TECHNICAL HIGHLIGHTS/ADMINISTRATIVE REPORT
FOR
THE NUCLEAR REGULATORY COMMISSION (NRC)
REACTOR SAFETY RESEARCH PROGRAM
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INTRODUCTION

Sandia National Laboratories, Albuquerque, New Mexico, is conducting a broad-based reactor safety research program on behalf of the U.S. Nuclear Regulatory Commission (NRC). The overall objective of the program is to provide NRC a comprehensive data base essential to (1) defining key safety issues, (2) understanding the controlling accident sequences, and (3) developing and verifying the complex computer models used in accident analysis and licensing reviews.

Together with other programs, the Sandia effort is directed at assuring the soundness of the technology base upon which licensing decisions are made and includes experiments and model and code development.

Priority is currently assigned to those tasks important to the resolution of issues raised as a result of the accident at TMI II. Phenomenological research is directed toward identification, quantification, and modeling those physical processes that determine containment loads and threats, as well as radioactive fission-product release and transport in the event of a severe accident. Many contemporary safety analysis computer codes are written without the benefit of experimental data to guide the developer in properly conceptualizing and quantifying "risk significant" phenomena. Analyses are currently conducted with these codes to support important safety-related decisions with only a vague understanding of uncertainty in results. It is the function of the research reported herein to quantify and reduce these uncertainties and to provide the safety analysis community with more robust and technically defensible analysis tools. It is important to keep in mind while reading this report that individual phenomena currently being quantified and modeled will not necessarily be important for each specific accident sequence at each plant. It is not the primary purpose of this research to make those judgments, but to provide data and models so that self-consistent, technically defensible predictions can be made on a case-by-case basis.

CONTENTS

1. CONTAINMENT LOADING AND RESPONSE

In the analysis of severe accidents, several scenarios lead to the release of molten-core material and the subsequent interactions involving molten fuel, coolant, structural material, potential in-core retention systems, the reactor vessel, and the reactor cavity boundary. The present program seeks to identify the results of these interactions. The results also will be used to support model development for the containment code CONTAIN, the melt progression code MELPROG, and the phenomenological models CORCON and VANESA and to provide data for their verification.

1.1 Ex-Vessel Core Debris Interactions

(D. A. Powers, J. E. Brockmann, E. R. Copus, J. E. Gronager, 6422)

1.1.1 Current Progress and Technical Highlights

1.1.1.1 Core Debris Interactions With Concrete

The predominant activity was the preparation of reports on experiments done during the year. Specifically,

1. The report on the TURC 1T and TURC 1SS tests was being finalized for publication,

2. Data analyses were completed for the reports on the SWISS 1 and SWISS 2 tests,

3. Work was begun on the report for the TURC 2 and TURC 3 tests, and

4. A draft report on the HSS tests was written.

The report on the TURC 1 test has been submitted to the NRC in draft form. The other reports are scheduled for submission to the NRC in September.

Another document prepared describes the experimental conditions for the TURC 2, SWISS 1, SWISS 2, and CC-2 tests and is entitled <u>Comparison of Code Predictions with Results of High Temperature Melt/Concrete Interactions Tests-- Description of Tests</u>. These descriptions are intended to be used as input data for "blind" predictions of the tests using computer models of core debris interactions with concrete. The NRC has organized an exercise to compare the predictions of the various computer models of core debris interactions with concrete (DECOMP, WECHSL, CORCON, and INTER) to the data obtained in these tests. The description of the test conditions has been sent to the developers and major users of these computer codes.

Two separate effects tests were run to qualify data from major tests and to prepare for future tests:

1. Power meters on the induction heating system used in the SWISS tests were calibrated. The instrumented, insulated block similar in size to the solidified metals found in the posttest dissection of the SWISS tests was used for the calibration.

2. Twenty-six kg of an UO_2-ZrO_2-Zr mixture were heated to a melting temperature while in contact with concrete using the IRIS technique. The purpose of the test was to qualify instrumentation techniques to be used in the large-scale SURC tests of urania/concrete interactions.

The HS-2 test was run. The HS series tests examined the interactions of core debris that is hot enough to ablate concrete but not hot enough to melt. The results of this research address issues concerning (1) the behavior of core debris following a transient quench or (2) the behavior of concrete long after the start of melt/concrete interactions when temperatures have fallen sufficiently for debris to solidify. The HS-2 test involved the interaction of steel with concrete.

In the HS-2 test, concrete ablation rates were observed to be about 1.5 cm/h when the steel was solid. Later in the test sufficient power was applied to the slug to melt it and ablation rates rose to about 6 cm/h. Ablation of the concrete was monitored both by thermocouples embedded in the concrete and by imaging using the gamma ray image-intensification system developed in this program.

Gas and aerosols evolved in the test were monitored. First examination of the gas composition data suggests the CO_2 evolved from the concrete was not reduced completely to CO as has been observed in the past. This suggests some kinetic inhibitions arise in gas reactions with core debris once the debris has solidified. Such inhibitions, if they occur in reactor accidents, would limit the amount of flammable gas released into the reactor containment during long-term core debris/concrete interactions.

Finally, the test setup was begun for a hot solid test of urania interactions with concrete. This test will be fully instrumented for heat fluxes, gas generation, gas composition, aerosol generation, and concrete erosion. Visual data on the erosion rate and pattern will be obtained using the gamma ray imaging system. Sustained heating will be by induction.

1.1.2 Anticipated Activity

Most of the attentions in the program will be devoted to finalizing the documents concerning the TURC 2 and TURC 3 tests, the SWISS 1 and SWISS 2 tests, and the HSS tests. The hot solid test with a urania slug will be run.

1.2 High-Pressure Melt Ejection and Direct Containment Heating

(W. Tarbell, J. Brockmann, D. Powers, 6422; M. Pilch, I. Cook, W. Frid, 6425)

A number of severe accident sequence analyses predict that the reactor pressure vessel (RPV) may fail while the primary remains at elevated pressure. These so-called "dispersive" accidents may result in molten debris being forcibly ejected from the RPV into the reactor cavity. The High-Pressure Melt Ejection and Direct Containment Heating Program (HIPS) is studying the phenomena that may arise during such accident sequences. Experiments show nearly complete removal of the material ejected into the cavity by extremely efficient dispersal of the highly fragmented debris. Analyses indicate that the energy imparted to the atmosphere may create potentially threatening containment pressure levels.

1.2.1 Current Progress and Technical Highlights

Activities in the program were directed toward preparing for future tests and documenting past tests. The most important of the test preparation activities was the continuing development of the SURTSEY test facility. The SURTSEY facility is a 1:10 scale model of a reactor containment shell. It is to be used in pressurized melt ejection tests to quantitatively determine the extent of atmospheric heating by core debris dispersed from a reactor cavity. In addition to the concrete foundation for the SURTSEY facility being poured, construction of the containment vessel was started. Also, design drawings for the containment model were received from the vendor. The design specifications supplied by the vendor are now being reviewed by structural engineers at Sandia with their primary attention focused on the support frame.

Prior to running tests in the SURTSEY facility, several tests that do not require a containment model need to be run. One such test (HIPS-8C) is to involve pressurized melt ejection into a concrete cavity vented both along the keyway and through an annular gap around the reactor vessel. Venting of the cavity around the reactor vessel has been neglected in the past tests. Debris expelled from the reactor cavity along vents around the vessel would emerge directly into the large volume of typical containment shells. Since the flow area around the vessel can be more than 25 percent of the flow area through the cavity keyway, it is entirely likely that a significant amount of core debris could be dispersed directly into the containment. The abundant oxygen in the containment would be readily accessible to this debris. As a result, the potential exists for the exothermic oxidation of this debris to be quite complete and heating of the containment shell atmosphere to be most extensive.

Hardware necessary for the HIPS-8C test has been designed and is now being fabricated. Execution of the HIPS-8C test has been delayed until September pending completion of the hardware construction.

The pressurized melt ejection tests done in this program with water-filled reactor cavity models have all resulted in overpressurization of the model cavities. During the report period two separate effects tests were run to better understand the interaction of melt streams with water. Melts were ejected from the pressure vessel into 8-in diameter pipes containing a water column 40 in deep. The air space between the melt generator and the water was 40 in. Driving pressures of less than 200 psi and about 500 psi were used. Posttest inspections and the outputs from pressure transducers and strain gauges in the water pipes indicated that melt/water interactions of violent nature took place near the surface of the water in both tests. This is not similar to what has been observed in pressurized melt ejection tests, where melt-coolant interactions took place after melt had been driven well below the surface of the water. It appears that water in the reactor cavity will complicate pressurized melt ejection processes. More careful examinations will be required before meaningful tests involving these complications can be run.

