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PROJECT TITLE: Severe Accident Sequence Analysis (SASA)

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Major work accomplished during November includes completion of the final version of the BWR-LTAS Users Guide and transmittal of the final version of the code to Sandia (SNL). The first revised version of the MARCON-2.0B code, in which the initial set of ORNL BWR models has been incorporated, was also transmitted to the SASA program personnel at SNL. Work continues to install the remaining ORNL-developed BWR response models into MARCON with a projected completion date of mid-January. Other work in progress involves preparation for the fission product transport calculations for the ATWS severe accident sequence, improvements to the BWR secondary containment response model, processing of the final draft of the second subcontractor (RPI) report concerning the MELRPI code, and checkout and testing of the MELRPI code at ORNL.

The personnel contributing to the SASA effort at ORNL are divided into three working groups. The individual group reports for progress during November are 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.

Peer review comments were received and the corresponding modifications have been made to create the final version of the NUREG/CR-3764 report, "BWR-LTAS: A Boiling Water Reactor Long-Term Accident Simulation Code." Liaison has been maintained concerning this code with A. Peterson (SNL) and L. Smith (SAI) of the SASA program at Sandia. A tape of the latest version of BWR-LTAS, which includes all updates, corrections, and improvements made since late July (when the last version was sent), was mailed to A. Peterson on November 15.

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

Implementation of MARCON-2.0B at ORNL (C. R. Hyman, L. J. Ott, T. L. Heatherly) Work is continuing to incorporate the ORNL-developed BWR models into the SNL code MARCON (essentially MARCH 2.0 with the INTER package replaced by CORCON MOD2). This code will be designated MARCON-2.0B upon incorporation of all currently existing ORNL-developed BWR response models.

On November 15, a tape containing the MARCON coding as modified to date was transmitted to A. Peterson at SNL. The BWR model upgrades included in this version are:

- channel boxes and control blades
- more extensive properties package for H_2 and H_2O
- new depressurization algorithm
- SRV actuation models
- explicit heat balance modeling for covered region of core
- pool boiling curve correlation application to covered section of core
- explicit modeling of invessel water inventory
- new equation of state for H_2, H_2O gas mixtures.

Ongoing work is dedicated to the incorporation of the remaining ORNL-developed BWR models into MARCON. During November, attention has been focused on incorporating the separated fuel/cladding model, as well as an updated model of the zirconium/steam reaction. We intend to have all of the remaining BWR models included in MARCON-2.0B by mid-January 1985.

An input deck for a modified Loss of Decay Heat Removal accident transient for Browns Ferry has been prepared for use with MARCON-2.0B; the deck has been debugged and successfully executed.

Corrections to the ZRWATR Subroutine (C. R. Hyman) In studying the updated zirconium/steam reaction model for MARCON, an error was found in the formulation of the heat source term for the cladding in the sub-timestep determination of the cladding temperatures used to calculate the oxidation rate. The error causes the calculated cladding temperature to be too high, and the magnitude of the error is proportional to the length of the BOIL timestep, which determines the number of sub-timesteps into which the BOIL timestep is divided. The error has been corrected in the MARCON-2.0B model.

This error has been determined to exist in the MARCH 2.0 coding as well. Dr. Mike Manahan at BCL has been so informed.

Investigation of Radiant and Volumetric Heat Sources in the BWR Steam Separators and Standpipes (J. C. Conklin, Dissertation) The formulation and scaling of the governing conservation equations of mass, momentum, and energy for the fluid flowing through the standpipes is being reviewed prior to casting the equations in finite difference form for numerical solution. Because steam is a participating medium for radiation heat transfer, an additional term arises in the conservation of energy equation for the flowing fluid. The effects of this additional term, as well as those due to viscous heat generation and pressure work are being addressed. The formal documentation of the computer modeling for the transient temperature distribution in the standpipe wall was completed and is being reviewed.

