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1994 MCAP ANNUAL REPORT*

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Los Alamos, NM 87545**

April 1995

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1994 MCAP Annual Report

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INTRODUCTION

MELCOR is an integrated, engineering-level computer code that models the progression of severe accidents in light water reactor (LWR) nuclear power plants. 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 (BWRs) and pressurized water reactors (PWRs). Its current uses include the estimation of severe accident source terms and their sensitivities and uncertainties in a variety of applications.

Independent assessment efforts have been successfully completed by the US and international MELCOR user communities. Most of these independent assessment efforts have been conducted to support the needs and fulfill the requirements of the individual user organizations.

The resources required to perform an extensive set of model and integral code assessments are large. A prudent approach to fostering code development and maturation is to coordinate the individual assessment efforts of the MELCOR user community. While retaining individual control over assessment resources, each organization using the MELCOR code could work with the other users to broaden assessment coverage and minimize duplication. In recognition of these considerations, the US Nuclear Regulatory Commission (US NRC) has initiated the MELCOR Cooperative Assessment Program (MCAP), a vehicle for coordinating and standardizing the assessment practices of the various MELCOR users. In addition, the user community will have a forum to better communicate lessons learned regarding MELCOR applications, capabilities, and user guidelines and limitations and to provide a user community perspective on code development needs and priorities.

As a part of the MCAP activities, the Reactor Design and Analysis Group (TSA-12) at Los Alamos National Laboratory (LANL) is responsible for producing an annual summary report of MELCOR assessment activities. Eleven MELCOR assessment reports have been received by LANL in the past year and are reviewed below. Last year, all the assessments were performed by staff from Sandia National Laboratories (SNL), the developers of the MELCOR code. We are happy to report that this year three of the assessments reviewed herein were performed outside SNL. While this balance may shift in the coming years, we will probably still see the majority of MELCOR assessments coming out of SNL. As we noted last year, it is natural for the institution that has the most familiarity with the code to be at the forefront of formally assessing the code. We are encouraged by the widespread interest in MCAP, as seen in the last three coordination meetings, and look forward to reviewing assessments from throughout the MELCOR user community in the years to come.

This second Annual Report builds on the foundation laid with the first Annual Report. We have incorporated changes suggested throughout the year, and as always, we

encourage suggestions of ways we can make this report more useful to the MCAP participants. Please contact the authors if you have suggestions.

UPDATE OF MCAP ASSESSMENT MATRIX

The MCAP Assessment Matrix has been updated to reflect MELCOR assessments and plant calculations performed to date. The matrix consists of four main segments. The first segment is a new addition to the assessment matrix, which lists fundamental assessment studies. The second main segment identifies in-vessel facilities and programs and the key severe accident phenomena explored in the experiments in those facilities and programs. Assessment calculations either planned, in progress, or completed are also entered into the In-Vessel Matrix. The third main segment identifies ex-vessel facilities and programs, and the key severe accident phenomena explored in the experiments in those facilities and programs. Assessment calculations either planned, in progress, or completed are also entered into the Ex-Vessel Matrix. The fourth main segment lists plant calculations either planned, in progress, or completed by MCAP participants. The current MCAP Assessment Matrix is provided in App. A.

Even a cursory review of the MCAP Assessment Matrix reveals a key challenge that must be faced by MCAP participants. That challenge is coming to a common view of which assessments and how many must be completed to demonstrate the adequacy of the MELCOR code. We hope that the MCAP participants will find the MCAP Assessment Matrix to be a useful tool in meeting this challenge.

SUMMARIES OF MCAP ASSESSMENT REPORTS

In the following sections, we present brief summaries of 11 MELCOR assessment reports issued in the past year. The summaries that follow are based solely on the contents of the cited reports. In reading these summaries, the reader should keep in mind the time lag involved in preparing a comprehensive summary report such as this one. The assessments summarized below were conducted with various versions of MELCOR from version 1.8.1 to 1.8.3. Most, if not all, of the code errors and other problems discovered in the summarized reports were corrected in later versions of the code.

In reviewing the assessment reports for this report, we have attempted to characterize the degree to which the MELCOR-calculated results agree with the test results. To better communicate this information, we use the standard set of code assessment descriptor definitions found in App. B. The defined assessment descriptors are "excellent," "reasonable," "minimal," and "insufficient" agreement. We recommend the reader review the definitions in App. B for a fuller understanding of our assessment judgments.

MELCOR 1.8.1 Assessment: PNL Ice Condenser Aerosol Experiments

For this report,¹ two ice condenser experiments conducted in the High Bay Test Facility at Pacific Northwest Laboratory (PNL)² were modeled with MELCOR version 1.8LF. The MELCOR results were compared with the experimental data and with CONTAIN calculations of the same experiments.^{3, 4}

The experimental apparatus consisted of a 14.6 meter vertical ice column with turning vane assemblies at each end. Steam, hot air, and ZnS aerosol were injected into an inlet downcomer and homogenized in mixing chambers. After passing through two right angle bends, the mixture entered the ice column, which contained vertical flow channels in which baskets of ice were stacked. The flow stream then passed through the exit turning vane and into a scrubber. Experiment 11-6 was a low-flow test that simulated conditions after the blowdown phase of a postulated severe accident. To simulate the decreased ice inventory

that would exist under those conditions, hot air and steam were circulated through the apparatus for about 30 minutes before aerosol injection began. Experiment 16-11 was a high-flow test and did not have the pretreatment of experiment 11-6.

The MELCOR models for these tests both had a total of 10 control volumes, 3 of which modeled the ice condenser section. Flow paths connected each section in series. In addition, an extra flow path was added to the entrance and exit of each control volume in the ice condenser section. This alleviated numerical problems in the model in which liquid water from melting ice blocked gas flow. Because some backflow was observed in the low flow test, a final flow path conducted 20% of the mass and energy from a mass/energy source control function in the uppermost section of the ice condenser to a mass/energy sink control function in the lowermost section. The 20% value came from an informal sensitivity study. This recirculation flow was zeroed out in the model of the high flow experiment. Most control volumes had one heat structure to model the mass of the piping components. Each control volume in the ice condenser section had 3 heat structures, representing the ice, the horizontal structures that could absorb heat and accept aerosol deposition, and the vertical structures that could only absorb heat. The control volume representing the turning vanes at the bottom of the ice condenser had both horizontal and vertical heat structures.

The reference MELCOR models of both experiments were in reasonable agreement with experimental data for temperatures in the ice compartments and aerosol retention. Figures 1 and 2 show representative results from experiments 11-6 and 16-11. Temperatures were measured at a number of locations corresponding to the three control volumes representing the ice condenser; the calculated temperature almost always fell within the band between the highest and lowest measured temperatures at that level. In addition, the calculated times at which sharp increases in the rate of temperature rise indicated complete melting of the ice at each level were in reasonable agreement with the data.

The MELCOR results were in better agreement with the particle retention data than were the CONTAIN results, as shown in Figures 3 and 4. It was difficult for the analyst to make comparisons of calculated temperatures because of differences in nodalization between the two sets of calculations. MELCOR temperatures tended to be too high in the diffuser inlet and outlet regions, and CONTAIN temperatures tended to be too low. Unfortunately, in the ice condenser itself, where the MELCOR results were in reasonable agreement with the data, no CONTAIN temperatures were available for comparison.

Obtaining reasonable agreement of calculated results with data requires good choices of modeling parameters by the analyst. Sensitivity studies were used in this assessment activity to set many of the important parameters, so this report should be reviewed by anyone attempting to model an ice condenser. Three parameters that had significant impact on the accuracy of the model were the temperature range of ice melting, the heat transfer coefficient multiplier, and the number of nodes used in the heat structures modeling the ice.

The analyst found that setting the range of ice melting to 274–284 K gave the best match with the observed temperatures and melt times. This temperature range sets the heat of reaction for ice melting. Setting the low value to 274 avoids convergence problems known to occur in the control volume thermodynamics (CVT) package when liquid water is present and the temperature is near the freezing point (273.15 K).

A previous report⁵ recommended a value of 1.2 for the heat transfer coefficient multiplier. This gave reasonable results for experiment 16-11 (the high flow experiment), but did not melt the ice quickly enough in experiment 11-6 (the low flow experiment with the premelted ice). For that experiment, a value of 1.8 in the lowest ice condenser control volume and a value of 3.8 in the middle control volume gave the best agreement with the observed time at which all the ice melted. No adjustment was needed in the top ice

condenser control volume of experiment 11-6, because all ice was melted there before the experiment started. These higher values were within the acceptable range, compared to the value of 5.0 used by the CONTAIN model.⁵

Finally, the analyst strongly recommended using only two nodes in heat structures representing ice. The problem with using more nodes arose when the material within one node had completely melted. At that point, MELCOR replaced the ice with a low density, high heat capacity substance that tended to insulate the remaining ice and retard further melting.

Numerical Results of MELCOR 1.8.2 with DEMO Problem

For this report⁶ the authors ran a standard "DEMO" problem (developed by S. Thompson at SNL) on three different workstations and with three different versions of MELCOR. The DEMO problem simulated a station blackout sequence at a typical BWR. They examined the calculated results for numerical effects caused by differences among the workstations, different time-step sizes, different compiler options, and different code versions. Most calculations were run on IBM RISC 6000/560, HP 9000/730, and DEC 3000/500 AXP workstations. A few additional calculations were run on a Sun SPARCstation 2. The code versions used were the released versions of MELCOR 1.8.1, 1.8.2, and a special version of 1.8.2 that used double precision in all calculations.

Plots of the steam dome pressure in the pressure vessel were used to compare the results of the various calculations. Two events show clearly on these plots: an abrupt decrease in pressure at about 6,300 s due to lower head penetration and a spike at about 11,000 s caused by a hydrogen deflagration. As shown in Figure 5, the calculated times of the pressure decrease differed by as much as 2,000 s, and the timing of the pressure spike differed by as much as 1,000 s in MELCOR 1.8.1 calculations done with the HP at four levels of compiler optimization, the IBM at two levels of optimization, and the Sun SPARCstation 2. In the MELCOR 1.8.2 calculations with the same time step (Fig. 6), the pressure decrease was calculated at nearly the same time by all three workstations regardless of compiler optimization level, but there were still large differences in the timing of the pressure spike. The timing of the deflagration spike as calculated by the different workstations and compiler options tended to converge on a single value as time-step size was reduced. Unfortunately, MELCOR also tended to terminate abnormally as the time-step size decreased, especially at higher levels of compiler optimization.

Finally, the authors created fully double-precision versions of MELCOR 1.8.2 for the IBM and HP using a compile-line option to promote REAL*4 items to REAL*8 and REAL*8 to REAL*16. This option was not available on the DEC because the DEC Alpha chip has no 16-byte instructions. As Figure 7 shows, doubling the precision of the MELCOR 1.8.2 calculations improved the timing of the deflagration spike.

MELCOR 1.8.2 Assessment: The DF-4 BWR Damaged Fuel Experiment

This report⁷ assessed MELCOR modeling of the DF-4 Damaged Fuel experiment, conducted in the Annular Core Research Reactor (ACRR) at SNL.⁸ In the DF-4 experiment, a fuel bundle representing a typical BWR D-Lattice geometry was driven to the point of core material melting, oxidation, and relocation by nuclear heating from the ACRR. MELCOR version 1.8NX was used in all calculations. The MELCOR results were compared with the experimental data and with results from four other severe accident codes.⁹⁻¹²

The test bundle (Fig. 8) consisted of 14 fuel rods, arranged on 2 opposite sides of a canister or channel box, and a control blade sheath surrounding steel-clad B₄C control rods. These components were surrounded by several layers of thermal insulation. Upward steam flow to the test bundle was arranged so that 80% of the flow entered the channel (fuel) region and 20% entered the bypass (control blade) region. Test instrumentation included Pt/Rh thermocouples welded directly to the apparatus, and W/Re thermocouples protected by ceramic zirconia sheaths. Because of the thermal insulation provided by the ceramic sheaths, the temperatures reported by the sheathed thermocouples lagged the bare thermocouples by approximately 300 K and 25 s. The test instrumentation also included a hydrogen recombiner to measure hydrogen production and an optical port through which the experiment was videotaped. The final state of the test bundle was determined in post-irradiation examinations (PIE).

Overall, the thermal-hydraulic response of the model was in reasonable agreement with the data. The calculated control-blade temperature was in reasonable agreement with the observed value for most of the test (Fig 9). The worst agreement of the calculated temperature was between 6500 and 6900 s, when it was approximately 100 K low. Some of that difference may be because the thermocouples reported surface temperatures, and the model calculated bulk temperatures. The calculated temperature rose to the melting point of the B₄C-stainless steel eutectic and remained there for about 60 s as the eutectic melted. After all the B₄C was consumed, the temperature rose again to the stainless steel melting point, and the control blade collapsed. The MELCOR-calculated failure time of the control blade was about 80 s later than observed at the 36.8-cm level shown in Figure 9.

The cladding temperature calculated by MELCOR at the same level was in excellent agreement with the data for the majority of the experiment (Fig. 10). MELCOR was about 50 s late in predicting the rapid oxidation near the failure point. The cladding remained at the zircaloy melting temperature for about 50 s, then the cladding failed.

Lower in the bundle, temperatures were affected by the arrival of debris from above. A comparison of the calculated and measured cladding temperatures at the 9.6-cm axial level (Fig. 11) showed that the temperatures rose rapidly at the lower levels as debris arrived from the upper levels. The model's timing of this debris arrival and of the cladding failure was in reasonable agreement with the data.

MELCOR had mixed success in predicting the relocation of core components as the core damage progressed. Overall, the agreement would have to be characterized as minimal, with the qualification that the behavior observed in DF-4 may not be typical of what would be seen in a real nuclear power plant accident. The model successfully predicted the relocation of the stainless steel components and the presence of large amounts of unoxidized zircaloy in the debris, both in reasonable agreement with the PIE photos. However, while MELCOR relocated the fuel pellets to the debris bed, the PIE showed that most of the fuel rod pellets remained stacked after the cladding material had oxidized or melted away. The model calculation of this phenomenon, then, is insufficient. The author of the report did not recommend adjustment of the MELCOR fuel relocation parameters because it is not clear if the fuel stacking observed in DF-4 and the CORA experiments¹³ would be seen in a nuclear plant accident.

The author performed sensitivity calculations on the MELCOR parameters that affect fuel relocation. Changing the oxide thicknesses needed to hold up molten materials had only a small effect. Changing the debris formation parameters in record COR00008 so that no unoxidized zircaloy or stainless steel were required to hold up their associated components resulted in the fuel pellets remaining in place, held by a layer of oxidized zircaloy. After this change, the agreement between the calculated and observed fuel relocation was reasonable.

This change was not, however, justified by the data, which clearly showed that oxidized zircaloy did relocate. The author's final recommendation was that when modeling experiments that exhibit fuel pellet stacking one should probably adjust these parameters.

The calculated hydrogen production was compared with that measured by the hydrogen recombiner and with the extrapolated hydrogen production based on the PIE (Fig. 12). MELCOR started hydrogen production much earlier than seen in the experiment. Rapid hydrogen production began at about the same time as observed, but at a higher rate. The final total production agreed well with the amount given by the PIE. Overall, the MELCOR prediction of the hydrogen production was reasonable.

The MELCOR results for test DF-4 were compared with the calculations of APRIL.MOD3, BWR SAR/DF4, MELPROG, and SCDAP/RELAP5. MELCOR-calculated lower-control-blade temperatures early in the test were lower than those predicted by the other codes. With that exception, MELCOR agreed well with both the experiment and with the other codes.

MELCOR 1.8.2 Assessment: IET Direct Containment Heating Tests

For this report,¹⁴ a number of integral effects test (IET) direct containment heating (DCH) experiments were modeled using MELCOR version 1.8NN (version 1.8NM was the release version of 1.8.2) to evaluate the new high-pressure melt ejection (HPME) model in MELCOR's fuel dispersal interactions (FDI) package. Seven of the modeled experiments were performed in the 1:10 linear scale Surtsey test facility at SNL. Another three counterpart experiments from the 1:40 corium-water thermal interactions facility at Argonne National Laboratory (ANL) were also modeled to examine the effects of scaling. Except as noted, the model descriptions and comparisons discussed below apply to the SNL model.

In the SNL Surtsey facility, a high temperature, chemically reactive melt can be ejected by high-pressure steam into a 1:10 linear scale model of a reactor cavity. Components of the facility model the Zion reactor pressure vessel, cavity, in-core instrument tunnel, and subcompartment structures. For the DCH tests, the reactor pressure vessel was simulated by a melt generator with a cast MgO crucible lined with steel. A 4-cm hole in the bottom of the crucible, initially sealed with a fusible brass plug, simulated a hole in the pressure vessel bottom head that would be produced by ejection of an instrument guide tube. To begin an experiment, an iron-oxide/aluminum/chromium thermite mixture in the crucible was remotely ignited. When increasing pressure signaled a successful ignition, the operator failed a burst diaphragm upstream of the crucible, admitting high-pressure steam from the steam accumulator. The steam forcefully ejected the molten thermite mixture into the cavity, which connected to the instrument tunnel leading to the Surtsey vessel. The instrument tunnel entered the Surtsey vessel at an inclination of 26° from the vertical, the same angle as in the Zion reactor. In the Surtsey vessel were 1:10 linear scale models of the subcompartment structures, including the crane wall, steam generators (SGs), reactor coolant pumps (RCPs), the seal table room (with holes in its floor for the instrument guide tubes), the biological shield, the refueling canal, the radial beams and the gratings at the RCP deck, and the operating deck. The inclined portion of the instrument tunnel was sized to 2.7 times its correctly scaled length to match the scaling of the ANL apparatus as discussed below.

The design of the ANL experiment apparatus was analogous to the SNL facility, duplicating all the key features of the SNL facility at the smaller 1:40 linear scale. The inclined portion of the instrument tunnel had to be extended to 2.7 times its correctly scaled length to allow the melt crucible to fit under the expansion vessel that contained the subcompartment structures.

The MELCOR models of the two facilities were very similar, both comprising 5 control volumes, 5 flow paths, and 12 heat structures. To account for the numerous flow paths between the subcompartment structures and the dome in the facility, each with the possibility of either inflow or outflow, each model included both a straight through flow path and a recirculation flow path between its subcompartment and dome control volume. Wherever possible, model dimensions were taken from the data reports. Most of the heat structure input and some of the flow path parameters for the SNL facility was taken from the CONTAIN model.¹⁵ In many cases, the input values for the ANL model had to be scaled ($\frac{1}{4}$ for heights and lengths, $\frac{1}{16}$ for areas) from the SNL model because of incomplete dimensional data in the ANL reports.

