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PROJECT TITLE: Severe Accident Sequence Analysis (SASA)

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Major work in progress during May includes completion of the final report for the ATWS accident sequence analysis study, preparation of a training seminar for I&E personnel concerning BWR accident sequences, and work to incorporate revisions in response to the peer review comments for the Loss of Decay Heat Removal fission product transport analysis report. Effort continues in the conversion of BWR-LACP to Fortran with preparation of a user's guide and toward the incorporation of modifications and improvements to MARCH-BWR in preparation for the degraded core calculations for the ATWS accident sequence. Preparations are also in progress for the fission product transport calculations for the ATWS sequence. Unit 1 of the Browns Ferry Nuclear Plant serves as the model plant for all studies.

The personnel contributing to the SASA effort at ORNL are divided into three working groups. The individual group reports for progress during May 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-LACP to study the plant response to operator actions.

The final version of the ATWS accident sequence analysis report was completed during May. Several important improvements in the report were incorporated as a result of the numerous peer review comments. Among these, the recommendation (Chapt. 5) that the HPCI turbine be tripped in the event of an MSIV-closure initiated ATWS was modified to specify that this should be done only if the SLC system fails to inject the sodium pentaborate solution into the reactor vessel. A discussion of the results of several new BWR-LACP runs that examine the effect of minimal operator action (e.g., only initiation of SLC and nothing else) was added to Chapter 5. A new appendix was added for the purpose of demonstrating the ease at which the steady-state reactor power can be calculated if the injection rate is specified as opposed to the difficulty of calculating reactor power if the reactor vessel water level is specified. The discussion of uncertainties in Chapter 6 was expanded to include reference to the recent preliminary BNL RAMONA results that predict much higher core power with the reactor vessel water level lowered to the top of the core than does BWR-LACP at ORNL or RELAP5 at INEL.

The conversion of the CSMP-based BWR-LACP code to its Fortran version BWR-LTAS (Boiling Water Reactor — Long-Term Accident Simulation) has progressed significantly during May with the completion and successful testing of Fortran programs for the reactor vessel thermohydraulics and injection systems, and for portions of the primary containment thermohydraulics. Several of the chapters for the user's guide have been completed. Liaison has been established with Lanny Smith, who will be responsible for the installation and operation of BWR-LTAS at Sandia.

A 90-min training presentation concerning important aspects of BWR reactor plant construction and containment and the results of ORNL SASA Program BWR severe accident studies was completed during May. The presentation is scheduled for delivery at three training seminars for personnel of the NRC Office of Inspection and Enforcement during June.

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.

MARCH Modifications for the In-Vessel Phase of the Browns Ferry ATWS Study (L. J. Ott) Action to provide separate models for fuel, gap and cladding in the BWR core models is continuing. As part of this effort, coding was implemented during May for a structure-gas radiation model that treats the gas as an absorbing/transmitting medium. It is expected that this routine will be applied to several locations within the reactor vessel in the future, but its current application is limited to the uncovered portions of the core.

A subroutine to permit calculation of the axial heat conduction in the fuel and the axial conduction in the cladding separately was also completed during May. Supporting subroutines, primarily for calculation of solid material physical properties were adapted from the information provided by NUREG/CR-0497.

It is expected that the major effort to provide separate modeling for the BWR fuel and cladding will be completed during June. Additional effort to provide modeling of the effects of the B_4C control rod pellets reaction with steam is in progress and should be completed in early July. Application of the code to ATWS calculations will begin at that time.

Implementation of MARCH 2.0 at ORNL (C. R. Hyman, J. J. Robinson) Because not all of the ORNL-BWR models were included in MARCH 2.0, effort is underway to incorporate the ORNL-BWR models into an SNL version of MARCH 2.0; this version also provides corium-concrete reaction calculations by the new code CORCON. Upon completion of this conversion, Sandia and ORNL will share a version of MARCH for BWR analyses that has both the latest in-vessel BWR models and the latest corium-concrete interaction models.