Improvements to the Browns Ferry Secondary Containment Model (S. A. Hodge) Several improvements to the ORNL-developed secondary containment model were implemented during November. The secondary containment model is driven by the flows from the drywell taken from the output of MARCH subroutine MACE and is used to calculate the thermal hydraulic conditions in the reactor building and refueling bay for the portion of each Severe Accident sequence studied that occurs after drywell failure. Systems represented include the standby gas treatment system (SGTS) and the reactor building fire protection system sprays.

Recent modifications to the secondary containment model involve provision for representation of the effect of plugging of the upstream HEPA filters in the SGTS by the aerosols generated in the corium-concrete reaction on the drywell floor and improvements to the calculation of condensation or evaporation at the surfaces of the walls and slabs and from the water pools in the reactor building and refueling bay. The Users Guide for the secondary containment model has also been significantly upgraded. These coding and documentation improvements were undertaken because the secondary containment response has been found to have an important effect upon fission product transport in all previous ORNL SASA program severe accident studies.

BWR Degraded Core, ECCS, and Lower Plenum Corium Progression Model Development

1. Implementation of MELRPI.MOD2 at ORNL (A. Sozer and T. L. Heatherly) The revised version of the MELRPI.MOD2 code received from RPI has been implemented onto the ORNL IBM computer system. The code was compiled successfully in double-precision form and a load module of the code has been prepared.

2. BWR Degraded Core Analysis and Testing of MELRPI.MOD2 (A. Sozer) Some recommended minor modifications and additions to the draft report "The Modeling of BWR Core Meltdown Accidents for Application in the MELRPI.MOD2 Computer Code," NUREG/CR-3889, were prepared as a result of the final ORNL review and sent to RPI for their comment.

   In order to check the reproducibility of results obtained from the revised version of MELRPI.MOD2 at RPI, six different runs with the revised version have been performed at ORNL, and the results compared to those of RPI. The calculations were performed for three different ECCS injection methods, bottom flooding, core spray and interstitial injection, each at 100% and at 50% of rated flow. All of these runs utilized restart parameters from a previous initial core uncovering calculation that had been performed with the initial version of MELRPI.MOD2.

   Except for the core spray cases, results are in good agreement with those of RPI indicating that MELRPI.MOD2 is transportable. (Core spray cases will be repeated after the initial core uncovering

portion of the accident sequence is rerun with the revised version of MELRPI.MOD2 to produce the corresponding restart parameters.) Additional work performed at RPI during November is reported in the following paragraphs.

3. Modeling of Lower Plenum/Head Failure (M. Z. Podowski, RPI) Based on the lumped parameter energy equations for the debris, reactor vessel wall, and control rod guide tubes, this model evaluates the propagation of melting across the guide tubes and the vessel wall and, following lower head failure, the flow of molten material from the reactor vessel lower plenum into the drywell. This new model has been implemented numerically as a FORTRAN subroutine and is being tested now.

4. Development of New Model for In-Vessel Thermal-Hydraulics (M. Z. Podowski, RPI) In order to account for transient flow and heat transfer phenomena leading to core heatup, and possibly melting, a new model has been developed for the in-vessel thermal-hydraulics of a boiling water reactor. The concept of the model is based on dividing the vessel into several control volumes, including: reactor core (multi-node model as used in MELRPI), upper plenum and riser/steam separator, steam dome and downcomer, and lower plenum. Each control volume is subsequently divided into two regions, single-phase/two-phase pool and dry steam/hydrogen region. Fluid parameters in each control volume (region) are described by time-dependent conservation equations for mass and energy, and the flows between neighboring volumes are evaluated from the lumped momentum equations. It is anticipated that this model will calculate the reactor response to such external actions as coolant leakage in the recirculation loop, safety relief valve (SRV) actuations, and water injection through an ECCS.

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.

Fission Product Transport Computation (C. F. Weber) Activities during November have focused on continued development of the fission product transport code. A correction was made to the organic iodide formation calculations, although the effect of this change on the atmospheric release of all forms of iodine was negligible. The main accomplishment this month has been the development and encoding of a new algorithm that will effectively calculate radioactive decay of all nuclides in each individual control volume, while taking only 10% more CPU time than the previous procedure, which calculated radioactive decay for the entire plantwide inventories. This new method is important in that now it is feasible to compare calculations utilizing a realistic treatment of radioactive decay with those which ignore possible effects of decay.

