

PROJECT TITLE: Severe Accident Sequence Analysis (SASA)
PROJECT MANAGER: S. A. Hodge
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Major work in progress during August includes optimization of the input and output subroutines of the BWR-LTAS code and preparation of the user's guide, the development of the final BWR response models necessary for the degraded core calculations for the ATWS accident sequence, the incorporation of 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. Checkout and testing of the CORCON MOD2 code on the ORNL IBM computer system and the preparation and submission of summaries for the five ORNL-SASA program papers to be presented at the 12th Water Reactor Safety Meeting (WRSF) were completed during August.

The personnel contributing to the SASA effort at ORNL are divided into three working groups. The individual group reports for progress during August 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 uncover, using the ORNL-developed simulation program BWR-LTAS to study the plant response to operator actions.

Improvements have been made to the initial version of BWR-LTAS (Boiling Water Reactor - Long-Term Accident Simulation), the Fortran-77 version of the CSMP-based BWR-LACP code. Operator action capabilities for the injection systems and containment cooling systems have been expanded and failure flags to automatically disable the appropriate equipment for events such as station blackout and scram discharge volume break have been provided for user input. The draft BWR-LTAS users guide was modified to maintain consistency with the coding improvements.

Liaison was maintained with Mr. L. Smith of the SNL-SASA program, who has received and successfully read into the Sandia computer system the initial version of BWR-LTAS. Compilation errors caused by our use of specific, rather than generic, names for certain library functions were experienced when the tape was run on the CDC computers at Sandia. These usages have been eliminated from the ORNL version, so the problem will not be repeated when the next version of BWR-LTAS is transmitted.

Support was provided to the Group II effort to extend the analysis of the unmitigated MSIV-closure-initiated ATWS event through fuel damage and events subsequent to containment failure. The present effort is to calculate (using the ORNL version of MARCH) the clad temperature and the extent of cladding and fuel damage during the power excursions of the

no-operator-action case presented in Chapter 3 of the ATWS accident sequence report, NUREG/CR-3470. This is necessary because BWR-LACP calculates an average fuel temperature and the gross magnitude of the power excursions, but does not calculate clad temperatures.

BWR-LACP output data was provided to Group II so that a power vs. water level relationship could be derived for input to the MARCH-BWR code. In examining the BWR-LACP output for this case, it was found that a coding error in BWR-LACP caused the calculation to incorrectly predict total core uncover at three times during the no-operator-action transient (see Fig. 3.2 in NUREG/CR-3470). This error was corrected, and the case rerun. The results, attached as Figure 1, show partial, but not total core uncover. Changes in the other system variables shown in Figures 3.1 and 3.3 through 3.7 of the ATWS accident sequence report were not significant. This is because the reactor power is at decay heat levels during the affected periods of core uncover both before and after the error correction and is therefore unchanged.

Additional information concerning the sequence of events for the sixth Browns Ferry SASA study, total loss of plant control air, has been received from TVA.

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

MARCH Modifications for the In-Vessel Phase of the Browns Ferry ATWS Study (L. J. Ott) During the adaptation of the ZRWATR subroutine (determines Zr/water reaction rate, etc.) from MARCH 2.0 into the ORNL version MARCH-BWR, several apparent errors in modeling and coding were detected. After liaison with Dr. Mike Manahan at BCL, a mutually agreed-upon set of coding was adopted. The nature of the coding involved is such that the errors would affect the calculated results only in unusual cases. BCL has indicated that these errors will be corrected in the final release version of MARCH 2.0.

Information Exchange with Utilities (L. J. Ott, S. A. Hodge) Several exchanges with staff of the New York Power Authority have been made with regard to ORNL's improved BWR models as implemented in MARCH-BWR. Copies of SASA reports and listings of several routines (pertinent to BWRs) have been sent to Mr. Richard Deams of NYPA. As a related matter, a complete set of ORNL-SASA program reports has also been provided to Mr. Fred Mogolesko of Boston Edison Company.

