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PROJECT TITLE: Severe Accident Sequence Analysis (SASA)

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Three improvements to the BWR-LTAS code to provide for its application to the general BWR MK II containment design were developed during [redacted] The improved version of the code will transmitted to the SASA program at Sandia National Laboratory (SNL) in early March for use there in support of the RMIEP program. The implementation of all existing ORNL-developed BWR models into MARCON 2.0B was completed during February and a written description of the input parameters was assembled. The MARCON 2.0B code will now be transmitted to the SNL SASA program for use 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. Checkout and testing 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.) In the fission product transport analyses group, preparations for the transport calculations for the ATWS accident sequence have been completed. Four presentations concerning ongoing or recently completed work at ORNL were delivered at the SASA Program Review Meeting at Silver Spring on February 20.

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-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. The first two improvements, which involve the Residual Heat Removal (RHR) system heat exchanger effectiveness model and the primary containment heat sink models have been completed. The third improvement involves models to permit the study of small-break LOCAs within the primary containment and is nearly completed. It is anticipated that the new version of the BWR-LTAS code with the above improvements will be transmitted to the SASA program at SNL during early March.

BWR-LTAS calculations for the Station Blackout accident sequence with the operator actions assumed by the IDCOR study were performed during February. The results were contrasted to those that result from the operator actions assumed by the ORNL SASA program in a presentation at the SASA Program Review Meeting on February 20.

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.

Development of MARCON-2.0B at ORNL (C. R. Hyman, L. J. Ott, T. L. Heatherly) Work to complete the incorporation of all existing ORNL-developed BWR models into the SNL code MARCON (essentially MARCH 2.0 with the INTER package replaced by CORCON MOD2) was completed during February. The current version of the code is designated MARCON 2.0B and has nine BWR modeling features not available in MARCH 2.0. These are:

1. Improved package for calculation of hydrogen and steam properties
2. Improved equation of state for hydrogen-steam mixture
3. Improved depressurization algorithm
4. Improved representation of safety/relief valve effects
5. Heat transfer correlations for covered region of core
6. Accurate calculation of reactor vessel water level
7. Corrected Zr/H_2O reaction algorithms
8. Separation of fuel and cladding
9. Separation of the water inventory into core, lower plenum, and downcomer regions.

The tables of information that describe the required input for MARCON-2.0B have been placed on the ORNL word processing system and modified as necessary to include the new input needed for the BWR models and to eliminate the requirement for input specific to PWR applications such as ice bed temperatures, etc.

A tape of the modified coding and a copy of the modified input description is being transmitted to the SASA program personnel at SNL. One final modification to be developed and implemented into the code is the coding necessary to implement 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.

The MARCON 2.0B code will be considered complete from the standpoint of special BWR modeling when it has been modified to include the B_4C reaction models. A tape of the coding and a copy of the associated input description will also be provided to the SNL SASA program at that time. Locally, the code will be first applied to the Browns Ferry ATWS accident sequence degraded core response studies.

Hydrogen Generation Calculations 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.

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 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 it is assumed that the steam cooling provided by ADS operation would cool the fuel to this extent. The calculations at ORNL have two broad purposes. These 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 accomplishing the first purpose have been completed during February. As expected, the results show that the minimum temperature reached by each axial node is a function of its height in the core. The node is cooled during the depressurization while it remains covered, but begins to heat up after it is uncovered. For example, the minimum temperature reached by the node at the core midplane is 336°F.

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*From the standpoint of maximizing the hydrogen generation rate.

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

Previous work by Ji Wu Yang at Brookhaven has indicated that the initial fuel temperature parameter is important to the calculated hydrogen generation rates. The question is complicated, however, by the timing of the subsequent restoration of reactor vessel injection that restores the steam supply.

It is expected that work to accomplish the second purpose will be completed during March. The hydrogen generation rates will be calculated by MARCON 2.0B.

Availability of the CONTAIN Code at ORNL (C. R. Hyman) The CONTAIN code is operational on the ORNL computer system and available for use by the ORNL SASA program. This is by courtesy of the NRC-sponsored LWR Aerosol Release and Transport Program at ORNL and involves no cost to SASA. C. R. Hyman will attend a workshop on the use of CONTAIN at SNL during the first week of March.