The updated version of MARCH 2.0 obtained from SNL in April has been successfully executed. Effort to incorporate the ORNL BWR in-vessel models is continuing. During May, both the hydrogen and water properties packages were replaced with the more extensive ORNL versions for BWR application. The MARCH 2 primary system depressurization algorithm PRIMP was also replaced, as necessary to incorporate the ORNL SRV models and the analytical Redlich-Kinong-Soave multigas equation of state models.

Grand Gulf ATWS Calculation (S. A. Hodge, L. J. Ott) Per the request of Ron Lipinski of Sandia, MARCH-BWR was exercised on a set of input data that had been used for the BMI-2104, Vol. III calculations for the TC sequence at Grand Gulf. It was necessary to provide some additional input at ORNL such as the dimensions of the channel boxes and the characteristics of the SRVs. The code output was sent to Dr. Lipinski for analysis.

Investigation of Radiant and Volumetric Heat Sources in the BWR Steam Separators and Standpipes (J. C. Conklin, Dissertation) Formulating and scaling of the governing conservation equations of mass, momentum, and energy have continued. A preliminary model is being developed to determine the axial fission product concentration for the flow, given an initial concentration and flow rate at the inlet, that will account for the plate-out on the standpipe inside wall. The information from this model will be used in the fluid computational model to locate the fission product heat source terms.

BWR Severe Accident Model Development at RPI (M. Podowski et al)

1. MELRPI Code Development (R. Taleyarkhan) A new model for the molten mass relocation has been implemented into a modified SLUG subroutine. This new model calculates the amount of molten/refrozen zirconium (from cladding and channel walls) and stainless steel (from control blades) during the relocation of these materials to the bottom of the core. Also, modifications have been made in the method used to calculate the mass of molten materials released from the core to the lower plenum. Improvements and modifications have been made in the main system of differential equations for the in-core heat transfer. Specifically, such variables as the nodal mass of oxidized and molten materials that were previously evaluated per unit length, now represent the total nodal values and can be easily summed over the entire core volume.

2. Lower Plenum Modeling (R. Taleyarkhan) The modeling of BWR reactor vessel lower plenum failure and melt ejection can be divided into three stages: (a) molten debris relocation along the control rod guide tubes, along with formation of a pool on the lower reactor vessel head; (b) meltdown and relocation of several sections of the CRD guide tube housing; and (c) molten debris release and consequent ablation of the reactor vessel. The first two component-models, (a) and (b), have been coded, debugged and tested. Model c has also been preliminarily coded and is presently being tested for bugs. The release of debris

material can occur either under gravity feed or due to a high reactor pressure.

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.

Chemical Change Effects (E. C. Beahm) A procedure has been derived to be used as part of the improvements to MARCH-BWR to calculate hydrogen and heat of reaction produced in the B_4C-steam reaction. The procedure allows calculations of the relative extent of the three in-vessel chemical reactions given below:

$$B_4C + 7H_2O = 2B_2O_3 + CO + 7H_2 , \quad (1)$$

$$B_4C + 8H_2O = 2B_2O_3 + CO_2 + 8H_2 , \quad (2)$$

$$B_4C + 6H_2O = 2B_3O_3 + CH_4 + 4H_2 . \quad (3)$$

The relative importance of these reactions depends upon the steam pressure, the initial hydrogen pressure, and the temperature. The calculations determine the number of mols of H_2, CO, CO_2, and CH_4 produced as well as the heat of reaction.

Thermochemical data in the form of free energies of formation, for fission product compounds are being inserted into a modified SOLGASMIX computer program. Calculations of fission product equilibria in the reactor vessel will be performed using the main subroutine which has the thermochemical data permanently in storage and receives input data consisting of temperature, pressure, and the amount of material (mols). These modifications will permit rapid calculation of equilibria between fission products, steam, hydrogen, and reactor vessel materials.