A valve with a time-dependent area controlled the rate of steam injection in the model. The valve opened at a constant rate between 0 and 1 s, then varied with time to match the observed steam accumulator depressurizations in the tests (the data indicated occasional debris plugging of the steam/thermite reaction product flow path). Although there was some variation from test to test, the valve model was kept the same for all SNL tests modeled.

While the debris and steam were injected together in the experiments, the debris in the HPME model is inserted directly at its final destination. The debris composition, temperature, amount, and distribution were input using the FDI HPME stand-alone mode. Aluminum and alumina had to be added to the allowable input materials for the HPME model for this study. This code modification is in MELCOR version 1.8MY and later, and is thus in the release version of 1.8.2 (1.8NM). The HPME model allows distribution into control volume atmospheres, onto surfaces, and into CORCON cavities. For this study all debris was distributed into control volume atmospheres and allowed to settle. A table was used to input the cumulative debris injection as a function of time, attempting to match the rate of pressure and temperature increase in the vessel. The debris input was kept the same for all experiments, again using one set of reasonable values to model all SNL tests even though the debris measured in the different experiments varied, in part due to partial melting of the crucible liner and brass plug.

The HPME model input includes characteristic times for debris settling, oxidation and heat transfer in atmospheres, and oxidation on surfaces. The debris-settling characteristic time input for the model was based on free-fall times for the various volume heights. The other characteristic times were selected using an iterative procedure. These characteristic times gave reasonable agreement with measured vessel pressures, subcompartment temperatures, and hydrogen production and combustion data. The report's author noted that the selected values are not necessarily the best set of values for representing the data. The iterative procedure found one possible set of characteristic times that gave reasonable agreement, but that set was selected for purely mathematical reasons (maximizing agreement with the data) and may not represent the actual physical processes involved.

The burn package was activated in most of the calculations, with burning suppressed in all control volumes except the vessel dome. This agreed with the experimental observation of a jet of flame at the outlet from the subcompartment to the dome. Combustion parameters were set to prevent the burning of preexisting hydrogen, only allowing hydrogen generated during the HPME to burn. These nonstandard burn parameters would not be appropriate for plant analyses before and after the HPME period, so changes were made in the MELCOR input allowing one set of burn parameters during HPME and another set at other times. This input change was made in MELCOR version 1.8NN and does not appear in the release version 1.8.2.

A final adjustment was made to the control volume flow areas, reducing them by approximately a factor of 10 from their default values. These areas are only used by MELCOR to calculate the volume velocities for use in calculating the convective heat transfer coefficient from the atmosphere to heat structure surfaces, and do not affect the flow path calculations at all. The analyst needed to enhance the heat transfer from the atmosphere to the surfaces to compensate for two effects. First, with the hot debris added to their final destinations instead of being transported with the steam, it tended to heat the downstream volumes and decrease the differential pressure driving the flow. If the debris were carried with the flowing steam, it would increase the upstream pressure and increase the differential pressure driving force. Second, MELCOR does not model radiation heat transport from the debris to the heat structures. Because hand calculations indicated this heat transfer mode was important, it was feared that too much energy would instead be transferred from the debris to the atmosphere. Increasing the flow, and consequently the convective heat transfer coefficient, allowed this excess energy to be carried to the heat structure, as the debris-to-heat-structure radiation would have if it were modeled. The capability to add an optional multiplier for volume velocities (used only during the HPME period) was added to MELCOR version 1.8NN and does not appear in the release version 1.8.2.

The model as described above gave reasonable agreement with the experimental data for pressures, temperatures, and hydrogen generation and combustion. As Fig. 13 shows, MELCOR correctly calculated the division of the pressure response into two families based on the presence or absence of hydrogen combustion, giving a peak pressure rise of about 100 kPa due to HPME and an additional pressure rise of about 150 kPa due to hydrogen combustion. The results also correctly calculated the lack of any significant effect caused by either preexisting hydrogen or basement condensate water.

In many cases, the analyst was only able to make a qualitative comparison of predicted and observed hydrogen production and combustion. In reducing their data, the experimenters assumed that all oxygen depletion was due to reaction with hydrogen, and that debris reacted only with steam, not with free oxygen. MELCOR, on the other hand, assumes that metals preferentially react with free oxygen before reacting with steam. The assessment report dealt with this discrepancy by presenting two sets of calculated hydrogen production and combustion values for each test modeled, both the values calculated by MELCOR, and the amounts that would be calculated using the initial and final amounts of oxygen and hydrogen from the MELCOR calculations in the same formulas used by the experimenters. Table 1 compares these two sets of MELCOR production and combustion with the data. In the tests that had significant amounts of free oxygen present, the two sets of MELCOR values for hydrogen production and consumption tended to bracket the experimental data. The report suggested the likelihood that the true answers lie somewhere between the extremes considered. While it is probably unlikely that no debris reacted with oxygen, MELCOR's treatment of the debris, inserting it directly at its final destination, exposed it to more oxygen than it would have seen had it been transported with the blowdown stream. In that case it probably would have arrived at its final destination later than the first arrival of the steam/hydrogen mixture carrying it.

The reference MELCOR calculations were compared with CONTAIN analyses of some of the same tests, as presented at the CONTAIN Peer Review.¹⁶ Despite the differences between the two codes (CONTAIN's DCH model is more detailed and mechanistic than MELCOR's) the results obtained were quite similar, both predicting a smaller pressure rise (on the order of 100 kPa) in cases with no hydrogen combustion, and a higher pressure rise (~200–250 kPa) with hydrogen combustion.

TABLE 1
Hydrogen Production and Combustion, IET DCH Tests

Experiment	Hydrogen (gm-moles)					
	Produced			Burned		
	Data ^a	MELCOR ^b	MELCOR ^c	Data ^a	MELCOR ^b	MELCOR ^c
IET-1	233	286	266	3	0	4
IET-1R	248	266	267	11	0	28
IET-3	227	232	352	190	188	313
IET-4	303	243	361	240	209	332
IET-5	319	240	313	53	0	91
IET-6	319	236	354	345	218	307
IET-7	274	229	351	323	223	350

a from gas grab bottle samples at 30 min

b actual end-of-calculation values at 20 s

c calculated from initial and final oxygen and hydrogen moles assuming only steam/metal reactions

MELCOR 1.8.2 Independent Assessment: MIST SBLOCA Test 3109AA

For this report,¹⁷ Multi-Loop Integral System Test (MIST) test 3109AA was modeled with MELCOR 1.8.2 (release version). The MIST facility, designed to simulate small-break loss-of-coolant accidents (SBLOCA) in Babcock & Wilcox (B&W) type PWRs, is a scaled facility with two hot legs and four cold legs. It was specifically designed to incorporate unique features of the B&W design, in particular the hot-leg U-bends and the once-through steam generators. MELCOR results were compared with the data¹⁸ and with TRAC¹⁹ and RELAP5¹⁸ calculations.

Test 3109AA started with the primary system in single-phase natural circulation, with the guard heaters in automatic operation, the reactor vessel vent valves (RVVV) manually closed, and the core power at 3.9% of scaled full power. The initiating event of the transient was a scaled 10-cm² break in the B1 cold-leg pump discharge. The steady-state control functions were maintained until the pressurizer level dropped to 0.305 m, at which time the steam generator secondary level control set point was increased to 9.63 m, full high-pressure injection flow was activated, core-power decay was started, RVVV automatic control was started, and abnormal transient operator guidelines (ATOG)-based steam generator secondary pressure control began.

The transient for test 3109AA can be divided into four major phases: subcooled decompression, intermittent circulation, loop stagnation, and refill. For the first 185 s, the primary was in single-phase natural circulation, the liquid was subcooled, and the pressure decreased rapidly as the liquid expanded through the leak. After saturation pressure was reached, the intermittent circulation phase began, wherein the natural circulation flow in a loop was governed by the hot-leg liquid level. Loop stagnation occurred after the final spillover in the broken loop. In this phase, the primary was cooled by feed and bleed from the high-pressure injection system to the leak, and by auxiliary feed water in the broken-loop steam generator.

The MELCOR model was based on boundary conditions and geometrical input for a TRAC model developed at LANL,¹⁹ but with coarser noding consistent with MELCOR modeling practice. The model consisted of 40 control volumes and 20 heat structures. The

steam generator secondary side pressure, the auxiliary feed water injection rates, and the high-pressure injection rate were all specified as functions of time only. As recommended in the FLECHT-SEASET natural circulation assessment,²⁰ finer noding was used in the primary side of the steam generator to better capture the natural circulation phenomena. However, sensitivity studies in this assessment showed that coarse noding of the steam generator primary (using only one control volume rather than four) gave adequate modeling of the natural-circulation and heat transfer phenomena in test 3109AA.

The model was in reasonable agreement with the data during the steady state and for the first 200 s of the transient, which was mostly single-phase. Table 2 shows the timing of key events in the experiment and in the MELCOR, TRAC, and RELAP5 models. MELCOR calculated both the time at which the A loop reached saturation temperature and the time at which natural circulation ceased in reasonable agreement with the data. The majority of the important parameters, including break flow (Fig. 14), primary system heatup, natural and intermittent circulation, loop stagnation, and refill (Table 2), were calculated with reasonable agreement. The calculated primary pressure (Fig. 15), however, was in minimal agreement with the data. Of the two boiler-condenser mode (BCM) events observed in the test, MELCOR successfully calculated the occurrences (Table 2), but it calculated the wrong timing and magnitude for the first event, and the wrong duration for the second. The liquid level in the A loop hot leg was also calculated with minimal agreement (Fig. 16).

The MELCOR results did not match the data as well as did the TRAC or RELAP5 calculations. This is the expected result, since MELCOR is a Tier 1 code with zeroth order models. Both TRAC and RELAP5 contain higher order models, and would be in Tier 2 if they were severe accident codes. There were, though, certain values that were calculated best by MELCOR. TRAC's calculated primary pressure (Fig. 17) was in better agreement with the data than either the MELCOR or RELAP5 primary pressure calculations, which were quite similar to each other. TRAC calculated the timing of the first BCM event better than either of the other two codes, while MELCOR came closest to the observed timing of the second BCM event (Table 2). The TRAC-calculated liquid levels in the hot legs and in the pressurizer were generally better than those calculated by either RELAP5 or MELCOR.

As new users outside of the MELCOR development group, the report's authors discovered some areas where the MELCOR user guidelines, output, and plotting support software could be improved, as well as possible code errors. They recommended including more examples and providing more detailed information about how to make the transition from a steady-state to a transient calculation. Other areas where the authors felt the guidelines could be improved were in explaining the interactions and perils involved in connecting time-specified components in series, and in explaining how to input power into a heat structure. Some other specific changes to improve the wording in the guidelines were also suggested. The authors felt that the HISPLT plotting routine should be given capabilities to plot (1) heat structure power curves (2) computer time used by each model component, (3) heat transfer and flow regimes, and (4) valve areas to automatically number plots, and to sum variables such as flow rates. The most significant code error reported was that, in some cases, the flows into and out of a liquid-full control volume were not equal.

TABLE 2
Timing of Key Events in MIST Test 3109AA

Event	Test (s)	MELCOR (s)	TRAC (s)	RELAP5 (s)
Break initiation	0.0	0.0	0.0	0.0
Core power decay, HIP and RVVV automatic control initiated, pressurizer level dropped to 0.305 m, and SG-secondary level reset	150.0	150.0 ^a	160.0	54.0
Intact-loop hot leg saturated	185.0	175.0	185.0	b
Intact-loop flow ended.	240.0	260.0	310.0	150.0
BCM began in intact-loop SG as SG-secondary level settled in on set point	565.0	1350.0	565.0	498.0
RVVVs closed in response to draining in vessel and downcomer.	1150.0	1150.0 ^c	1480.0	b
RVVVs opened, allowing core-generated steam to pass from vessel to downcomer. Repressurization ended as steam passing through RVVVs contacted cold HPI water in downcomer and condensed. Intermittent broken-loop flow ended as steam no longer pushed liquid over hot-leg U-bend.	1575.0	RVVV opened at 1575 s. ^c Intermittent broken-loop flow ended at 1700 s.	1840.0	1050.0
SG secondary pressure set point decreased to intact-loop SG secondary pressure. Steaming of SG secondary began, AFW came on to maintain level, BCM heat-transfer event resulted. Primary depressurization rate increased. HPI flow rate exceeded leak flow rate for remainder of transient.	4000.0	BCM was computed at 4100 s. HPI flow rate exceeded leak flow rate at 3700 s.	3400.0	2826.0
Calculation terminated	—	7000.0	7000.0	4002.0

^a The MELCOR pressurizer level did not drop to the specified level at 150 s, but Test 3109AA actions at 150 s were modeled into the input deck.

^b Not included in Ref. 18.

^c RVVV opened based on area-vs-time table.

MELCOR 1.8.2 Independent Assessment: MIST SBLOCA Test 3404AA

For this report,²¹ MIST test 3404AA was modeled with MELCOR 1.8.2 (release version). Test 3404AA was a simulated break of 10 tubes in the B Loop SG. It differed from test 3109AA (assessment summarized above) in the break location and in the steady-state primary pressure (11.99 MPa in 3109AA, 14.79 in 3104AA). The MELCOR calculation was compared with the data¹⁸ and with results calculated with TRAC.²²

Test 3404AA started with the primary system in single-phase natural circulation, with the guard heaters in automatic operation, the reactor vessel vent valves (RVVV) manually closed, and the core power at 3.9% of scaled full power. The transient was initiated by opening a connection between the B SG inlet plenum and the secondary side just below the upper tube sheet, transferring the RVVVs to automatic control, and actuating the core-power decay ramp. A second set of control actions was to be taken as the subcooling dropped below 28.7 K. These actions were delayed by almost 2 min in the test, but did take place before the system reached saturation. These actions included actuating the high-pressure injection system, raising the control point for the secondary side level in the intact-loop SG to 9.63 m (the 95% point), and initiating a 55.56 K/h secondary cooldown in both SGs. Another operator-initiated control action was to open the low-flow-rate steam control valve in the broken-loop SG when the level in that SG reached 6.096 m and was still rising, to prevent overfill. The SG secondary side pressure was automatically controlled to mimic the actions specified in the ATOG. When the primary pressure dropped to 6.55 MPa, the operator isolated the broken-loop SG secondary. The operator did not open the broken-loop SG high flow rate steam pressure control valve as planned when the pressure in that SG exceeded 8.96 MPa. When that pressure rose to 9.48 MPa, the test was aborted to avoid blowing the secondary safety release valves.

The transient for test 3404AA can be divided into three major phases: continuous natural circulation, depressurization to isolation, and refill. The natural-circulation phase lasted until the initial interruption of the broken loop at 220 s. The second phase lasted until about 800 s, as the primary was depressurized rapidly in response to several mechanisms. The refill phase lasted from the isolation of the broken loop SG to the end of the test.

The MELCOR model for test 3404AA was developed from the model for test 3109AA described above, changing the break location and the initial and boundary conditions. Again, the fine noding of the primary-side SG recommended in the FLECHT-SEASET natural circulation assessment²⁰ was used, and for test 3404AA, as in 3109AA, this fine noding proved not to be necessary.

In contrast to the model of test 3109AA, the model of test 3404AA was beset with code failures. The analyst had to make 35 different calculations to arrive at an input deck that would run to the end of the test at 2700 s. To get the code to run the problem, the analyst had to change the k-factor at the break from one that matched the observed pressure profile within 2 to 5% in the first 800 s, to one that only matched within 5 to 10%, and simultaneously had to change the high-pressure injection rate to the upper bound of the experimental uncertainty range.

With the above changes made to obtain a stable calculation, MELCOR was able to calculate the steady state with excellent agreement. For the transient calculation, seven variables were chosen as key parameters: vessel (Fig. 18) and hot-leg (Fig. 19) levels, primary pressure (Fig. 20), and the break (Fig. 21), intact-loop (Fig. 22), and broken-loop (Fig. 23) flow rates. MELCOR calculated reasonable values for all seven key parameters. Two BCM events were observed in the test, and while MELCOR calculated the correct pressure drop due to BCM cooling and the correct BCM duration, it calculated incorrect

initiation times for both events. It would appear from this result that MELCOR's models for evaporation/condensation and BCM need improvement.

The TRAC model also gave overall reasonable agreement with the data. In some cases, the MELCOR results were in better agreement with the data than were the TRAC results, for instance in the primary pressure from 0 to 1000 s and from 2000 to 2700 s (Fig. 20). The TRAC-calculated times for the two BCM events were in much better agreement with the data than were the MELCOR results. The data indicated that on at least 30 occasions, break flow reversed, going from the secondary side to the primary side. Neither code calculated this break flow.

The analyst repeated the same suggestions for code and documentation improvement reported in the 3109AA assessment, and added three more to the list. In addition to the lack of code robustness already noted, the facility to select interactive restarts can cause the user to lose data. If the user answers "no" to the prompt for restarting from the final dump, the code cycles back to the first dump, but no longer informs the user which dump is being offered. Finally, the message sent when the code fails sometimes fails to inform the user what condition caused the failure.

Air Ingression Calculations for Selected Plant Transients Using MELCOR

The effects of air ingression on the consequences of various severe accident were studied in two sets of MELCOR calculations.²³ Both sets of calculations modeled the Surry plant, a three-loop Westinghouse PWR. One set of calculations analyzed a station blackout with surge line failure prior to vessel breach, starting from nominal operating conditions using a modified version of SNL's Surry TMLB' input deck.²⁴ The other set of calculations analyzed a station blackout occurring during shutdown (refueling) conditions. The input deck for this set of calculations was modified from a mid-loop configuration Surry deck developed at Brookhaven National Laboratory (BNL)²⁵ from SNL's Surry TMLB' deck. For both accident scenarios, a basecase calculation was done. The calculation was then repeated, modeling air ingression from containment into the core region following core degradation and vessel failure. These calculations were performed with a version of MELCOR, based on version 1.8.2, in which ruthenium release rate was enhanced by making it explicitly dependent on the partial pressure of oxygen. This enhanced ruthenium release rate does not appear in the release version of 1.8.2 (version 1.8NM), it was first introduced into the production code in version 1.8NX.

In addition to the two sets of analyses done for this report, a similar air-ingression sensitivity study was done as part of a low-power/shutdown probabilistic risk assessment (PRA) study.²⁶ That PRA study analyzed a station blackout occurring during shutdown (refueling) conditions at a different plant, the Grand Gulf plant, a BWR/6 with Mark III containment. While the Grand Gulf calculations will not be formally documented until the completion of the PRA program, the NRC contract monitor for that program agreed to permit a brief summary of the Grand Gulf calculations in the subject report.