BWR Containment Response Under Accident Conditions (C. R. Hyman) Existing MARCON documentation was reviewed for applicability to modeling the drywell sumps for the BWR MK I containment. Models are available that analyze thermal radiation exchange among surfaces enclosing a reactor cavity in a PWR and the ablation of concrete walls surrounding the cavity. These models are not currently available for BWR drywell sump analysis. ORNL has requested that SNL make these models operational for drywell sump analysis in a forthcoming version of MARCON 2B.

MARCH 2.0 (V151) at ORNL (C. R. Hyman) On February 7, 1985, ORNL received a package from Roger Wooton of Battelle Columbus Laboratories concerning his diagnosis of ORNL's problems with running MARCH 2.0 (V151) at ORNL (ref: Oct. 1984 SASA Monthly Report). ORNL is studying this package at this time.

BWR Degraded Core, ECCS, and Lower Plenum/Lower Head Failure Model Development (A. Sozer) The results of the MELRPI sample problems for ECCS initiation after significant core degradation have been reproduced on the ORNL computer system using the current version of the MELRPI code. The final version of the subcontractor report describing the models for degraded core response to sudden ECCS restoration is being published as NUREG/CR-3889. Additional work at RPI during February is reported in the following paragraph:

Modeling of Lower Plenum/Head Failure (M. Z. Podowski) Model development for BWR lower head failure by melt release into the lower plenum is nearly completed. Testing of the FORTRAN subroutine is underway.

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) Final planning for the communications links between MARCON 2.0B and the ORNL SASA program fission product transport code was established during a meeting with L. J. Ott and C. R. Hyman of the ORNL SASA Program Group II on February 20.

The Effect of Precursor Decay upon BWR Fi~ ion Product Transport (C. F. Weber) Calculations were performed during February in preparation for a brief presentation at the SASA Program Review Meeting by S. D. Clinton concerning the precursor effects phenomena. Briefly, unless consideration of the effects of radioactive decay of Tellurium is included in the calculations of fission product transport for BWR severe accident sequences, a sneak pathway for radioactive iodine to bypass the pressure suppression pool will be overlooked.

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. Since the Browns Ferry SGTS blowers are capable of sustaining a pressure drop equivalent to 16 inches of water across the upstream HEPA filter that would be loaded by aerosols under severe accident conditions, we have asked that the tests demonstrate the effects of this pressure drop across the filters. In response to this requirement, additional in-series axial vane fans were added to the test facility so as to approach the required differential pressure.

The dry latex aerosol HEPA filter loading test was terminated during the last week of February at a pressure drop across the filter of 15 inches of water because of signs of potential damage to the ducting of the test facility should the test continue. However, careful inspection of the HEPA filter itself revealed no signs of damage. It seems reasonable to conclude that the type of HEPA filter installed at Browns Ferry would withstand the maximum pressure drop that could be imposed by the SGTS blowers and would therefore ultimately become completely plugged during the latter portion of a Severe Accident sequence. Correlations based upon the test results will be incorporated into the ORNL SASA program secondary containment model.

MEETINGS AND TRIPS: A. Sozer attended the BWR technology course for the BWR 4 MK I plant represented by Browns Ferry offered by the NRC Office of Inspection and Enforcement at Chattanooga, TN (Feb. 11–15). This course is well worthwhile and is recommended for anyone planning to perform studies concerning accident sequences at BWR facilities.

A. Sozer visited the RPI subcontractor at Troy, NY on February 18 and 19 to discuss the status of the lower plenum/lower head failure models and the coolant dynamics models under development for the MELRPI code.

S. A. Hodge, S. D. Clinton, R. M. Harrington, and A. Sozer attended the SASA Program Review Meeting at Silver Spring, MD on February 20–21 and each made a presentation concerning ongoing or recently completed work.

REPORTS, PAPERS, AND PUBLICATIONS: The paper "Tellurium Precursor Effects on Iodine Transport in a BWR Accident" by C. F. Weber has been accepted for presentation at the forthcoming ANS meeting at Boston, MA on June 9–14, 1985.

The final version of the report "The Modeling of BWR Core Meltdown Accidents — For Application in the MELRPI.MOD2 Computer Code," NUREG/CR-3889, was submitted for makeup and reproduction on February 7.

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