Aerosol Transport Analysis (A. L. Wright) In aerosol transport analyses performed for previous SASA accident sequences, the only mechanism by which aerosols were assumed to be deposited in the reactor building was by gravitational settling. However, if the fire protection sprays are actuated in the building, collision of the spray drops with aerosols and subsequent settling of the spray drops would provide a mechanism for additional aerosol deposition. Accounting for aerosol removal by the water sprays is important, because the greater the aerosol deposition in the reactor building, the less likelihood there is that the Stand-By Gas Treatment system filters will plug and tear due to aerosol deposition.

As a first step to determining the possible influence of building sprays on aerosol deposition, scoping calculations were performed this month to compare aerosol decontamination factors due to sprays to aerosol decontamination factors due to aerosol settling. The formulation used to estimate spray decontamination factors was that published in

the NUREG-0772 report and used in the NAUA-Mod4 code; the settling decontamination factor was calculated based on the Stokes settling velocity.

The reactor building fire protection system provides ceiling sprays that are actuated upon local high temperature. As the sprays are initiated, they have a large effect on reactor building temperatures, and a surprisingly low injection flow will reduce the building ambient temperature to the point where no additional sprays are actuated. Based on discussions with Steve Hodge, a range of spray flows between 50 and 400 GPM was assumed. The diameter of the spray droplets (which is a critical parameter to the calculations) was assumed to be 750 microns (based on discussions with fire protection spray vendors), and aerosol radii r of 0.1, 1.0, and 10 microns were assumed.

The table below presents ratios of calculated spray decontamination factors to settling decontamination factors for varied aerosol sizes and water spray rates. These results indicate that aerosol removal by sprays can be significant, even at the lowest spray flow assumed. The calculation of aerosol removal by sprays is very sensitive to water drop size, which at present is only approximately known for the Browns Ferry building sprays.

Decontamination factor ratios for various conditions

(Assumes water droplet size of 750 microns)

Spray flow rate (GPM)	Spray/settling DF ratio (r = 0.1 microns)	Spray/settling DF ratio (r = 1 micron)	Spray/settling DF ratio (r = 10 microns)
40	0.001	3.9	1.4
500	0.01	31.3	10.9

Future efforts in this area will concentrate on obtaining improved estimates of spray drop sizes, and also on determining the most appropriate way to model spray removal of aerosols in the ATWS sequence calculations.

<u>Analysis of the Standby Gas Treatment System (SGTS)</u> (S. D. Clinton) The BWR standby gas treatment system (SGTS) is of prime importance in severe accident sequences because the system is the final secondary containment barrier to the atmosphere. Experimental data are needed on the aerosol loading capacity and potential failure mode of high-efficiency particulate air (HEPA) filters as the flow rate decreases to 1 to 5% of the design value. Six tests have been requested at the New Mexico State University filter plugging facility to supply the following information:

(1) structural integrity of the HEPA filter at pressure drops up to 16 in. of water, and (2) air flow and pressure drop data as a function of mass accumulation on the filter media. The volumetric flow will be set initially at the design value (about 1000 ft^3/min) and then allowed to decrease as aerosol accumulates on the filter. Polystyrene latex particulates (mean diameter of 0.3 to 0.4 m) have been selected to be the aerosol material for preliminary tests. The moisture content of the air might be critical in the determination of structural integrity since liquid water can weaken the filter media and promote tearing. On the other hand, water vapor might have an insignificant effect on the filter media strength, and consequently, the air temperature (80 to 220°F) is an important variable.

MEETINGS AND TRIPS:

S. A. Hodge attended an NRC-sponsored meeting at Chicago on May 10. The meeting purpose was to prepare for the subsequent NRC/IDCOR interface meeting.

S. A. Hodge attended the NRC/IDCOR meeting on integrated analysis of severe accident containment loads on May 15, 16, and 17 and presented a summary of the NRC and NRC contractor positions on May 17.

REPORTS, PAPERS AND PUBLICATIONS:

The final version of the report *ATWS at Browns Ferry Unit One — Accident Sequence Analysis*, NUREG/CR-3470, ORNL/TM-8902 was submitted for makeup, reproduction, and distribution on May 29.

PROBLEM AREAS:

None.