Preparations were completed to conduct some small-scale separate effects tests dealing with pressurized melt ejection. These tests will be used to validate a model of the disruption of melt streams by the effervescence of dissolved gas. This model has been devised by an attache from the Swedish Power Board working on the program. The tests will also offer an opportunity for qualifying experimental techniques for (1) using hydrogen to pressurize melts and (2) using real-time, high-speed image-intensification to obtain visual data on expelled core debris.

A computer model describing the discharge of melt from a pressurized vessel has been developed in this program, and when completed will be incorporated into the EJECT module of the MELPROG code. Predictions of this model were compared to data obtained from the HIPS tests that used 80 kg melts and the SPIT tests that used 10 kg melts. The model predictions are in excellent agreement with data from the HIPS tests. The model overpredicts the rate of melt discharge observed in the SPIT tests.

1.2.2 Anticipated Activity

The test of venting around the reactor vessel (HIPS-8C) will be run. The rest of the work in the program will be devoted to documenting the results of the past experimental and analytic research. A topical report is being prepared on HIPS tests in which melts were discharged from pressurized vessel into dry, scale models of reactor cavities. A topical report on the model of melt discharge from a pressurized vessel is also being prepared. Both of these topical reports will be available in draft form by the end of September.

1.3 CORCON and VANESA Code Development

(D. A. Powers, 6422; D. R. Bradley, 6425)

One of the important elements of severe accident analyses is the prediction of loads on the containment and release of radionuclides brought about by the interaction of core debris with concrete. CORCON is the state-of-the-art model for predicting the nature of high-temperature core debris attack on concrete. This model yields predictions of the rate and extent of concrete erosion, the nature of flammable gas and noncondensible gas production, and the partitioning of heat from the core debris into the concrete and into the containment atmosphere. The VANESA model is a state-of-the-art model of aerosol generation and radionuclide release during core debris/concrete interactions. It yields predictions of the aerosol composition, particle size, and density as well as aerosol generation rates.

1.3.1 Current Progress and Technical Highlights

1.3.1.1 VANESA Development

A draft copy of the VANESA manual was prepared and submitted to the NRC for review. This massive work delineates the technological basis for predicting the generation of aerosols and release of fission products during core debris interactions with concrete. The model involves descriptions of the thermodynamics and kinetics of vaporization. The manual also presents the available technology for describing mechanical aerosol generation and the formation of aerosols by condensing vapor. Entrapment of aerosols by a water pool overlying core debris is described. Finally, implementation of the model as a computer code is described. This computer code is being given to interested users.

The documentation of the VANESA model demonstrates the superior technical foundation that exists for the predictions of radionuclide release during core debris interactions with concrete relative to, for example, in-vessel release. The technology and data base available for these predictions have not been recognized in some recent reviews of the severe reactor accident source term and NRC-sponsored analyses of this source term.

1.3.1.2 CORCON Development

The primary attention in the CORCON development area has been focused on:

1. Completing the CORCON sensitivity study, and

2. Comparing CORCON predictions to results of core debris/concrete interaction tests.

The CORCON sensitivity study has been completed. Results are being analyzed and a report is being written. The results of the sensitivity study show that predictions with the current version of the model are sensitive to initial conditions supplied to the model. These initial conditions are obtained from models of the in-vessel phases of a reactor accident. The state of development of these in-vessel models means their predictions are quite uncertain.

Comparisons of predictions by CORCON to the results of tests have shown excellent agreement in some cases but not in others. Attentions have focused on the models of heat transfer to the concrete in order to explain the instances of poor agreement. Agreement is best when it can be assured that no crust forms on contact between the melt and cold concrete. Poor agreement seems to arise when crust formation occurs such as in the TURC 2 test.

A model of crust formation, growth, and stability has been formulated. The model is now being implemented into CORCON for comparisons to test data.

Decay heat schemes used by models of reactor accident phenomena are always a question. Recently, this question has arisen in connection with welding CORCON to the MARCH code. The CORCON and MARCH codes use different treatments of decay heating. The decay heat scheme in CORCON was formulated based on results concerning radionuclide inventory obtained with the ORIGEN code for equilibrium cycle fuel. The decay scheme recognized that as burnup increases, the fission energy comes from both ^{235}U and ^{239}Pu and not just ^{235}U as is assumed in the ANS standard decay heat curve. It appears that the CORCON scheme is preferable to those based on the ANS standard decay heat curve. Users involved with uniting CORCON to other codes are being advised to retain the CORCON decay heating scheme rather than using the ANS standard.

1.3.1.3 Joint Project with the KfK BETA Program

A series of melt/concrete interaction tests are being done at the BETA facility operated by the Kernforschungszentrum

Karlsruhe in West Germany. The tests sponsored by West Germany at this facility have used siliceous concrete. In order to obtain more data for validation of CORCON, the NRC has entered a joint effort with KfK to test limestone concrete at the BETA facility. The agreement with KfK is being carried out through the CORCON and VANESA development program.

Two types of concrete have been sent to Germany--limestone concrete and limestone/common sand concrete. The concrete has been used to fabricate crucibles appropriate for the BETA tests. A suggested test plan has also been formulated and transmitted to KfK. The test with limestone/common sand concrete is scheduled for September and will be observed by investigators from the CORCON development program.

1.3.2 Documentation

D. A. Powers, J. E. Brockmann, and A. W. Shiver; <u>VANESA: A Mechanistic Model of Radionuclide Release and Aerosol Generation During Core Debris Interactions</u>, to be published as a NUREG/SAND report.

1.3.3 Anticipated Activity

The VANESA model documentation will be finalized for publication once comments from reviewers have been obtained.

A users letter for CORCON will be issued. The letter will briefly outline the status of code development and modifications that have been made to CORCON.

Documentation of the CORCON Sensitivity Study will continue.

Crust formation and stability models will be incorporated into CORCON and tested.

The attache supported by this program at the KfK BETA project will return. Investigators from the program will go to the BETA facility to observe the testing of the limestone/common sand concrete crucibles.

1.4 Molten Fuel-Coolant Interactions
(B. W. Marshall, Jr., M. Berman, 6427)

The objective of this program is to develop an understanding of the nature of Fuel-Coolant Interactions (FCIs) during hypothetical accidents in LWRs. The understanding of FCIs achieved in this program is expected to resolve key reactor safety issues for both terminated and unterminated accidents. Models are being developed to quantitatively determine:

1. The rates and magnitudes of steam and hydrogen generation,

2. The degree of mixing and coarse fragmentation of the fuel,

3. The degree of fine fragmentation of the individual droplets composing the coarse mixture, and

4. The fraction of the available thermal energy which is converted into mechanical energy.

Experiments are being conducted to determine the influence of the important independent variables, which can be divided into three classes: thermodynamic conditions (temperature of fuel and coolant and ambient pressure); scale variables (amount of fuel and coolant initially involved); and the boundary conditions (pour diameter and rate, shape and degree of confinement of the interaction region, presence of structures, water depth, and fuel-coolant contact mode). Primary experimental measurements include: photographic and x-ray observation of the FCIs, pressures generated in the coolant and the cover gas, steam and hydrogen generation, and the resulting debris characteristics.

1.4.1 Current Progress and Technical Highlights

1.4.1.1 FITSD Experiments
(B. W. Marshall, Jr., M. Berman, 6427)

No experiments were conducted in the FITS (Fully Instrumented Test Site) vessel. The FITSD series will be continued after the high-explosive simulation of a steam explosion is completed and the water phase pressure measurements are evaluated.

1.4.1.2 EXO-FITS Experiments
(B. W. Marshall, Jr., J. Fisk, 6427)

No experiments were conducted at the EXO-FITS facility. The evaluation of the pressure transducers and mounting hardware used during an FCI experiment continued and is described in Section 1.4.1.4.