One other a complishment this month was the computation of Browns Ferry Cycle 6 core inventories for most major radioactive nuclides. These results were obtained from a locally developed code that provides nuclide inventories for SASA calculations by weighing and interpolating data output from the ORIGEN code and have been provided to TVA personnel for use with their ongoing probabilistic risk assessment (PRA) analyses.

Aerosol Production and Transport (A. L. Wright) The CORCON-MOD2 core-concrete interaction code, developed at Sandia National Laboratories, was made operational at ORNL a few months ago. Calculations were performed in August 1984 with CORCON-MOD2 using input conditions developed for SASA Loss of Decay Heat Removal (LDHR) accident-sequence calculations. Since the SASA LDHR core-concrete interaction calculations were initially performed with the CORCON-MOD1 code, this provided us the opportunity to compare results predicted by the two codes for the same code input conditions. A summary of the calculated total amounts of CO, CO_2, H_2, and H_2O releases for the two codes for two different sets of code input are summarized in the table below. For the first case, 13% of the available Zircaloy fuel-rod cladding was assumed to be oxidized at the start of the core-concrete interaction; for the second case, 90% of the Zr was assumed to be initially oxidized. The results illustrate significant differences in the gas release calculated by the two versions of CORCON. Although not summarized here, there are also substantial differences in the metal and oxide layer temperatures calculated by the two codes.

Comparisons of total gas release calculated with the CORCON-Mod1 and CORCON-Mod2 codes for SASA LDHR accident sequence conditions

	CORCON-Mod1 result	CORCON-Mod2 result
Case 1: 13% of Zr oxidized at start of core-concrete interaction		
Total CO release (kg)	0.9	6716
Total CO_2 release (kg)	6456	8423
Total H_2 release (kg)	101	850
Total H_2O release (kg)	2098	2738
Case 2: 90% of Zr oxidized at start of core-concrete interaction		
Total CO release (kg)	3570	5520
Total CO_2 release (kg)	7387	2111
Total H_2 release (kg)	203	314
Total H_2O release (kg)	2404	694

Analysis of the Standby Gas Treatment System (SGTS) (S. D. Clinton) The first (water spray only) of three HEPA filter loading tests at New Mexico State University was terminated after 20 hours of operation. At the design airflow capacity (1000 cfm and 1 in. of water resistance), water droplets with an estimated size range of 5 to 15 μ were injected at a constant rate of 300 g/min. The HEPA filter pressure drop increased to a steady state value of 5 in. of water. This filter has been placed in a standby condition for further water tests at pressure drops up to 16 in. of water. The pressure test conditions in Military Specification MIL-F-51068E have been discussed with the personnel responsible for the experiment.

The second HEPA filter loading test with dry-latex aerosol was halted after 20 days due to a continuing problem with the aerosol generator. With the initial design conditions, the HEPA filter pressure drop had only increased to 1.5 in. of water. The dry-latex test will continue when the repairs have been completed (estimated shutdown time of 2 days). Personal observations from a site visit on November 7th and 8th were positive despite the encountered operational problems. Most of the delays and setbacks can be attributed to the facility being inoperative for the past two years.

MEETINGS AND TRIPS: S. D. Clinton visited the filter plugging test facility at New Mexico State University, Las Cruces, NM, to witness the ongoing loading tests of the Browns Ferry-type standby gas treatment system HEPA filters.

Vance Behr, Al Benjamin, Eric Haskin, and Dave Kunsman of the SARRP program at SNL visited ORNL on November 15 for the purpose of discussions with SASA program personnel concerning BWR plant response under severe accident conditions.

S. A. Hodge attended an NRC-sponsored meeting at the Wiltsie Building, Silver Spring, MD on November 19 for the purpose of discussions concerning the results of the ongoing Accident Source Term (BMI 2104) study.

REPORTS, PAPERS, AND PUBLICATIONS: None.

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