SASA MONTHLY

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PROJECT TITLE: Severe Accident Sequence Analysis (SASA)

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TECHNICAL HIGHLIGHTS:

BWR-LTAS calculations are being performed in support of the Browns Ferry Loss of Control Air accident sequence study. MARCON 2.0B calculations are being performed for the Browns Ferry Station Blackout accident sequence. Checkout of the MELRPI code on the ORNL computer system continues. (This code permits study of the core response after significant degradation of the original geometry.) MELRPI calculations will also be performed for the Station Blackout accident sequence. The fission product transport analyses group is ready to perform the fission product transport calculations as soon as the thermal-hydraulic results from MARCON 2.0B and MELRPI are available.

The personnel contributing to the SASA effort at ORNL are divided into three working groups. The individual group reports for progress durin[redacted]re presented below with a brief initial statement of the purpose of each group.

Group I: (R. M. Harrington) Determines and analyzes the events of the accident sequence that would occur prior to core uncovery, using the ORNL-developed simulation program BWR-LTAS to study the plant response to operator actions.

During an hours-long accident sequence after reactor shutdown and MSIV closure, all decay heat steam discharged from the reactor vessel by the safety relief valves (SRVs) would have to be condensed in the pressure suppression pool. The pool cooling mode of the RHR system provides four heat exchangers to remove decay heat from the pool--more than enough cooling to prevent excessive pool temperatures. For the Loss of Control Air accident sequence, one or more of the RHR coolers might not be available. Two BWR-LTAS runs were completed to help determine if one RHR pool cooler would be sufficient to prevent pool temperature from exceeding the threshold that would result in inadequate RHR pump net positive suction head (NPSH). This threshold pool temperature depends on the total pressure of the wetwell atmosphere: at atmospheric pressure (i.e., no containment backpressure), the pool temperature should not exceed 190°F, whereas the corresponding limit for 4 psig of backpressure is more than 200°F.

The first BWR-LTAS run simulated 4 hours at hot standby (full reactor pressure), followed by depressurization at the normal operational rate (100 F/h) to a target pressure of 100 psia. Reactor vessel injection needs were supplied from the condensate storage tank by the steam-driven RCIC system. The peak pool temperature for this transient (182°F occurring at 12 h after shutdown from full power) would not compromise the minimum required NPSH for the RHR pump even without

#1

containment backpressure. Considering the 10 psig containment pressure calculated at 12 h, the NPSH would be well in excess of any reasonable estimate of the actual minimum requirement.

The second BWR-LTAS run simulated just 10 minutes at hot standby, followed by an emergency depressurization to about 50 psia by manual opening of four SRVs. Reactor vessel injection needs were supplied from the pressure suppression pool by intermittent use of one RHR system pump. The peak pool temperature predicted for this transient is 185°F after 14 h--higher by 3°F than that of the first run because of the depressurization to a lower vessel pressure and because the pool mass remains approximately constant due to the use of the pool as the source for injection to the reactor vessel.

In order to assure the quality of the BWR-LTAS calculation of the 185°F peak pool temperature, a closed-form analytical solution for the long-term pool temperature response was developed. This is possible because of the steady-state nature of the reactor vessel response during the long, slow pool heatup. Input to the analytical model was the same as for the BWR-LTAS run, including the use of a heat exchanger cooling coefficient of 234 Btu/s*F [10,000 gpm shell-side heat exchanger flow and 4500 gpm tube-side secondary (river) flow]. This coefficient was calculated from basic information provided by the TVA for the RHR heat exchangers, including fouling allowances for both tube and shell sides; it compares conservatively to the 233 Btu/s*F demonstrated by the data of Table 4.8-1 of the Browns Ferry FSAR for degraded shell-side flow of 6500 gpm. The temperature of the secondary flow was assumed to be 90°F. For the analytical model it was determined that depressurization of the reactor vessel results in a 21°F increase in pool temperature.

