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PROJECT TITLE: Severe Accident Sequence Analysis (SASA)

PROJECT MANAGER: S. A. Hodge

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BWR-LTAS calculations are being performed in support of the Browns Ferry Loss of Control Air Accident sequence study. The improved version of the BWR-LTAS code with modifications to permit application to BWR MK II containment analyses has been transmitted to the SASA program at Sandia National Laboratory (SNL) and successfully executed there. The current version of the MARCON 2.0B code has also been provided to the SNL SASA program for execution there and for upgrading of the subroutine CORCON models for corium-concrete reaction. 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. The MARCON 2.0B is being exercised locally on the Browns Ferry Station Blackout accident sequence. Checkout of the MELRPI code on the ORNL computer system continues. (This code permits study of the core response after significant degradation of the original geometry.) The fission product transport analyses group is ready to perform the fission product transport calculations as soon as the thermal-hydraulic results from MARCON 2.0B are available.

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

It was agreed during January that three coding improvements would be made to BWR-LTAS at ORNL to permit the application of the code to the general BWR MK II containment design. These modifications have been completed and involve the Residual Heat Removal (RHR) system heat exchanger effectiveness model, the primary containment heat sink models, and the provision of models to permit the study of small break LOCAs within the primary containment. A tape of the new coding has been provided to the SASA program at SNL and they have successfully reproduced the sample run included on the tape.

Calculations with BWR-LTAS have begun for the Browns Ferry Loss of Control Air accident sequence. One set of calculations was intended to determine the period of time between reactor scram (and vessel isolation) and receipt of a high drywell pressure signal (at 2.5 psig) if the drywell coolers are not operating. The best estimate is 11

minutes, obtained from a calculation with realistic heat losses from the primary system and an assumed 5 gpm leak from the primary system to the drywell atmosphere. It is important to note that the recirculation pump motors were assumed to be deenergized at the time of scram as would most likely be the case. (If the recirculation pumps continued to run, they would contribute as much heat to the drywell atmosphere as all of the other heat sources combined). The calculation also includes consideration of the large temporary additional heat source from the scram outlet lines after scram.

A second set of calculations has been performed to determine how many RHR system coolers are required after reactor scram and isolation to prevent loss of the RHR pumps on low Net Positive Suction Head (NPSH). The cooler heat transfer coefficient (Btu/s-°F) was established after consultation with Bill Brock of TVA. The calculations involve modeling of the HPCI operation as pumping at full flow (5000 gpm) from the condensate storage tank (CST) back to the CST with the required amount of injection (about 400 gpm) bled into the reactor vessel as required to maintain the desired level. This mode of operation prevents safety relief valve (SRV) cycling since the HPCI turbine steam demand at full flow is equivalent to about one-fourth of the steam flow to one SRV at full reactor pressure. [It should be noted, however, that this method is not feasible if the drywell coolers are not operating because the HPCI system will automatically realign for full injection to the reactor vessel upon receipt of a high drywell pressure signal.] Although calculations are not yet completed, it is believed that operation of one RHR system cooler will be found to be sufficient to maintain NPSH.

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

<u>Development of MARCON-2.0B at ORNL</u>: (C. R. Hyman, L. J. Ott, T. L. Heatherly) A package consisting of a magnetic tape of the Fortran source listing of the ORNL-revised MARCON 2.0B code, a sample problem input deck, paper output produced at ORNL from the Fortran source and the sample problem input, a cross-correlation listing of subroutines and common blocks, and a written description of the required code input parameters was transmitted to Andy Peterson of the SNL SASA Program on March 20. The BWR severe accident analyses model upgrades included in this package are:

- separated fuel and cladding models
- axial conduction
- improved heat transfer correlations for uncovered regions of core
- corrected energy balance for covered portion of core during pressure transitions caused by reactor vessel relief valve cycling
- improved modeling of the effects of relief valve actuation

- new reactor vessel pressure algorithm
- new equation of state for hydrogen-steam mixture
- specific calculation of reactor vessel water level as a function of vessel internal structure and the liquid inventory within the vessel
- multi-node analysis of the states of the in-vessel water inventories
- new boiling and flashing algorithm
- new Zr/steam reaction models
- new modeling for ECC injection systems
- extensive reformatting of the printed output for the primary system calculations to highlight parameters important to BWR severe accident analyses.

It is emphasized that there has been no attempt to maintain PWR analyses capability for the code. In fact, every effort has been made to strip specific PWR models involving ice condensers, steam generators, etc. from MARCON 2.0B in the interest of streamlining and ease of future code modifications for BWR applications.

