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PROJECT TITLE: Severe Accident Sequence Analysis (SASA)

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Much of the October SASA program effort was devoted to the preparation of the five papers delivered at the 12th Water Reactor Safety Research Information Meeting (WRSM). Other major work continuing in progress involves the testing of the final BWR response models necessary for the degraded core calculations for the ATWS accident sequence, the incorporation of the ORNL-developed BWR response models into the MARCON code provided by the SASA program at SNL, and preparation for the ATWS fission product transport calculations.

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

Recent calculations using RAMONA-3B at BNL and RETRAN-02 at EPRI have indicated that the reactor power under MSIV-closure ATWS conditions with the vessel at pressure and the water level lowered to the top of the core would be in the neighborhood of 20%. Since the BWR-LACP code used for the calculations reported in the recent ORNL SASA program report* predicted a power level of about 9% under these conditions, the BWR-LTAS code (FORTRAN version of BWR-LACP) was exercised on several runs during October to scope the effect of higher power level on the major ATWS accident sequences. Three cases of MSIV-closure-initiated ATWS were considered: (1) no operator action, (2) mitigation by a single operator action to initiate sodium pentaborate injection, and (3) mitigation only by level and pressure control with pressure suppression pool cooling (i.e., no boration).

In order to cause BWR-LTAS to calculate a power level of 19% with the downcomer water level at the top of active fuel and the reactor vessel at pressure, a modifier was added to decrease the negative reactivity feedback due to core voiding in excess of the normal full power voiding.

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*R. M. Harrington and S. A. Hodge, "ATWS at Browns Ferry Unit One — Accident Sequence Analysis," NUREG/CR-3470 (July 1984).

Comparison of the calculated results for the nominal and the higher power cases reveals significant differences. For example, the no-operator-action case with feedback modified for higher power becomes a high pressure boil-off with core melt before primary containment failure instead of a low pressure boiloff with core melt after containment failure. The effect on the case with operator level/pressure control and with pool cooling is very sensitive to assumptions concerning operator actions. If the vessel remains pressurized, then the primary containment would reach overpressure failure at about 1.6 h instead of 6 h. With depressurization (as would be required by the suppression pool heat capacity/temperature limit of the EPGs), the difference is obscured because the depressurization lowers the power levels in both cases; however, a preliminary estimate is four hours to primary containment failure versus a much greater time to failure for the base case.

For the ATWS sequence with operator action limited to boration, the case with void feedback modified for higher power level has the same overall result as the case with nominal void feedback; severe accident consequences are prevented in both cases. The effect of the higher initial power level is counterbalanced by the greater initial rate of natural circulation sweeping the injected sodium pentaborate into the core.

The draft preliminary report for comment "Assessment of the TRAC-BD1/MOD1 Containment Model" by Philip Wheatley, was reviewed per request of T. R. Charlton of INEL.

The paper "An Efficient Method of Simulation of BWR Severe Accident Sequence Events Before Core Uncovery" was prepared and presented at the 12th WRSM at Gaithersburg on October 23.

Liaison was maintained with L. Smith of the SNL SASA program regarding the planned use of BWR-LTAS in support of the RMIEP program.

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.

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 was reviewed prior to casting the equations in finite difference form for numerical solution. A buoyancy force, in addition to that induced by the volumetric heating of the fission products, is due to the plate-out of volatile fission products on the inside standpipe wall. The effect on the momentum conservation equation due to this mass transfer is being addressed. The formal documentation of the computer modeling for the transient temperature distribution in the standpipe wall has been prepared.

Implementation of MARCH 2.0 (V151) at ORNL (C. R. Hyman, T. L. Heatherly) During October, ORNL has implemented MARCH 2.0 (V151) onto the ORNL IBM computing system. Several modifications were required in order to make the code operational, none involving correction of non-standard use of the FORTRAN-77 language. A BWR Mark III sample problem was executed and the output was compared satisfactorily with results included in the transmittal package.

ORNL SASA prepared an input deck for MARCH 2.0 (V151) for a Browns Ferry loss of decay heat removal calculation for use in a paper presented at the 12th WRSM. However, we were unable to successfully execute the problem. After discussions with BCL personnel, we incorporated several modifications to the input deck as suggested by R. Wooten of Battelle. None of the suggestions appears to have addressed the problem whereby the calculation gets trapped in an infinite loop inside the MACE containment calculation. On October 10 and 17, ORNL sent copies of the input and output of this problem, as well as a magnetic tape containing the input deck, to Battelle for their continued examination. On October 17, ORNL informed Battelle that no further ORNL effort will be spent on MARCH 2.0 (V151) until this problem has been resolved.

Loss of Decay Heat Removal (DHR) Accident Sequence Calculations for Presentation at the 12th Water Reactor Safety Research Information Meeting (L. J. Ott, C. R. Hyman) A paper entitled "Effects of Improved Modeling on Best Estimate BWR Severe Accident Analysis" was prepared and presented at the 12th WRSM. It was originally our intention to compare the ORNL MARCH-BWR code predicted responses for the Loss of DHR sequence with the MARCH 2.0 (V151) predictions; however, we could not get MARCH 2.0 to execute this problem. Therefore, comparisons were made to the old ORNL version of MARCH used for NUREG/CR-3617.* As can be seen in Table 1, the capabilities of the 'old' ORNL version of MARCH are very similar to those of MARCH 2.0 (V151).

Use of the improved BWR models extant in the current ORNL version of MARCH produces significant differences in the calculated results — for instance, the core uncovers one hour later. (This uncovery time has been confirmed by hand calculations.) Comparisons of several variables (reactor vessel water level, clad temperatures, etc.) were included in the WRSM paper.