1.4.1.3 Analysis of the FITSD Experiments
(O. P. Seebold, B. W. Marshall, Jr., 6427)

A preliminary data analysis of the FITSOD experimental data was completed and a memo issued. The preliminary analysis on the FITS5D data was also completed during this period. The memo summarizing the FITS5D experiment is currently being reviewed and should be issued during the upcoming bimonthly period. The analysis of the FITS8D experiment has also continued and should be completed in the near future.

1.4.1.4 High-Explosive Simulations of a Steam Explosion (B. W. Marshall, Jr., J. W. Fisk, 6427; M. F. Young, 6425)

The high-explosive simulation of a steam explosion continued. We conducted several straight high-explosive (SHE) experiments in both air and water. In the water experiments, we continued to observe symmetrical readings in arrival time but not in pressure magnitude. Since more experience exists with measuring pressures generated from a high explosive suspended in air, we performed a few experiments in air to determine whether the asymmetrical pressures were still observed. We continued to note the asymmetrical pressure magnitudes in these experiments, while the timing of the pressure pulses was again relatively symmetric.

In our efforts to reduce and interpret the experimental data taken to date, we have found the need to numerically filter the data. The actual pressure signature can be partially masked by "noise" recorded by the gauge due to pressure waves that propagate through the boundary material, reflections from other boundaries, and relief waves from the air and other surfaces. To assist in the interpretation of the data, we have developed an adaptive interference cancelling routine. This code numerically develops a filter function for a given "noise" signal. This adaptive filter can then be used to cancel the "noise" from the actual pressure signature.

To assist this effort, we conducted two experiments using a point source detonator (the primacord line charge was unavailable at the time, but will be used as soon as it is available). In the first experiment, we mounted the gauges flush with the inner wall of the cylinder and recorded the pressure signature resulting from firing a detonator. In the second experiment, we constructed identical mounting fixtures to those used in the first test, except that the gauges did not see the pressures generated in the water. Instead, the gauge response was assumed to represent the "noise" due to vibrations of the cylinder walls, etc. To experimentally verify the numerical filtering technique, we developed and installed gauge mountings which were isolated from the cylinder walls (i.e., the gauge mountings did not come in contact with the cylinder walls). Two gauges, in the upper and lower position, were used in the two detonator experiments described above. Comparisons of the filtered signal with the response of the isolated gauge revealed interesting but inconclusive results. In one case, the filtered response appeared very similar to that of the isolated gauge indicating that noise was partially masking the true pressure trace. In contrast, the filtered response of a second gauge located at the same axial position did not compare favorably with the isolated gauge response. In future work, two isolated gauges opposing each other will be used to ensure symmetry and to compare the response of these gauges to the filtered response of the wall-mounted gauges.

During the SHE and detonator experiments, we have observed very rapid pressure rise times and short pulse durations. In the SHE experiments the pulse width at half max was typically ~30 μs. During the two detonator experiments, we observed a pressure pulse width of ~6 to 10 μs. The resulting pressure histories from such short pulse widths are very difficult to predict analytically, but may be calculated with a hydrodynamics code such as CSQII. Further, this type of pressure rise history may be difficult to record undistorted with the current experimental hardware. We have measured the frequency response of the amplifiers and cable lines used in these tests to be approximately 50 kHz with no distortion. Therefore, a pressure pulse width smaller than ~20 μs most likely will be distorted and not be representative of the actual pressure magnitude. The pressure pulse width for the detonator tests was much faster than this time and, therefore, may cause some distortion and inconsistencies between different gauges. Further work in this area is ongoing and will be reported.

1.4.1.5 Upgrade of Experimental Measurements
(B. W. Marshall, Jr., O. P. Seebold, 6427; M. F. Young, 6425)

As reported previously, we are currently pursuing additional experimental techniques that will enhance our ability to determine the governing parameters of FCIs. We have analyzed high-speed photographs taken during an FCI experiment using image processing techniques available within Sandia. We are hopeful that enhancement techniques can be defined that will provide additional information about the mixture region.

Early results indicate that a quick count of melt-particle sizes and the mixture-zone shape can be achieved with established techniques. In addition, we are pursuing further techniques in an attempt to enhance the details of the inner zone of the mixture region. More film footage of past FCI experiments are being prepared to see if such enhancement techniques are possible. Presently, we are waiting for the conversion of high-speed films of various CM experiments into an acceptable format for image processing.

1.4.1.6 Suppression of Steam Explosions by Increasing the Viscosity of the Aqueous Coolant Phase
(L. S. Nelson, K. P. Guay, 6427)

We have conducted a brief series of scoping experiments to investigate the suppressive effects of increased coolant viscosity on steam explosions. Molten 12-g globules of tin at ~927 K were poured into 2 kg of aqueous coolants that contained a water-soluble thickener. The coolant temperature was ~298 K. The vigorous spontaneous steam explosions that normally occur in water alone were completely suppressed when the viscosity of the coolant exceeded 15 mPas (15 cp).

It is possible that this effect could be applied directly to the prevention of damaging steam explosions in certain accident situations where melts and aqueous coolants might come together. Moreover, incorporation of this effect into our current models should lead to a substantially increased understanding of how steam explosions are triggered and propagate.

1.4.1.7 Monte Carlo Computer Code (O. P. Seebold, M. Berman, 6427)

The Monte Carlo code used by the authors of SAND83-1438 has been implemented on the VAX computer. The code has been bench tested and compared with the results obtained in the previously mentioned report. The results of the original version of the code and the version on the VAX compare rather well.

A single case was run with 10,000 and 100,000 iterations to estimate the cost of running the code. The former case used 58 s of VAX CPU time costing ~\$1.60 at \$100/CPU·h, while the latter case ran for 137 s of VAX CPU time resulting in a cost of ~\$3.80. Therefore, the cost of running the code on the VAX is extremely small and the code should provide a useful tool for analysis in the future.

1.4.1.8 Modeling of Explosion Propagation and Structural Loading Using CSQII (K. L. Schoenefeld, 6425)

The hydrodynamic code CSQII was used to investigate the use of gauge blocks as mounting devices for pressure transducers during a steam explosion experiment. Also, the effect of a single wrap of PETN primacord (as in the HEX experiments) upon the pressure magnitude at a radial position was investigated.

The purpose of the gauge block analysis was to investigate the effect of using recessed gauge blocks to mount the pressure transducer, instead of flush mountings, for pressure measurements during a steam explosion experiment. Of particular interest was a comparison of the predicted pressure traces inside the cavity of the gauge block to those on the chamber wall. Several cases were run with variations in hole diameters (0.8 to 6 cm), depths (2.5 to 9.8 cm), radial location of the hole (at the center and midway between the center and the side wall), and pressure wave source (planar and hemispherical waves). In general, it was concluded that an initial pressure wave impinging upon the cavity entrance was doubled due to the influence of the cavity boundaries. This doubled pressure wave continued down the cavity and again increased by a factor of roughly 1.5 to 1.7 when it hit the cavity bottom. As a result of these calculations, we might expect pressure magnitudes in recessed cavities to be about 1.5 to 1.7 times higher than those measured by gauges flush with the chamber wall. Further, the diameter, depth, and radial location of the recessed cavity (within reason) do not seem to influence this conclusion.

In the HEX tests, PETN primacord was wrapped in a spiral around a cylindrical block of wood to create a burn time (or detonation velocity) similar to that of a steam explosion. As previously reported, we observed significant variations in the pressure magnitudes during these experiments. In order to determine whether these asymmetrical pressure measurements were due to the spiral wrapping, CSQII was used to model a top view of a single wrap inside a chamber of water. The code results indicate that pressure magnitudes along the chamber wall can vary by as much as 40 percent between locations 180 degrees apart. It also showed that both the magnitude and shape of the pressure pulses are affected by the detonation velocity of the PETN, the diameter of the wrap, and the speed of sound of the material around which the primacord is wrapped. It therefore appears that this may not be a desirable configuration to use for evaluating the response of pressure gauges used in steam explosion experiments.

1.4.1.9 Modeling of Coarse Fragmentation and Hydrogen and Steam Generation (TEXAS)
(M. L. Corradini, C. Chu, U of W)

The major effort has been writing the documentation for the TEXAS code--the preliminary thesis for C. Chu. Following is a summary of the work as applied to the EXO-FITS experiments.