The closed-form analytical model also predicted a peak pool temperature of 185°F. This independent check establishes the quality of the BWR-LTAS result. To determine whether a delay in depressurization of the reactor vessel might result in higher peak temperatures, the analytical model was run for a range of depressurization times between 10 min and 20 h, and for no depressurization at all. As shown on Table 1, the highest peak pool temperature of 201°F would be obtained by delaying the rapid depressurization until 16 h. This peak temperature is acceptable from the standpoint of NPSH considerations as long as abnormal leakage from the containment does not prevent the expected buildup of internal pressure. It should be noted from Table 1 that the time of peak temperature corresponds to the time of depressurization for depressurization times of 10 hours or greater.

Group II: (L. J. Ott) Determines and analyzes the events of the accident sequence that would occur following core uncovery, including core melt and containment failure.

Development of MARCON-2.0B at ORNL: (C. R. Hyman, L. J. Ott, T. L. Heatherly) Work has continued in the debugging of MARCON 2.0B with ORNL modifications as described in the March SASA monthly report. Runs were successfully made which analyze the Station Blackout and Loss of Decay

Table 1. Peak pressure suppression pool temperature as a function of time of depressurization for one RHR heat exchanger in operation at time 10 min. Reactor vessel is isolated at time zero and steam generated by decay heat is condensed in the pressure suppression pool. Vessel is depressurized to 50 psia.

Time of Depressurization		Peak Temperature (°F)	Time of Peak Temperature	
Minute	Hour		Minute	Hour
None		180.0	990.0	16.5
20.0		184.5	810.0	13.5
60.0	1.0	184.9	780.0	13.0
120.0	2.0	185.6	750.0	12.5
180.0	3.0	186.4	720.0	12.0
240.0	4.0	187.3	710.0	12.0
300.0	5.0	188.3	690.0	11.5
360.0	6.0	189.6	660.0	11.0
480.0	8.0	192.7	570.0	9.5
600.0	10.0	196.7	600.0	10.0
720.0	12.0	199.3	720.0	12.0
840.0	14.0	200.5	840.0	14.0
960.0	16.0	201.0	960.0	16.0
1200.0	20.0	200.5	1200.0	20.0

Heat Removal for the Browns Ferry Unit One Plant and a TQUV accident sequence for the Grand Gulf plant. A model for calculating heat transfer from the primary system to the drywell atmosphere was added.

Work was begun on making the plotting package operational for MARCON 2.0B. The in-vessel portion of the plot code has been finished, and work is continuing for the containment part of the plot code.

Work has been completed for outputting of variables pertinent to secondary containment analysis and for fission product transport analysis.

Status of MARCH 2.0 (V151) at ORNL: (C. R. Hyman) During April, ORNL attempted to use the NT = 2 option for modeling of the heat transfer from the primary system to the drywell atmosphere. Three runs were made to test the operability of this containment event option. Comparisons of calculated drywell temperatures show no effect due to heat transfer from the primary system. ORNL contacted Battelle and discussed the matter with Roger Wooten and on April 18 forwarded a package consisting of the MARCH 2.0 (V151) output as well as the input data file used to make the runs.

BWR Degraded Core, ECCS, and Lower Plenum/Lower Head Failure Analysis and Model Development. Related work performed at ORNL and at RPI is reported in the following paragraphs:

1. BWR Core Uncovery Analysis for the Loss of Decay Heat Removal (LDHR) and Loss of Injection (TQUV) Accident Sequences. (A. Sozer) MELRPI was developed for use in degraded core analyses for mechanistic modeling of melting, relocation, and vessel failure processes. The purpose of the LDHR and TQUV calculations is to provide comparisons to the MARCON results for these sequences and to determine the radial distribution of the two-phase levels. Necessary modifications to MELRPI.MOD2 determined by the LDHR and TQUV calculations performed during March and April have been completed. A small timestep is required in order to dampen fluctuations in steam generation rates and yield realistic dryout times. Modifications were made to the calculation of steam generation rates in individual radial zones and to the calculation of core flow area reduction due to crust formation.