All three studies lead to the same conclusions. The ruthenium release in-vessel increased from only trace amounts with no air ingestion to 50–80% of initial inventory with moderate air ingression rates of 10–100 mole/s. There was only 10–20% increase in clad oxidation degree and energy because most of the oxygen calculated to enter the core region escaped before it was consumed.

The enhanced ruthenium release with air ingression was expected because of the addition discussed above of enhanced ruthenium release rate to the MELCOR fission product model. The relatively small changes in core temperatures, hydrogen production, steam

consumption, and oxidation energy were not expected. The greater oxidation energy, due to reaction of Zircaloy with oxygen, did cause core temperatures to rise more quickly than for oxidation only with steam. Those higher temperatures then caused the remainder of the core material to melt, relocate and be lost to the cavity sooner than predicted with no air ingress into the core. Oxidation of Zircaloy with air rather than with steam, for the relatively short period of time that the clad remained in-vessel, did not significantly affect the overall steam consumption, hydrogen production, and total oxidation energy, because these were dominated by the long-term oxidation of structural stainless steel in the core and especially in the lower plenum.

The assumed air ingress did not significantly affect most of the accident scenario. There was very little change in the release of the volatile species, *i.e.*, noble gases, Cs, I and Te, which are released at lower temperatures ($T \leq 2000\text{K}$); most of their initial inventory was released before vessel breach and air ingress. In most cases, there was 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. There was usually a decrease in the release of refractory species, *i.e.*, Ce and U, which were released only at very high temperatures ($T > 2500\text{-}3000\text{K}$), possibly due to the cooling effect of ingesting relatively cool containment air into the core region.

The predicted changes in radionuclide release reflected only the effects of air ingress changing the calculated core temperatures and relocation history. If other species are also sensitive to the oxygen potential, there may be other changes in fission product release with air ingress; however, any such additional effects air could have on the release and transport characteristics of other radionuclides were not modeled in MELCOR for these initial calculations.

These calculations demonstrate the potential of air ingress 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.

MELCOR 1.8.2 Assessment: Surry PWR TMLB' (with a DCH Study)

This report²⁴ presents the results of a series of calculations of a station blackout (TMLB') accident at Virginia Electric Power Company's Surry Power Station, Unit 1. The nuclear reactor of Surry Unit 1 is a 2441 MWt three-loop Westinghouse design. Earlier, this accident was calculated as a demonstration for the MELCOR Peer Review process using MELCOR version 1.8.1.²⁷ That earlier calculation has been used as a standard test problem to investigate problems identified by the Peer Review. The calculations reported here allow direct comparison with the earlier calculation, showing the impact of model improvements, model extensions, and new models in version 1.8.2. Most of the new calculations reported here were performed with MELCOR version 1.8NM, the release version of MELCOR 1.8.2. The single exception is the sensitivity study of the new direct containment heating (DCH) model incorporated into MELCOR version 1.8NN, developed as a result of Sandia's assessment analysis of the IET DCH experiments.¹⁴

The MELCOR model for these calculations was developed from a MELCOR input deck developed by Idaho National Engineering Laboratory (INEL),²⁸⁻³⁰ which was itself based on a SCDAP/RELAP5 input deck from INEL.³¹ The model has 21 control volumes (6 for the reactor vessel and internals, 4 for the primary system loops, 4 for the steam generator secondaries, 6 for the containment, and 1 for the environment); 33 flow paths (13 internal to the reactor coolant system (RCS), 7 associated with the steam generator secondary side,

10 internal to the containment, and 3 connecting the RCS to the containment); and 88 heat structures (64 for the RCS and 24 for the containment). All control volumes were specified to be vertically oriented volumes using nonequilibrium thermodynamics. All heat structures used the steady-state temperature self-initialization option. The control volumes were specified so that the highest and lowest points in the piping (at the top of the U-tubes and the bottom of the pump loop seal) occurred at the boundaries between control volumes. This was done to allow resolution of the vertical differentials driving natural-circulation flow.

No input changes were required to run the MELCOR 1.8.1 input model on MELCOR 1.8.2. Input changes were made, however, to take advantage of new and upgraded models. These included using step functions in valve area tables and reducing surge-line flow-path interfacial momentum exchange length. The new eutectics model, which is not used by default, was not enabled for the comparison of 1.8.2 with 1.8.1. The new debris radial relocation model is enabled by default, so no input change was required.

Table 3 compares the timing of key events in the MELCOR 1.8.2 and MELCOR 1.8.1 models. The early-time behavior of the 1.8.2 model was generally very similar to 1.8.1 for the steam generator secondary inventory boiloff, pressurizer filling and venting, core uncover, and initial clad failure and gap release. MELCOR 1.8.2 calculated vessel failure approximately one hour earlier than 1.8.1. About half that difference was due to correcting the "levitating water" problem diagnosed and corrected in the LOFT LP-FP-2 assessment.³² The other half was due to the correction in 1.8.2 of the code logic in 1.8.1 that prevented failure of a blocked component (the core plate at ring 1 in this case) until the blockage material reached its melting point.

TABLE 3
Timing of Key Events, MELCOR 1.8.1 and 1.8.2 Calculations of Surry
PWR TMLB'

Event	Time	
	MELCOR 1.8.2	MELCOR 1.8.1
Gap Release		
Ring 1	10,226.4 s (2.84 hr)	9,998.9 s (2.78 hr)
Ring 2	10,336.4 s (2.87 hr)	10,095.6 s (2.80 hr)
Ring 3	10,666.4 s (2.96 hr)	10,418.9 s (2.89 hr)
Core Plate Fails		
Ring 1	11,663.2 s (3.24 hr)	11,260.9 s (3.13 hr)
Ring 2	12,754.4 s (3.27 hr)	11,961.7 s (3.32 hr)
Ring 3	12,246.4 s (3.40 hr)	13,970.5 s (3.88 hr)
LH Penetration Fails		
Ring 1	11,708.8 s (3.25 hr)	14,998.0 s (4.17 hr)
Ring 2	11,792.7 s (3.28 hr)	15,698.4 s (4.36 hr)
Ring 3	14,182.9 s (3.94 hr)	17,715.1 s (4.92 hr)
Debris to Cavity	11,786.6 s (3.27 hr)	15,021.4 s (4.17 hr)
Deflagrations Start	12,703.2 s (3.53 hr)	—
Deflagrations End	14,413.7 s (4.00 hr)	—
CORCON layer flip	~23,000 s (6.4 hr)	~27,000 s (7.5 hr)
Deflagrations Start	32,222.7 s (8.95 hr)	46,622.9 s (12.95 hr)
Deflagrations End	36,992.3 s (10.28 hr)	49,751.5 s (13.82 hr)

As Table 4 shows, there was a significant difference in the core state at the time of vessel failure in the two calculations. Much of the difference can be attributed to the new radial debris relocation model in MELCOR 1.8.2. In both models, the core plate supporting the middle, highest-power ring failed first. The mass of debris in the lower plenum calculated by MELCOR 1.8.1 corresponds well with the mass of material initially present in the inner ring of the core, while in the MELCOR 1.8.2 calculation about half the initial mass of the core region was relocated to the lower plenum before bottom head failure. In the MELCOR 1.8.2 calculation, most of the material in the core region was "intact," either still in its original position or frozen into place on intact components. Most of the debris in the MELCOR 1.8.2 calculation was relocated radially into the middle ring by the radial relocation model, and thence into the lower plenum. In contrast, most of the material in the core region of the MELCOR 1.8.1 calculation at vessel failure was particulate debris. This

TABLE 4
Core State at Vessel Failure, MELCOR 1.8.1 and 1.8.2 Calculations of
Surry PWR TMLB'

	MELCOR 1.8.2		MELCOR 1.8.1	
	Intact	Debris	Intact	Debris
Active Fuel Region Masses (kg)				
UO ₂	11,430	6,748	5,553	62,196
Zircaloy	6,648	1,299	1,444	9,227
Zirc Oxide	2,249	729	280	3,296
Steel	1,485	20	298	18
Steel Oxide	65	24	95	11
CRP	993	340	884	
Total	22,870	9,160	11,074	74,748
Core Plate Masses (kg)				
UO ₂		1,196		31
Zircaloy		165		153
Zirc Oxide		40		57
Steel	1,815	3	1,525	18
Steel Oxide				
CRP				754
Total	1,815	1,404	1,525	1,013
Lower Plenum Masses (kg)				
UO ₂		62,268		12,853
Zircaloy		4,164		2,035
Zirc Oxide		2,126		707
Steel	32,675	175	32,990	
Steel Oxide	5	108	31	
CRP		672		368
Total	32,661	69,514	33,021	15,963
Average Debris Temperature (K)				
		~2175		~2850
Fraction Oxidized				
	Zircaloy ~24%	Steel ~0.4%	Zircaloy ~20%	Steel ~0.3%

"stacking" phenomenon has been seen in many other MELCOR 1.8.1 calculations, with debris stacking up in columns and not migrating radially.

Significantly more hydrogen was burned in the cavity in the MELCOR 1.8.2 calculation than in the MELCOR 1.8.1 calculation (Fig. 24). As shown in Figure 25, the masses of hydrogen burned in the two calculations are roughly proportional to the number of hydrogen burns calculated. These differences are likely due to numerical effects discovered in the burn package of both code versions. Figure 26 compares the hydrogen production calculated by the MELCOR 1.8.1 and MELCOR 1.8.2 models. MELCOR 1.8.2 predicted slightly less hydrogen production in the cavity and more in the core than predicted by MELCOR 1.8.1, but the total production calculated by the two models agreed within about 5%.

The MELCOR peer review²⁷ raised some questions about the failure of the pressurizer to drain in the MELCOR 1.8.1 calculation until the vessel depressurized after it failed, arguing that this result was counterintuitive and might reflect a code problem. Members of the peer review committee felt the failure of the pressurizer to drain might result from an inadequacy of the MELCOR momentum exchange model, which would lead to an incorrect two-phase countercurrent flow limit (CCFL). Because of this problem, a change was made in MELCOR that treats the interfacial momentum exchange length as a separate variable from the inertial length. The default value for the exchange length is the buoyancy force characteristic dimension, but the user can override that value if desired. A sensitivity study (discussed in the Sensitivity Studies section below) showed that the new input option does allow a wide variation in CCFL and associated pressurizer drainage rates, but was inconclusive as to the correct value of the parameter.

In addition to the calculation discussed above that directly compared the MELCOR 1.8.1 and MELCOR 1.8.2 calculations, a comprehensive set of calculations was run with MELCOR 1.8.2, examining the effects of a number of input parameters to arrive at a reference calculation that best modeled the accident. In comparing the MELCOR 1.8.1 and MELCOR 1.8.2 calculations, the analyst made the two input models as similar as possible to gauge the changes wrought by the new code version, leaving many new models out of the MELCOR 1.8.2 calculation. The MELCOR 1.8.2 reference model added the eutectics model, HPME debris distribution, and DCH interactions to the MELCOR 1.8.2 model compared to MELCOR 1.8.1. Table 5 shows the timing of key events in the MELCOR 1.8.2 reference model.

The eutectic model generally had only a small effect on the calculations. Earlier core support plate failure and earlier lower head penetration were calculated with the eutectic model enabled, but the difference was less than 5%. The biggest difference was in the lower plenum structure. Without the eutectic model, about 80% of the steel structure in the lower plenum melted and fell into the cavity. With the eutectic interaction modeled, Zr and stainless steel debris in the lower plenum melted at lower temperatures and flowed to the cavity sooner. The combination of lower debris residence time and less heating of the steel structure led to a higher retention of the lower plenum steel structure (~70%). The smaller amount of steel transferred to the cavity with the eutectic model led to decreased concrete ablation and lower containment pressure at late times.

The increased steel retention in the lower plenum exposed a weakness in the VANESA code, which calculates ex-vessel releases in MELCOR. The VANESA code has no provision for a disappearing metallic layer. As the metallic layer in the cavity approaches zero, the release of radionuclides from that layer increases exponentially. This is a known problem with the VANESA formulation. Such problems are more likely to occur with MELCOR 1.8.2 than with MELCOR 1.8.1 both because of the higher structure retention

TABLE 5
Timing of Key Events, MELCOR 1.8.2 Reference Calculation of Surry PWR TMLB'

Event	Time
Loss of Electrical Power	0
Pump Coastdown Complete	~300 s (5 min)
SG Secondary SRV First Open	0
SG Secondary Side Dried Out	~5000 s (1.4 hr)
Pressurizer PORV First Open	~5250 s (1.5 hr)
TAF (Top-of-Active-Fuel) Core Uncovery	~7200 s (2 hr)
Natural Circulation Stopped	~7850 s (2.2 hr)
Start of Zr Oxidation/H ₂ Production	~10,000 s (2.8 hr)
Gap Release	
Ring 1	10,235.0 s (2.84 hr)
Ring 2	10,334.4 s (2.87 hr)
Ring 3	10,667.9 s (2.96 hr)
Core Support Plate Fails	
Ring 1	11,177.9 s (3.10 hr)
Ring 2	11,906.3 s (3.31 hr)
Ring 3	13,062.9 s (3.63 hr)
Lower Head Penetration Fails	
Ring 1	11,219.3 s (3.12 hr)
Ring 2	13,029.5 s (3.62 hr)
Ring 3	13,842.3 s (3.85 hr)
Start of Debris Ejection to Cavity	11,219.3 s (3.12 hr)
HPME/DCH Starts	11,219.3 s (3.12 hr)
HPME/DCH Ends	11,251.7 s (3.13 hr)
Intermittent H ₂ Deflagrations Start in Cavity	12,863.2 s (3.57 hr)
Intermittent H ₂ Deflagrations Stop in Cavity	13,766.2 s (3.82 hr)
CORCON Layer Flip	~24,000 s (6.7 hr)
Cavity Dried Out	~28,200 s (7.8 hr)
Intermittent H ₂ Deflagrations Start in Cavity	32,970.0 s (9.16 hr)
Intermittent H ₂ Deflagrations Stop in Cavity	14,778.9 s (12.44 hr)
Calculation Ended	90,000.0 s (25 hr)

due to the eutectic model seen in these calculations and because the greater robustness of MELCOR 1.8.2 allows calculations to be carried to later times. Users are cautioned to carefully examine radionuclide releases when vanishing metallic layers are predicted in CORCON.

The modeling of DCH in the Surry TMLB' analysis drew on the previous experience modeling DCH in the IET experiments.¹⁴ The calculations showed brief pressure and temperature spikes when HPME and DCH began, as shown in Figure 27. Only about 15% of the available core material was predicted to be ejected during HPME in the reference calculation, so the effect was not extreme. Sensitivity studies described below examined the effects of increasing HPME to 60% of available core material, of turning the HPME and/or DCH models off; of varying the relative amounts of debris deposited directly in the cavity,

into the various containment atmospheres, and onto heat structures; of varying the characteristic interaction times specified for oxidation, heat transfer, and settling of airborne debris; and of enabling hydrogen combustion below the default mole fraction ignition limit during HPME/DCH periods.

MELCOR 1.8.2 Assessment: The MP-1 and MP-2 Late Phase Melt Progression Experiments

MELCOR 1.8OF was used to calculate tests MP-1 and MP-2, two late-phase melt progression experiments conducted in the ACRR facility at SNL.³³ For these experiments, sealed packages representing degraded reactor cores were driven by nuclear heating in the ACRR facility. These test bundles consisted of three regions: a rubbleized debris bed (the debris region) resting on a metallic crust surrounding 32 fuel rods (the crust region) and a bottommost region of 32 bare fuel rods (the stub region). The two test assemblies were not identical, the major differences being a longer stub region in MP-2 and different radial barrier/insulation regions surrounding the test regions. The MP-1 test resulted in a partial melting of the debris bed; the MP-2 test was heated for a longer time, resulting in partial melting and relocation of the crust to the bottom of the test section.

Both of the tests were modeled as three control volumes for the three regions, connected by flow paths. Because the experiments modeled began with partially damaged cores rather than the intact cores usually modeled, new input options for particulate and conglomerate debris were added to MELCOR. These added options exist in the default MELCOR code beginning with version 1.8OD and so are not in MELCOR 1.8.2 (version 1.8NM). MELCOR version 1.8OF was used for these calculations. The core package input for MP-1 consisted of 3 radial rings and 13 axial levels, while the MP-2 model added 2 axial levels because of the longer stub region in that experiment. The MP-1 model contained 14 heat structures, 1 radial boundary structure for each core level plus an upper axial boundary structure. The MP-2 model contained 16 heat structures, again because of the longer stub region.

A modeling compromise had to be made to calculate heat losses through the lower part of the core. It would not have been appropriate to model the lower grid spacer as part of the lower head because the COR package assumes an adiabatic boundary at the bottom of the lower head. In the experiments, a substantial amount of heat was conducted away from the core by the relatively massive grid spacer. Accordingly, the grid spacer was modeled as the first axial level of the core package. The "other structure" (OS) component of the outer ring of the first axial level was linked to the radial heat structure at that level by the core boundary conduction option. While this was the most appropriate modeling strategy given MELCOR's capabilities, it still did not represent the actual grid spacer. Because MELCOR does not model conduction from inner ring OS components to the outer ring OS component, heat loss from the grid spacer was underpredicted.

The total fission power used in the MP-1 model was calculated from region-wise coupling factors given in the experimental report³⁴ and the total ACRR power. The power distribution in the MP-2 model was calculated using control functions that used the actual UO₂ masses in each region to account for the relocation of UO₂ during the experiment. The axial peaking factors were calculated from shape factors and region-wise coupling factors given in the experimental reports.^{34, 35} Input values for the radial shape factors for both experiments were averages of the individual values for each axial region, since MELCOR does not allow different radial shape factors at each axial level.

For both experiments, MELCOR was generally in reasonable agreement with the data. The debris temperatures calculated for MP-1 were underpredicted by up to 500 K (Fig.

28), but peak temperatures were within 250 K of the correct values at all levels. The crust temperatures were overpredicted by 200 K or less at all levels (Fig. 29). This combination of overpredicted crust temperatures and underpredicted debris temperatures may be due to inaccurate peaking factor values, skewing power from the debris region to the crust region. MELCOR also did not calculate the melting observed in the debris region, also possibly because of an erroneous skewing of power away from the debris region. Stub region temperatures in MP-1 were substantially overpredicted at all levels (Fig. 30). The data showed a significant axial temperature gradient in the stub region because of efficient heat conduction through the relatively massive grid spacer to the cooling jacket. As discussed above, MELCOR does not calculate radial conduction from inner ring OS components to the outer ring OS component. This discrepancy was not considered severe by the analyst because the heat sink effect of the grid spacer is atypical of what would be encountered in a real accident scenario.

