PROJECT TITLE: Severe Accident Sequence Analysis (SASA)

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Major work in progress during MARCH includes completion of the draft report of the fission product transport study for the Loss of Decay Heat Removal (DHR) accident sequence, completion of the draft report for the pressure suppression pool thermal mixing study, continuing modifications and improvements to MARCH-BWR in preparation for the degraded core calculations for the ATWS sequence, conversion of BWR-LACP to Fortran and preparation of a users' guide, preparation for the fission product transport calculations for the ATWS sequence, and work to incorporate revisions in response to the peer review comments into the ATWS accident sequence analysis report. Unit 1 of the Browns Ferry Nuclear Plant serves as the model plant for all studies. The draft fission product transport report for the Loss of DHR accident study was issued for peer reveiw on March 8 with comments requested by April 30. The draft report "Pressure Suppression Pool Thermal Mixing" was issued for peer review on March 21 with comments requested by May 15.

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

Activities during March involved tasks necessary for the completion of the ATWS accident sequence investigation and for conversion of the BWR-LACP code from the presently employed CSMP simulation language into Fortran and preparation of a users' guide.

D. C. Cook's two-dimensional pressure suppression pool model was successfully compiled and run with BWR-LACP for several ATWS transient cases for which the assumption of a well-mixed pool is not justified. Since the suppression pool surface temperature calculated with Cook's model is significantly higher than the bulk pool temperature used previously, the containment pressure increases more rapidly with the new model. For example, the time to trip of the RCIC turbine on high backpressure at 40 psia containment pressure occurs at 194 min with the bulk model and at 177 min with the new model. The final version of the ATWS accident sequence analysis report will include the new results where applicable.

The no-operator-action case discussed in Chapter 3 of the draft ATWS report has been recalculated to correct for the effects of an erroneous modeling assumption used previously. As part of his review

of the draft report, L. Claassen of GE has pointed out (and we have confirmed) that once opened, the ADS valves will not automatically shut when the reactor vessel water level is restored. (The Browns Ferry simulator models the opposite.) With the ADS valves remaining open, the new calculations indicate lower power spikes when the reactor vessel is reflooded by the low-pressure ECCS systems. For example, the peak power spike shown on Fig. 3.1 of the draft report is 200%. With the ADS valves remaining open, this is reduced to 178%. However, the low-pressure injection can enter the reactor vessel sooner when the ADS valves remain open. The net effect on the progression of the accident sequence is small, reducing the primary containment failure time from 41 min to 37 min for the no-operator-action sequence. The corrected transient results will be provided in the final version of the report.

The task of converting BWR-LACP to Fortran and providing a users' guide is underway. This activity is intended to produce a working, documented Fortran 77 version within three months. Mr. Len Fuller, who will assist in this task, has been introduced to the structure and use of the present version of the code. A benchmark accident sequence that exercises all of the important case models has been selected and the calculational results have been archived. A draft of several chapters of the documentation has been completed.

<u>Group II</u>: Determines and analyzes the events of the accident sequence that would occur following core uncovery, including core melt and containment failure.

<u>MARCH Modifications for the In-Vessel Phase of the Browns Ferry ATWS Study</u> (L. J. Ott) Action to provide separate models for fuel, gap and cladding in the BWR core models is continuing. First, these new models require an improved heat transfer package for the covered and uncovered portions of the core. The package has been developed, implemented, and debugged in MARCH-BWR. A sample run has been made (Loss of DHR accident sequence) to determine the effect on the accident event timing — a comparison is given in the following table:

	Original time (min)	New time** (min)	New time*** (min)
Containment failure	0*	0*	0*
Core uncovery	149	218	239
Start melt	357	436	449
Core slump	442	489	503

*All times relative to containment failure at 2050 min.

**New ORNL pressure control and water level determination routines.

***Same as (**) plus new heat transfer package.

The time of 239 mins to core uncovery in the sample run compares favorably with hand calculations (±2M). During the implementation of the package in MARCH-BWR, an apparent error was found in the original MARCH 1.1 modeling which incorrectly accounts for changes in the internal energy of the core structures — the result being a faster boiloff of the reactor vessel water inventory. For instance, before this error was corrected, the time to core uncovery in the DHR sequence was ~10% less.

The mathematical models for the separation of fuel and cladding have been formulated and are in the process of implementation.

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. In the process, several subroutines have been identified which either inadequately model BWR features or do not provide the necessary thermophysical properties needed for proper heat transfer calculations. At the present time, we intend to completely remove some of these subroutines and replace them with subroutines extant in the ORNL MARCH-BWR code. These subroutines are: STMH2P, STEEM, THTRPH, SPEEDY, THTRPS, and POLATE. Other subroutines will be either modified or replaced in order to incorporate ORNL BWR models. These include AXIALC, HRSTM, PRIME, CR1, EFCRIT, PROPS, RANGE, BOILNT, BOILEX, BOILPR, BOILP2, and BOILLP. Also, two new subroutines from MARCH-BWR will be added to perform safety relief valve modeling and the primary system pressure calculations.