1.4.1.9.1 Thesis Summary

If adequate cooling water is not supplied to the core of a light water reactor, fission-product decay heat will eventually cause the reactor fuel and cladding to melt. This could lead to slumping of the molten core materials into the lower plenum of the reactor vessel, possible failure of the vessel wall, and ejection of the molten core materials into the reactor cavity. If the molten core materials enter either region, there is a strong possibility of molten core contact with water. This fuel-coolant contact mode is of interest for possible energetic (steam explosion) or nonenergetic (steam spike) fuel-coolant interactions. According to some previous studies, a necessary initial condition for a large-scale steam explosion in a core meltdown accident in the LWR is the formation of a coarsely predispersed mixture of the molten fuel and water. On the other hand, a nonenergetic steam spike may occur with poor initial fuel-coolant mixing. Hence, a mechanistic treatment of the fuel-coolant mixing process is necessary for fundamental understanding of fuel-coolant interactions. The balance of this subsection describes a new transient model for fuel-coolant mixing and fuel fragmentation during mixing.

Some past work has concentrated on the limit for which mixing could occur based on steady-state models. Henry and Fauske demonstrated that when fuel is mixed with water coolant, the resulting steam velocity quickly reaches a value corresponding to the fluidization limit of water drops so that the required initial conditions (large fuel masses >5 to 10 tonnes) for an energetic explosion that might threaten the containment is difficult to achieve. Corradini suggested that the limit to fuel-coolant mixing was determined by fuel- or coolant-liquid fluidization. This steady-state limit may never be reached because it takes time for the fuel to dynamically break up to this size and a premature energetic fuel-coolant interaction or agglomeration of the fuel on the pool base may occur first. Theofanous first identified this concept of the time for transient jet breakup as a limit to mixing. More recently, Corradini and Moses proposed a simplified transient model which allowed for radial expansion of the falling fuel as it fragments. The fuel particle diameter and volume fraction were taken to be empirical functions of dimensionless time obtained from the FITS experiments. This model can be used to calculate the rate of hydrogen and steam generation and the initial condition for an explosion. Bankoff and Han developed a one-dimensional, unsteady fuel-coolant mixing model, using the PHOENICS code. They assumed that prefragmented fuel particles with a uniform constant diameter fall into and mix with a steam-water mixture at thermodynamic equilibrium. They concluded that there is a rapid level swell, with most of the fuel drops being fluidized and swept upwards in a region of high steam volume fraction.

In this work a one-dimensional three-field fluid model was developed to predict fuel-coolant mixing processes with a relative velocity-induced fragmentation model. It is based on the TEXAS model, a Lagrangian-Eulerian hydrodynamics code. Two Eulerian fields (coolant vapor and liquid) and one Lagrangian particle field (fuel) are employed in the present model. The model is currently limited to non-explosive fuel-coolant interactions. The numerical solution technique is based on the SIMMER-II pressure iteration method with a number of improvements; e.g., convective energy and pressure work terms are included in the pressure iteration. In addition, a complete set of constitutive relations was developed for the interfacial mass, momentum, and energy transport term; e.g., a virtual mass model for the rapid relative acceleration between the components. The key constitutive relation in the computer model is a transient fuel fragmentation model based on Rayleigh-Taylor instabilities and a dynamic pressure deformation mechanism. This model considers the fuel to be dynamically fragmented from its initial entry diameter to smaller sizes due to instabilities and deformation generated by relative velocity as it falls through the water pool. This model can be used to analyze the fuel-coolant mixing phase of the FITS FCI tests and similar accident situations.

1.4.1.10 Modeling of the Steam Explosion Phases
(M. L. Corradini, M. Oh, U of W)

The major effort has been the documentation of M. D. Oh's computer model for the analysis of the FITS tests. The computer code and the theory behind its development will be issued as Oh's Ph.D. thesis at the end of August.

1.4.1.11 Modeling of Film Collapse and Fuel Fragmentation
(M. L. Corradini, B. J. Kim, U of W)

We have completed the documentation of the results of B. J. Kim's work. His efforts have resulted in a theoretical model for analyzing Nelson's single droplet experiments. The document, Kim's Ph.D. thesis, will be available for publication in September.

1.4.2 Presentations, Visits, and Meetings

On July 9th and 10th a meeting was held with the Aluminum Association to discuss an industry-supported study of steam explosions in molten metals.

R. Curtis of the NRC and M. Berman of Sandia visited the Winfrith Laboratory of the United Kingdom Atomic Energy Authority for three days from June 3-5. The purpose of this visit was to discuss United States and United Kingdom research on fuel-coolant interactions and to review the status of the United Kingdom report on Phase 2 of the PWR Severe Accident Containment Study. The exchange of information was highly informative, with both sides benefitting.

1.4.3 Documentation

On June 28, 1985, M. Berman sent a memorandum to distribution summarizing his trip to Winfrith Laboratories (UKAEA).

On July 24, 1985, O. Seebold sent a memorandum to the FCI staff reviewing the preliminary data reduction and interpretation of the FITSOD experiment.

1.4.4 Anticipated Activity

The high-explosive evaluation of the pressure gauges is expected to continue. We expect to develop a slower burning charge for use in these experiments and to develop a technique of mounting gauges which will not be influenced by vibrations in the cylinder wall. The adaptive filter routine developed will be tested and used extensively during this period to assist in the interpretation of the data.

We will have an FCI working group meeting on August 28-29, 1985, in Madison, WI.

1.5 Hydrogen Behavior
(J. T. Hitchcock, M. Berman, 6427)

The major concerns regarding hydrogen in LWRs are that the static or dynamic pressure loads from combustion may breach containment or that important, safety-related equipment may be damaged due to either pressure loads or high temperatures. In order to assess the possible threats, it is necessary to understand how hydrogen is produced, how it is transported and mixed within containment, and how it combusts.

The objectives of this program are (1) to quantify the threat to nuclear power plants (containment structure, safety equipment, and the primary system) posed by hydrogen combustion; (2) to disseminate information on hydrogen behavior and control; and (3) to provide programmatic and technical assistance to the NRC on hydrogen-related matters.

1.5.1 Current Progress and Technical Highlights

1.5.1.1 HECTR Analysis and Code Development
(C. C. Wong, 6427)

The HECTR (Hydrogen Event: Containment Transient Response) code is a reactor accident analysis tool designed to calculate the transport and combustion of hydrogen and the transient response of the containment. It was developed to meet urgent NRC licensing needs and continues to be a major tool for predicting both local and global conditions during combustion sequences. It has been successfully applied to BWR Mark III, PWR ice condenser, and PWR large dry containments.

At the request of the NRC, we have initiated a cooperative analysis program to address several differences between the NRC and IDCOR on the hydrogen combustion issue. A standard problem has been defined to compare the hydrogen combustion modeling between the HECTR code and the MAAP (Modular Accident Analysis Program) code. The standard problem involves a S2HF accident sequence in a PWR ice condenser containment. The S2HF accident scenario consists of a small break (0.5 to 2-in diameter) loss-of-coolant accident with failure of emergency coolant and containment spray recirculation. All of the water inventory from the sprays, which are only operated in the injection mode, is assumed to be trapped in the upper compartment because of the failure to remove the upper-to-lower compartment drain plugs. This causes the reactor cavity to remain dry throughout the transient. Incomplete hydrogen burns initiated by the deliberate ignition system will occur in the lower and upper compartments. When the reactor vessel fails, the molten fuel will slump onto the floor of the cavity and result in a core-concrete interaction. This interaction will generate substantial amounts of combustible gases, some of which may continuously recombine or generate flames which can propagate into the lower compartment. The extent of this recombination depends upon the flow of oxygen into the cavity from the lower compartment. Our primary interest is to investigate these three important physical phenomena: (1) natural circulation between the reactor cavity and lower compartment, (2) the effects of continuous recombination of hydrogen in the reactor cavity and flame propagation, and (3) the incomplete burning in the lower and upper compartments. For the purpose of consistency, HECTR will use the hydrogen, steam, and carbon monoxide sources generated from the MAAP code. To analyze this standard problem, HECTR will use two different noding systems. The first one will duplicate the noding system which IDCOR used in their calculation. The other will be based on the best engineering judgment of NRC and Sandia personnel.