The debris temperatures calculated for MP-2 were also lower than observed, by up to 500 K (Fig. 31). Crust region temperatures were within 100 K of the observed values in the outer two rings, and within 200 K in the inner ring. As in MP-1, the calculated stub region temperatures were overpredicted, again due to the heat sink effect of the grid spacer, which could not be adequately modeled by MELCOR. The material relocation calculated within the debris region and from the crust to the stub region was in reasonable agreement with the observed final material state.

Fifteen sensitivity studies were performed on the model input parameters. These studies revealed no unexpected sensitivities in either the MP-1 or MP-2 model.

The MELCOR calculations for MP-1 and MP-2 were compared with calculations performed using DEBRIS and TAC2D.^{34, 35} These codes gave better temperature predictions than MELCOR, especially in the stub region, due to their two-dimensional heat transfer modeling. The other two codes also gave more accurate predictions than MELCOR of axial heat transfer through one of the axial insulating layers to the grid spacer, again due to their two-dimensional capabilities. DEBRIS calculated slightly better material relocation than did MELCOR (TAC2D performed only heat transfer calculations).

MELCOR 1.8.2 Calculations of Selected Sequences for the ABWR

This report³⁶ summarizes the results from MELCOR calculations of severe accident sequences in the General Electric Co. (GE) designed Advanced Boiling Water Reactor (ABWR) and provides a code-to-code comparison with results calculated for the same sequences with the Modular Accident Analysis Program (MAAP). Although the ABWR results themselves may be of interest to many of the MCAP members, the code-to-code comparison results are of particular importance because differences in the calculated results for the same scenario by the two codes focus attention on the basic assumptions and code models that contribute to those differences. Given those differences, readers can gain insights about the sensitivity of key calculated parameters based upon the differences in basic assumptions and code models. In addition, this work also provides insights into the ability of MELCOR to simulate integrated accidents that progress from the initiating event to the release of radionuclides to the containment and environment. Finally, sensitivity calculations were performed to evaluate the impact of limestone rather than basaltic concrete on core-concrete interactions.

The purpose of this analysis activity was to compare the results of GE-prepared MAAP analyses of ABWR accident sequences³⁷ with analyses of the same accident sequences using MELCOR. Five sequences were analyzed and these are briefly described in Section 3 of the report. All sequences were initiated by a loss of all core cooling, and in all sequences

the passive flooders automatically floods the lower drywell. Vessel failure occurred at low pressures in two of the sequences, LCLP-PF-R-N and LCLP-FS-R-N. Vessel failure occurred at high pressures in the remaining three sequences, LCHP-PF-P-M, LCHP-FS-R-N, and LCHP-PS-R-N. Some of the features varied among the transients were (1) containment depressurization via a relief rupture disk or leakage through movable penetrations, (2) use of firewater spray, or not, and (3) use of the drywell spray, or not.

The ABWR is a next-generation LWR, and commercial operation of these reactors is expected in the 1990s. The ABWR is an evolutionary plant based on the existing BWR technology. The author provides a general description of the ABWR in Section 2 of the report. The author also provides several cautions related to the MELCOR ABWR input model used in the calculations. In addition to stating that the input deck was not developed by SNL, the author notes that the ABWR plant design has changed since the deck was developed some years ago. The MELCOR ABWR input deck has not been reviewed against the current plant design, although several differences between the current design and the input model were identified during the study. These differences are documented in the report. The MELCOR ABWR input model consists of 11 control volumes (6 for the primary, 4 for the containment, and 1 for the environment), 21 flow paths, 35 heat structures, and a 52-cell core. The input model is described in Section 4 of the report.

A good deal of information and insights are conveyed in this 250-page report. Each of the sequences is described in detail in Section 5 of the report. Both the sequence phenomena and the code-to-code comparison of MELCOR and MAAP results are presented. Additional sensitivity studies for which no MAAP counterpart exist are presented in Section 6 of the report. The author's summary of results is provided in Section 7 of the report, and several of these are presented below, sometimes in near-verbatim reproductions of the author's statements.

1. The event sequences calculated by MELCOR and MAAP were in general agreement, albeit with timing shifts. The major differences identified were in core degradation and vessel failure time, and in core-concrete interaction and containment depressurization time. In all cases, the MELCOR-predicted time of core uncover was later than predicted by MAAP. The MELCOR-predicted core degradation process also occurred slower than predicted by MAAP. One measure of code-to-code similarity is whether sequence-to-sequence trends are calculated. The author notes that both MELCOR and MAAP predict vessel failure to occur earlier in sequences with the automatic depressurization system (ADS) depressurizing the primary system than in scenarios where the vessel fails at pressure.

2. The absence of an ex-vessel debris quenching model in MELCOR was the source of significant differences between the MELCOR and MAAP calculated late-time sequences. Lacking such a model, core debris in the cavity remains unquenched and hot in the MELCOR calculations, resulting in significant and continued core-concrete interactions. This in turn resulted in faster containment pressurization and earlier rupture disk opening, which are the result of noncondensable gas generation arising from extended core-concrete interactions and continued boiling of cavity pool water by contact with hot, unquenched debris. Such interactions are not predicted in MAAP, which assumes ex-vessel debris quenches. However, sensitivity studies were performed to assess the effect of quenching debris in the cavity compared to having hot, unquenched debris present. Using input workarounds to quench the debris in the cavity resulted in slower containment pressurization and later rupture disk opening. These trends are consistent with the MAAP results.

3. 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 depressurizing containment through the rupture disk. As previously stated, one measure of code-to-code similarity is whether sequence-to-sequence trends are calculated. Both MELCOR and MAAP predict much greater releases for most fission products, with the exception of the noble gases, if the containment leaks through movable penetration seals than if the containment depressurizes as intended through the rupture disk in the wetwell vapor space. This behavior, as calculated by two independently-developed codes, provides strong evidence of the benefit of suppression pool scrubbing on reducing the source term to the environment.

MELCOR 1.8.3 Assessment: GE Large Vessel Blowdown and Level Swell Experiments

For this report,³⁸ six blowdown experiments conducted by GE in their large blowdown vessel were modeled using MELCOR version 1.800. This version of MELCOR includes an implicit bubble separation algorithm added since the release version of MELCOR 1.8.2, which was MELCOR 1.8NM. The experiments modeled were a set of separate effects tests studying the level swell phenomenon for BWR transients and LOCAs.

The large blowdown vessel is a vertical cylinder with hemispherical end caps. A horizontal blowdown line with a flow-limiting venturi exits from near the bottom of the vessel. A rupture disk assembly at the outlet of the blowdown line is used to initiate a transient test. Attached to the inside end of the blowdown line is a vertical dip tube extending to near the top of the vessel, allowing top-level blowdowns with initial steam flow to be studied. This dip tube can be removed for studying bottom-level blowdowns with initial liquid flow.

At the beginning of a test the vessel is partially filled with saturated liquid water at over 7.14 MPa (1000 psia), the remainder of the vessel contains saturated steam. Absolute pressure, differential pressures at several levels in the vessel, and temperatures are measured during the course of the tests. Four of the modeled tests were top-level blowdowns using four different throat sizes for the flow-limiting venturi. The other two tests were bottom-level blowdowns, with two different venturi throat sizes and two different initial liquid levels.

The base case MELCOR model of the blowdown vessel consisted of two control volumes modeling the vessel itself and the outside environment, one flow path, and one heat structure. Two segments were modeled in the flow path for the top blowdown experiments, while only one segment was used for the bottom blowdown experiments. Nonequilibrium thermodynamics with bubble rise were used in the vessel control volume.

For the top blowdown tests, the MELCOR calculations were in overall reasonable agreement with the test data (Fig. 32). The calculated depressurization rates for the four tests followed the overall trend of the observed rates. MELCOR did miss an effect seen early on in the experimental data, where the observed pressure dropped sharply at the beginning of the test, followed by a pressure recovery as the liquid, now supersaturated, began to boil. This delay is thought to be caused by delayed nucleation in the liquid phase, as bubbles take some time to form and grow before they can rise through the liquid phase and be released into the vapor phase. MELCOR does not model delayed nucleation. As the venturi diameter (and consequently the blowdown rate) increased, this effect became more and more significant in the test.

The initial calculated swell of the liquid level was in excellent agreement with the data for each of the top blowdown tests (Fig. 33). Both the rate at which the level increases and the

absolute level were essentially identical with the observed values until MELCOR reached an internal limit of 40% for the pool bubble fraction (this limit is imposed to improve the stability of MELCOR calculations). Before the MELCOR pool bubble fraction limit was reached, MELCOR correctly predicted the faster rate of pool swell at higher blowdown rates. Beyond that point, the calculations diverged from the data. The observed swelled levels continued to increase with time, then the swelling rate flattened out and dropped off as the pool was cooled by the boiling. As venturi size (and blowdown rate) increased, the maximum observed liquid level also increased, and remained above the level observed at the same time at lower blowdown rates for the remainder of the test. The calculated maximum levels were the same for all four tests. In all cases the level rose until the 40% limit was reached, at which time the level suddenly and discontinuously started to drop. Because of this sharp decrease, the calculated level at a given time was lower at higher blowdown rates, the opposite of the effect observed in the tests. A sensitivity study, in which the maximum pool bubble fraction was varied above and below the default 40% value, demonstrated that a better match with experiment can be achieved by using higher values (increasing as the throat diameter increases). Even then the model tended to maintain a constant pool bubble fraction once it hit this maximum, whereas the test data indicated that the pool bubble fraction begins to decrease once it hits its maximum value.

For the two bottom blowdown tests, the calculated results were in reasonable to excellent agreement with the data. The pressure first dropped slowly as liquid was discharged through the blowdown line. At the same time the blowdown flow rate was higher than in the top blowdown tests. When the blowdown line was uncovered, the rate of pressure decrease increased and the blowdown flow rate decreased. MELCOR successfully predicted both of these effects (Fig. 34). At the smaller throat diameter, the calculated pressure and swollen liquid level (Fig. 35) were in excellent agreement with the data. The agreement was still reasonable at the larger throat diameter. The slight degradation of model agreement at the higher blowdown rate was probably due to the form loss and discharge coefficients being compromise values used for all bottom blowdown tests, as shown by the sensitivity study where those coefficients were varied.

SENSITIVITY STUDIES

PNL Ice Condenser Aerosol Sensitivity Studies

In the reference calculation, the injected ZnS aerosol was input as a second aerosol component, with water as the first. Five aerosol sections were used; a value of 2500 kg/m³ was used for solid density in the aerosol. Finally, a value of 1.0×10^{-7} m was used for the minimum particle diameter. Sensitivity studies examined the effects of changing these values: ZnS was combined with water for a single aerosol component (the MELCOR default), the number of aerosol sections was increased to 10 and to 20, aerosol solid density was decreased to 1000 and increased to 5000 kg/m³, and the minimum particle diameter was increased and decreased by a factor of 10.

These aerosol sensitivity studies tended to confirm the validity of the modeling decisions made in developing the reference model. Combining the ZnS and water into a single aerosol component degraded the agreement of the calculated particle retention with the data while only saving about a minute of CPU time. As the number of aerosol sections increased, the particle retention values appeared to be converging on values 1–2.5% lower than in the reference calculation (presumably more accurate results), but at the cost of greatly increased CPU time. Because MELCOR gives all aerosol components the same density, the default value of 1000 kg/m³ was felt to be too low. The sensitivity studies in

which the aerosol density was changed confirmed that the 2500 kg/m^3 value used in the reference calculation gave the best agreement with observed particle retention data. Finally, the sensitivity studies in which the minimum particle diameter was varied confirmed the validity of the value used in the reference model.

PNL Ice Condenser Thermal-Hydraulic Sensitivity Studies

Some of the thermal-hydraulic modeling used in the reference model was changed to observe the effects on the calculated results. Flow loss coefficients were doubled and halved from the reference model values. This had negligible effect on the results, as the pressure drop was relatively low in these experiments. Another sensitivity study looked at using equilibrium thermodynamics instead of the nonequilibrium thermodynamics used in the reference model. This lowered the vapor temperature and thus the melting rate of the ice, and increased the particle retention at late times. The nonequilibrium thermodynamics gave a better match of calculated results with the data. A final sensitivity study turned on the SPARC (suppression pool aerosol removal code) bubble rise model, which was off in the reference model. The SPARC model changed the particle retention results, but it did not really represent anything physical happening in the experiment. It is recommended that the SPARC model be left off for ice condenser simulations.

PNL Ice Condenser Ice Studies

The sensitivity studies conducted in developing the reference models for the two PNL ice condenser experiments identified three parameters that had significant impact on the accuracy of the model. These were the temperature range of ice melting, the heat transfer coefficient multiplier, and the number of nodes used in the heat structures modeling the ice.

The analyst found that setting the range of ice melting to 274–284 K gave the best match with the observed temperatures and melt times. This temperature range sets the heat of reaction for ice melting. Setting the low value to 274 avoids convergence problems known to occur in the CVT package when liquid water is present and the temperature is near the freezing point (273.15 K).

A previous report⁵ recommended a value of 1.2 for the heat transfer coefficient multiplier. This gave reasonable results for experiment 16-11 (the high flow experiment), but did not melt the ice quickly enough in experiment 11-6 (the low flow experiment with the pre-melted ice). For that experiment, a value of 1.8 in the lowest ice condenser control volume and a value of 3.8 in the middle control volume gave the best agreement with the observed time at which all the ice melted. No adjustment was needed in the top ice condenser control volume of experiment 11-6, because all ice was melted there before the experiment started. These higher values were within the acceptable range, compared to the value of 5.0 used by the CONTAIN model.⁵

Finally, the analyst strongly recommended using only two nodes in heat structures representing ice. The problem with using more nodes arose when the material within one node had completely melted. At that point, MELCOR replaced the ice with a low density, high heat capacity substance that tended to insulate the remaining ice and retard further melting.

DF-4 BWR Damaged Fuel Experiment Sensitivity Studies

The analyst performed a number of sensitivity studies on MELCOR parameters. As discussed above, many sensitivity calculations varied the MELCOR parameters that affect fuel relocation to try to duplicate the fuel pellet stacking seen in the experiment. Changing the oxide thicknesses needed to hold up molten materials had only a small effect. Changing

the debris formation parameters in record COR00008 so that no unoxidized zircaloy or stainless steel were required to hold up their associated components resulted in the fuel pellets remaining in place, held by a layer of oxidized zircaloy. After this change, the agreement between the calculated and observed fuel relocation was reasonable. This change was not, however, justified by the data, which clearly showed that oxidized zircaloy did relocate. The author's final recommendation was that when modeling experiments that exhibit fuel pellet stacking, one should probably adjust these parameters.

Deactivating the eutectic model predicted much different behavior for the B₄C and decreased hydrogen production. Changing the melting point of the B₄C/stainless steel eutectic by ± 50 K had little effect on the results. Using the default, bulk-temperature model for heat transfer between the fluid and solid components, instead of the model based on the finer resolution dT/dz model, gave higher temperatures and earlier failure of components. Other studies showed sensitivities to zircaloy properties, COR component view factors, allocation of canister mass, candling heat transfer coefficient, debris size and porosity, and nodalization. No sensitivities were found to the minimum component mass, B₄C oxidation modeling, and outer boundary temperature.

IET Direct Containment Heating Test Sensitivity Studies

Many sensitivity studies were performed in the IET DCH assessment, mostly confirming the parametric choices made in model development. Varying the characteristic time for airborne debris oxidation showed that better agreement with the hydrogen production and combustion data (using the experimenters' assumption of no oxygen/metal reactions) was obtained using a longer characteristic time than was used in the reference calculations. The MELCOR analysis of the scale effects comparing the 1:10-scale SNL experiments and the 1:40-scale ANL experiments were inconclusive. The calculated pressures of the SNL and ANL counterpart tests compared reasonably well when both the geometry and the characteristic time scales were scaled. However, the assessment report found suggestions of better DCH energy-transfer efficiency, less pressurization due to hydrogen combustion, and greater effect of preexisting hydrogen at the smaller scale, differences that were not reproduced in the MELCOR analyses. Other sensitivity studies suggested that some of the greater pressurization observed in the experiment could be associated with the different scales of the facilities. These factors are differences in: (1) heat transfer, (2) recirculation flow area, and (3) burn parameters.

MIST 3109AA Sensitivity Studies

In one sensitivity study, the break k-factor was reduced and increased by 7% from the value used in the base model. Increasing the k-factor by 7% had the expected result, with decreased break flow resulting in higher liquid levels. Reducing the k-factor by 7% had the unexpected result of causing a break flow discontinuity: from about 2200 to 3000 s, the break flow decreased sharply from 0.07 to 0.02 kg/s, then increased sharply back to 0.09 kg/s. Another sensitivity study changed the choked-flow discharge coefficient from the MELCOR default of 1.0 used in the base model to 0.9. This resulted in smaller break flow and higher liquid levels. An earlier MELCOR assessment³² had shown better prediction of natural-circulation flow in steam generators modeled with finer noding than is usual in MELCOR models; therefore, four vertical CVs were used in the SG primary of the base model. However, for this model, using only one CV in the SG primary yielded results that were within 1% of the four-CV model.

MIST 3404AA Sensitivity Studies

As in the MIST 3109AA study, the break k-factor was reduced and increased from the value used in the base model. The break-flow discontinuity discussed above was not seen in this study. Increasing and decreasing the k-factor had the expected results, decreased break flow and higher primary liquid levels at the higher k-factor, and increased break flow and lower primary liquid levels at the lower k-factor. Changing the choked-flow discharge coefficient from the MELCOR default of 1.0 used in the base model to 0.9 resulted in smaller break flow and higher liquid levels, as observed in the MIST 3109AA study. Finally, as done in the MIST 3109AA study, four vertical CVs were used in the SG primary of the base model to improve natural-circulation flow. With this model, as with the MIST 3109AA model, using only one CV in the SG primary yielded results that were within 1% of the four-CV model.

Surry PWR TMLB' Interfacial Momentum Exchange Length/CCFL Sensitivity Study

The MELCOR peer review²⁷ questioned the failure of the pressurizer to drain until the system depressurized following vessel failure in the MELCOR 1.8.1 Surry TMLB' calculation. There was general agreement that this appeared to violate physical intuition. Members of the peer review committee expressed concern that the failure of the pressurizer to drain was a result of an inadequate momentum exchange model in MELCOR, leading to an incorrect two-phase CCFL. Two changes, introduced in MELCOR since version 1.8.1, impact the prediction of countercurrent flow. The first change, in update 1.8JQ, separated the momentum exchange length from the inertial length in the momentum exchange model. Instead of assuming the momentum exchange force acted over the entire inertial length as in MELCOR 1.8.1, the default momentum exchange length was taken as the range of elevations encompassed by the junction openings for vertical flow paths, and as the inertial length for horizontal flow paths. User input can be used to override these values if desired. The other change, in update 1.8LJ, prevented misapplication of the algorithm preventing phase depletion as a result of outflow of pool or atmosphere from a junction. In certain cases the calculated void fraction could be miscalculated, incorrectly precluding countercurrent flow.

