number of coding errors being corrected before distribution to external users. Assessment analyses have also been completed for several of the IET direct containment heating experiments, with comparison to test data and to CONTAIN, with generally good results.

30.6 Plant, Integral, Calculations

Reactor coolant system thermal/hydraulic response, core heatup and degradation, and fission product and aerosol release and transport in a PWR geometry all were studied at full plant scale in the TMI-2 accident analysis, and are important in LOFT LP-FP-2.

However, there is no experiment (not even the TMI accident) which represents all features of a severe accident (*i.e.*, primary system thermal/hydraulics; in-vessel core damage; fission product and aerosol release, transport and deposition; ex-vessel core-concrete interaction; and containment thermal/hydraulics, and hydrogen transport and combustion), and only the TMI accident is at full, plant scale. It is therefore necessary for severe accident codes to supplement standard assessment against experiment (and against simple problems with analytic or otherwise obvious solutions) with plant calculations that cannot be fully verified, but that can be judged against expert opinion for reasonableness and internal self-consistency (particularly using sensitivity studies) and also can be compared to other code calculations for consistency. Table 30.6.1 summarizes the plant analyses done with MELCOR mentioned in this assessment survey report, with sensitivity studies and/or code-to-code comparisons. Again, only analyses that are completed or already underway are included; analyses scheduled but not yet begun are not included.

30.7 Identified Needs

This review of MELCOR assessment to date reveals that most of the severe accident phenomena modelled by MELCOR have received or are receiving some evaluation. However, in many of these areas, the assessment to date does not cover all phenomena of interest, or is based on a limited number of experiments and analyses which may be insufficient to cover the scale(s) of interest and which may be insufficient to allow identification of experiment-specific problems vs generic code problems and deficiencies.

There has been no assessment at all of MELCOR for ex-vessel melt phenomena such as core/concrete interactions or debris bed coolability (although the core/concrete interaction has had some "second-hand" assessment in the standalone CORCON assessment activities done), although assessment using the SURC-2 data will begin as soon as the implementation of CORCON-Mod3 in MELCOR is complete. Furthermore, there has been no assessment as yet for fission product scrubbing by pools, sprays and filters, although several analyses in this area are now in progress or planned at Sandia.

Although SNL has assessed the new ice condenser and direct containment heating models, to date there has been no assessment against test data done for hydrogen burns or for engineered safety features such as containment sprays and/or fans (except for a few limited MELCOR/HECTR plant-analysis code-to-code comparisons done some time ago). There has also been no assessment done evaluating MELCOR's capability of modelling passive containment cooling or flooded-cavity behavior, features important in new reactor designs.

Table 30.6.1. MELCOR Plant Calculations

Plant	Plant Type	Scenarios Analyzed	Codes Compared	Owner
TMI-2	B&W PWR		??	SNL
LaSalle	BWR/5, Mark II	Station blackout (SBO)		SNL
Surry	3-loop PWR	TMLB' w/ and w/o DCH	SCDAP/RELAP5,	SNL
			MELPROG/TRAC,	SNL
			CONTAIN (DCH)	SNL
		AG, S_2D, S_3D	STCP	SNL
Surry	3-loop PWR	TMLB' w/surge-line-break	SCDAP/RELAP5,	UK AEA
			CONTAIN	UK AEA
Peach Bottom	BWR/4, Mark I	Station blackout	STCP	BNL
Oconee	B&W PWR	LOCA, TMLB'	STCP	BNL
Calvert Cliffs	CE 3-loop PWR			BNL
Zion	4-loop PWR		MAAP	BNL
Peach Bottom	BWR/4, Mark I	Station blackout	BWRSAR/CONTAIN	ORNL
		LBLOCA		ORNL
Point Beach	2-loop PWR	SBO	MAAP	UWisc
Peach Bottom	BWR/4, Mark I	TQUX, AE		NUPEC
Browns Ferry	BWR/?, Mark I	S_2E SBLOCA	THALES-2, STCP	JAERI
TVO	BWR	TB, MSLBreak	MAAP	VTT
		10% SBLOCA	MAAP, SCDAP/R5	VTT
Mühleberg	BWR/4, Mark I	SBO w/ and w/o ADS, V-sequence, SBLOCA		HSK
Beznau	2-loop PWR	SBO, V-sequence, SGTR, HL SBLOCA, IBLOCA, LBLOCA	MAAP	
Gösgen	3-loop PWR	SBO		HSK
Leibstadt	BWR/6, Mark III			HSK
Ascó II	3-loop PWR	AB-and V-sequence, SGTR		Spain
Garoña	BWR/3, Mark I	SBO	MAAP	Spain

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