We continued to work on the development of a diffusion flame model for the HECTR code. Our work mainly concentrated on analyzing the Nevada Test Site (NTS) continuous injection experiment, Test CO3, using an empirical approach to model the diffusion flame. This approach uses empirical correlations to characterize the diffusion flame and also the flow induced by the flame. For these calculations, we used a three-compartment model so that HECTR could predict the amount of hydrogen (together with the ambient gases) entrained into the standing flame more accurately. The preliminary results show that HECTR predictions compare reasonably well against the experiment. However, more work is needed to better represent the standing flame structure and its effects.

The assessment of HECTR against the large-scale hydrogen combustion experiments performed at NTS continued. We have completed 21 HECTR calculations of the premixed hydrogen combustion tests using a single compartment model. The HECTR predictions are being compared to the measured data. Our preliminary findings show that for lean hydrogen burns, HECTR overpredicts the flame speed and combustion completeness, and thus overpredicts the peak pressure and temperature. For those tests with high concentrations of steam, HECTR underpredicts the flame speed, which leads to a longer burn time and lower peak pressure.

1.5.1.2 The FLAME Facility
(M. P. Sherman, S. R. Tieszen, 6427; W. B. Benedick, 1131)

The FLAME facility is a large horizontal channel used to study hydrogen combustion problems relevant to nuclear reactor safety including flame acceleration and transition to detonation. The degree of transverse venting along the top of the channel can be varied. Obstacles can be attached to the side walls and floor. The facility was built to be a half-scale model of the ice condenser upper plenum region. The first series of 20 tests has been run and the results are to be presented at the Tenth International Colloquium on Dynamics of Explosions and Reactive Systems, August 4-9 at Berkeley.

On June 28, 1985, the first test with obstacles was conducted in FLAME. The obstacles were simple plywood sheets mounted perpendicular to the two side walls. Fifteen obstacles were placed on each wall about 1.2 m apart and were symmetrically opposed. They extended from floor to ceiling, a distance of 2.44 m. Since each obstacle extended 30 cm into the channel, and the channel width is 1.83 m, the blockage was one-third. The top of FLAME was fully closed with steel plates, i.e., there was no transverse venting.

The hydrogen mole fraction for this test was to have been 13 percent. Due to a technician error, the mixing fans in the channel did not operate, and the hydrogen-air mixture was not mechanically mixed. As in all our tests, hydrogen gas entered the channel through three ports near the bottom, one port near the center of the channel, and one port near each end of the channel. We estimate the hydrogen entered the channel with a speed of about 28 m/s. The gas mixture was allowed to sit for 30 min prior to taking gas samples and igniting the mixture. Mass-spectroscopic laboratory results indicated about 15 percent hydrogen near the ceiling and about 10 percent near the floor. The mixture was not well mixed. The inhomogeneity of the mixture may, however, be <u>typical of what would occur in some accidents</u>.

The effect of the obstacles on the flame speeds and overpressures was dramatic. For 15 percent hydrogen in FLAME without obstacles and with no top venting, the overpressure was about 10 kPa (1.5 psig) and the equivalent planar flame speed was about 60 m/s. In the latest test with obstacles, F-21, the peak overpressure was about 600 kPa (90 psig), and the equivalent planar flame speed was over 500 m/s. The loads generated were sufficient to destroy all the obstacles, the plywood end door, two plywood blowout roof panels, and the thermocouple rakes.

It is well known that obstacles in the flow path of a premixed gas flame can cause flame acceleration. Small-scale tests of hydrogen-air mixtures carried out at McGill University have shown such flame acceleration. It is also known, from our previous FLAME tests and the work of other researchers, that flame acceleration is larger at larger scale. The indication from our latest test is that in tests with obstacles and no transverse venting, we will see much higher overpressures, faster flame speeds, and transition to detonation at lower hydrogen concentrations than in comparable tests without obstacles. It will be important to see if the same is true in tests with obstacles and top venting. The results of our current test series should quantify these results for various hydrogen concentrations and degrees of transverse venting.

The current test series uses simple obstacles. It permits comparison with other experimental data at small-scale, can be modeled using the CONCHAS-SPRAY code at Sandia Livermore, and shows the effect of obstacles on flame acceleration and transition to detonation. The next test series will use more complex obstacles more closely related to the geometry in the ice condenser upper plenum region. From small-scale tests, one might expect that the more complex obstacles, with their greater length of shear layers but lower blockage ratio, will cause greater flame acceleration than the simple obstacles.

The changes recommended by the Safety, Health, and Environmental Committee (SHEAC) in the hydrogen delivery system for FLAME located near the hydrogen trailer were carried out. These included the addition of a burst disk and alteration in the nitrogen purge lines to eliminate all air in the hydrogen lines.

1.5.1.3 CONCHAS-SPRAY Simulation of FLAME Experiments: Preliminary Considerations (K. D. Marx, 8363)

We are developing a CONCHAS-SPRAY simulation of FLAME experiments with venting. This requires the implementation of outflow boundary conditions at periodically spaced segments along the boundary of the computational domain. The first experiment to be studied is FLAME Experiment F4, a burn in a 28 percent hydrogen-air mixture with 50 percent venting.

In simulations such as these, the combustion and turbulence models used in the code represent the most difficult aspects of the calculation. The reason is that they cannot be determined from first principles, but must be determined from plausible physical arguments coupled with tuning to experimental data.

In the initial attempt at a FLAME simulation, the Magnussen-Hjertager combustion model was used to describe the burning of the mixture. In this model, the combustion rate is assumed to be proportional to the minimum of appropriately weighted chemical mass fractions divided by the turbulence time scale. A quenching criterion is also implemented.

The turbulence model that we use employs a single transport equation to determine the transport of turbulent kinetic energy. A constant turbulent length scale is specified. These permit the determination of the eddy viscosity and turbulent time scales.

These models are the same as those used in a previous simulation of small-scale McGill University experiments. (The turbulent length scale was increased appropriately to account for the much larger dimensions of the FLAME experiment.) However, when applied to the FLAME simulation with no further tuning, the results are unsatisfactory. Experimentally, the configuration quickly approaches a steady-state burning mode in which the flame front is quite planar and tipped in the forward direction. The computed flame front is very non-planar and burns rapidly along the top of the domain, where the vents are located. Furthermore, the burn does not progress down toward the bottom (unvented) boundary rapidly enough.

No serious attempt has yet been made to improve the performance of the code with the present combustion and turbulence models. Instead, we are proceeding to implement the k-epsilon turbulence model. This is a more sophisticated (two-equation) model in which an additional equation is used to account for the transport of the rate of dissipation of turbulent kinetic energy. This work is currently in the debug stage. When it is completed, we will return to the FLAME simulations.

In recent months, we have made some improvements to the code. We have changed part of the algorithm for use in situations in which the flow velocity is of the order of the speed of sound or greater. This results in a reduction in computer time by a factor of 2 to 3 for some problems. Some new graphics output has also been implemented.

1.5.1.4 Heated Detonation Tube
(S. R. Tieszen, M. P. Sherman, 6427; W. B. Benedick, 1131)

The purpose of the Heated Detonation Tube (HDT) is to develop an experimental data base on H_2-air-steam detonability. These data can be used to develop models to assess the possibility of detonation inside containment. A detonation wave consists of a complex, three-dimensional, cellular structure formed by the multiple interactions of the shock waves. The characteristic cell width, λ, of this structure is a key to determining the important detonation parameters such as critical initiation energy and propagation limits.

The HDT, with its 43-cm ID and capability of operating at temperatures above 100°C, is a unique facility. The primary objective of the current test series is to measure detonation cell sizes for H_2-air-steam mixtures which are predicted to occur following a reactor accident. A second objective is to provide data to modelers to assess and calibrate predictive codes. This is accomplished by investigating the separate effects of temperature, pressure, and diluent concentration in addition to the H_2 concentration.

The main effort has been the revision of the topical report on the HDT. The uncertainty analysis chapter has been updated. The chapter will not include the digital image processing technique which is still under development, but will reflect the current concern over the uncertainty involved in determining detonation cell width. The results chapter has been reviewed and journal-quality figures of the data have been generated. We plan to release the completed document for internal Sandia review in early August.

Considerable effort has been expended to address concerns raised by an internal safety audit of the HDT facility by SHEAC. To achieve the increased steam concentrations to be tested in the next series, the operating pressure in the HDT will be higher than has previously been tested. Therefore, a static pressure test of the facility will be required before the next test series.