2. MARCON Data Reduction and Coupling of MARCON with MELRPI MOD2. (A. Sozer) A routine has been developed to calculate time-averaged values of MARCON results for pressure, two-phase level, and total core steam generation rate for use as input to MELRPI. Alternatively, the routine can be used to calculate the arithmetic average of each parameter during periods in which the steam flow rate varies by a user-specified amount. This approach is simple to use and valid for periods in which no rapid changes occur in the MARCON output data.

A second approach has been developed for use when the MARCON output is rapidly changing which allows much tighter coupling of the two codes. For this case, MELRPI.MOD2 has been modified to read MARCON generated pressures, steam generation rates, two phase levels and associated times directly, data point by data point. Several runs have been performed in this manner and the RESTART option has also been used successfully in these cases.

3. Station Blackout Transient Analyses. (A. Sozer) A preliminary MELRPI calculation has been performed for 1 h of problem time.

4. Models for Corium Interaction with Structural Materials. (A. Sozer) Two problem statements including boundary conditions, average properties, heat generation rates, and geometry definitions for an isolated analysis of the interaction of corium with the core plate and control rod guide tubes in the lower plenum have been provided to ORNL Engineering Physics Division personnel.

5. Testing and Verification of MELRPI Modeling. (M. Z. Podowski, RPI) Several MELRPI test runs were made, including both the analysis of various scenarios of BWR accidents and comparison with the PBF tests results. In the latter case, various modifications were made in the code, in order to account for significant differences existing between the geometry of a typical BWR and that of the PBF test facility.

6. Modeling of Lower Plenum/Head Failure and In-Vessel Thermal Hydraulics (M. Z. Podowski, RPI). Work continues on various improvements and numerical verification of the lower plenum/lower head heatup and failure models. The main effort is focused on developing, numerically implementing and testing a simplified version of the model for mass and heat exchange in the reactor vessel during vessel depressurization. Preliminary results of the test runs seem to be satisfactory; however, they still need further verification. Work is also underway on the development of a more sophisticated version of the model that will evaluate the in-vessel mass transfer due to the natural circulation of steam between the core and the downcomer.

Group III: (R. P. Wichner) Determines the magnitude and timing of fission product release from the fuel, establishes the various pathways for fission product release to the atmosphere, and performs the fission product transport calculations for each Severe Accident sequence analyzed.

Standby Gas Treatment System (SGTS) Effectiveness Calculations: (S. D. Clinton) The results of recent studies by various organizations have shown that the BWR MK I Primary Containment would fail either by overpressure or overtemperature not long after reactor vessel failure under severe accident conditions. Therefore, it is most important that

the capability of the BWR secondary containment to mitigate the concomitant releases of fission products to the atmosphere be assessed.

Previous ORNL SASA program studies have shown that the effectiveness of the SGTS system is a major factor in mitigation of fission product releases for postulated severe accident sequences at Browns Ferry. Accordingly, we have sponsored tests at the filter test facility at New Mexico State University to determine how long the installed HEPA filters would perform their function under severe accident conditions. The progression of these tests over the last several months has been discussed in the applicable monthly reports. The current status is reported in the following paragraphs.

The pressure drop, air flow rate, and mass accumulation data were received from New Mexico State University during April for the water spray and dry latex aerosol HEPA filter experiments. The latex particle (0.3 to 0.4 micron) loading test was operational for 860 h during a 93-d period. Due to evidence of potential damage to the ducting of the test facility, the dry latex experiment was terminated at a HEPA filter pressure drop of 15 in. of water with an air flow rate of 920 cfm and a latex loading of 932.2 g (2.06 lb). (It should be noted that previously reported latex loadings were only estimates.) The filter flow resistance (ratio of pressure drop to air flow rate) can be approximated as a linear function of the latex mass accumulation with an intercept of 1.0×10^{-3} in. water/cfm and a slope of 7.5×10^{-3} in. water/cfm·lb of latex (see Fig. 1).