For the reference MELCOR 1.8.2 Surry calculation, momentum exchange lengths in the pressurizer surge line and in the hot leg were set equal to the pipe diameters instead of using the default values of the flow path length for these horizontal lines. Figure 36 shows that the pressurizer drained slightly faster in the MELCOR 1.8.2 calculation before vessel failure with the code changes and the momentum exchange length input used. Sensitivity calculations were performed where the hot-leg, surge-line, and cold-leg momentum exchange lengths were individually set to 10% of their pipe diameters, and in which all three of the smaller values were used. As seen in Figure 36, changing the pressurizer surge line momentum exchange length had the greatest effect on pressurizer drainage. While this study showed that a considerable range of pressurizer drainage flow could be achieved by varying the momentum exchange length, the question of a "correct" value remains open.

Surry PWR TMLB' Eutectic Model Sensitivity Study

As mentioned in the discussion above of the Surry TMLB' calculation, the eutectic model introduced in MELCOR 1.8.2 was not used in the comparison of the MELCOR 1.8.1 and MELCOR 1.8.2 models. The eutectic model was, however, used in the MELCOR 1.8.2 reference calculation. A sensitivity study compared the results obtained using the MELCOR 1.8.2 reference model with and without the eutectic model enabled.

The eutectic model generally had only a small effect on the calculations. Earlier core support plate failure and earlier lower head penetration were calculated with the eutectic model enabled, but the difference was less than 5%. The biggest difference was in the lower plenum structure. Without the eutectic model, about 80% of the steel structure in the lower plenum melted and fell into the cavity. With the eutectic interaction modeled, Zr and stainless steel debris in the lower plenum melted at lower temperatures and flowed to the cavity sooner. The combination of lower debris residence time and less heating of the steel structure led to a higher retention of the lower plenum steel structure (~70%). The smaller amount of steel transferred to the cavity with the eutectic model led to decreased concrete ablation and lower containment pressure at late times.

Surry PWR TMLB' Fission Product Release Model Sensitivity Study

The reference calculation in the Surry PWR TMLB' study was done using the CORSOR release model with the surface/volume correction term. A set of sensitivity calculations was performed using the other five options available in MELCOR 1.8.2: CORSOR without the surface/volume correction, CORSOR-M with and without surface/volume correction, and CORSOR-Booth with high and low burnup coefficient sets. In-vessel, the CORSOR and CORSOR-M options resulted in similar releases of Xe, Cs, and I volatiles. CORSOR gave higher releases for Ba, Ru, Mo, Ce, La, Cd, and Sn, while CORSOR-M gave higher releases for Te and no release at all of Mo, La, or Cd. CORSOR-Booth gave lower releases for the most volatile species (Xe, Cs, and I), as well as for Ba, Te, and U, than either CORSOR or CORSOR-M, while some other species were intermediate between CORSOR and CORSOR-M. The differences in overall release produced by using the various CORSOR options were much smaller, because the later ex-vessel release somewhat compensated for in-vessel differences.

The Class 5 release in-vessel appeared surprisingly high. MELCOR includes a model to hold up Te by unoxidized Zircaloy clad in the form of ZrTe, reducing the release by a factor of $\frac{1}{40}$ if sufficient clad remains unoxidized. It would appear that this holdup model should have been activated; however, inspection of the coding by the developers showed that only intact Zircaloy in the clad component was used to calculate the degree of Te holdup. Unoxidized Zircaloy, in particulate or conglomerate debris, were not included in the holdup calculation, so the calculated degree of Te release was higher than expected. This unexpectedly high Te release has been reported in other cases,³⁹ probably due to the same cause.

Using the different release models also affected the timing of key events predicted by MELCOR. Release of small amounts of Classes 7, 9, 11, and 12 prior to clad failure and gap release were sufficient to affect the code time step, which caused a small but noticeable divergence of system response early in the transient. The different releases of radionuclides calculated with the various CORSOR options also affected the local distribution of decay heating, further perturbing the transient sequence.

Surry PWR TMLB' HPME/DCH Sensitivity Study

The amount of melt available for HPME is determined by a number of factors, including the amount of early core damage, core plate failure criteria, falling debris heat transfer, possible quenching of debris in the lower plenum, lower head penetration heat transfer, and lower head failure criteria. MELCOR default values were used for the core plate and bottom head penetration failure temperatures and the falling debris and lower head penetration heat transfer coefficients. In limited sensitivity studies, some of those values were varied, but given the lack of data for guiding users in selecting proper values for those parameters, the

analysts chose a different method to allow comparison of DCH behavior with different amounts of HPME. By adjusting the peaking factors in the core, they were able to increase the material ejected during HPME to approximately 60% of the available core material. Other HPME/DCH studies looked at the effects of (1) turning the HPME and/or DCH models off; (2) varying the relative amounts of debris deposited directly in the cavity, into the various containment atmospheres, and on heat structures; (3) varying the characteristic interaction times specified for oxidation, heat transfer, and settling of airborne debris; and (4) enabling hydrogen combustion below the default mole fraction ignition limit during HPME/DCH periods. The changes made in HPME had the expected results. Depositing more debris directly into the cavity or onto heat structures reduced the magnitude of the pressure/temperature excursion, while increasing the amount of debris deposited in the containment atmosphere increased the magnitude of the pressure/temperature excursion. Specifying more debris into a relatively small volume, like the cavity atmosphere, resulted in a larger pressure/temperature excursion in that atmosphere.

MP-1 and MP-2 Sensitivity Studies

A number of sensitivity studies were performed in the course of the MP-1 and MP-2 modeling studies. Many of these were done in preliminary studies to determine nondefault input options to use in the reference model discussed above. The sensitivity studies were grouped into four main areas: general COR package input options, heat structures package studies, COR and CVH nodalization for MP-1, and the time step and machine dependency studies reported below with the numeric effects studies.

MP-1 and MP-2 COR Package Sensitivity Studies

Nine studies were done on the COR package input options. The first looked at the effect on MP-2 calculations of turning off the axial heat conduction. Axial heat conduction was important in these tests because of the steel grid spacer that acted both as a heat sink and as a conduction path to the cooling jacket. As expected, turning off axial heat conduction increased debris- and crust-region temperatures, and decreased stub region temperatures (although MELCOR does not model axial heat conduction in the particulate debris component, the debris temperature was raised by radiation from the hotter crust region). The calculation without axial conduction was not able to run to completion, instead dying when large-scale component failure occurred. The second COR package input option examined was candling. The reference calculation for MP-2 used IDRP=2, which allows individual materials to candle from the conglomerate debris. Using the MELCOR default candling (IDRP=0) and turning candling off completely (IDRP=1) resulted in more material held up in the cladding component in the crust region. In addition, turning off candling held all molten debris material in the debris region.

Two COR package options that had little if any effect on the results were the candling model refreezing heat transfer coefficient and the convection heat transfer coefficient. Changing them by $\pm 50\%$ from the default values had negligible effects on the results. Increasing and decreasing the diameter of particulate debris by a factor of two also had no effect on the results.

Changing core component view factors for radiation heat transfer did affect the results. Decreasing them in the MP-2 model increased temperatures, and increasing them decreased temperatures. The changes were greater as the view factors decreased than as they increased, indicating that radiation heat transfer was already near its maximum. At the lowest view factor examined, the calculation did not run to completion because of a control volume filling with debris, a condition not allowed by the CVH package.

Another COR package option studied was the eutectics model, which was not activated in the reference models for either test, since the experimental results showed no evidence of large-scale eutectics effects. Because the conglomerate debris for the experiments was formed by pressing together a mixture of component powders rather than by melting them together, the conglomerate debris components melted individually in the tests. Activating the eutectics model in the MP-2 model lowered crust temperatures somewhat, and led to a poorer agreement with final material disposition.

The final COR package options studied were the minimum component mass and the minimum component mass for temperature control. These were set to 10^{-7} and 10^{-6} kg, respectively, from the default values of 10^{-3} and 10.0 kg, which are meant for modeling full-size reactors. The default values were used in a sensitivity study of the MP-1 model. Changing the minimum component mass had no effect on the results because none of the component masses were below 10^{-3} kg. The component masses were, however, all under 10.0 kg, so changing the minimum component mass for temperature control caused the crust region temperatures to increase by 125 K, and the stub region temperatures to decrease by 75 K. The debris region temperatures did not change.

MP-1 and MP-2 Heat Structure Sensitivity Studies

There were three sensitivity studies on heat structure package options. In the base calculations, gamma heating was neglected because it was considered low enough to be insignificant compared with the fission heating. For this study, gamma heating was activated in the MP-2 model according to heating rates obtained from the personnel performing the MP experiments. There were slight changes in component temperatures when gamma heating was activated, but they were insignificant.

Another sensitivity study looked at gaps in the insulation layers between the fuel regions and cooling jackets. Because these components did not fit together without gaps, the gaps filled with helium. These gaps were ignored in the base models. For this study, helium gaps were inserted between all heat structure interfaces inside the ZrO_2 layer in the MP-2 input model. The ThO_2 layer in the debris region was split in two, and another helium gap was inserted between the layers. The total helium gap inserted amounted to about 3% of the heat structure thickness. Adding these gaps increased component temperatures inside the test bundle by about 150 K. Unfortunately, the calculation did not run to completion because massive relocation resulted in the CVH problem discussed above in the MP radiation heat transfer view factor study.

A final heat structure sensitivity study looked at changing the outer boundary temperature. Relative to the thermocouple measurement taken on the inside wall of the cooling jacket, temperatures were changed by ± 50 K. For the temperature decrease, a lower limit of 300 K was invoked. These changes had only very small effects on the calculated results.

MP-1 and MP-2 COR and CVH Nodalization Sensitivity Studies

As stated above, the COR nodalization in the MP-1 basecase input model used 3 rings and 13 axial levels. The CVH nodalization used 3 control volumes and 2 flow paths. This nodalization was chosen as a compromise between accuracy and problem run time. For the nodalization sensitivity study, a fine nodalization case was run with 3 rings, 25 axial levels, and 9 control volumes. A coarse nodalization case was also run with 1 ring, 4 axial levels, and 1 control volume. The fine nodalization case took approximately 7 times as long as the basecase to run, while the coarse nodalization case took only one-fifth as much execution time. The results of the fine nodalization case, while more accurate than the basecase,

probably did not justify the extra run time. The coarse nodalization case, by lumping all regions of the test into just one, gave significantly poorer results than the basecase.

ABWR Sensitivity Studies

Sensitivity studies were performed on two of the calculations to calculate the impact of using limestone-based concrete rather than basaltic concrete. With limestone concrete, the containment pressurized faster, and the rupture disk opened earlier, due to more generation of noncondensables in core-concrete interactions. Another sensitivity study looked at the effect of quenching core debris in the cavity compared with having hot, unquenched debris present. Having quenched debris in the cavity resulted in slower containment pressurization and later rupture disk opening, which was in better agreement with the MAAP calculations. Neither varying the concrete type nor varying the debris temperature significantly affected the fission product release.

GE Large Vessel Blowdown Sensitivity Studies

A set of sensitivity studies were conducted as a part of this study. These can be classified as CVH studies, FL studies, and numeric studies. The first two classes of studies are documented here, the numeric studies are reported below with the other numeric studies.

GE Large Vessel Blowdown CVH Sensitivity Studies

Using equilibrium thermodynamics rather than nonequilibrium thermodynamics gave almost identical break flows and depressurizations in the top blowdown experiments. However, since the equilibrium thermodynamics do not allow bubbles in the pool, the swollen liquid levels were identical with the collapsed liquid levels, and were in much poorer agreement with the data. There were only minor differences in the bottom blowdown depressurizations, flow rates, and levels with equilibrium and nonequilibrium thermodynamics. As discussed above, a sensitivity study in which the maximum pool bubble fraction was varied above and below the default 40% value demonstrated that a better match of top-blowdown swollen liquid level with the data can be achieved by using higher values of the maximum bubble fraction (increasing as the throat diameter increases). Even then the model tended to maintain a constant pool bubble fraction once it hit this maximum, whereas the test data indicated that the pool bubble fraction begins to decrease once it hits its maximum value. Because no one value proved best for all experiments, the default 40% value was used in the reference calculations.

Another sensitivity study examined the effects of changing the bubble rise velocity from the MELCOR default of 0.3 m/s. Velocities of 0.03, 0.1, 1, and 3 m/s were examined. In the top blowdown tests, at lower bubble rise velocities, the liquid bubble fraction tended to equilibrate at the maximum allowable value. For each exit venturi size, there was a bubble rise velocity beyond which the bubble fraction equilibrated at a value below the maximum. Beyond that velocity, increasing velocity lowered the equilibrium bubble fraction. Vessel depressurization and break flow were insensitive to the bubble rise velocity in the top blowdown tests. In the bottom blowdown tests, the break flow and depressurization did change with changing bubble rise velocity. Vessel level swell also depended on the bubble rise velocity. There was no single bubble rise velocity other than the default value that provided improved agreement with data.

A final CVH sensitivity study subdivided the vessel into 13 axial control volumes connected by flow paths. Blowdown under these conditions resulted in a "layer cake" of alternating swollen liquid and vapor phases. With increasing level in the top blowdown

tests, the bubble fraction decreased, counter to physical intuition. The single control volume in the reference model gave a better qualitative fit to the swollen level data. Finally, the calculation time increased substantially for the finer noded model. There appeared to be no benefits found in subdividing the vessel into multiple, stacked control volumes.

GE Large Vessel Blowdown FL Sensitivity Studies

The choked-flow discharge coefficient is a multiplier for the critical flow calculated by the MELCOR choked-flow model. The default MELCOR value is 1.0 for both forward and reverse flow. A sensitivity study varied this coefficient below and above the default value. Varying the coefficient had the expected result on vessel depressurization: higher values allow faster depressurization. While no single value was found that was best for all top and bottom blowdown tests, this study concluded that a value of 0.85 gave the best overall fit to data for the top blowdown tests, and 0.75 worked best for the bottom blowdown tests.

Another coefficient available to the modeler is the flow loss coefficient, which accounts for form loss from bends and sudden area changes. Unlike the choked flow discharge coefficient, which affects the maximum flow under choking conditions, the flow loss coefficient is always active, and acts as a loss proportional to the flow velocity in the momentum equation. In the top blowdown tests, varying the flow loss coefficient between 0.0 and 2.0 had negligible effect on vessel depressurization, break flow, or vessel level swell. This is to be expected, since usually only steam was flowing in the top blowdown tests, and the effect of its relatively low density on the overall momentum equation was minimal. Changing the flow loss coefficient in the bottom blowdown tests, however, did have an observable effect on the results. A value of the flow loss coefficient of 1.5 was chosen for the reference calculations.

The SPARC bubble rise physics model was turned off in the reference calculations. In sensitivity studies, this model was turned on or off in the blowdown flow path, in the inlet side, in the outlet side, and in neither or both sides of the flow path. There were no significant differences in the calculated response in any of the tests analyzed.

NUMERIC EFFECTS IN MELCOR CALCULATIONS

Most of the assessment reports summarized herein contain sensitivity studies on numeric effects that can affect the outcome of MELCOR calculations. Many of these effects are the results of varying the time-step size and of performing identical calculations on different hardware platforms. Summarized below are the numeric effects sensitivity studies from the assessment reports.

Numeric Effects in the PNL Ice Condenser Study

The reference calculations for both PNL ice condenser experiments was run on a Cray X-MP-24, a DEC VAX 8650, a 486 PC, and a Sun SPARCstation 2 to compare the results with those calculated with the IBM RISC 6000 Model 550 used in the majority of the calculations. There were some differences in the last significant digit of the printed outputs, but the plotted data could not be distinguished.

Time-step studies looked at decreasing the maximum time step by factors of 2, 5, 10, and 100. Time steps were not increased for this study because the maximum time steps in the reference calculation were being limited by the CVH package. As the maximum time step decreased in the calculations of experiment 11-6, the calculated particle retentions increased, as shown in Figure 37. The retentions seemed to be converging on a value about 2% higher than that calculated in the reference model. Similar results with smaller changes

were seen in the calculated temperatures for experiment 11-6 and in the particle retentions and temperatures calculated for experiment 16-11.

A third numerical sensitivity study was performed in the ice condenser study to see the effect of removing heat structure numerical damping. The Heat Structure Package contains a model that stabilizes numerical oscillations that can occur when heat is transferred to a control volume with a small thermal inertia. The reference calculations for the two experiments used the default value for the 4070 coefficient that controls numerical damping. For this sensitivity study, the 4070 coefficients for both models were set to zero. This had negligible effect on the calculated results for experiment 16-11. In experiment 11-6, removing this damping resulted in a smaller heat transfer coefficient, slower ice melting, and more particle retention.

Numeric Effects in the DEMO Problem Study

This study looked at the differences in a DEMO problem calculation performed on different machines (IBM RISC 6000/560, HP 9000/730, DEC 3000/500 AXP, and Sun SPARCstation 2). The study demonstrated that many of the machine differences seen in MELCOR 1.8.1 calculations were much smaller in MELCOR 1.8.2 calculations. The MELCOR 1.8.2 results were brought into even better agreement by using double-precision calculations.

Numeric Effects in the DF-4 BWR Damaged Fuel Study

The DF-4 base calculation used a maximum time step of 0.25 s between 7200 and 8000 s (the time when core degradation took place) and 1.0 s at all other times. For the time-step sensitivity studies, the time-step sizes during the 7200–8000 s period were set to 0.1, 0.5, and 1.0 s. A fourth sensitivity calculation was performed with the time-step size set so high that MELCOR always chose a default time-step size below the input value. The calculated times for cladding and control blade failure varied by about 100 s in these time-step studies.

The base calculation and other sensitivity studies were performed on a DECstation 5000/240. The base calculation was repeated on an HP 755, an IBM RS/6000 Model 550, a Sun SPARCstation 2, a Cray X/MP-24, and a 486 PC. Very little variation was seen in the results calculated by the different machines beyond a small difference in final material distribution.

Numeric Effects in the IET Direct Containment Heating Study

A time-step sensitivity study in the IET DCH study simultaneously changed the maximum time-step size and the initial time-step size for HPME injection from the base case values of 1 s and 1×10^{-4} s. Other values input for the two time-step sizes were 0.5 and 5×10^{-5} , 0.1 and 1×10^{-5} , 0.05 and 5×10^{-6} , and 0.01 and 1×10^{-6} s. For the experiments without hydrogen burn, there was little effect of time-step size on the results, with the single exception of the IET-1R calculation. The pressures calculated for that test with the largest time-step size were approximately 8 kPa higher than those calculated at the lower time steps. The Sun SPARCstation 2 calculations for that test at that time step were not fully converged. In most of the cases with hydrogen burn, the total pressure and amount of hydrogen burned depended on the time-step size, with a tendency to converge as the time step was decreased.