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

1. MELRPI Computer Code Implementation and Testing at ORNL (A. Sozer) Reviews of the published, preliminary, and revised reports related to BWR core degradation and ECC system modeling work at RPI, and determination of areas in these reports for

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*R. P. Wichner et al., "Noble Gas, Iodine, and Cesium Transport in a Postulated Loss of Decay Heat Removal Accident at Browns Ferry" (Aug. 1984).

Table 1. Identification of model differences between current and old ORNL versions of MARCH. "Yes" indicates the model is included, while "No" means it is not

Model designator	ORNL-current	ORNL-old	MARCH 2.0
Separated fuel & cladding	Yes	No	No
Axial conduction	Yes	No	Yes
Improved heat transfer correlations for uncovered region of core	Yes	No	No
Mechanistic rod models in water covered region	Yes	No	No
BWR cannister and control blades	Yes	Yes	Yes
BWR SRV models	Yes	Yes[a]	No
New pressure calculation algorithm	Yes	No	No
BWR level routine	Yes	No	No
Multi-node analysis of invessel water inventory	Yes	No	No
New boiling and flashing algorithm	Yes	No	No
New Zr/steam reaction model	Yes	No	Yes[c]
"Melt" models for cannisters and control blades	Yes	Yes	Yes
New H_2/H_2O physical properties package	Yes	Yes[b]	Yes[b]

[a]Improved over original MARCH, but still not as mechanistic as in the current ORNL version.

[b]Old version was improved over original MARCH, but still not as comprehensive as in the current ORNL version.

[c]Version in MARCH 2.0 at present has deficiencies.

further clarification have been performed. Work on tests and related input for further verification of the MELRPI.MOD2 computer code has been started. Additional work performed at RPI during October is reported in the following paragraphs.

2. MELRPI Code Testing and Verification (M. Z. Podowski, RPI) Some additional changes and modifications have been made to the emergency core cooling system modeling and numerics (ECCS subroutine). A revised version of the report "The Modeling of BWR Core Meltdown Accidents for Application in the MELRPI.MOD2 Computer Code" has been prepared, implementing the modifications recommended by the peer reviewers.

3. Modeling of Lower Plenum/Head Failure (M. Z. Podowski, RPI) The modeling concept for the lower head failure analysis has been recently modified. The new version has a simplified spatial structure, with the emphasis focused on the melting/freezing phenomena leading to CRD guide tube failure, vessel wall melting (ablation of the initial opening) and melt release. This modified model is being tested now. Also, some further modifications/extensions of the MELRPI modeling are being studied, concerning the possible inclusion of a model calculating the melt inflow into the lower plenum. (So far, the development of the lower plenum model has been based on an assumed inflow from the core.)

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 October have involved a total revision of the procedures for calculating fission product transport and retention in the reactor vessel. Deposition calculations for iodine and cesium have been improved and the chemical behavior of tellurium is now modeled in the reactor vessel. Incorporation of the SOLGASMIX program (for calculation of gas and liquid phase species distributions) is complete.

Aerosol Production and Transport (A. L. Wright) Previous calculations have indicated that spray washout of aerosols in the reactor building due to the operation of fire-protection sprays may be important in the ATWS fission-product transport analysis. Efforts continued this month to include an aerosol transport code. This month Ken Lee from Battelle-Columbus Laboratory provided the coding changes necessary to include spray washout in NAUA-MOD4. These coding changes have been included in the ORNL version of the code, and debugging of the modified version of the code is now underway.

Analysis of the Standby Gas Treatment System (SGTS) (S. D. Clinton) The first of three HEPA filter loading tests is in progress at the New Mexico State University filter plugging facility. The scheduled tests at ambient temperature are: (1) water spray only, (2) dry-latex aerosol, and (3) wet-latex aerosol. A site visit has been scheduled for November 8th to observe the latter stages of the dry latex loading experiment. The test objectives are to determine (1) the structural integrity of the HEPA filter (MSA) 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.

Absorption of Gaseous Iodine by Water Droplets (M. F. Albert, Thesis Work) The spray model was compared to the Containment Systems Experiments, BNWL-1244, runs A-3, A-4, A-6, A-7, and A-8. These experiments cover a range of operating conditions and provide a good test of the model. The comparison shows a relative error range of 120 to 7%. A majority of the tests show an error of 38% or less with only a few tests showing an error of greater than 30%. Those tests which show an error of greater than 30% will be reviewed again to see if there is any explanation for this discrepancy.

A report on the absorption of gaseous iodine by water droplets has been started and a draft will be completed in a couple of weeks.

MEETINGS AND TRIPS: S. A. Hodge attended meetings at San Jose, California concerning the question of hydrogen generation in the BWR core under severe accident conditions for application to MK III Containment response studies. A meeting on October 2 involved onl, NRC and pertinent subcontractor representatives. The subsequent meetings on October 3 and 4 were with the Hydrogen Control Owners Group (HCOG). Results from previous BWR Severe Accident Sequence Analyses (SASA) program studies were useful at these meetings.

Ed Beahm, Dave Cook, Mike Harrington, Cliff Hyman, Larry Ott, and Bob Wichner attended the 12th Water Reactor Safety Research Information Meeting at Gaithersburg, Maryland, and gave papers.

REPORTS, PAPERS, AND PUBLICATIONS: Five ORNL SASA Program papers were delivered on October 23 at the 12th Water Reactor Safety Research Information Meeting; these will be published in the proceedings of the meeting.

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