These data show excellent agreement with a similar New Mexico State experiment performed previously using dry stearic acid particles and a HEPA filter by the American Air Filter Company for which the conditions at test termination were 5.7 in. of water, air flow rate of 500 cfm, and a stearic acid loading of 1.0 lb. The slope of the flow resistance-stearic acid curve is 10.4×10^{-3} in water/cfm·lb of stearic acid, and correcting for the density ratio of stearic acid to latex (0.85 to 1.05), the slope becomes 8.4×10^{-3} in. water/cfm·lb of latex. The 12% increase in resistance demonstrated by the current test with latex particles could be easily accounted for by differences in particle rigidity and particle size distribution. Adjusting the flow resistance-mass slope for the aerosol density expected in a severe accident, the Browns Ferry HEPA filter (Mine Safety Appliances Company) flow correlation has been incorporated into the ORNL SASA program Secondary Containment Model.

After reviewing the filter loading data from the water spray test (300 g/min), the most significant result is that the HEPA filter showed no damage or apparent decrease in particulate removal efficiency at a pressure drop of 15.2 in. of water. Compared to the dry latex loading test, the filter flow resistance increases slowly with water accumulation until the mass of water on the filter is in the order of 10 lb. Further water loading results in a sharp increase in the filter flow resistance (see Fig. 2). The times required for the two water spray experiments (maximum filter pressure drop of 5.0 and 15.2 in. of