The base calculation and most of the sensitivity studies were performed on a Sun SPARCstation 2. Machine dependency calculations were run on an IBM RISC-6000 Model 550, an HP 755, a Cray Y-MP8/864, and a 486 PC. The failure of the SPARCstation 2

calculation of IET-1R to fully converge at the time-step size used in the base calculation was the only significant machine dependency seen in this study. All the other machines were able to give a fully converged answer at that time-step size.

Numeric Effects in the MIST 3109AA and 3404AA Studies

The calculations for these studies were run on a Sun SPARCstation 2 and a Cray Y-MP. Only small differences were seen in the calculated results. Changing the maximum time-step size from 1.1 to 10.0 s had minimal effect on the CPU time and no effect on the calculated results. Decreasing the time-step size below 1.1 s resulted in a sharp increase in CPU time, again with no difference in the calculated results.

Numeric Effects in the Surry PWR TMLB' Study

Numeric effects sensitivity studies looked at the effect of running the reference calculation on different machines. The machines used were Cray, Sun SPARCstation 2, HP Model 755, IBM RISC-6000 Model 550, and 486 PC. Time-step studies were also performed, using either code-selected time steps or with the maximum time step constrained by user input to 5, 2.5, and 1 s. These numeric studies found minor differences in valve cycling in the steam generator secondary and the pressurizer relief valve, the timing of the clad failure, gap release, and penetration failure of the core support plate and lower head, amounts of radionuclides released, magnitude and timing of the cavity and containment response, and hydrogen burn frequency and duration. No significant branching into different response modes was found in any of these numeric effects studies. This study and a Brookhaven National Laboratory study⁴⁰ showed a significant reduction of the numerical sensitivity of MELCOR 1.8.2 compared to that of MELCOR 1.8.1.

The valve cycling differences seen in these sensitivity studies were caused by differences in over- and undershooting the valve controller setpoint pressures with different time steps and/or different machine accuracies. Two code changes were tested. The tabular function logic was modified to allow step function input, which should minimize the effect of valves caught in a part-open state interpolating between table entries. The other change tested was a time-step controller that limits the time step when a valve pressure setpoint is being approached. These changes decreased this numerical sensitivity significantly, but other contributing effects remain to be identified.

There were significant differences in the machine- and time-dependency calculations in the times that hydrogen burned in the containment, and in the amount of hydrogen burned. An additional set of sensitivity study calculations adjusted down by an order of magnitude the default overshoots allowed in the combustion ignition mole fractions. This gave no visible improvement in the scatter of the results. This numerical sensitivity prevented study of the effects of enhanced hydrogen ignition during HPME/DCH, because the numerical sensitivity was great enough to dominate and cover up the physical effect studied.

Numeric Effects in the MP-1 and MP-2 Studies

The basecase calculations for test MP-2 were run on an IBM RS/6000 model 550 workstation. To test the machine dependency of this calculation, sensitivity study runs were performed on a DECstation 5000/240, a Hewlett Packard 755, a Sun SPARCstation 2, a Cray Y-MP 8/864, and a 486 PC. There were slight differences between the results calculated by the different machines, including differences of up to 5% blocked volume fraction in the upper portion of the debris region. These differences were never enough to change the blockage state (fully or partially blocked) in any case. Time-step studies were

done on the MP-2 calculation, changing the maximum time-step size from the 0.5 s used in the base calculation, to 0.1, 0.25, and 1.0 s. The core component temperatures were generally insensitive to time-step size. The failure timing of the uppermost debris region level varied by several hundred seconds, and the crust region temperatures varied slightly, but the variations were considered slight and had little effect on the overall calculation.

Numeric Effects in the GE Large Vessel Blowdown Studies

The reference calculations were performed on a 486 PC. The basecase calculations were rerun on an IBM RISC-6000 Model 550, an HP 755, a Sun SPARCstation 2, and a Cray Y-MP8/864. No significant differences were seen for either the top or bottom blowdown calculations on any of these machines. The maximum time step in the reference calculations was set to 2.0 s. Reducing the maximum time step to 1.0, 0.5, 0.25, and 0.1 s had negligible effect on the top blowdown tests. Somewhat larger effects were seen in the bottom blowdown tests, but the time-step dependence was still quite small.

A final numeric effect study compared the results obtained with version 1.800 and the release version 1.8.2, which was 1.8NM. The significant difference between the two codes for these studies is the implicit bubble separation algorithm in version 1.800. In most cases there was no significant effect on the results of using the implicit bubble separation algorithm. The single exception was in the bubble rise velocity study discussed above. In version 1.8NM the vessel pool bubble fraction always rose to the maximum allowable value, albeit more slowly at the higher bubble rise velocities. As discussed above, in version 1.800, with the implicit bubble separation algorithm, as bubble rise velocity increased, a maximum velocity was reached at which the pool bubble fraction equilibrated at a value below the maximum allowed value. As the velocity increased above that maximum, the equilibrium pool bubble fraction became lower.

USER GUIDELINES

In this section we present significant additions to the existing MELCOR user guidelines as reported by the authors of the reports summarized above. The guidelines are listed in separate subsections for each report that included new user guidelines.

PNL Ice Condenser Ice Studies

This report found three parameters that had significant impact on the accuracy of the model. These were the temperature range of ice melting, the heat transfer coefficient multiplier, and the number of nodes used in the heat structures modeling the ice.

The analyst found that setting the range of ice melting to 274–284 K gave the best match with the observed temperatures and melt times. Values of the heat transfer coefficient multiplier between 1.2 and 3.8 gave the best predictions of ice melt times and were consistent with previous studies.⁵ Finally, the report strongly recommended using only two nodes in heat structures representing ice. The problem with using more nodes arose when the material within one node had completely melted. At that point, MELCOR replaced the ice with a low density, high heat capacity substance that tended to insulate the remaining ice and retard further melting.

IET Direct Containment Heating Study

This study obtained reasonable agreement with experimental data and would be a good starting point for modeling HPME and DCH. The HPME model input includes characteristic times for debris settling, oxidation and heat transfer in atmospheres, and oxidation on surfaces. The debris-settling characteristic time input for the model was based

on free-fall times for the various volume heights. The other characteristic times were selected using an iterative procedure. These characteristic times gave reasonable agreement with measured vessel pressures, subcompartment temperatures, and hydrogen production and combustion data. Because of the empirical nature of their selection, they are not necessarily the best set of values for representing the data. The iterative procedure found one possible set of characteristic times that gave reasonable agreement, but that set was selected for purely mathematical reasons (maximizing agreement with the data) and may not represent the actual physical processes involved.

Another important aspect of the model's agreement with the data was the use of special combustion parameters during the HPME period. These parameters were set to prevent the burning of preexisting hydrogen, only allowing hydrogen generated during the HPME to burn. These nonstandard burn parameters would not be appropriate for plant analyses before and after the HPME period, so changes were made in the MELCOR input allowing one set of burn parameters during HPME and another set at other times. This input change was made in MELCOR version 1.8NN and does not appear in the release version 1.8.2.

A final aspect of the model was the need to enhance heat transfer from the atmosphere to heat structure surfaces during the HPME period. This was done by reducing the control volume areas by approximately a factor of 10 from their default values. These areas are only used by MELCOR to calculate the volume velocities for use in calculating the convective heat transfer coefficient from the atmosphere to heat structure surfaces, and do not affect the flow path calculations at all. Because MELCOR adds hot debris to its final destination instead of transporting it with the steam, it tended to heat the downstream volumes and decrease the differential pressure driving the flow. If the debris were carried with the flowing steam, it would increase the upstream pressure and increase the differential pressure driving force. Second, MELCOR does not model radiation heat transport from the debris to the heat structures. Because hand calculations indicated this heat transfer mode was important, it was feared that too much energy would instead be transferred from the debris to the atmosphere. Increasing the flow, and consequently the convective heat transfer coefficient, allowed this excess energy to be carried to the heat structure, as the debris-to-heat-structure radiation would have if it were modeled. The capability to add an optional multiplier for volume velocities (used only during the HPME period) was added to MELCOR version 1.8NN and does not appear in the release version 1.8.2. Because of the code changes made specifically to enhance HPME modeling, users should be sure to use version 1.8NN or later if HPME/DCH phenomena are to be modeled.

MIST Tests 3109AA and 3404AA

As new users outside of the MELCOR development group, the authors discovered some areas where the MELCOR user guidelines could be improved from the viewpoint of first-time readers. They recommended including more examples, for instance explaining what is meant by "two-character code version" in the "RUNONLY" command. The authors also would have liked to see more detailed information about how to make the transition from a steady-state to a transient calculation. The User Guidelines did not clarify how to make that transition, requiring a telephone call to SNL to solve that problem. Another area where the authors felt the Guidelines could be improved were in explaining the interactions and perils involved in connecting time-specified components in series. The authors modeled a mass-flow-rate boundary condition using a time-specified CV connected to a time-specified FL. This overspecification of boundary conditions resulted in unphysical results in the CV. The User Guidelines, in the authors' opinions, should warn of situations like these that could result in overspecified boundary conditions. Another area where the

User Guidelines could be more helpful is in explaining how to input power into a heat structure. For a model with 45 heater rods and a total power of 45 MW, for example, the values of variables "hsmult" (the number of identical heat structures being modeled) and "isrc" (the internal power source type) should be 45.0 and 1.0, respectively. It was not clear from the User Guidelines that was the case.

Some other specific changes to improve the wording in the guidelines were also suggested. The authors felt the last sentence in the introductory paragraph of Section 4, page EXEC-UG-15, "The executive has required and optional input," was not clear. The TEND description should be moved before the TIMEk description to keep the input commands in alphabetical order. Finally, in the "ZPOL" input on page CVH-UG-19, the authors believe "ZPOL" should be replaced with "VOID" in the sentence, "If absent, no pool will ever be present, and input of TPOL or ZPOL is not permitted."

A potential user pitfall was discovered in the interactive restart mechanism. If this option is chosen, the computer prompts the user, giving the time of the first found restart dump, and asking if the user wants to restart from that dump. If the user answers "N" for no, the computer presents the next dump for consideration, continuing in that manner with each answer of "N" until it presents the last dump. If the user answers "N" to this prompt, the computer cycles back to the first restart dump, but stops presenting the dump time of the selected dump, only asking the question, "DO YOU WANT TO RESTART FROM THIS DUMP (Y/N)." If the user had wanted to restart from the last dump, but had accidentally hit the RETURN key or answered "N", the user might mistakenly assume that, since no dump time was presented, the computer was still offering the last dump rather than the first dump as the point to restart from. Answering "Y" at this point would lose all the calculated results between the first and last restart dump. The Guidelines should warn of this peculiarity, and ultimately the code should be changed to avoid the problem.

A final problem reported was that in certain cases when a simulation failed to continue running, a message saying "Calculation terminated by: Executive detected abort signal - see above for reason" appeared on the computer screen. In these cases, no reason for the failure was printed on the screen or in any of the diagnostic files.

MP-1 and MP-2 Experiments

For these experiments, the best compromise between accuracy and execution time was achieved with 3 CVH components (one for each region: stub, crust, and debris) and a COR with 3 radial rings and 13 axial levels for MP-1 (15 levels for MP-2 to account for the longer stub region). Additional heat loss from the grid spacer was modeled using the core boundary conduction option on input record COR00011 to link the "other structure" component in the first axial level and outermost ring directly to the radial heat structure at that level. The intact material porosities at all axial levels was set to an artificially high 0.75 to force the fuel rods in the crust and stub regions to support particulate debris. The critical minimum thickness for zircaloy was set to 10^{-5} m compared with the MELCOR default of 10^{-4} m to allow the cladding to support conglomerate debris in the crust region. The minimum component mass and minimum mass used for temperature control were set smaller than the MELCOR defaults, as discussed above in the sensitivity studies. Parameters in the sensitivity coefficient array 1302 were set to allow UO_2 appearing in the cladding and particulate debris components to absorb fission power. By default, only UO_2 in the fuel component is allowed to absorb fission power. The properties of zircaloy were set to the values given by the MATPRO material properties package.⁴¹ The densities and thermal conductivities of UO_2 , ZrO_2 , and stainless steel were changed from the MELCOR defaults to the values in the MP-1 experimental report.³⁴

GE Large Vessel Blowdown and Level Swell Experiments

The sensitivity studies for these experiments contain a number of helpful user guidelines for modeling blowdown effects. Nonequilibrium thermodynamics are preferred to equilibrium thermodynamics. The default values for maximum pool bubble fraction and bubble rise velocity can be used. A single control volume is preferable to multiple stacked control volumes. A choked-flow discharge coefficient between 0.75 and 0.85 is preferable to the default value of 1.0. Finally, a flow loss coefficient of 1.5 best reproduced the data.

MCAP USER COMMUNITY PERSPECTIVE ON CODE DEVELOPMENT

One of the general goals of the MCAP is to develop MCAP perspectives on code development needs and priorities and submit them to the NRC to help assure that the most serious concerns are being addressed within the limited development budget. MCAP participant suggestions regarding 1994 MCAP development perspectives were solicited at the second MCAP technical meeting held April 27-29, 1994, in Bethesda, Maryland. The process for developing MCAP user community perspectives on code development is still evolving. For this second round of MCAP perspectives, the following process was followed. First, the MCAP participants prepared their list of suggestions in advance of the meeting. Second, the suggestions were collected, printed, and distributed to the meeting participants. Third, the suggestions were reviewed by MELCOR development staff members at SNL and comments were provided to the meeting participants during a discussion session. Fourth, following the meeting, the comments by the MELCOR development staff during the discussion session were summarized and incorporated into the ballot. Fifth, the ballot was sent to the MCAP participating institutions and other interested MELCOR users. The objective of the ballot was to assess the priority of the various development suggestions. Those receiving the questionnaire were encouraged to assign no more than one-third of the development suggestions a "high" priority. Finally, the results were tabulated and some general trends identified. The raw questionnaire results and some observations are reported in the remainder of this section.

Thirteen institutions responded to the questionnaire. Although, this was an encouraging response, seven fewer institutions responded this year than last. This significant response from the MELCOR user community provides some assurance that the MCAP perspectives will be carefully studied by the US NRC and the code development staff at SNL. While these user community perspectives will be considered in the development efforts, they will not be a primary source of improvement needs identification. The NRC does not expect to respond to these suggestions either singly or collectively, however, they do require the code developers to review them and to comment as appropriate.

The questionnaire form (slightly revised to condense its length) and the tabulated results are presented below. We have attempted to review the results of the balloting from several different screening perspectives. First, we have focused on those development suggestions receiving the greatest number of HIGH votes. The most HIGH votes received by any suggestion was 8 for the suggestion that a cladding ballooning model be provided. All suggestions receiving 7 or more votes were selected as having a HIGH priority by the MCAP institutions. Second, we have considered those MELCOR development suggestions receiving the highest number of HIGH and MEDIUM votes. If a suggestion received a near unanimous positive vote of either HIGH or MEDIUM priority, it was identified as a MELCOR development priority. Thus, those suggestions receiving 12 or more votes were collected in this category. Many of the suggestions satisfied both screening perspectives; thus, we believe that these suggestions have the very highest priority.

SUGGESTIONS SATISFYING BOTH SCREENING PERSPECTIVES

(e.g., received 7 or more HIGH *and* 12 or more HIGH and MEDIUM votes).

- Implement a full restart/change capability (i.e., the ability to change variables in a restart deck).
- Upgrade HS Package to permit oxidation of heat structures.
- Improve MELCOR numerics. Two specific issues were identified. See detailed description in Table 6, Nuclear Research Institute Rez, Czech Republic, Item 1.

SUGGESTIONS RECEIVING SEVEN OR MORE HIGH VOTES

- Provide the capability to model heat transfer from the pressure vessel bottom head to the cavity in the same manner as currently available for heat transfer structures.
- Change input requirements so that bottom head penetrations are not mandatory (e.g., VVER reactors have no penetrations in bottom heads).
- Provide a cladding ballooning model (received largest number of HIGH votes).

SUGGESTIONS RECEIVING TWELVE OR MORE HIGH and MEDIUM VOTES

- Alter the lower vessel head model so that failure areas and temperatures for the vessel and penetrations can be modeled separately. This would permit modeling of light water reactors having no penetrations in the lower head.
- Provide the capability to model axial heat conduction between neighboring or adjacent heat structures (e.g., sought for improved modeling of a pressure vessel bottom head).
- Provide the capability to model radiation between heat structures in the same volume.
- Revise the heat structure model so that surface temperature above and below the water level could be different (on both sides). This would also allow heat transfer to both pool and atmosphere simultaneously. Currently this is possible but NOT recommended.

We urge you to take some time to look at the raw results (Table 6) for additional insights regarding the votes. This may be particularly helpful if you have a specific interest in one of the proposals that did not meet the screening priority, e.g., a suggestion that received 6 HIGH votes or 11 HIGH and MEDIUM votes. You may wish to resubmit these development candidates with additional justification to the other MCAP participants at the 1995 Technical Meeting.

The ballot prepared this year appears to be too large. This is an outcome of having each MCAP participating institution submit 2-3 development suggestions. For the next ballot, the MCAP Coordinator will provide the opportunity for a prescreening vote during the annual technical meeting. Those suggestions of least interest to the MCAP participants following a discussion session with the SNL MELCOR development staff will be eliminated from the ballot.