Degraded Core Analysis for the ATWS Accident Sequence (L. J. Ott) Efforts are underway to perform an analysis using MARCH-BWR of the no-operator action case of the MSIV-closure initiated ATWS (Chapter 3 of NUREG/CR-3470). The results compare very favorably with BWR-LACP calculations until the power/pressure spikes start. Because of the model MARCH uses for the primary system energy balance, it is impossible for the code to respond correctly (as compared with BWR-LACP and RELAP5)

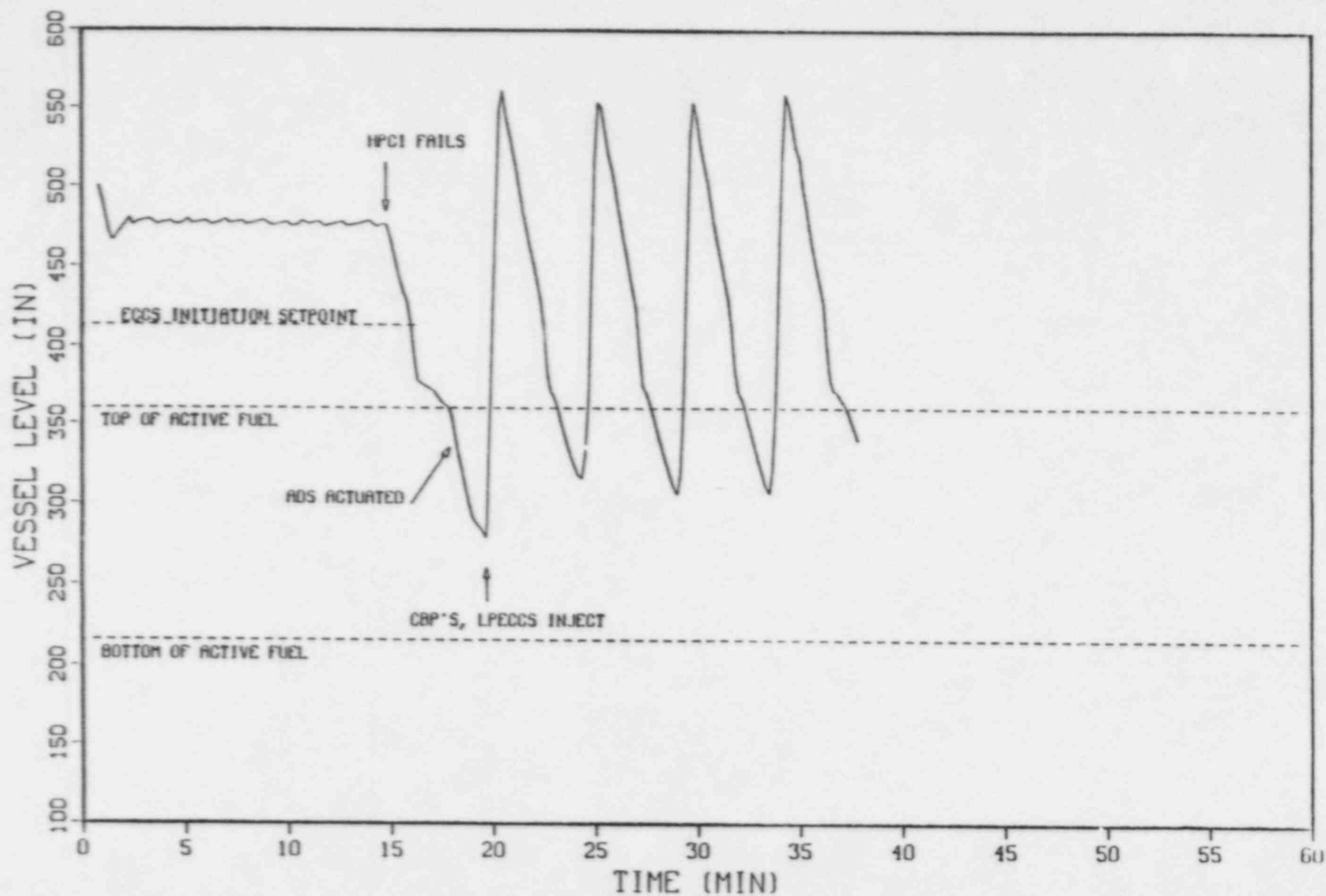


Fig. 1. Reactor vessel water level for the MSIV closure-initiated ATWS accident sequence without operator action.

at this point in the transient. Accordingly, mathematical models have been developed which essentially separate the one liquid control volume provided by the original version of MARCH into several smaller volumes (more representative of the reactor vessel configuration). This will permit better steaming and pressure predictions.