Investigation of Radiant and Volumetric Heat Sources in the BWR Steam Separators and Standpipes (J. C. Conklin, Dissertation) Formulation and scaling of the governing conservation equations of mass, momentum and energy have continued. Formulation of the governing equations revealed two time scales that might arise due to the radioactive decay of the fission products. The first time scale will identify the low frequency buoyant effects due to relatively low values of volumetric heat source strength. The second time scale will identify the high frequency acoustic effects due to relatively high values of volumetric heat source strength. The mathematical character of the resulting scaled equations will be different for each time scale, requiring different numerical solution considerations. The volumetric heat source strengths of the Beta-emitting fission products entrained in the flow through the steam separators and standpipes will identify the appropriate analysis considerations.

Documentation of the Browns Ferry Secondary Containment Model (S. A. Hodge) Effort is underway to provide adequate documentation for the secondary containment model developed at ORNL to calculate the thermal hydraulic conditions in the reactor building and refueling bay under accident conditions. The model is driven by the flows from the drywell taken from MARCH code output.

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

1. MELRPI Code Development (R. Taleyarkhan) The new model for separate treatment of the melting and relocation of the control blades has been implemented into the main program and is currently being tested. Also, work is underway to model the molten masses released from a rubblized bed using the slug relocation model and to correctly model the hydrogen generation.

2. Coding and Documentation of the ECCS Models The digital computer subroutine ECCRPI has been implemented (and debugged) into the main body of MELRPI. It can be used to evaluate increasing/decreasing two-phase swollen levels and a single bypass level, for various injection modes, for both the intact and rubblized core geometries.

Several runs were made using ECCRPI for different coolant injection modes. Some of the results obtained were used in a paper entitled "The Modeling of Emergency Core Cooling Systems (ECCS) in a Degraded BWR Core." A summary of this paper has been accepted for the 1984 ANS Meeting in New Orleans, and the full paper has been submitted for publication in the Proceedings on Nuclear Thermal-Hydraulics.

3. Model Development for the Lower Head (R. Taleyarkhan) A new model has been developed, for meltdown, relocation and failure of components in the lower plenum. A subroutine (LPFRPI) has been written for the numerical implementation of this model. Presently, LPFRPI is being debugged, and test runs made to check numerical efficiency of the subroutine.

Group III: 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) In discussions with sequence analysts, we have jointly selected the specific ATWS sequence for study characterized by the following early events:

Initiation: MSIV closure, no scram
No LOSP; i.e., coolant injection available from the condensate booster pumps
Successful trip of recirculation pump
No operator action

The resulting events, up to an early drywell failure at 37 min, have been described in Chapter 3 of the draft NUREG/CR-3470. The sequence parameters following drywell failure are currently being developed. For these we have tentatively selected an assumption that the failure of the drywell simultaneously causes a large break LOCA and loss of coolant injection.

We are now beginning to revise the fission product transport calculation to handle this sequence. Principally this involves (1) obtaining

a revised initial fission product inventory to account for the 37 min of fission power from initiating event to drywell failure, and (2) including some additional short-lived precursors to the nuclide inventory.

In addition, we will add tellurium transport models to the transport calculations as they are developed.

Chemical Change Effects (E. C. Beahm) A model for iodine transport and chemistry has been obtained from Peter Hosemann of KfK. This fission product release model considers successive stages of an accident scenario including release from the fuel, release from the primary system into the containment, reactions and behavior inside the containment, and release from containment. In its present form, the model employs iodine as user-selected relative amounts of CsI or I_2 leaving the primary system. The CsI is assumed to follow aerosol behavior as calculated by the NAUA code. The I_2 interacts with sump water by hydrolysis, using a selected iodine partition coefficient (concentration of iodine in liquid phase divided by the concentration of iodine in the gas phase). Organic iodine is assumed to form at a rate that is dependent on the I_2 concentration until a preselected time when the organic iodine remains a constant fraction of the gaseous I_2 concentration.

This model has been loaded on the computers at ORNL and successfully run. Initially, the model will be used to calculate the sensitivity in the concentration of gaseous I_2 and organic iodine to different input ratios of CsI and I_2, and also to different values of the iodine partition coefficient. After the initial calculations, alternate descriptions of iodine reactions in containment will be inserted into the model.

Tellurium Transport Model (R. P. Wichner) A tellurium transport model is being developed based on tellurium behavior observed in NRC research programs, literature observations, and predicted chemical equilibria with various materials.