TABLE 6

**1994 Ballot - Raw Questionnaire Results
MCAP Perspectives on MELCOR Code Development**

DEVELOPMENT NEED	HIGH	MEDIUM	LOW
AEA Technology			
Reduce the size of restart and plot files by providing an option to plot only user-specified variables (less than a full set).	0	3	10
Reduce the size of restart and plot files by using automatic compression/decompression algorithms on restart/plot files.	2	2	8
Implement a full restart/change capability (i.e., ability to change variables in restart deck).	7	5	1
Introduce interactive capabilities into HISPLT.	1	7	5
Report status against MELCOR Peer Review findings.	1	7	3
Brookhaven National Laboratory			
Upgrade HS Package to permit oxidation of heat structures.	7	6	
ECN			
Define Cadmium vapor pressure in the RH package.		3	6
Improve/correct implementation of "class reaction or RNRCT records"		3	6
Revise the models for condensation of steam on aerosols and the evaporation of fog so that when there is an excess of steam available for condensation on aerosols, the code does not divide the aerosol into smaller particles to permit the condensation of all steam.	4	6	2
The thermal accommodation coefficient in MELCOR is not consistent with that in CONTAIN and VICTORIA. Determine the preferred value, and implement a consistent value in MELCOR.	3	4	4
Correctly generalize the interpolation of the heat transport coefficient (or Nusselt numbers) - See Attachment 6 to minutes of 2nd MCAP Technical Meeting held April 27-29, 1994 for details.	1	6	2

DEVELOPMENT NEED	HIGH	MEDIUM	LOW
Gesellschaft fur Anlagen und Reaktorsicherheit (GRS) mbH			
1. Remove the differentiation between plot and control variables.	4	4	5
2. Modify the control function logic so that it includes the capability for specifying a delay time (i.e., a clock is started following activation of the control function but the action is delayed until the time increases to the sum of the initiating and delay times).	3	5	6
3. Alter the lower vessel head model so that failure areas and temperatures for the vessel and penetrations can be specified separately. This would permit modeling of light water reactors having no penetrations in the lower vessel head.	6	7	1
4. Provide the capability to model heat transfer from the pressure vessel bottom head to the cavity in the same manner as currently available for heat structures.	7	4	1
5. Provide the capability to model heat transfer from penetrations outside the pressure vessel bottom head (e.g., BWRs and PWRs) to the cavity in the same manner as currently available for heat structures.	3	5	3
6. Provide the capability to model axial heat conduction between neighboring or adjacent heat structures (e.g., sought for improved modeling of a pressure vessel bottom head).	3	10	
7. Provide the capability to plot the temperature of each axial node of the pressure vessel bottom head and use these values within control functions.	2	8	2

DEVELOPMENT NEED	HIGH	MEDIUM	LOW
8. Improve the failure/error messages currently written by the code in relationship to pressure vessel failure. A. Differentiate between local failures (penetrations) and global failures (bottom head). B. Currently, the message "INITIAL DIAMETER OF HOLE" is written at times when there is not ejection of melt from the vessel to the cavity. The user suggests that the message be written at the time when the vessel is breached and ejection of melt from the vessel to the cavity occurs. C. Print the message "THE INNER NODE OF VESSEL BOTTOM HEAD IN RING X REACHES FAILURE TEMPERATURE" when this circumstance occurs. D. Include the message "CORE MELT PROCESS BEGINNING IN RING X" at such time as the fuel reaches the melting temperature in any node of ring X and melt relocation begins. E. Write the message "OUTER RINGS BLOCKED/INNER NOT" in the <u>message</u> file as well as the diagnostic file.	2	8	3
GPU NUCLEAR			
1. RN Package: Add ability to input plant specific radionuclide inventory using the default elements and default decay tables. Basis: Save user time by not requiring the input of information already in the code.	2	4	5
2. Permit renodalization on restarts. Basis: this helps the user capture important phenomena that require more detailed nodding, particularly when such phenomena last a short time.	4	4	5
3. Provide the capability to start a code calculation post scram, e.g., one day after trip, which will assist the user to calculate post-shutdown scenarios.		3	8
4. Improve the calculational efficiency of the CVH package (too slow).	4	7	1
5. During core degradation and relocation, the code often fails on minimum time step ($<10^{-5}$ s) because there is too much heat available to the atmosphere on successive time steps. Address this problem.	3	8	1
6. Develop core spray model for BWRs [Note: this recommendation was made in a letter from GPU Nuclear to F. Eltawila dated August 25, 1994 and was not, therefore, available for comment by SNL at the time the other recommendations were provided.	1	2	8

DEVELOPMENT NEED	HIGH	MEDIUM	LOW
Japan Atomic Energy Research Institute			
1. The heat transfer coefficient for in-vessel debris falling into a pool has a strong effect on accident progression. A recommended value (other than zero) should be included in the COR Users Guide.	4	5	3
2. The solid debris ejection model and radial relocation model also have a strong effect on severe accident progression. Guidance should be provided about whether these models should be used for best-estimate analysis.	2	8	2
3. Keep previous models as an option in the next update so that comparative calculations can be made with the updated code version.	1	3	9
4. Reduce the fraction of total CPU time spent in the CVH package (User notes that current MELCOR uses too much CPU time for probabilistic safety analysis).	5	5	1
5. In addition to the source and sample input, include output of sample problems with the code release package so that the adequacy of the code installation can be verified.	3	6	1
KAERI			
1. Review the adiabatic expansion model during the blowdown phase of an accident. The code seems to improperly handle the rapid drop of coolant and accumulator temperature during this phase. The calculation fails when the control volume temperature is too low.	1	5	6
2. When an attempt was made to simulate the global failure of the reactor vessel lower head, the calculated temperature in the lower plenum did not seem physically reasonable following the failure. This outcome should be investigated. In a certain case, hydrogen burning occurred in the lower plenum. There was no oxygen flow from the cavity to the lower plenum, but the burn criteria was satisfied.	5	6	1
3. See if the time-step control can be improved. The time interval associated with improvement number 2 was given as an example.	2	4	5

DEVELOPMENT NEED	HIGH	MEDIUM	LOW
KEMA (Netherlands)			
1. dT/dz model: In MELCOR 1.8.1, the dT/dz model caused unreasonably low (freezing) and high ($T > 3000K$) temperatures in the core region. Correct the model. In MELCOR 1.8.2 the recommendation of one (core shroud) heat structure per core level also corrected the functioning of the dT/dz model, but a modified dT/dz model is preferred by the user.	5	4	3
2. CORCON: In several MELCOR 1.8.1 calculations with MCCI, the CORCON package showed a problem. Some time after the start of MCCI, the density of steel is similar to that of the heavy oxides (HOX), a layer flip is modeled, and the HOX transfer into LOX. After some time, however, the LOX transfers back to HOX. This flipping of the layers continues until almost all the steel is reacted. Model improvement is recommended.	4	3	2
3. Critical flow: Based upon a code-to-code benchmark with a TRAC calculation of a feedwater line rupture, it is recommended that the critical flow model be improved. MELCOR calculated an initial flow of approximately 1500 kg/s while the TRAC value was 350 kg/s and the character of the change flow was smoother.	3	6	2
4. Material properties: Using a delta function for material properties (e.g., in the melting/solidifying region) can cause problems. If time step is too large, it is possible that the delta peak is missed and the integral function is not corrected. MELCOR does not check on this behavior. It is recommended that such a check be instituted.		6	5
5. Core modeling: To simulate the quality of the water around an 8-level core model, the one surrounding CVH volume contains part water and part steam. MELCOR models, however, the water in the pool. During steady state calculations, the top of the core is always uncovered. Separate CVH volumes, one per core level, can be used but this is costly in CPU time. A new model is preferred.	2	4	6

DEVELOPMENT NEED		HIGH	MEDIUM	LOW
KfK/INR				
(All suggestions relative to modeling of two-phase flow in a duct with a component)				
1. The 1-D momentum equations (including the unsteady terms) of the two phases should refer to the fully open cross-sectional area of a flow path in all cases where the length of the flow path is considerably larger than the component dimension. This facilitates a more physical representation of two-phase flow in a channel, e.g., the momentum exchange function and the wall friction can be formulated using channel void fractions and velocities (and not the corresponding values in the local restriction).	2	4	4	
2. For valves with time-varying area fractions, introduce time-dependent hydraulic loss coefficients K, e.g., as MELCOR control functions. If the flow is computed for the unobstructed cross-sectional area, as suggested above, the calculation needs only K as input, which is directly controlled by the fraction open. Inconsistencies between the area fraction open and the K value can be avoided by the approach. Complex components need the hydraulic loss coefficient K as a function of the flow state.	2	6	4	
3. Introduce momentum exchange functions representing a range of flow regimes, e.g., use a quadratic dependence on the phasic velocity difference, which better represents, in general, the experiments for obstructed and unobstructed flows.	1	5	4	
4. Introduce simple flow regime modeling, which allows the code to determine the momentum coupling coefficients in the improved exchange function for a given flow on a rational basis.		5	5	
KFKI/AEKI - Hungary				
1. Provide a HSTRACE capability similar to the CVHTRACE capability currently in the code. In some applications the number of heat structures exceeds the number of control volumes and it is difficult to identify the heat structures involved in a code failure.		6	7	

DEVELOPMENT NEED	HIGH	MEDIUM	LOW
2. Provide sensitivity coefficients specific to identified heat structures. In some cases the effect can be provided with control functions, but the characteristic heat transfer changes several times and control functions cannot be used. The capability would be used, for example, in modeling steam generators.	2	7	3
3. Provide the capability for modeling more than a single core plate. Basis: there are designs where more core plates are assumed for the proper simulation of the core degradation, e.g., VVER-440 reactor).	1	6	6
4. Provide the capability to model radiation between heat structures in the same volume.	5	7	
5. Provide a new control function that can be used to solve a user-defined differential equation. The example given is a dynamic simulation of the main circulation pump. A second example given was the simulation of a controller in a safety system.	1	3	8
6. Provide user guidelines about the acceptable limits for various sensitivity coefficients.	6	3	3
7. Provide user guidelines about the proper use of a mass source in the CVH package. There are some hints in the CORA13 assessment report but more information is needed.	2	3	7
8. Provide more information about the debugger version of MELCOR. Debugger was not usable because of memory requirements.		3	9
Nuclear Research Institute Rez, Czech Republic See Attachment 6 to minutes of 2nd MCAP Technical Meeting held April 27-29, 1994, for figures provided with original discussion.			
1. Improvement of numerics (User has encountered two major problems, which are believed to be related to numerics) A. MELCOR sometimes fails when there is very small amount of atmosphere in a control volume and a flow path is connected to the top. Reflood of pressurizer surge line is one of the examples I have seen recently. There must be some singularity in numerics.	8	5	

DEVELOPMENT NEED	HIGH	MEDIUM	LOW
B. There is a problem with the loop seal on hot legs of VVER 440 primary system. To model this, at least two control volumes have to be used in MELCOR (see following figure). After LOCA there is only steam in the first volume (CV 1) and water column in the second one (CV 2). Since there is only little pressure difference on the steam/water interface, the flow direction is unstable and it may change several times in each time step. This is probably the cause of the terrible slow down of calculation speed during this period of accident.	2	9	2
2. Change of heat structure model so that surface temperature above and below water level could be different (on both sides). This would also allow heat transfer to both pool and atmosphere simultaneously. Currently it is possible but NOT recommended.	6	7	
3. Change of heat structure package model to allow definition of a bundle of (cylindrical) horizontal heat structures. This would allow (together with above mentioned suggestion) to better model horizontal heat exchangers, e.g. VVER steam generators.		4	8
4. Inclusion of external mass sources and sinks to calculation of volume-centered velocity. It would be more realistic and it might save control volumes and flow paths sometimes. Currently only flow path mass flows are taken into account. The user may not define external mass source directly to a target control volume if the user needs to correct the heat transfer to/from heat structures in this control volume. The user must define one more CV and flow path instead. See following examples, which are quite different from the heat transfer to walls point of view even if they may be equivalent from the mass flow point of view.		9	3
5. Take into account differences in partial pressures of various materials in the atmosphere when calculating mass flows. Currently, if neighboring control volumes have equal total pressure, there is no flow between them, even if concentration of some material differs significantly.	1	4	6
6. Change input requirements so that bottom head penetrations were not mandatory. (VVER reactors have no penetrations in bottom heads).	7	4	2

DEVELOPMENT NEED	HIGH	MEDIUM	LOW
7. Change COR package to allow canister to be defined also for PWRs. It would facilitate MELCOR calculations of VVER 440 reactors, which have canisters (and bypass) even if they are PWRs.	2	5	6
8. Change of special edit control function treatment. Currently the edit is written whenever the control function is .TRUE. Consider writing the edit when this control function changes its value.	2	3	7
Ruhr - Universitat Bochum			
1. Provide for local modification of the heat transfer coefficient for special flow situations. Specific example was the desire to modify the coefficient in the region of the CORA inlet cross flow but the sensitivity coefficients act on the total bundle length.	3	5	5
2. Improve flexibility of the electrical heater rod model so that ease of application to different experiments will be increased. Currently, changes have to be made by modifying the coding. Provide documentation of electrical heater rod model.	1	3	9
3. Base ZrO ₂ oxide crust removal model on mechanistic approach. A two-stage effort is proposed. The first stage would permit changes via the input deck. The second stage would be to implement and document the more mechanistic model. Provide documentation of the ZrO ₂ oxide crust removal model.	6	5	1
4. Provide a clad ballooning model.	8	2	2
TRACTEBEL			
1. Improve MELCOR documentation A. Specified model descriptions B. Specified correlation descriptions C. Provide guidance for preparing an input deck and recommended PWR values D. Provide guidance for parameters having defaults that are <u>not</u> recommended E. Provide examples in the User's Guide that are representative of a nuclear power plant F. Provide a programmer's guide	4	4	4
2. Replace default model that assumes a stratified flow through junctions with a flow map		5	6
3. Some variables can only be plotted while others can be used as control functions.	3	6	4

DEVELOPMENT NEED	HIGH	MEDIUM	LOW
4. Improve guidance regarding choice of junction location and opening height, as these modeling choices can strongly influence the results and required CPU time.	5	5	3

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Fig. 1.
Temperatures in the Middle Ice Condenser for Experiment 11-6 of the
PNL Ice Condenser Experiment.

Fig. 2.
Temperatures in the Bottom Ice Condenser for Experiment 16-11 of the
PNL Ice Condenser Experiment.

Fig. 3.
Particle Retention for Experiment 11-6 of the PNL Ice Condenser Experiment.

Fig. 4.
Particle Retention for Experiment 16-11 of the PNL Ice Condenser Experiment

Fig. 5.
MELCOR 1.8.1 Steam Dome Pressures
for DEMO problem.

Fig. 6.
MELCOR 1.8.2 Steam Dome Pressures
for DEMO problem.

Fig. 7.
MELCOR 1.8.2 Double Precision Steam
Dome Pressures for DEMO problem.

Fig. 8.
Cross Section of DF-4 Test Bundle.

Fig. 9.
DF-4 Control Blade Temperatures at the 36.8 cm Level.

Fig. 10.
DF-4 Cladding Temperatures at the 36.8 cm Level.

Fig. 11.
DF-4 Cladding Temperatures at the 9.6 cm Level.

Fig. 12.
DF-4 Hydrogen Production.

Fig. 13.
SNL/IET Vessel Pressures.

Fig. 14.
MIST 3109AA Break and HPI Flow Rates.

Fig. 15.
MIST 3109AA Primary and Secondary Pressures.

Fig. 16.
MIST 3109AA A- and B-Loop Hot-Leg Collapsed Liquid Level.

Fig. 17.
MIST 3109AA TRAC, MELCOR, and Data Primary Pressure.

Fig. 18.
MIST 3404AA Vessel Collapsed Liquid Level.

Fig. 19.
MIST 3404AA Hot Leg Liquid Level.

Fig. 20.
MIST 3404AA MELCOR, TRAC, and Data Primary Pressure.

Fig. 21.
MIST 3404AA HPI and Break Flow.

Fig. 22.
MIST 3404AA Loop A Total Flow.

Fig. 23.
MIST 3404AA Loop B Total Flow.

Fig. 24.
MELCOR 1.8.1 and 1.8.2 Mass of Hydrogen Burned in Cavity of Surry PWR TMLB'.

Fig. 25.
MELCOR 1.8.1 and 1.8.2 Number of Hydrogen Burns in Cavity of Surry PWR TMLB'.

Fig. 26.
MELCOR 1.8.1 and 1.8.2 Surry PWR TMLB' Hydrogen Production.

Fig. 27.
MELCOR 1.8.2 Surry PWR TMLB' Containment Pressure.

Fig. 28.
MP-1 Debris Temperatures at Axial Level 11.

Fig. 29.
MP-1 Crust Temperatures in Ring 1.

Fig. 30.
MP-1 Stub Temperatures in Ring 1.

Fig. 31.
MP-2 Debris Temperatures at Axial Level 10.

Fig. 32.
GE Large Vessel Top Blowdown Pressures.

Fig. 33.
GE Large Vessel Top Blowdown Two-Phase Levels.

Fig. 34.
GE Large Vessel Bottom Blowdown Pressures.

Fig. 35.
GE Large Vessel Bottom Blowdown Two-Phase Levels.

Fig. 36.
MELCOR 1.8.1 and 1.8.2 Surry PWR TMLB' Pressurizer Liquid Levels.

Fig. 37.
Particle Retentions at Different Maximum Time Steps for Experiment 11-6 of the PNL Ice Condenser Experiment.

APPENDIX A
MCAP ASSESSMENT MATRIX

MELCOR ASSESSMENT MATRIX - Part I
Fundamental Assessments
 April 1995

Reference	Focus	Brief Abstract
1	Comparison of MELCOR 1.8.1 to analytical solutions	The MELCOR code has been used to simulate a number of simple gedanken (thought) problems, with results compared with analytical solutions. Problems described in Ref. 1 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. Sensitivity studies examined time-step effects, machine dependencies, nodding, and code options.
2	Platform studies with MELCOR 1.8.2	Calculations for a demonstration problem were performed on various workstations with various compiler optimization levels. The demonstration problem is a simplified, 5 control volume model of a BWR containment system. Calculations performed with MELCOR versions 1.8.1 and 1.8.2 were compared. MELCOR 1.8.2 shows improved transportability among workstations.
3	Basic studies of MELCOR radionuclide package.	The MELCOR code (version not identified) was used to compare results from small demonstration calculations with direct solutions of the same problem and to demonstrate that the models yielded the expected results. The focus of this work was on the radionuclide package. Calculations included fission product release from the core, fission product aerosol deposition, fission product vapor condensation, and revaporization. Code problems identified during the effort were minor.

MELCOR ASSESSMENT MATRIX - Part I (continued)
Fundamental Assessments
 April 1995

4	Compare MELCOR results to analytical and experimental results for two-phase flow through ducts.	The goals of the work were to (1) investigate flow rates and related pressure drops for stationary two-phase flow through components, (2) generalize single-phase methods, (3) perform MELCOR calculations using input parameters justified by physics, (4) compare MELCOR results to analytical and experimental results, and (5) suggest guidelines for MELCOR calculations and possible code improvements to better represent two-phase mass flow rates and pressure drops across duct components.
5	The impact of selected heat and mass transfer correlations on results of key severe accident signatures calculated by MELCOR were assessed.	Concerns about the heat and mass transfer correlations in MELCOR were raised during the MELCOR Peer Review. The existing MELCOR heat and mass transfer correlations were reviewed. New or improved correlations were identified and were test-implemented in MELCOR. Sensitivity studies were performed for two accident scenarios using a MELCOR model of Surry. Sensitivities to (1) maximum time step, (2) free and forced convection correlations in the HS Package, (3) boiling correlations in the COR package, (4) debris bed heat transfer correlation in the COR Package, and (5) condensation heat transfer correlations in the HS Package are discussed.
6	MELCOR and RELAP5/SCDAP calculations of SDS valve tests conducted at the Wylie Laboratory are presented.	The capability of MELCOR 1.8.2 to predict valve flow rates was assessed using data from a full flow steam compatibility flow test conducted at Wyle Laboratory in 1993. Lessons learned during the assessment effort are listed.

