SASA MONTHLY
DEC. 1984

PROJECT TITLE: Severe Accident Sequence Analysis (SASA)

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NRC B&R NO./FIN NO.: 60 19 01 3 0/B0452

One major task was completed during December. The BWR operator action plant response code BWR-LTAS was placed in its final form and the associated User's Guide was submitted for makeup, reproduction, and distribution; copies of the final version of the code and the manual have been mailed to the immediate users at Sandia. Ongoing work includes implementation of the remaining ORNL-developed BWR response models for the portion of an accident sequence after permanent core uncovery into MARCON-2.0B with an expected completion date of 15 January. Subsequently, existing models for the reaction of the B_4C control rod powder with steam will be implemented and tested and calculations will be made for the degraded core portion of the Browns Ferry ATWS accident sequence. Checkout and testing of the MELRPI code on the ORNL computer system continues. In the fission product transport analyses group, an analysis of the effect of consideration/non-consideration of the radioactive decay of fission products was completed during December and preparations for the calculations attendant to the fission product transport portion of the Browns Ferry ATWS accident sequence are in the final stages.

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

Processing of the final version of the NUREG/CR-3764 report "BWR-LTAS: A Boiling Water Reactor Long-Term Accident Simulation Code" has been completed and the publication mats sent to reproduction and distribution.

Two final modifications to the BWR-LACP code were completed during December. The safety relief valve (SRV) simulation was modified so that the automatic opening of the SRVs now occurs when the pressure differential between the reactor vessel and the wetwell is equal to or exceeds the setpoint. This change is necessary to correctly model the operation of the two-stage Target Rock valves installed at Browns Ferry. The previous model provided for SRV automatic opening when the reactor vessel pressure exceeded the setpoint. The difference in modeling, of course, only has significant effect when the containment pressure is high.

The second modification installed during December involves the primary containment model, which did not properly account for the condensation of water vapor from the flow from the drywell atmosphere through the downcomers into the pressure suppression pool during periods when the drywell pressure exceeds the wetwell pressure by more than 1.50 psi (the gases in the flow bubble up through the pool and enter the wetwell atmosphere). The corrected model now accurately represents the condensation of water vapor from the flow, including the requisite transfer of mass and energy from the gas stream to the pressure suppression pool. This correction would have negligible effect on the previous accident sequences studied because in these sequences* there was either very little vapor in the drywell atmosphere or the flow was predominently from the wetwell to the drywell. However, the correction is necessary for possible future code application such as small break loss-of-coolant accidents.

These final code modifications have been transmitted to the SASA program personnel at Sandia as well as has the final version of the User's Guide. The current state of the code now becomes the Archive version. This version will immediately be applied at ORNL to the study of the loss of control air accident sequence for Browns Ferry and will be applied at Sandia to the study of various accident sequences for La Salle. It is certain that various new models and capabilities will have to be added to the code by both working groups as the code is applied to new situations. Accordingly, the personnel expecting to use the code at Sandia have been invited to come to Oak Ridge at their convenience for a seminar to discuss the inner structure of the code.

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.

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.

Ongoing work is dedicated to the incorporation of the remaining ORNL-developed BWR models into MARCON. During December, work has concentrated on incorporating the separated fuel/clad model and the recently corrected zirconium/steam reaction models. These models have been placed in MARCON 2.0B and are now in the process of being debugged. We intend to have all of the remaining BWR models included in MARCON-2.0B by mid-January 1985 and will transmit a tape of the modified coding to the SASA program at SNL at that time.

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*Station Blackout, Small-Break LOCA Outside Containment, Loss of Decay Heat Removal, Loss of Injection, and ATWS.

BWR Degraded Core, ECCS, and Lower Plenum Corium Progression Model Development (A. Sozer)

1. BWR Degraded Core Analysis and Testing of MELRPI.MOD2 at ORNL
The local version of MELRPI.MOD2 has been modified so that the ORNL SASA plotting routine can be utilized with selected output parameters. Comments resulting from local operations with the MELRPI.MOD2 computer code have been provided to Dr. Michael Podowski at RPI. The interaction between the ORNL and RPI staffs have proven very helpful in enhancing the transportability of MELRPI.

2. Modeling of Lower Plenum/Head Failure (M. Z. Podowski, RPI) Based on the lumped parameter energy equations for the corium, 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 planum into the drywell. This model will be transmitted to ORNL during January.

