PROJECT TITLE: Severe Accident Sequence Analysis (SASA)

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

TECHNICAL HIGHLIGHTS:

Major work in progress during February includes completion of the draft report for the Anticipated Transient Without Scram (ATWS) accident sequence study, continuing modifications and improvements to MARCH-BWR in preparation for the degraded core calculations for the ATWS sequence, and completion of the draft report of the fission product transport study for the Loss of Decay Heat Removal (DHR) accident sequence. Unit 1 of the Browns Ferry Nuclear Plant serves as the model plant for all studies. The draft ATWS accident sequence report was issued for peer review comments on February 14 with comments requested by March 30. The fission product transport report for the Loss of DHR sequence will be issued for peer review in early March.

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

Most of the effort during February was devoted to completion of the draft report for the ATWS accident study, which was distributed for peer review comment in February. The BWR-LACP calculations of suppression pool temperature reported in the draft report are based on the assumption of uniform suppression pool temperature. For some of the accident sequences that involve discharge through a small number of SRVs, this may not be a good assumption. Therefore an effort is underway to utilize David Cook's 3-dimensional suppression pool model to calculate pool temperature and drywell pressure buildup for these cases. The results of these calculations will be included in the final ATWS study report.

The task of documenting the BWR-LACP code was initiated in February.

Group II: 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) The implementation of new BWR reactor vessel pressure control (replacement of subroutine PRIMP) and water level determination routines in MARCH-BWR affect the event timing for the Loss of DHR accident sequence as follows:

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

*All times relative to containment failure at 2050 min.

Thus these improvements have important effect on the calculations. A tape of this latest code version has been provided to the General Electric Company per their request. Action to provide separate models for fuel, gap, and cladding has been implemented.

Implementation of MARCH 2.0 at ORNL (L. J. Ott, C. R. Hyman) After accommodation of three intrinsic functions not available on the IBM system, MARCH 2.0 was successfully compiled. However, although about 5% of the sample problem has been run, successful execution has been thwarted by several underflow errors.

The actual incorporation of the ORNL BWR models in MARCH 2.0 has been carefully reviewed with the following findings:

1. The BWR core models provided to BCL in December were added as a separate subroutine which could be called from BOIL (the PWR calculations are done in BOIL separate from the BWR calculations).

2. The ORNL physical properties package for steam and hydrogen, necessary for BWR calculations because it is accurate at saturated conditions, was included in MARCH 2.0; however, it can be accessed in only two places: in BOIL in the PWR calculations, and in EXITQ. Specifically, this package can not be accessed by the BWR calculational routine.

3. Improved in-core heat transfer correlations are available for PWR calculations in BOIL, but these improvements were not incorporated in the BWR calculational routine.

4. The quench models (i.e., mechanistic models for core structures recovered during periods of level swell) were not incorporated in either the PWR or BWR calculational sections.

5. The BWR safety relief valve models were not incorporated in PRIMP (the routine which calculates reactor vessel leakage and pressure).

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. For flow without volumetric heat sources, a thermal entrance length along the standpipe axis will exist before an invariant radial temperature profile is established. Inclusion of volumetric heat sources in the flow will increase this entrance length, with the increase being a function of Reynolds Number as well as heat source strength. This increase in entrance length will affect the ratio of bouyancy to inertial forces along the standpipe axis and will be considered for investigation of temperature effects due to fission product behavior.

MARCH Modifications for Containment Analyses in the ATWS Study (C. R. Hyman) Two modifications have been made to MARCH-BWR. The first corrects the improper modeling of T-quencher discharge from the SRVs; for depressurized containments and input parameter NT = -7, the SRVs now continue to discharge into the pressure suppression pool. The second modification corrects two algebraic errors in subroutine HEAD and affects the calculation of the debris temperature.

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

1. MELRPI Code Development (R. Taleyarkhan) Several structural modifications are underway with regard to implementation of a new model for control blade melting and relocation. Instead of one grid for each radial zone, two independent grids have been introduced. The first grid tracks motion of the channel walls and fuel/clad in a node; whereas the other is used for the control blades. The original SLUG routine has been modified to account for molten stainless steel relocation. The logic necessary to evaluate neighboring structural and fluid temperatures for radiation and convective heat transfer calculations has been worked out and independently tested.