Division 1131, which runs the 9920 explosive site where the HDT is located, has obtained capital improvement funds and has purchased and installed three additional Tektronix 7612D Transient Digitizers (~$100K) which double the current capability to digitize in the megahertz range. The HDT used the existing 7612s (also purchased on internal Sandia funds) and will make use of the additional channels in the next test series. Additional electronic power supplies to power the pressure gauges used to measure detonation velocity are also available for our use.

1.5.2 Presentations, Meetings, and Documentation

Dr. Ding Jing from the Beijing Institute of Technology, Beijing, China, toured the experimental facilities on July 19, and held discussions with the staff.

A presentation was made by J. T. Hitchcock at the National Association of Manufacturers Conference on Prevention and Control of Catastrophic Gas Releases on June 4-6, 1985, in Washington, DC.

S. R. Tieszen, M. P. Sherman, W. B. Benedick, and M. Berman, Detonability of H_2-Air-Diluent Mixtures, SAND85-1263, Draft Report.

A. C. Ratzel, S. N. Kempke, J. E. Shepherd, and A. W. Reed, SMOKE: A Data Reduction Package for Analysis of Combustion Experiments, SAND83-2658, Draft Report.

M. P. Sherman, S. R. Tieszen, W. B. Benedick, J. W. Fisk, M. Berman, and M. Carcassi, The FLAME Facility-Test Series 1: The Effect of Transverse Venting on Flame Acceleration and Transition to Detonation, Draft Report.

1.5.3 Anticipated Activity

Calculations of the NRC/IDCOR standard problem using HECTR will begin. Test Series 2 in FLAME with simple obstacles will continue. Work will also continue on preparing the HDT for the final test series on high steam concentrations and separate effects of pressure and temperature.

1.6 Hydrogen Mitigative and Preventive Schemes
(L. S. Nelson, M. Berman, 6427)

This program is directed toward understanding the behavior and consequences of operating deliberate ignition systems in the containment of nuclear power plants during various phases of hypothetical hydrogen-producing accidents. Since deliberate ignition can be performed safely with lean mixtures of hydrogen in air (6 to 8 v/o), we have concentrated essentially all effort in this range of compositions. We are investigating (1) how resistance-heated igniters are affected by the operation of the water spray system in a containment, (2) air flows that might be produced by these sprays, (3) ignition and flame propagation in lean hydrogen-air mixtures in the presence of various densities of water fogs (related to both water sprays and condensing steam), (4) effects of oxidic and metallic aerosols on the combustion of lean mixtures, (5) chemical changes hydrogen combustion might produce in fission-product containing aerosols, and (6) concepts for constructing nonpowered igniters that would function during station blackouts where resistance-heated igniters would be disabled.

1.6.1 Current Progress and Technical Highlights

1.6.1.1 Behavior of Hydrogen Igniters in the Presence of Water Sprays and Gas Flows
(L. S. Nelson, K. P. Guay, 6427)

We continued construction of an enclosed outdoor chamber in which we will study the combined effects of water sprays and gas flows on helical and cylindrical hydrogen igniters.

1.6.1.2 Air Currents Driven by Sprays in the Sequoyah Containment Geometry
(K. D. Marx, 8363)

We have previously reported on the computation of air flows driven by water sprays in reactor containment buildings. It was subsequently determined that some of the parameters used in our most realistic calculation deviated (in most cases by quite small amounts) from those which actually occur in the Sequoyah and Catawba reactors. Hence, the following changes were made in the computation to more nearly correspond to the actual Sequoyah configuration:

1. The radius of the inner wall of the steam generator enclosures was moved out from 5.21 to 6.25 m.

2. Similar small changes were made to the height of the steam generators, the radius and height of the ice condenser, and the dimensions of the containment building.

3. The spray header configuration was changed from a group of six rings uniformly distributed over radii of 5.8 to 9.9 m to two rings at radii of 5.8 and 7 m, respectively. The flow rate from each header was taken to be the same, whereas it was previously proportional to radius. The heights of the rings were altered slightly.

4. The total flow rate was changed from 590 to 600 kg/s (9340 to 9500 gal/min).

5. The angular distribution of spray injection velocities was altered to correspond to Sequoyah specifications.

6. The inlet air velocity at the top of the ice condensers, which was used to simulate the effect of the air fans, was changed from the deliberately high value of 0.60 m/s (2 ft/s) to the more realistic value of 0.10 m/s (0.32 ft/s).

All of these changes are relatively small except for those in the spray header configuration (3 above) and the inlet velocity (6). The fans have negligible effect on the air flow configuration in the containment, so the change in fan inlet velocity did not influence the results. However, the changes in spray header configuration, coupled with the change in steam generator inner radius noted in 1, have a significant effect on the <u>peak</u> air flow velocity.

In the previous calculation, the peak air velocity was about 7 m/s; with the changes noted above it is 14 m/s. This peak occurs on the axis in both cases. The difference is entirely due to changes 1 and 3. In the new configuration a greater percentage of the water spray is permitted to fall all the way to the bottom of the containment building, whereas most of it landed on the top of the steam generator in the prior calculation. Previous analysis has shown that the peak air velocity generally increases with the spray fall distance. The present result is consistent with that concept.

Even though the peak velocity has proven to be sensitive to these changes, there still remain large regions over the steam generator enclosures and the ice condensers where the flow velocities remain small (<3 m/s). In fact, in some of these regions, the flow velocities diminish significantly. Hence, the original conclusion that the presence of structures in the containment building generally reduces air flow velocities is still true. These recent results serve to point out that localized peaks in flow velocity can vary greatly when constraints on the spray and airflow are imposed.

1.6.1.3 Hydrogen Combustion in Water Sprays and Condensing Steam
(L. S. Nelson, K. P. Guay, C. J. Richards, 6427)

With high-speed photography we have continued to examine the formation of drops by rotating disc generators. The films indicate the following:

1. A minimum water flow below which the drop generation becomes unstable. Above this minimum, the thin layer of water flowing outward on the surface of each rotating disc extends continuously to the edge where it breaks up to eject small, monodisperse drops. Below the minimum flow, however, this layer of water breaks up unstably before it reaches the edge of the disc, producing large polydisperse drops. Our original concern of flooding the discs to unstable drop production does not occur, at least up to the maximum flow of 16 L/min for our 12-disc generators.

2. The drop diameter produced by the rotating discs is proportional to the surface tension of the liquid supplied to the discs. This was predicted by theory. Thus, the addition of 2.5×10^{-3} w/o of a surfactant to the supply water, which produced a solution with a surface tension one-third that of water, reduced the minimum drop diameter obtainable with our discs from 100 microns to about 35 microns. The smaller diameters produced with surfactants are expected to bridge the drop sizes now produced in the FITS chamber by the generators with water alone (100 and 200 microns) and those anticipated to form during the condensation of steam.

Preparations are being made to repeat the eight hydrogen burns at 6.5 v/o (balance air) in the presence of water drops in the FITS chamber. Uncertainties in mixture preparation encountered in these earlier experiments will be minimized with new gas handling hardware now being installed. The water drop generators have been refurbished mechanically; this involved primarily installation of new bearings and seals. Careful effort has been made to reduce leaks in the FITS chamber to increase the reliability of the measurements associated with the burns. Moreover, real-time on-site quadrupole mass spectrometric gas analyses will be available for the new series of experiments. This will provide an independent check on the pre- and posttest bottle sampling used in previous combustion experiments in the FITS chamber.

1.6.1.4 The Effects of Aerosols on Hydrogen Combustion
(L. S. Nelson, 6427; W. B. Benedick, 1131; J. E. Shepherd, 1512; J. H. Lee, R. Knystautas, M. Garg, McGill University)

We studied the ignition behavior of iron aerosols dispersed in air alone and in lean hydrogen-air mixtures. We performed experiments in both the 0.18 m^3 chamber at McGill University and in the 5.1 m^3 VGES chamber at Sandia.

We found that the lean limits for ignition of the iron aerosol dispersed in air alone were ~400 and ~275 g/m^3 for the smaller and larger chambers, respectively, suggesting a possible scale effect. As small amounts of hydrogen were added to the air, the minimum concentration of iron needed to sustain combustion decreased roughly as predicted by the LeChatelier rule for mixtures of fuels. Deviations from the rule were greater for the larger chamber than the smaller chamber, again suggesting a scale effect.