The general character of tellurium behavior may be outlined as follows:

1. Under the general reducing conditions expected in the reactor vessel prior to failure, Te is expected to exist principally as Te_2 in the vapor phase and as metallic Te liquid in the condensed form. As such, Te has a high tendency to dissolve in available metals, most strongly in Zr, secondarily in Ni and Cr, and to a lesser degree in Fe. Therefore, with unoxidized cladding or steel present, there will be a strong vapor pressure reduction due to dissolution.
2. The tendency for Zr to be oxidized by steam overwhelms its attraction for Te. Therefore, Te alloyed in cladding will be largely evolved when the cladding oxidizes.
3. The observed Te interaction with steel is generally one where initially a Ni-Te surface layer is observed, followed by an in-diffusion to combine with Cr-rich phases at later times. Thus,

while metallic Te has a relatively high volatility, under reactor vessel conditions there will be a tendency to sequester Te in either steel or cladding material.

4. Under containment conditions, elemental Te is expected to oxidize to form TeO_2. As such, it is expected to be transported in association with the mass of aerosol material generated by the corium/concrete reaction.

Review of the Loss of DHR Fission Product Transport Analysis (S. D. Clinton, R. A. Lorenz) The subject report, NUREG/CR-3617, is undergoing internal review. External review is also in progress.

Absorption of Gaseous Iodine by Water Droplets (M. F. Albert, Thesis Work) A well-mixed drop model for the adsorption of gaseous iodine by a water drop has been completed. This model assumes (1) a well-mixed drop, (2) the drop always falls at its terminal velocity, (3) no heat transfer occurs, and (4) the drop remains spherical at all times. Since the available data is from large scale containment spray tests (such as the containment spray experiments, BNWL-1244), the model for a single drop has been expanded for a spray system. Additional assumptions to those listed above are: (5) all drops are of the same size, and (6) there is no interaction between drops.

Some problems have been encountered in comparing the spray model to the data since the data has a number of sources of error (over 40% of the iodine released into the containment is unaccounted for). Some of the sources of error involve the removal of iodine by the walls, paint, piping, and insulation; other errors exist in the accounting techniques. To a large degree, measurement error was introduced due to the large, wetted structures and walls in the experiment which provided a far greater surface area for the adsorption to take place than did the total surface area of all of the drops in the containment. Another source of error (which might not be significant) is the absorption of iodine by the spray solution collected in the bottom of the containment. Approximate correction for the effect of these errors can be applied to permit a comparison of the spray model with the results. Preliminary results of the well-mixed drop spray model show errors ranging from insignificant to greater than 100%.

Observations as to the role of the chemistry on the absorption of iodine have been made. Chemical reactions lower the concentration of molecular iodine in the liquid, which increases the concentration driving force for mass transfer. Therefore, rapid kinetics would enhance the rate of mass transfer. According to the well-mixed drop model, the number of moles transferred from the gas phase to the liquid does show this increase, but resistance to mass transfer exists in the gas and liquid phases which diminishes its effect. The effect of the chemistry is significant for buffered solution of pH greater than 7 and at higher temperatures when the partition coefficient is small. For unbuffered solutions of pH less than 7, the iodine hydrolysis reactions tend to a steady-state value very quickly due to the production of hydrogen ions

(H^+) and the reactions nearly cease for the time interval of interest. Iodine hydrolysis rates therefore do not influence the rate of mass transfer for this case. The effect of chemical kinetics will assume greater significance as the concentration of iodine in the drop increases, such as for recirculatory sprays or as the time span gets larger. Therefore there are times when the chemical kinetics is an important factor in determining the rate of mass transfer of iodine.

MEETINGS AND TRIPS:

S. A. Hodge attended the NRC Containment Loads Study Group Meeting at Rockville, Maryland on March 13 and 14, and an NRC-sponsored meeting concerning severe accident unresolved issue prioritization, also at Rockville, on March 15 and 16.

S. A. Hodge, L. J. Ott, and C. R. Hyman attended the First NRC Review Meeting on Best-Estimate Severe Accident Analysis Codes at Silver Spring on March 28.

S. A. Hodge attended a meeting at TVA headquarters, Knoxville, on March 30 to discuss the ORNL-developed Browns Ferry secondary containment model.

REPORTS, PAPERS AND PUBLICATIONS:

M. Podowski and R. Taleyarkhan delivered a presentation at the NRC review meeting on best-estimate severe accident analysis codes covering the SASA program subcontracted work at RPI. This work involves the development of the MELRPI code for BWR severe accident analyses.

The draft report "Noble Gas, Iodine, and Cesium Transport in a Postulated Loss of Decay Heat Removal Accident at Browns Ferry," NUREG/CR-3470, ORNL/TM-9028, was distributed for peer review on March 8.

The draft report "Pressure Suppression Pool Thermal Mixing," NUREG/CR-3471, ORNL/TM-8906, was distributed for peer review on March 21.

PROBLEM AREAS:

None.