Implementation of CORCON MOD2 (C. R. Hyman, J. J. Robinson) Efforts to provide double precision capability for IBM Fortran 77 programs have been completed and a checkout run was made using the pre-release version of CORCON MOD2. Most numbers printed out agree with those on the sample printout sent by Mr. R. Cole of SNL to four significant digits. The code continues to experience underflow problems which have been found to originate in subroutine RLUD. ORNL SASA staff believe the underflows contribute insignificantly to calculated results and consequently have arranged for the variables that are the source of the underflows to be set equal to zero with subsequent normal execution of the program. This approach has not resulted in division by zero.

A letter requesting the NRC-released final version of CORCON MOD2 and a blank computer tape were sent to Mr. Dave Bradley of SNL on August 10, 1984.

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 continuing in the incorporation of these models into the SNL code MARCON (essentially MARCH 2.0 with the INTER package replaced by CORCON MOD2). During August, a detailed study of MARCON subroutines BOIL, MELT, BOILEX, BOILPR, BOILP2, and BOILLP was completed. These subroutines along with QSLUMP comprise the analogous coding extent in the BOIL subroutine of the ORNL version of MARCH 1.1. Efforts will continue in the incorporation of ORNL-BWR models into these subroutines of MARCON.

Investigation of Radiant and Volumetric Heat Sources in the BWR Steam Separators and Standpipes (J. C. Conklin, Dissertation) Computer coding to solve for the transient temperature distribution of the standpipes is being written. Special consideration is being given for the following situations: volumetric heating in the standpipe wall due to the gamma ray emitting fission products, radiation heat transfer on the inside surface, prescribed temperature on the outer wall, fission product plateout on the inside wall, and temperature dependent thermal properties. An iterative solution scheme will be necessary due to the nonlinearities introduced by the temperature dependent properties and radiant heat transfer at the inside wall.

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

1. MELRPI Code Modifications. In response to the peer review comments concerning the report "The Modeling of BWR Core Meltdown Accidents - for Application in the MELRPI MOD2 Computer Code," an effort has been made to improve both the structure of the MELRPI code and some of its models. In particular:

(a) Modification of the structure of the MELRPI input file. The old input file has been modified so that the user can specify, via input parameters, whether and when any of the existing ECCS models should be turned on. Consequently, the entire transient, including both phases, before and after ECCS initiation, will be calculated as a single run.

(b) Modification of the model evaluating the coolant overflow conditions in a rubblized core. An improved model has been developed to account for the effect of different heights of individual radial zones containing rubblized nodes on the core overflow following the ECCS action.

(c) The models evaluating the swollen pool level(s) inside fuel canisters are being checked again. Some modifications are underway concerning the modeling of water evaporation rate in the quenching zone.

2. Modeling of Lower Plenum/Head Failure. Testing of the LPFRPI subroutine, including modifications of some of its models, has been continued. The results obtained indicate that the basic structure of LPFRPI, similar to that developed for MELRPI, is both consistent and numerically efficient. However, the models used for individual phenomena, such as heat transfer between the molten/solidifying debris and the lower plenum structures, lower head failure, etc., require an extensive parametric analysis. This is mainly because these models are based on several simplifying assumptions, which must be tested for various input conditions.

In the test cases run so far, a given (time-dependent) inflow of molten debris from the degraded reactor core into the lower plenum was used as an input parameter. The LPFRPI code evaluated such time-dependent phenomena as freezing/remelting of the debris, melting of the control rod drive tubes leading to lower head failure, and debris discharge through the ablating holes in the lower head. These preliminary numerical results are being examined now. Further tests and, if necessary, modifications will be performed as the next step.