1.6.1.5 Consequences of Hydrogen Combustion in the Presence of Aerosols
(L. S. Nelson, 6427; G. D. Valdez, 6449)

We continued to analyze data recorded during the burns performed in the VGES chamber with combustions between 6.5 and 29.6 v/o hydrogen in air; each burn was performed with 1 kg of 10 w/o CsI-90 w/o Al_2O_3 mixed powders dispersed throughout the combustion volume.

Theoretical modeling of the consequences of results such as these in an accident situation was continued.

1.6.1.6 Nonpowered Hydrogen Igniters
(L. R. Thorne, J. V. Volponi, 8353; L. S. Nelson, K. P. Guay, 6427)

A platinum catalytic igniter has been prepared for evaluation in several static-dry hydrogen-air mixtures in the 5.6 m^3 FITS combustion chamber.

We have designed and are fabricating the hardware for safely inserting the platinum igniter into the FITS chamber while filled with a combustible mixture of hydrogen and air.

1.6.2 Presentations, Meetings, and Documentation

On July 11, 1985, we met with the Aluminum Association to discuss an industry-supported study of metal aerosol (dust) explosions in the metals industries.

A draft report, Platinum Catalytic Igniters for Lean Hydrogen/Air Mixtures, by L. R. Thorne, J. V. Volponi, and W. J. McLean, was submitted to the Nuclear Regulatory Commission for comments.

1.6.3 Anticipated Activity

We will finish the tests of igniters in combined gas flows and water sprays.

We will revise a draft copy of the report entitled Behavior of Resistance-Heated Hydrogen Igniters During the Operation of Water Sprays in Containment submitted to the NRC in early April.

We will continue to reduce the data obtained from the FITS water drop experiments performed between December 1984 and March 1985. More burns are planned at 6.5 percent hydrogen in air with and without sprays. After these experiments are completed, we will attempt burns in condensing steam.

We will do more diagnostics on water drop sizes and densities in the FITS chamber with optical techniques. We will study drops produced with the spinning disc generators and by the condensation of steam.

Theoretical modeling of the effects of aerosols on hydrogen burns will continue.

Analyses of the six CsI aerosol/hydrogen combustion experiments performed in November and December 1984 will continue. In collaboration with the Containment Modeling Division (6449) efforts will be made to cast these results into the context of source term modification via the CONTAIN code.

We expect to test the ability of the platinum igniter to ignite static-dry hydrogen-air mixtures in the 5.6 m^3 FITS chamber. Compositions of 10 and 6.5 v/o hydrogen in air will be used.

2. FISSION-PRODUCT SOURCE TERM

2.1 High-Temperature Fission-Product Chemistry

(D. A. Powers, R. M. Elrick, 6422; R. A. Sallach, 1846)

The purpose of this program is to obtain data on the chemistry and processes that affect the transport of fission products under accident conditions. The program now consists of three tasks related to one another. Baseline thermodynamic and reactivity data are being collected for compounds of fission-product elements of particular interest. An experiment facility has been built to allow the chemistry of fission products in prototypic steam-hydrogen environments to be studied. The interaction of fission products with reactor materials such as stainless steel can be examined in this facility. Results of these experimental studies are compared to predictions of thermochemical models to determine if reaction kinetics play an important role in fission-product transport.

2.1.1 Current Progress and Technical Highlights

Activities in the High-Temperature Fission-Product Chemistry Program have focused on the analysis and interpretation of experimental results. Specific activities have included the following:

1. Development of a kinetic model to describe the interaction of cesium-bearing vapors with oxides on stainless steel to form cesium silicate.

2. Development of a model of the kinetics of B_4C oxidation in steam.

3. Thermodynamic analysis of the Ag-Cd-In-Te system.

4. Development of a revaporization model.

The "irreversible" adsorption of cesium on stainless steel to form cesium silicate has been a significant finding of this program. Cesium silicate formation has been observed recently in the dissection of the TMI-2 core. Empirical kinetics of the reaction have been measured in this program. The nature of the product requires, however, that there be silica present. Silica is formed from a minor constituent in stainless steel as a result of steam oxidation of the steel. The rate of cesium silicate formation can be controlled either by transport of cesium to the silica or by formation of silica. Attempts are under way to formulate a model that takes into account these two potential rate-controlling mechanisms.

The rate of silica formation is proportional to the rate of steel oxidation. Steel oxidation rates are known and these rates have been verified by experiments in this program.

The rate of transport of cesium to the silica is complicated by the duplex oxide that forms on steel. The cesium must pass through an outer layer of Fe_3O_4 to reach the silica-bearing inner layer. Some controversy exists on how this transport might occur. Two possibilities that are being considered in the modeling are (1) fast ion diffusion and (2) Knudsen diffusion.

Examination and modeling of B_4C oxidation is an outgrowth of our studies on the effects of oxidation products on fission-product chemistry. Data from the chemistry tests are being used to formulate an oxidation rate model. Early results indicate paralinear rather than the more familiar parabolic kinetics may be appropriate. Linearity in the kinetics appears after initial parabolic oxidation rates because the oxide product vaporizes in steam as boric acid. The model being developed here will be of use to those individuals modeling core degradation in boiling water reactors. The oxidation of B_4C could be a potent source of both hydrogen and energy during core degradation.

In depth studies of the thermodynamics of the silver-indium-cadmium system are being extended to include tellurium. The computer code SOLGASMIX PV is being used for this work. Early results indicate cadmium telluride can be a condensed, but not a vapor form of tellurium. Silver is more reactive toward tellurium as a liquid than as a solid. These results show that tellurium released from reactor fuel may react to form aerosols and need not react just with structural steels. Tellurium bound to aerosols could contribute to an accident source term whereas tellurium bound to structures may not.

Modeling of revaporization is focusing on revaporization of:

1. Cesium bound to surfaces as a silicate,
2. Cesium adsorbed but, perhaps, chemically transformed,
3. Tellurium bound as tellurides, and
4. Cadmium deposited on surfaces.

Vapor pressures are being calculated as functions of both temperature and the ambient composition of the gas. Some difficulties are encountered when calculating thermodynamic vapor pressures for cesium silicate because there are insufficient data. A model of the cesium-silica system has been formulated based on silica polymerization. The model predicts well the Cs_2O-SiO_2 phase diagram. It is interesting in that activities are dominated by entropic effects rather than enthalpic effects of mixing as is usually the case for high-temperature systems.

Vaporization kinetics are controlled primarily by (1) surface vaporization and (2) gas phase mass transport. For cesium silicate there is the additional problem that vapors must migrate through an external Fe_3O_4 oxide layer. This migration is being modeled as Knudsen diffusion.

Experimental activities during this reporting period include:

1. Pressure drop measurements made in the Fission-Product Reaction Facility for calibration purposes.

2. Tests run to ascertain if temperature measurements in fission-product generators for the Fission-Product Reaction Facility are accurate. A consistent error of about 45°C has been found. Appropriate modifications to the system to correct this error are being made.

3. Preparations initiated to test SnTe deposition rates on steel. This test is being done to assist in the analysis of test results from PBF done by EG&G, Idaho. The test was suggested by work on tellurium adsorption on zircaloy done at the Battelle Columbus Laboratory.

2.1.2 Documentation

K. Vinjamuri, R. A. Sallach, D. W. Akers, D. J. Osetek, and R. R. Hobbins, Tellurium Chemistry, Tellurium Release and Deposition During the TMI-2 Accident, EGG-TMI-6894, EG&G, Idaho, Idaho Falls, ID, August 1985.

V. F. Baston, K. J. Hofstetter, G. M. Bain, and R. M. Elrick, Cesium Retention on Reactor Material Surfaces Summary of Accident and Test Data, AIME/ASM Meeting, New Orleans, LA, March 1986.

V. F. Baston, K. J. Hofstetter, G. M. Bain, and R. M. Elrick, A Comparison of TMI-2 and Laboratory Results for Cesium Activity Retained on Reactor Material Surfaces, invited paper for ANS Winter Mtg., November 1985.

2.1.3 Anticipated Activity

An investigator from the program will participate in the Specialists Meeting on Iodine Chemistry to be held in Harwell, England. He will present a paper on the effects of ionizing radiation on the high-temperature chemistry of CsI.

The test of SnTe deposition will be run.