Results obtained from MELRPI for a TQUV sequence with a stuck open relief valve have been compared with those obtained from MARCH. An option allowing for independently specifying the steam inflow to the dry region via input, has been implemented in MELRPI. The two-phase level, system pressure and total steam inflow as evaluated by MARCH, were used as input to MELRPI. A comparison of results from these two codes indicates that the MELRPI results are more conservative and, furthermore, that once a region of the core started melting, the MARCH code results are not physically consistent. This is so because of the inherent structural deficiencies of MARCH, which lead to coolant temperature fluctuations of several thousand degrees Fahrenheit. Since essential aspects of degraded cores, such as modeling of clad failure and rubble

bed formation are not accounted for by MARCH, the need for a code like MELRPI is apparent. Some results of the comparison between MELRPI and MARCH have been used in the summary of a paper submitted for the 5th International Meeting on Thermal Nuclear Reactor Safety.

2. Coding and Documentation of the ECCS Models. After final testing of the ECCS subroutine in conjunction with MELRPI, several runs were made to check out the effectiveness. The results obtained have been documented, and submitted as a paper for the ANS 1984 Annual Meeting.

Work is also under way to document the ECCS modeling scheme. The preliminary draft is under revision and will be published at ORNL as a NUREG/CR report upon completion.

3. Model Development for the Lower Head (R. Taleyarkhan) Work continues on the development of a model for the relocation of molten materials in the lower plenum, and for the reactor vessel bottom head failure. Two basic options have been established, one for a gradual relocation of small molten masses, and the other for a large mass of corium suddenly slumping on the lower head.

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 Analysis (R. P. Wichner, E. C. Beahm, C. F. Weber, A. L. Wright) The draft report "Noble Gas, Iodine, and Cesium Transport in a Postulated Loss of Decay Heat Removal Accident at Browns Ferry" (NUREG/CR-3617) has been completed and is in the process of distribution for internal and external review. The summary and conclusions of the report are summarized below:

1. There is little barrier for krypton movement through the reactor system. Steam evolution from the reactor vessel and suppression pool and concrete degradation gases evolved in the drywell effectively flush krypton. However, significant decay of krypton radioactivity occurs due to the long delay time in this sequence. Of the 4939 Pq present at scram, only ~20 PBq of krypton are projected to enter the atmosphere, principally due to radioactive decay. This amount of krypton activity represents 82% of the current activity.

2. Xenon transport results parallel those for krypton except with a smaller degree of radioactive decay. In addition, an appreciable degree of Xe radioactivity builds in from decay of iodine precursors. About 4700 PBq of Xe are projected to be released to air, which represents ~82% of the Xe activity at that time. The remainder resides principally in the fuel elements remaining in place, the reactor building air space, and dissolved in water leaked into the condenser through the shut MSIVs.

3. A far smaller degree of iodine is predicted to enter the atmosphere than noble gas because of the chemical reactivity of iodine. About 0.05 PBq of iodine are predicted to be in the atmosphere, which represents 7.3×10^{-4}% of the iodine activity existing at the end of the calculation; this is 1.8×10^{-5}% of the amount existing at reactor scram. The principal repositories of iodine at the end of the calculation are: plateout on reactor vessel surfaces (88%), retention in unfailed fuel (5.7%), as dissolved material in the reactor building water pool (3.5%), and plateout on drywell surfaces (2.6%).

4. The principal iodine pathway to the atmosphere is the following: (a) retention in the fuel during core degradation and passage with fuel rubble onto the drywell floor, (b) spraging release from the core debris on the drywell floor, (c) convective transport through the drywell and reactor building both as gaseous iodine and as chemisorbed iodine on aerosols, the convection of aerosols produced by the core/concrete reaction, and (d) passage through the SGTS filter/absorber system.

5. The suppression pool does not capture a large share of the iodine evolved from fuel because the reactor vessel surfaces capture most of the iodine via plateout (of various types) when the suppression pool is in the direct path, i.e., times prior to reactor vessel failure.

6. The assumed degree of aerosol trapping in the reactor vessel does not greatly affect the calculated iodine release because the subsequent sparging release from core rubble on the drywell floor overwhelms any variation in release created by varying assumptions concerning aerosol trapping in the reactor vessel.