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) Several tasks have been completed this month related to the completion last month of the ORIGEN calculations of nuclide inventories in the Browns Ferry Cycle 6 code. A code has been written to fold the ORIGEN data so as to calculate initial inventories for each of the 100 core nodes (the core is divided into 10 radial and 10 axial zones). The new inventories have

been tested in the Loss of Decay Heat Removal (LDHR) accident sequence, which was previously analyzed using fission product inventories based on the Cycle 4 core; the new inventories lead to slightly lower releases. Another code has been written to allow a more quantitative selection of nuclides. Isotopic contributions to the total element decay heat are used to eliminate isotopes whose radioactivity is negligible (see below).

Because the ATWS sequence requires a totally new nuclide library, the radioactive decay routine of the fission product transport code has been replaced. The new routine uses the 4th order Adams predictor-corrector numerical scheme, whereas the old routine calculated the exact solution of the decay differential equations. The new routine allows much easier changes in the nuclide list and runs in about the same CPU time. Results agree to within 0.1%.

Work has recently begun on a complete revision of the routine that calculates chemical interactions in the reactor vessel. This is being done to incorporate the SOLGASMIX code into the calculations and to include tellurium interaction models.

Selection of Nuclides for the Transport Model (R. P. Wichner) An important feature of the transport model employed in the SASA program is the use of individual nuclides for the determinations of the net transfer rate of fission product elements. In this way, the effects of radioactive decay due to holdup and precursor transport are automatically taken into account.

Application to the ATWS sequence, however, has required a re-evaluation of the nuclide list employed for the earlier cases. The original motivation was (1) the shorter decay times which require the addition of some shorter half-life nuclides and (2) the addition of tellurium which, of course, requires the addition of tellurium nuclides.

This re-examination has led to an objective method for selecting the significant fission product nuclides based on ORIGEN-derived values of the individual nuclide decay powers. The use of decay power, rather than half-life, curies or mass, as a selection criterion, we feel is an improvement. In particular, the estimated curie-level is not always a valid indication of nuclide importance since some high-curie nuclides have quite low-level emissions (this seems especially true for several tellurium nuclides).

The specific selection rule we have adopted is the following: We select those nuclides which contribute more than 3% of the fission product group decay power at either 1 hour or 30 hours after shutdown. This criterion has led to the following nuclide list, which is being incorporated into the transport model.

Krypton: 85, 85m, 87, 88

Xenon: 133, 135, 135m, 138

Iodine/bromine: ^{131}I , ^{132}I , ^{133}I , ^{134}I , ^{135}I
(no bromines)

Cesium/rubidium: ^{88}Rb , ^{89}Rb , ^{134}Cs , ^{136}Cs , ^{137}Cs , ^{138}Cs

Tellurium: 129, 131, 131m, 132, 133, 133m, 134

Aerosol Production and Transport (A. L. Wright) For the LDHR accident sequence, drywell core-concrete interaction calculations were performed using the CORCON-MOD1 code, developed at Sandia National Laboratories. Since then, an improved version of the code, CORCON-MOD2, has been developed and released by Sandia for general use. Present plans are to use the new version for calculations for the ATWS accident sequence and we are testing the MOD2 version for conditions applying to the LDHR sequence; in this way we can produce comparisons of MOD1 and MOD2 calculations. One calculation has been performed for the so-called "base case" LDHR code input parameters. In comparing the MOD1 and MOD2 results, we have observed significant differences in calculated gas evolution rates, layer temperatures, and Zircaloy oxidation rates. Calculations will be performed for other assumed LDHR input conditions (primarily variations in the initial amount of oxidized Zircaloy clad in the core melt) during the next month.

Chemical Change Effects (E. C. Beahm) The SOLGASMIX computer program has been modified to permit rapid calculation of equilibria between fission products, steam, hydrogen, and reactor vessel materials. The main subroutine of the program has been retained, but the input-output routines have been changed and a data subroutine has been installed. The data subroutine contains free energies of formation for 26 chemical species representing forms of cesium, iodine, boron, carbon, hydrogen, and oxygen expected under accident conditions.