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MELCOR ASSESSMENT MATRIX - Part II

April 1995

IN-VESSEL TESTS

CORA	CRL anneal exp.	DC in ACRR	DF in ACRR	EJET	FALCON
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IN-VESSEL PHENOMENA

Primary System Thermalhydraulics (see Note 1)						
Fuel Rod Heatup and Ballooning (see Note 2)	X			X		
In-Core Metal/Water Reactions and Materials Interactions (see Note 3)	X			X		
In-Vessel Melt Behavior (see Note 4)	X			X		
In-Vessel Degraded Core Geometry and Debris Characterization (see Note 5)	X		X	X		
Fuel/Coolant Interaction (see Note 6)					X	
In-Vessel Fission Product or Aerosol Release (see Note 7)		X				X
Fission Product/Aerosol Transport in the Reactor Coolant System (see Note 8)						X
Reactor Coolant System Failure (Note 9)						

MELCOR Assessment Status and Reference	Ref. C-1/P Ref. C-3/C Ref. C-4/C Ref. C-5/IP Ref. C-6/IP			Ref. D-2/C Ref. D-1/P		Ref. F-4/C Ref. F-10/C
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Ref. X/Y where X is the reference number, Y=C denotes completed, Y=P denotes planned, and Y=IP denotes in progress.

IN-VESSEL TESTS (continued)

FARO	FIST	FLECHT- SEASET	FLHT in NRU	HEVA	LOFT
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IN-VESSEL PHENOMENA

Primary System Thermalhydraulics (see Note 1)		X	X			X
Fuel Rod Heatup and Ballooning (see Note 2)				X		
In-Core Metal/Water Reactions and Materials Interactions (see Note 3)				X		
In-Vessel Melt Behavior (see Note 4)				X		
In-Vessel Degraded Core Geometry and Debris Characterization (see Note 5)	X			X		
Fuel/Coolant Interaction (see Note 6)						
In-Vessel Fission Product or Aerosol Release (see Note 7)					X	
Fission Product/Aerosol Transport in the Reactor Coolant System (see Note 8)						
Reactor Coolant System Failure (Note 9)						

MELCOR Assessment Status and Reference		Ref. F-9/C	Ref. F-1/C	Ref. F-2/C Ref. F-3/C Ref. F-5/C Ref. F-6/P Ref. F-7/P Ref. F-11/IP		Ref. L-3/C
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Ref. X/Y where X is the reference number, Y=C denotes completed, Y=P denotes planned, and Y=IP denotes in progress.

IN-VESSEL TESTS (continued)

LOFT FP-2	MAGICO	MIST	MP in ACRR	PACTEL natural circulation	PBF - SFD
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IN-VESSEL PHENOMENA

Primary System Thermalhydraulics (see Note 1)	X		X		X	
Fuel Rod Heatup and Ballooning (see Note 2)	X					X
In-Core Metal/Water Reactions and Materials Interactions (see Note 3)	X					X
In-Vessel Melt Behavior (see Note 4)	X					X
In-Vessel Degraded Core Geometry and Debris Characterization (see Note 5)				X		X
Fuel/Coolant Interaction (see Note 6)		X				
In-Vessel Fission Product or Aerosol Release (see Note 7)						X
Fission Product/Aerosol Transport in the Reactor Coolant System (see Note 8)						X
Reactor Coolant System Failure (Note 9)						

MELCOR Assessment Status and Reference	Ref. L-3/C Ref. L-2/P		Ref. M-1/C Ref. M-2/C	Ref. M-3/C	Ref. P-1/P Ref. P-11/IP	Refs. P-2/C Ref. P-3/C
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Ref. X/Y where X is the reference number, Y=C denotes completed, Y=P denotes planned, and Y=IP denotes in progress.

IN-VESSEL TESTS (continued)

PHEBUS-FP	PHEBUS-SFD	PMK	Semiscale	SIGMA	ST in ACRR
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IN-VESSEL PHENOMENA

Primary System Thermalhydraulics (see Note 1)			X	X		
Fuel Rod Heatup and Ballooning (see Note 2)		X				
In-Core Metal/Water Reactions and Materials Interactions (see Note 3)		X				
In-Vessel Melt Behavior (see Note 4)		X				
In-Vessel Degraded Core Geometry and Debris Characterization (see Note 5)		X				
Fuel/Coolant Interaction (see Note 6)					X	
In-Vessel Fission Product or Aerosol Release (see Note 7)	X					X
Fission Product/Aerosol Transport in the Reactor Coolant System (see Note 8)	X					
Reactor Coolant System Failure (Note 9)						

MELCOR Assessment Status and Reference	Ref. P-4/P Ref. P-8/P Ref. P-9/IP Planned US NRC (SNL)	Ref. P-5/C Ref. P-6/C Ref. P-7/C	P-10/C	Ref. S-3/C		Ref. S-2/C
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Ref. X/Y where X is the reference number, Y=C denotes completed, Y=P denotes planned, and Y=IP denotes in progress.

IN-VESSEL TESTS (continued)

STEP	TMI-2	VI at ORNL	GE Large Vessel Swell		
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IN-VESSEL PHENOMENA

Primary System Thermalhydraulics (see Note 1)		X		X		
Fuel Rod Heatup and Ballooning (see Note 2)		X				
In-Core Metal/Water Reactions and Materials Interactions (see Note 3)		X				
In-Vessel Melt Behavior (see Note 4)		X				
In-Vessel Degraded Core Geometry and Debris Characterization (see Note 5)		X				
Fuel/Coolant Interaction (see Note 6)						
In-Vessel Fission Product or Aerosol Release (see Note 7)	X	X	X			
Fission Product/Aerosol Transport in the Reactor Coolant System (see Note 8)		X	X			
Reactor Coolant System Failure (Note 9)						

MELCOR Assessment Status and Reference		Ref. T-1/C	Ref. V-1/C Ref. V-2/C	Ref. G-1/C		
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Ref. X/Y where X is the reference number, Y=C denotes completed, Y=P denotes planned, and Y=IP denotes in progress.

Detailed Phenomena Lists

- Note 1: Forced circulation; natural circulation; pressure differences; evaporation; condensation; break flow; fluid-structure heat transfer.
- Note 2: Conduction; fluid-structure heat transfer, fission product generation; clad swelling.
- Note 3: Zircaloy clad oxidation; stainless steel clad oxidation; structural steel oxidation; grid spacers oxidation; fuel channel/control blade oxidation; impact of reducing/oxidizing atmosphere; impact of geometry changes; impact of clad rupture; steam availability/starvation; fuel dissolution; eutectics formation; thermal interactions/melting.
- Note 4: Melt relocation modes (candling, slumping, melt paths); control material relocation; spacer grid material relocation; Zircaloy relocation; stainless steel relocation/ fuel/eutectic relocation; blockage formation//breakup; melt freezing/crust formation/remelting; molten pool behavior.
- Note 5: Fuel pins; channel wall/blades; lower core structures; upper core structures; core baffle/barrel/thermal shield; debris bed formation; debris composition; debris geometry; porosity; pools debris coolability; debris oxidation; debris/structure interaction.
- Note 6: Premixing; triggering; propagation; expansion; geometry effects; pressure effects.
- Note 7: Clad failure release; fuel pellet fragmentation release; molten fuel release; debris bed release; chemical interactions; chemical specie; physical form; control material release; other core material release.
- Note 8: Vapor condensation on aerosols; vapor condensation on structures, chemical reactions, diffusion, agglomeration (Brownian, turbulent, gravitational); turbulent impaction/inertial effects; chemical reactions with structures; revolatilization; resuspension; effect of decay heat.
- Note 9: Lower head thermal attack; lower head chemical attack; lower head ablation; lower head failure; effect of penetrations; piping failure; upper head thermal failure; explosion induced failure.

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MELCOR ASSESSMENT MATRIX - Part III
April 1995

EX-VESSEL TESTS

ABCOVE	ACE	ATT in Marviken-V	Battelle- Frankfurt	Battelle Model Containment	BETA
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EX-VESSEL PHENOMENA

Core Release/Ejection from Vessel or Behavior in Reactor Cavity (Note 1)						
Direct Containment Heating (Note 2)						
Containment Response (Note 3)						
Pool Scrubbing Phenomena (Note 4)						
Corium/Concrete/Water Interactions (Note 5)		X				X
Hydrogen/CO Phenomena in Containment (Note 6)				X	X	
Aerosol/Fission Product Phenomena in Containment (Note 7)	X		X			
Containment Leak and Failure (Note 8)						

MELCOR Assessment Status and Reference	Ref. A-1/C	Ref. A-3/P	Ref. A-4/C	Ref. B-1/C	Ref. B-2/C	
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Ref. X/Y where X is the reference number, Y=C denotes completed, Y=P denotes planned, and Y=IP denotes in progress.

EX-VESSEL TESTS (continued)

DCH in 1:40 CWTI (ANL)	DCH in 1:10 Surtsey (SNL)	Demonia	EPSI (Japan)	FITS (SNL)	HDR
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EX-VESSEL PHENOMENA

Core Release/Ejection from Vessel or Behavior in Reactor Cavity (Note 1)	X	X				
Direct Containment Heating (Note 2)	X	X				
Containment Response (Note 3)	X	X				X
Pool Scrubbing Phenomena (Note 4)				X		
Corium/Concrete/Water Interactions (Note 5)						
Hydrogen/CO Phenomena in Containment (Note 6)	X	X			X	X
Aerosol/Fission Product Phenomena in Containment (Note 7)	X	X	X			
Containment Leak and Failure (Note 8)						

MELCOR Assessment Status and Reference	Ref. D-2/C	Ref. D-2/C	Ref. D-1/C	Ref. E-1/P		Refs.H-1/C Ref. H-2/C Ref. H-3/C Ref. H-6/P
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Ref. X/Y where X is the reference number, Y=C denotes completed, Y=P denotes planned, and Y=IP denotes in progress.

EX-VESSEL TESTS (continued)

HYP (ANL)	LACE	LFP in 1:10 Surtsey	MACE	NUPEC hydrogen tests	PNL Ice Condenser
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EX-VESSEL PHENOMENA

Core Release/Ejection from Vessel or Behavior in Reactor Cavity (Note 1)			X	X		
Direct Containment Heating (Note 2)			X			
Containment Response (Note 3)			X			
Pool Scrubbing Phenomena (Note 4)						
Corium/Concrete/Water Interactions (Note 5)				X		X
Hydrogen/CO Phenomena in Containment (Note 6)			X		X	
Aerosol/Fission Product Phenomena in Containment (Note 7)		X				
Containment Leak and Failure (Note 8)				X		

MELCOR Assessment Status and Reference		Ref. L-1/C Ref. L-2/C LA1/P LA3/P LA5/P LA6/P (SNL)		Ref. M-1/P	Ref. N-1/P Ref. N-2/IP	Ref. P-1/C
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Ref. X/Y where X is the reference number, Y=C denotes completed, Y=P denotes planned, and Y=IP denotes in progress.

EX-VESSEL TESTS (continued)

Sizewell 1:25 Reactor Cavity	SNL 1:6 Scale Containment	SPARTA	SURC	Surry 1:42 Reactor Cavity (BNL)	SWISS-2
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EX-VESSEL PHENOMENA

Core Release/Ejection from Vessel or Behavior in Reactor Cavity (Note 1)	X				X	
Direct Containment Heating (Note 2)						
Containment Response (Note 3)						
Pool Scrubbing Phenomena (Note 4)			X			
Corium/Concrete/Water Interactions (Note 5)				X		X
Hydrogen/CO Phenomena in Containment (Note 6)				X		
Aerosol/Fission Product Phenomena in Containment (Note 7)				X		
Containment Leak and Failure (Note 8)		X				

MELCOR Assessment Status and Reference				Ref. S-1/IP		Ref. S-3/P
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Ref. X/Y where X is the reference number, Y=C denotes completed, Y=P denotes planned, and Y=IP denotes in progress.

EX-VESSEL TESTS (continued)

VI at ORNL	Watts Bar 1:42 Reactor Cavity (BNL)	WC in 1:10 Surtsey	WETCOR	Zion 1:10 Reactor Cavity	Zion 1:42 Reactor Cavity
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EX-VESSEL PHENOMENA

Core Release/Ejection from Vessel or Behavior in Reactor Cavity (Note 1)		X	X	X	X	X
Direct Containment Heating (Note 2)			X			
Containment Response (Note 3)						
Pool Scrubbing Phenomena (Note 4)						
Corium/Concrete/Water Interactions (Note 5)				X		
Hydrogen/CO Phenomena in Containment (Note 6)			X			
Aerosol/Fission Product Phenomena in Containment (Note 7)	X		X			
Containment Leak and Failure (Note 8)						

MELCOR Assessment Status and Reference	Planned US NRC (ORNL)					
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Ref. X/Y where X is the reference number, Y=C denotes completed, Y=P denotes planned, and Y=IP denotes in progress.

MELCOR ASSESSMENT MATRIX - Part IV
April 1995

PLANT CALCULATIONS

Plant/Sequence	Reference	Comments
PWR AB BWR Station Blackout	1	Planned
VVER type 440/type 213	2	Planned
PWR Station Blackout (TMLB') with DCH	3	Complete
MELCOR Applications to Power Upgrade Studies for a BWR/6 Mark-III Nuclear Power Plant	4	Planned.
<u>Plants</u> C-E 2-loop PWRs: Yong-gwang 3&4 for total loss of feedwater) C-E 2-loop PWRs: Ulchin 3&4 for Station Blackout (TMLB')	5a 5b	Planned, Will include code-to-code comparisons
PWR Station Blackout (Oconee)	6,7	Complete
Peach Bottom	6,7	Complete
B&W Plant	8,9	Complete
W-PWR 900 MWe, 3-loop	10	Complete
N-Reactor	11	Complete
BWR	12	Complete
PWR	13	Complete
Grand Gulf, PWR	14,16	Complete
LaSalle County Unit 2	15	Complete
BWR-4	17	Complete
Brown's Ferry BWR	18	Complete
TVO Power Plant	19,20	Complete
Muhleberg, GE BWR/4 (Mark-I double Torus)	21	Complete

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PLANT CALCULATIONS

Plant/Sequence	Reference	Comments
Beznau, W 2-loop PWR (Large Dry Containment)	22	Complete
Gosgen, Siemens-KWU 3-loop PWR (Large Dry Containment)	23	Complete
Point Beach, 2-Loop PWR	24	Complete
PWR Station Blackout (TMLB')	25	Complete
C-E Calvert Cliffs	26	Complete
TMI-1 Seal LOCA	27	Complete
Surry Air Ingression Studies	28, 29	Complete
ABWR	30	Complete
ALWR PCCS Studies	31	In Progress/Presented MCAP mtg.
German PWR	32	In Progress/Presented MCAP mtg.
BWR Mark I	33	In Progress/Presented MCAP mtg.
Forsmark 3 and TVO I/II BWRs	34	In Progress/Presented CSARP mtg.
VVER-440/213	35	In Progress/Presented CSARP mtg.

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APPENDIX B

CODE ASSESSMENT DESCRIPTOR DEFINITIONS

The descriptors will be used to provide an overall characterization of how MELCOR predicted the phenomenological behavior in the experimental facility. Four descriptors are used to characterize the degree of agreement and the application consequences of either the agreement or lack of agreement. The four descriptors are excellent, reasonable, minimal, and insufficient agreement. Each of these descriptions will be defined below, along with the consequences for future application of the code in the given area being characterized and the perceived need for additional code development.

Excellent agreement applies when the code exhibits no deficiencies in modeling a given behavior. Major and minor phenomena and trends are correctly predicted. The calculated results are judged to agree closely with the data. The calculation will, with few exceptions, lie within the uncertainty band of the data. The code may be used with confidence in similar applications. The term "major phenomena" refers to the phenomena that influence key parameters, such as rod cladding temperature, pressure, differential pressure, mass flow rate, and mass distribution. Predicting major trends means that the prediction shows the significant features of the data. Significant features include the magnitude of a given parameter through the transient, slopes, and inflection points that mark significant changes in the parameter.

Reasonable agreement applies when the code exhibits minor deficiencies. Overall, the code provides an acceptable prediction. All major trends and phenomena are correctly predicted. Differences between calculation and data are greater than deemed necessary for excellent agreement. The calculation will frequently lie outside but near the uncertainty band. However, the correct conclusions about trends and phenomena would be reached if the code were used in similar applications. The code models and/or facility noding model should be reviewed to see if improvements can be made.

Minimal agreement applies when the code exhibits significant deficiencies. Overall, the code provides a prediction that is only conditionally acceptable. Some major trends or phenomena are not predicted correctly, and some calculated values lie considerably outside the uncertainty bands of the data. Incorrect conclusions about trends and phenomena may be reached if the code were used in similar applications, and an appropriate warning needs to be issued to users. Selected code models and/or facility noding need to be reviewed, modified, and assessed before the code can be used with confidence in similar applications. Assessment results should be forwarded to the NRC, with a recommendation that the NRC have the code developer review the user's input and calculations to confirm whether the NRC, should direct that a user's warning be issued. Pending that outcome a warning may be issued to the MELCOR community stating that users applying the code in similar applications risk drawing incorrect conclusions. If the NRC issues such a warning, the warning should stay in force until the identified review, modification, and limited assessment activities are completed and the resultant characterization descriptor is "reasonable" or better.

Insufficient agreement applies when the code exhibits major deficiencies. The code provides an unacceptable prediction of the test. Major trends are not predicted correctly. Most calculated values lie considerably outside the uncertainty bands of the data. Incorrect conclusions about trends and phenomena are probable if the code is used in similar applications, and an appropriate warning needs to be issued to users. Selected code models and/or facility noding need to be reviewed, modified, and assessed before the code can be used with confidence in similar applications. Assessment results should be forwarded to

the NRC, with a recommendation that the NRC have the code developer review the user's input and calculations to confirm whether the NRC should direct that a user's warning be issued. Pending that outcome, a warning may be issued to the MELCOR community stating that users applying the code in similar applications risk drawing incorrect conclusions. If the NRC issues such a warning, the warning should stay in force until the identified review, modification, and limited assessment activities are completed and the resultant characterization descriptor is "reasonable" or better.