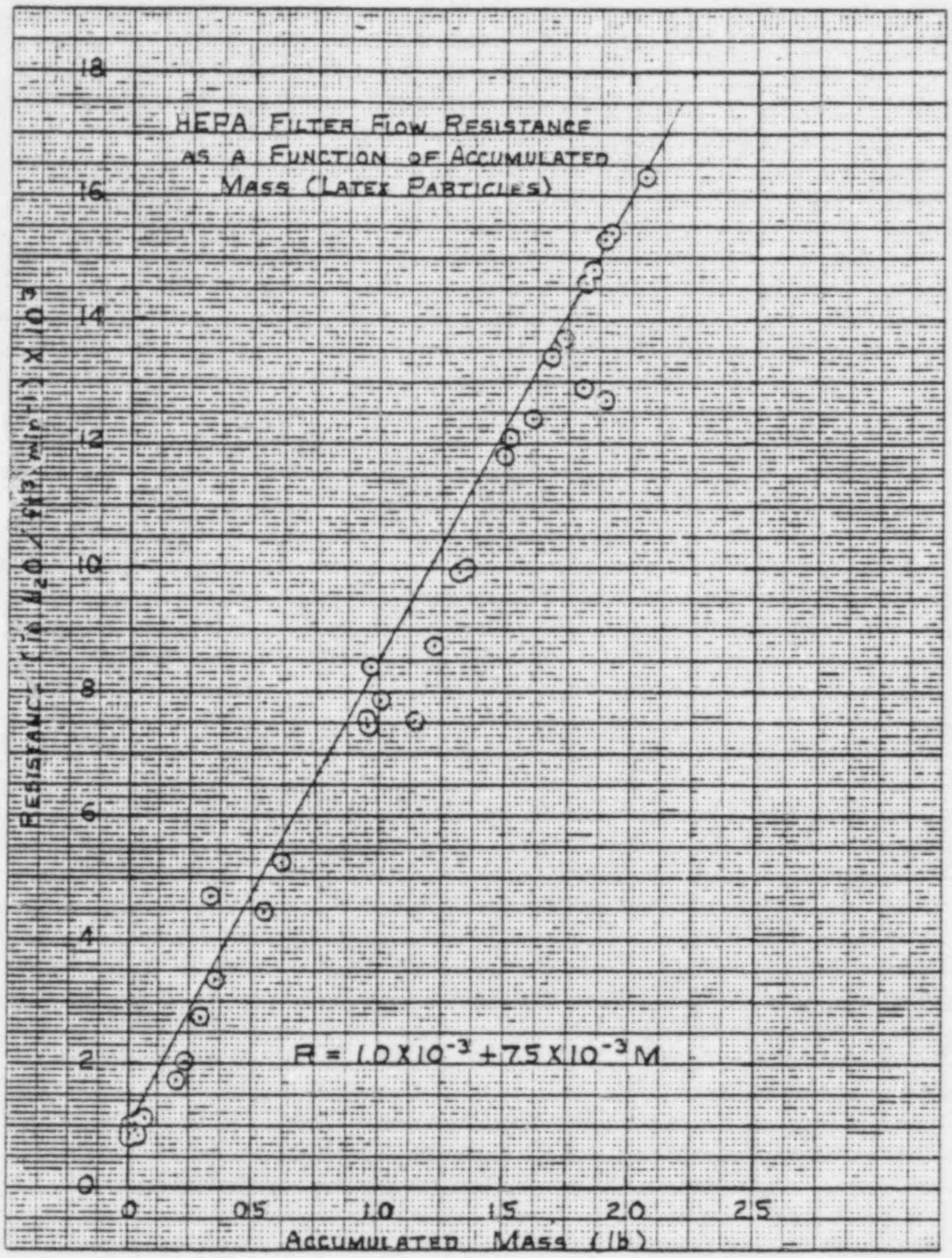

Fig. 1. HEPA Filter Flow Resistance as a Function of Accumulated Mass (Latex Particles).