The input-output routines have been set up so that the program can be easily used as part of the overall computer system used in calculating fission product transport analysis. At each time interval calculation, the equilibria program receives a temperature, a total pressure, and the total amount of each element Cs, I, B, C, O, and H. The output from the equilibria program gives the amounts (mols) of the chemical species at equilibrium to permit efficient computation of the equilibrium and the mole amounts from the previous calculation are used as the initial guess of the species distribution. This equilibrium program has been tested separately and is now being tested as part of the overall fission product transport system.

Analysis of the Standby Gas Treatment System (SGTS) (S. D. Clinton) The three HEPA filter loading tests proposed for the New Mexico State filter plugging facility have been delayed until September due to a delay in the transfer of funds.

As the pressure drop across the SGTS upstream HEPA filter approaches 16 inches of water, the filter media will either rupture (reactor building and refueling bay air will continue to exhaust through the 180-m stack) or plug, resulting in an increased leakage to the environment through the refueling bay blowout panels. In the LDHR study, the upstream HEPA filter was predicted to fail after 2770 min into the accident sequence, and the downstream filter ruptured after 2920 min. At an accident sequence time of 3100 min, the SGTS charcoal filter had accumulated 4.9 PBq of iodine with 0.050 PBq of iodine discharged to the stack. Also after 3100 min, 162 kg of core-concrete aerosol remained on the SGTS filters with an additional 87 kg having leaked through the ruptured filters. If, instead of failing, the SGTS upstream HEPA filters plugged after 2770 min, about 3.3 PBq of iodine activity would be available for transport through the refueling bay with 1.6 PBq (less than a 200-watt heat load) remaining on the charcoal bed. In addition, 168 kg of aerosol could flow into the refueling bay with some fractional release to the environment. Therefore, with the slow progression of the accident sequence encountered in the LDHR study, the release of activity to the environment may not be significantly different for either a ruptured or plugged HEPA filter.

Absorption of Gaseous Iodine by Water Droplets (M. F. Albert, Thesis Work) The computer library numerical integration program, LSODE, has been incorporated into the kinetic, drop, and spray models. In the kinetic model, LSODE is very fast under all initial conditions and results agree with the previous model. Results of the drop model vary from the previous model by showing that the elemental iodine in the drop is being removed by the chemical reactions faster than it is being transferred from the gas. In the spray model, LSODE is extremely inefficient and impractical because such a multistep method is inherently slow for small integration times.

A single-step method is now being incorporated into the models. This method is a Runge-Kutta-Fehlberg routine which has a variable step size and error control.

MEETINGS AND TRIPS:

A. L. Wright attended the NRC-sponsored workshop meeting for users of the CONTAIN code at Silver Spring, MD on August 7.

C. R. Hyman and A. L. Wright attended the NRC-sponsored workshop meeting for users of CORCON MOD2 at Silver Spring, MD on August 8.

Ray DiSalvo, Mark Leonard, and John Wreathall of Battelle Columbus Laboratories (BCL) and Tom Walker of the NRC met with the ORNL SASA team at ORNL during the morning of August 22 to discuss the concept of Accident Management and to resolve questions raised during the BCL review of the completed SASA studies. Dr. Walker is the NRC technical monitor for the ORNL-SASA program.

Tom Walker also met with S. A. Hodge on the afternoon of August 22 and during the morning of August 23 to discuss the current status of and the future plans for the ORNL SASA program.

S. A. Hodge attended the NRC/IDCOR interface meeting at Rockville, MD on August 28 and 29.

REPORTS, PAPERS AND PUBLICATIONS:

Summaries of the five ORNL SASA program papers to be presented at the 12th WRSN were submitted during August.

The final version of the report "Pressure Suppression Pool Thermal Mixing," NUREG/CR-3471, was submitted for makeup and reproduction on August 21.

The final report "Noble Gas, Iodine, and Cesium Transport in a Postulated Loss of Decay Heat Removal Accident at Browns Ferry," NUREG/CR-3617, ORNL/TM-9028, was distributed on August 28.

PROBLEM AREAS:

None.