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) Two recent calculations based upon the Browns Ferry Loss of Decay Heat Removal (LDHR) accident sequence have addressed the importance of consideration of radioactive decay in fission product transport calculations. For the first calculation (Method I), the radioactive decay of each nuclide is computed in each control volume at each time step and the fission product inventories are adjusted accordingly. For the second calculation (Method II), radioactive decay during the accident sequence is ignored so that the total nuclide inventories remain constant. The average activity per unit mass of each element (PBq/gmol) is determined using the nuclide mass inventories obtained from ORIGEN results at accident time 3000 min and is multiplied by the total elemental mass (gmol) released to the atmosphere to yield the activity (PBq)* released to the atmosphere. The results of the two calculations are shown in the following table.

The activities of Kr, Xe, and Cs do not vary appreciably between Method I (with decay) and Method II (without decay). However, the iodine inventories, both gas phase and on aerosols, are significantly higher for the case of Method I, where radioactive decay is considered.

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*Petabecquerel = 10^{15} Bq = 27,027 curies.

Table 1. Comparison of activity released to the atmosphere calculated by two different methods

	Atmospheric activity (PBq) at 3000 min	
Element	Method I	Method II
Kr	23.90	23.62
Xe	4629	4480
I (gas)	12.64	2.89
I (aerosol)	0.140	3.89×10^{-5}
Cs (gas)	1.45×10^{-3}	1.61×10^{-3}
Cs (aerosol)	2.26×10^{-5}	2.39×10^{-5}

To understand the reason for the higher iodine releases to the atmosphere for the case with consideration of radioactive decay, it is necessary to evaluate the isotopic contributions to the total iodine release calculated by Method I. In particular, the contributions of ^{132}I are illustrated in the following table.

Table 2. Contributions of ^{132}I to total iodine inventories

		Fraction contribution of ^{132}I (%)	
Time (min)	Location	Activity (% PBq)	Mass (% gmol)
1800[a]	Fuel	38.9	0.039
2590[b]	Atmosphere gas	35.1	0.028
3000[c]	Atmosphere gas	86.1	0.29
	Atmosphere aerosol	88.6	0.36
	Total	42.8	0.032

[a]As determined by ORIGEN code before fuel failure.
[b]Immediately prior to pressure vessel failure.
[c]Near end of accident sequence.

It can be seen that the iodine activity in the atmosphere is due disproportionately to ^{132}I. We recall also that the half-life of ^{132}I is only 2.30 h, whereas the other dominant radioactive isotopes ^{131}I and ^{133}I are much longer-lived, with half-lives of 8.04 d and 20.8 h, respectively. Unlike the other isotopes, the ^{132}I which exists early in the accident will decay before the occurrence of the significant atmospheric releases that follow pressure vessel failure. However, the supply is replenished by the decay of ^{132}Te, which is released in large

quantities after pressure vessel failure from the drywell rubble. That is, the atmospheric release of ^{132}I depends more on the behavior of tellurium in the drywell and reactor building than it does on the behavior of iodine!

These results demonstrate that the physical and chemical behavior of precursors, as well as their radioactive decay, are important considerations in fission product transport calculations.

Analysis of the Standby Gas Treatment System (SGTS) (S. D. Clinton) The second HEPA filter loading tests at New Mexico State University have continued; however, the facility will be unavailable for two weeks during the semester break starting December 19. After completion of maintenance on the aerosol generator system, the dry-latex test was restarted and continued until the filter pressure drop reached 5.7 in. of water (air flow rate of 650 cfm and mass accumulation on the filter of 640 g). In order to achieve the test objective pressure drop of 16 in. of water, the filter test facility was shut down and two additional axial vane fans were installed in series with the existing fan. These rather extensive modifications should be completed by January 7, 1985, and the second filter loading test will be resumed.

MEETINGS AND TRIPS: R. M. Harrington attended the NRC/IDCOR meeting on Baseline Risk Profile and Operator Procedures at Rockville, Maryland, on December 13 and 14.

T. S. Kress made the presentation "Severe Accident Sequence Analysis (SASA) Studies at ORNL" to D. F. Ross during the visit of the latter to Oak Ridge on December 14, 1984.

Dr. T. J. Walker visited ORNL on December 17 and met with S. A. Hodge in regard to SASA program planning matters.

REPORTS, PAPERS, AND PUBLICATIONS: The final version of the report "BWR-LTAS: A Boiling Water Reactor Long-Term Accident Simulation Code," NUREG/CR-3764, was submitted for makeup and reproduction on December 12.

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