7. An estimated 0.04 PBq of cesium activity is projected to enter the atmosphere by the end of the sequence; this represents about 1.9×10^{-4}% of the initial core inventory. Virtually all of this release is predicted to occur after the time of HEPA filter failure in the SGTS.

8. The principal cesium pathway differs somewhat from that of iodine due to its much lower volatility in the drywell and reactor building. Cesium transport in these areas occurs almost exclusively as material condensed on aerosols, whereas iodine transport occurs both as a gas and associated with aerosols. Hence, the critical factor in cesium release to air is the time for HEPA filter failure; virtually all cesium release occurs subsequently. Summarizing the principal cesium pathway: (a) retention in the fuel and passage onto the drywell floor with the core rubble, (b) sparging release from the core debris on the drywell floor, (c) transport through the drywell and reactor building associated with aerosol particles generated by the core/concrete reaction, and (d) passage through the SGTS following HEPA failure.

9. The principal cesium repositories at the end of the sequence are: (a) plated material on the reactor vessel upper surfaces, both as condensed CsOH and cesium associated with deposited aerosols (2.7%), (b) in the portion of the original fuel assemblies which are predicted to remain intact (0.32%), and (c) plated on drywell walls (0.20%). Very little cesium appears in the suppression pool ($2.2 \times 10^{-3}\%$) due to the effective trapping on reactor vessel surfaces of condensible materials. About half this amount resides in the reactor building basement pool due to aerosol settling on the building and washdown by the fire-protection spray water. (The above percentages are activity levels referred to the core activity level at scram.)

Although no formal sensitivity analysis was performed, the principal areas of uncertainty are believed to involve the SGTS failure model and the assumptions concerning iodine volatility over water, plateout and condensation in the reactor vessel, and the rate of sparging release from the core rubble on the drywell floor. These uncertainties are discussed in Section 6.3 of the report.

Chemical Change Effects (E. C. Beahm) Thermochemical equilibrium calculations of the chemical form of fission product iodine and cesium are being expanded to examine reactions with reactor vessel material and to estimate the range of uncertainty based on uncertainty in thermochemical data. For purposes of modeling a given SASA sequence, the calculations are done using equilibrium constant expressions which are easy to include in a model. In addition, thermochemical equilibria are calculated using the SOLGASMIX program modified at ORNL to give uncertainty in calculated results. Linear programming is used to check calculations where large uncertainties in the results are expected.

The equilibrium constant calculations are based on the equation:

$$CsI + H_2O = CsOH + 1/2H_2 + I \ .$$

Equilibria for other species such as HI, $CsBO_2$, Cs_2CrO_4 are then included to give the distribution of iodine between CsI, HI, and I as a function of temperature, and pressure of steam and hydrogen. Interval arithmetic is used to determine the range of values for the equilibrium constants.

Absorption of Gaseous Iodine by Water Droplets (M. F. Albert, Thesis Work) An evaluation of the iodine hydrolysis reactions has been completed and a kinetic model for the absorption of iodine by a water droplet has been chosen. The kinetic model is similar to the kinetic rate equations outlined in NUREG/CR-1900, except it is assumed that the reversible reaction between iodine (I_2) and iodine (I^-) and tri-iodine (I_3^-) is at equilibrium at all times. This reaction is extremely fast and the reaction reaches equilibrium in the time step taken.

The kinetic model is now being used in a mass transfer model, which assumes a completely mixed drop. At the present time, Higbie's penetration theory is used to determine the liquid-side mass transfer coefficient.

MEETINGS AND TRIPS:

C. R. Hyman and L. J. Ott attended the Containment Loads Study Group Meeting at Palo Alto, California, on February 1 and 2.

L. J. Ott attended a meeting concerning the NRC-sponsored QUEST program at Sandia on February 23 and 24 (not SASA-funded).

REPORTS, PAPERS AND PUBLICATIONS:

The draft report "ATWS at Browns Ferry Unit One — Accident Sequence Analysis," NUREG/CR-3470, ORNL/TM-8902, was distributed for peer review on February 14.

PROBLEM AREAS:

None.