Because the current model upgrades are so extensive, we have not yet been able to confirm the correct operation of all input options listed in the provided input description. In addition, there are some known deficiencies in the current models that will be corrected in the next revision of the code; this revision will incorporate the B_4C reaction models discussed in the next paragraph and will be provided to SNL (SASA) in about 90 days. The current version of the code is provided to SNL at this time so that modeling revisions to the CORCON and associated routines of this joint code can begin now. We have also asked that SNL repeat the sample problem at their facility and provide us a copy of the printed output so that we can assess the effect of the higher computer precision normally available at SNL, but only available at ORNL if we run the code at "double-precision." In other words, we wish to know if an effort to convert MARCON 2.0B to run in double-precision at ORNL would be worthwhile.

One final modification to be implemented into the code at ORNL is the coding necessary to employ the models for the reaction of control rod B_4C powder with steam that have been developed by Ed Beahm of the ORNL SASA Program Group III in previous work. Local fission product transport calculations both before and after the incorporation of the B_4C reaction models are planned for the purpose of demonstrating the importance of consideration of this reaction in BWR fission product transport studies.

<u>Status of MARCH 2.0 (V151) at ORNL</u> (C. R. Hyman) We were grateful to receive a letter from Roger Wooten of Battelle's Columbus Laboratories on January 30 that diagnosed the cause of our difficulties during the fall of 1984 in attempting to run the final release version of MARCH 2.0 (V151) on a hypothetical Browns Ferry accident sequence. We learned that the code has an internal timestep control algorithm that reduces the computational timestep to 0.0001 minute every time the vessel break

size increases from zero. Since SRV operation is treated as a vessel break and only one SRV is required to control vessel pressure, the timestep is severely reduced each time the SRV cycles from closed to open. Since we obtained no output after long computational times, we had assumed that the calculation must be stuck in some sort of infinite loop. We now know that it was merely a case of the calculational timestep continually being set back to 0.0001 minute.

In order to circumvent the problem, Roger Wooten suggested that we set the reactor vessel break area parameter to 1.0×10^{-20} so that the total reactor vessel break area would never be zero during the BWR relief valve cycling. This "fix" works and we are now able to perform MARCH 2.0 calculations that involve BWR transients and the concomitant relief valve cycling.

A smaller source of difficulty in our previous attempts to run MARCH 2.0 on BWR problems was also resolved by Roger Wooten's letter. MARCH 2.0 assumes that the relative humidity of the wetwell airspace must be 1.00. If the user exercises the available option to input a relative humidity less than 1.00, the MARCH 2.0 program will model the immediate vaporization of enough pressure suppression pool water to bring the relative humidity in the wetwell airspace to 1.00, with the concomitant artificial pressure increase.

We have also learned that the gas fractions WCMO, WHYD, WNTR, and WOXY input to MARCH 2.0 are to be taken as fractions of the non-condensibles, not fractions of the total gas-steam mixture.

Hydrogen Generation Calculation for Grand Gulf (S. A. Hodge, L. J. Ott) Calculations to determine the maximum hydrogen generation rates that might occur under severe accident conditions, not inherently resulting in core melt, at Grand Gulf have been completed by the Hydrogen Control Owner's Group (HCOG) staff using the industry MAAP code. At the request of Mark Wigdor of the NRC (NRR) staff, equivalent calculations are being performed by the ORNL SASA program using the special BWR models available here. Results are provided to Ji Wu Yang at Brookhaven National Laboratory.

The accident sequence that results in the maximum hydrogen generation rates is reactor scram with MSIV closure accompanied by a total loss of injection (TQUV). It is assumed that the operator manually initiates the Automatic Depressurization System (ADS) to open eight safety relief valves (SRVs) and these remain open throughout the sequence. The reactor vessel water level decreases as the initial water inventory is boiled and flashed away and the temperature of the uncovered fuel increases rapidly. At precisely the right moment,* injection of water to the reactor vessel is restored - not enough to

*From the standpoint of maximizing the hydrogen generation rate.

reflood the core and terminate the accident, but just enough to supply the steam necessary to fuel the $Zr-H_2O$ reaction. The modeled injection rate is 300 gpm, which represents the rate available by operation of two control rod drive (CRD) hydraulic system pumps.*

The HCOG calculations were performed for a constant reactor vessel pressure of two atmospheres and an initial fuel temperature of 250°F (saturation temperature corresponding to two atmospheres) since they assume that steam cooling provided by ADS operation would cool the fuel to this extent. The calculations at ORNL are:

1. to provide a calculation of the depressurization transient so as to determine the fuel temperatures after depressurization and thereby provide a check to the HCOG assumption of initial fuel temperature, and
2. to calculate the subsequent hydrogen generation rates and thereby provide a check to the HCOG results.

Results of the hydrogen generation calculations at ORNL show that the nature of the sudden vessel depressurization determines the consequent fuel node temperatures; these are a function of core height and need to be calculated. Nevertheless, the hydrogen generation rates predicted by the ORNL version of MARCH closely approximate the IDCOR code predictions. This fact was not known at ORNL until an independent comparison of the results of the two codes was completed at Brookhaven.