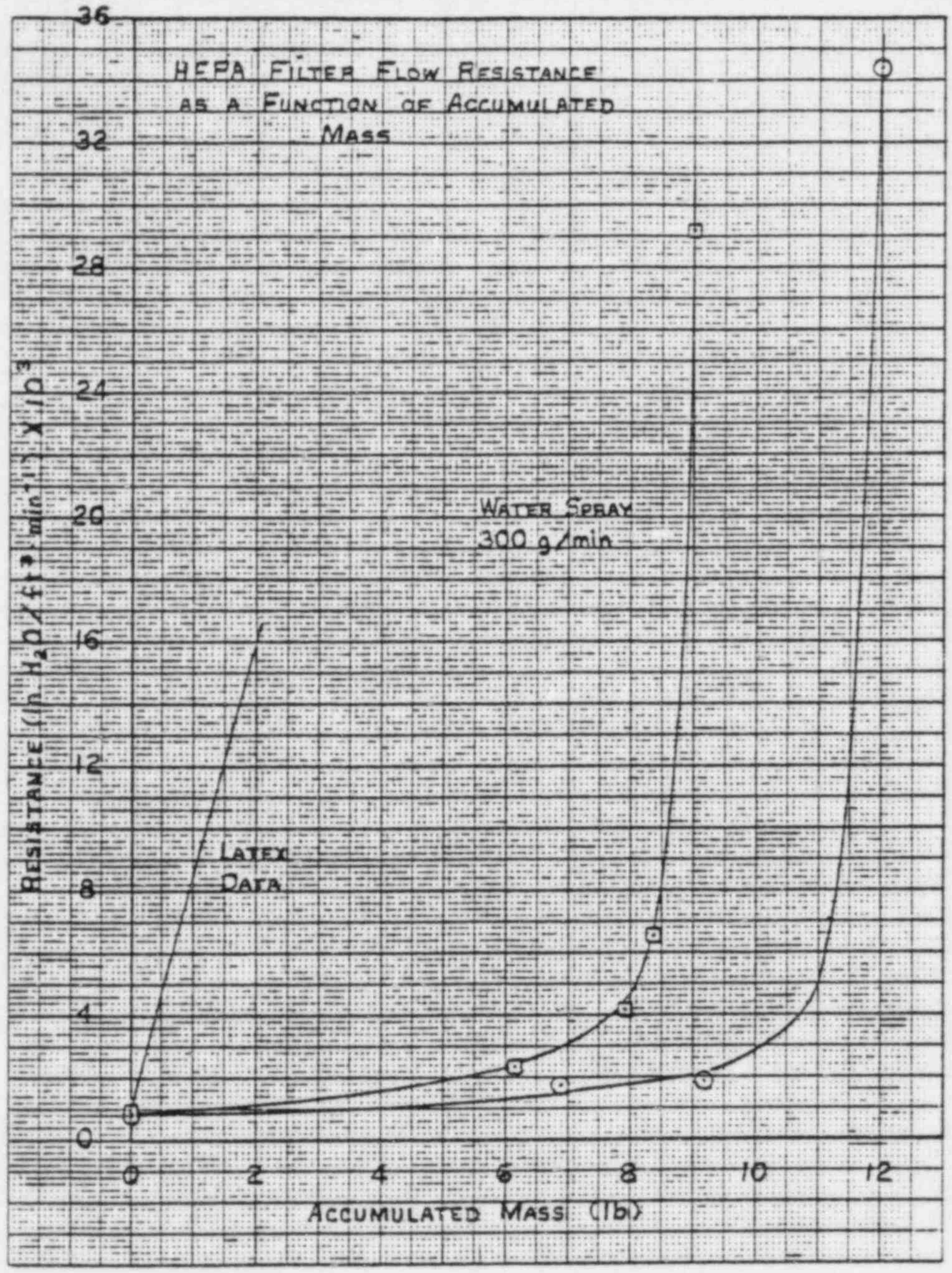

Fig. 2. HEPA Filter Flow Resistance as a Function of Accumulated Mass.

water) were 7.8 and 5.4 h, respectively. Consequently, the loading of latex particles on the planned third HEPA filter test which would involve simultaneous loading by latex and water spray would be <0.02 lb. Based on this information, the proposed third test has been postponed, and New Mexico State has agreed to store the third HEPA filter until a more desirable test condition can be negotiated.

MEETINGS AND TRIPS: Greg Slovik of Brookhaven National Laboratory visited ORNL on April 5 to view the television tapes of the SASA program ATWS drills at the Browns Ferry Control Room Simulator and to obtain other information concerning manual control rod insertion under ATWS conditions.

S. A. Hodge and A. Sozer visited RPI on April 9 to discuss future plans for the development and use of the MELRPI code.

A. Sozer worked with graduate students B. R. Koh and S. H. Kim at RPI during the period April 10-12 in performing trial calculations and analyzing the calculational methodology of the MELRPI code.

Alan Kolaczkowski of SAI, Albuquerque, visited ORNL on April 12 to discuss ORNL SASA program studies of Browns Ferry accident sequences. He will be performing equipment qualification analyses in support of the Accident Management Program at INEL.

S. A. Hodge attended the Severe Fuel Damage and Source Term Research Program Review Meeting at Idaho Falls on April 16 and 17.

S. A. Hodge attended an informal meeting at Albuquerque, NM on April 18 and 19 to discuss the Surry and Peach Bottom accident sequences analyzed in BMI-2104.

Dr. Gary Mueller of the University of Missouri at Rolla visited ORNL on April 19 for discussion with A. Sozer concerning plans for his work this summer in support of the ORNL SASA program. This effort will be sponsored by the Oak Ridge Associated Universities Program.

REPORTS, PAPERS, AND PUBLICATIONS: The final report, The Modeling of BWR Core Meltdown Accidents - For Application in the MELRPI.MOD2 Computer Code, NUREG/CR-3889, ORNL/Sub/81-9088/21, was distributed on April 19, 1985.

PROBLEM AREAS: At the current funding level, the fission product transport studies portion of the ORNL SASA program remains significantly curtailed.