<u>BWR Degraded Core, ECCS, and Lower Plenum/Lower Head Failure Analysis and Model Development</u> (A. Sozer) Related work performed at ORNL and at RPI is reported in the following paragraphs:

1. <u>MELRPI.MOD2 Coding Analysis</u> (A. Sozer, T. L. Heatherly) The MELRPI.MOD2 coding and its models have been transmitted to ORNL and are being modified, improved, and corrected as necessary, and made more efficient wherever this can be done.

2. <u>BWR Core Uncovery Analysis</u> (A. Sozer) For the first time, the MELRPI-MOD2-ECCS models developed for flooding the core have been utilized in core uncovery calculations for accident sequences (Loss of Decay Heat Removal and Loss of Injection) with no injection into the vessel. The purpose of these calculations is to provide comparisons to MARCON results and also to provide radial distribution of the two-phase levels in different zones. The results from Loss of Injection (TQUV) calculations indicate that the timestep size has a significant impact on the time to dryout and hand calculations have been performed to approximately determine the correct time. Further TQUV runs are underway. Trial Loss of Decay Heat Removal runs have

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*Operation of just one CRD hydraulic system pump would not prevent core melt.

resulted in modifications to the code. Results have been obtained for approximately three transient hours, and are under scrutiny.

Problems occurred in using the restart option of the code. A sample output with the input description has been sent to RPI.

3. Modeling of Lower Plenum/Head Failure and In-Vessel Thermal-Hydraulics (M. Z. Podowski, RPI) We have been working on further improvements and verification of the modeling assumptions used, as well as on the computer implementation of the overall model.

   A multi-node model for the analysis of flow and heat transfer transients inside the reactor vessel has been developed. Recently, we have been working on the computer implementation of this model, and testing the numerical method of integrating the resulting system of differential equations.

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.

Absorption of Gaseous Iodine by Water Droplets (M. Albert) A draft report was distributed for peer review during March which documents work recently accomplished under the sponsorship of the SASA program at ORNL to provide a methodology for predicting the rate at which gaseous molecular iodine is absorbed by water sprays. The new model was developed by Mike Albert, until recently a graduate student at the University of Tennessee, who performed this research in fulfillment of the requirements for a Master's degree.

This work is important to the ORNL SASA program since it will be used to calculate the effectiveness of the Browns Ferry reactor building fire protection system sprays in removing gaseous molecular iodine from the secondary containment under severe accident conditions.

Standby Gas Treatment System (SGTS) Effectiveness Calculations (S. D. Clinton) The results of recent studies by various organizations have shown that the BWR MK I Primary Containment would fail either by overpressure or overtemperature not long after reactor vessel failure under severe accident conditions. Therefore, it is most important that the capability of the BWR secondary containment to mitigate the concomitant releases of fission products to the atmosphere be assessed.

Previous ORNL SASA program studies have shown that the effectiveness of the SGTS system is a major factor in mitigation of fission product releases for postulated severe accident sequences at Browns Ferry. Accordingly, we have sponsored tests at the filter test facility at New Mexico State University to determine how long the

installed HEPA filters would perform their function under severe accident conditions. The progression of these tests over the last several months has been discussed in the applicable monthly reports. The current status is reported in the following paragraphs.

After terminating the dry latex aerosol HEPA filter loading run discussed in the last report, the HEPA filter from the first water spray experiment was returned to the test facility. With the new provision of three axial vane fans in series, the water spray test (300 g of water per minute) was continued until the pressure drop across the filter reached 15 inches of water. Subsequent examination of the HEPA filter revealed no damage and no apparent decrease in the particulate removal efficiency. It can be concluded that the type of HEPA filter installed in the SGTS at Browns Ferry would probably retain its structural integrity during a Severe Accident sequence in the presence of complete loading by either water spray or dry aerosol.

The pressure drop, flow rate, and mass accumulation data taken for the tests completed to date are being analyzed by D. L. Fenton of the New Mexico State University, Mechanical Engineering Department, and will be provided to the ORNL SASA program prior to commencement of the third HEPA filter test.

MEETINGS AND TRIPS: C. R. Hyman attended a two-day workshop at SNL on the use of the CONTAIN code on March 4 and 5.

R. M. Harrington and S. A. Hodge visited the TVA Headquarters at Knoxville on March 21 to obtain information needed for the Browns Ferry Loss of Control Air Accident Sequence Study.

REPORTS, PAPERS, AND PUBLICATIONS: The draft report Absorption of Gaseous Iodine by Water Droplets, NUREG/CR-4081, ORNL/TM-9438, was distributed for peer review on March 26. Comments have been requested by May 1 so that the final report can be distributed in early June.

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