A draft topical report on experiments with B_4C and the effects of B_4C oxidation on fission-product chemistry will be written.

The draft topical report on revaporization will be finished.

2.2 ACRR Source Term Tests

(K. O. Reil, P. S. Pickard, 6423; R. M. Elrick, 6422; J. Grimley, 6425; H. Stockman, 1543)

Release of radionuclides during fuel degradation in a core uncovery accident is the first stage in the determination of the amount and nature of the radioactive release from the damaged nuclear plant. Current estimates of the release of the principal fission products over the range of relevant accident conditions are subject to significant uncertainty (e.g., QUEST). A key element in reducing the uncertainty in predicted releases is an improved understanding of fission products from the fuel under severe fuel damage conditions. The ACRR Source Term Program is being developed to provide a data base for fission-product release over a range of fuel temperatures, system pressures, and fuel damage states, where little or no data currently exists, to allow the development of improved fission-product release models for use in consequence evaluation (VICTORIA). This program is related to out-of-pile programs at ORNL and BCL.

2.2.1 Current Progress and Technical Highlights

The primary activities of the Source Term (ST) Experiment Program during this period were the detailed design of ST-1 and ST-2 capsules, the continuation of the filter sampler tests to include lower substrate temperatures, and preparation for the sampler demonstration test. Preliminary VICTORIA calculations and laboratory chemistry analysis of combinations of simulated fission-product species were initiated.

2.2.1.1 Filter Sampler Tests

Additional tests were conducted to explore the design parameters of the sampler train that is being developed for the source term experiments. Several stages in the train have been proposed to selectively capture and isolate fission-product vapors--tellurium on nickel, cesium species on silica (SiO_2), and iodine and HI on silver or copper--and to capture particles with a high degree of efficiency in fiber filters designed from a filter model. Initial tests with tellurium on nickel, cesium iodide, and cesium hydroxide on SiO_2, and iodine on silver and copper have been completed and previously reported. The rates of reaction were measured at maximum temperatures anticipated during sampling in the source term experiments. The fiber filter model was also verified for particles from about 2 to 20 μm in size--the size range where the greatest uncertainty exists in the model. It previously has been reported that the reaction of CsOH with SiO_2 is considerably faster than the reaction of CsI with SiO_2--an extremely slow reaction. Examination of the silica by electron microprobe showed that most of the cesium from the CsOH had been deposited on the first millimeter or so of the silica coupon, indicating that the reaction was extremely fast and limited only by the rate at which the CsOH could diffuse to the coupon. Two earlier tests have been repeated but at the lower end of the temperature range anticipated during source term sampling.

The reaction of tellurium vapor with nickel coupons at a temperature of about 625°C appeared to be equally as fast as the reaction measured at about 925°C which was diffusion limited. Also, the reaction of CsOH with SiO_2 at about 670°C appeared, by visual inspection, to be controlled by the diffusion of the hydroxide to the silica. (The earlier test was conducted at 900°C.)

2.2.1.2 Source Term Experiment Capsule Design

The design of the experiment capsule for the first ST experiment is complete. The overall assembly drawings and detail drawings of individual components are being finalized. The procurement of commercial components for the capsule has been initiated. The design work for the shield plug assembly for the ST experiment has begun.

The prototype system to permit use of the bellows sealed compressor at high pressure (up to 30 atm in the second ST experiment) has been assembled and will be tested in the near future.

2.2.1.3 Hot Cell Modifications

Several modifications of the Area V Hot Cell are required for assembly, disassembly, and posttest analysis of the ST experiments. These include completion of a vertical entrance into the cell, construction of an external work area to mate a transfer cask with the vertical entrance, installation of a crane to move the transfer cask from the work area to a truck, and construction of a floor vault and elevator to permit assembly and disassembly of the long ST package within the limited height of the hot cell. The architectural and engineering design of these modifications is nearly complete. Construction is expected to begin in September.

The requirements for the transfer cask have been defined and the design drawings are essentially complete.

2.2.2 Anticipated Activity

The reaction of cesium metal with SiO_2 will be examined in a hydrogen environment and a systems test will be conducted where all stages of the sampler train will be exposed simultaneously to cesium iodine and tellurium species.

Drawings for the ST experiment capsule will be completed and fabrication of the ST-1 capsule and the out-of-pile test vehicle will be initiated. All remaining orders for commercial components for those vehicles will be placed. Construction of the hot cell modifications will begin. The BR-3 fuel rods for the ST experiments will be shipped to Sandia. The development of hot cell fixturing and procedures for assembly and disassembly of the ST experiment will continue. The design of the shield plug structure will be completed.

3. LWR DAMAGED FUEL PHENOMENOLOGY

Sandia's LWR Damaged Fuel Phenomenology Program includes analyses and experiments that are part of the integrated NRC Severe Fuel Damage (SFD) Research Program. We are investigating, both analytically and in separate-effects experiments, the important "in-vessel" phenomenology associated with severe LWR accidents. This investigative effort provides for two related research programs: the Damaged Fuel Relocation (DFR) Program and the Degraded Core Coolability (DCC) Program. The focus of these activities is to provide a data base and improved phenomenological models that can be used to predict the progression and consequences of LWR severe core damage accidents. The DFR experiment program provides unique data on in-vessel fuel damage processes that are of central importance in determining the release and transport of fission products in the primary system. The DCC experiment program provides data on the ultimate coolability of damaged fuel configurations. Models coming from both programs are used directly in the MELPROG code.

3.1 ACRR Damaged Fuel Relocation and Quench
(A. C. Marshall, P. S. Pickard, 6423)

The focus of the LWR DFR experiment program is directed toward providing separate-effects phenomenological data on important severe in-vessel fuel-damage processes to aid in the development of second generation severe accident analysis codes. The core damage configuration, hydrogen generation, and fission-product release are the primary areas of interest. The DF test series uses photography to record the damaged fuel configuration during an in-pile experiment in which accident conditions are simulated in a small LWR rod bundle. The decay heating in these experiments is simulated by fission heating of the fuel in the ACRR. Steam conditions, similar to expected accident conditions are provided.

3.1.1 Current Progress and Technical Highlights

The assembly of the DF-3 test capsule is essentially complete. The remaining leak testing, calibration, and installation should be completed soon. The DF-3 test is scheduled for the first half of September depending on reactor schedules and the conduct of the DCC-3 test in late August. The DF-3 test will examine the effect of Ag-In-Cd control rods on fuel damage behavior. Of particular interest are (1) the effect of Ag-Zr dissolution on Zr oxidation and fuel liquefaction and (2) the time dependence and magnitude of Cd and Sn aerosols formed during the transient.

3.1.1.1 DF-2 Posttest Examination

The test section from the DF-2 test has been sectioned and photographed. More extensive fuel rod damage was found than that indicated in the posttest radiographs. However, the damage (liquefaction and relocation) is less extensive than in DF-1. All of the zircaloy cladding is oxidized to ZrO_2, or has been dissolved into a corium (U-Zr-O) ceramic. Thickness and length of oxidized cladding remnants have been measured and results show the higher initial oxidation obtained for DF-2. The remaining fuel pellets show extensive dissolution due to molten zircaloy. Measurements of the remaining uneroded fuel area are underway to determine fuel liquefaction.

A test proposal for the DF-4 experiment was prepared and sent to a group of experts knowledgeable in BWR accident behavior. Detailed recommendations have been received from several groups and these comments are being used in refining the experiment plan.

The data from the zirconia shrouded tungsten thermocouples are distorted by heat losses down the thermocouple, thermal resistance between shroud and thermocouple, and the heat capacity of the thermocouple components. A thermocouple computer model is being developed to permit correction of these data. Correction of several hundred Kelvins is anticipated, based on earlier thermocouple modeling within DFRMOD.

Uncertainties in the thermal conductivity of the fibrous zirconia insulation used in the DFR experiments have created additional uncertainties in experiment analysis. Detailed measurements of that conductivity have been received. They will be compared to results of DFR test calibrations.

3.1.2 Anticipated Activity

The assembly of the DF-3 capsule will be completed. The DF-3 Ag-In-Cd test will be performed and the DF-4 BWR B_4C experiment plans will be finalized.

3.2 <u>ACRR LWR Degraded Core Coolability</u>

(K. R. Boldt, 6222; A. W. Reed, 6425; T. R. Schmidt, 6421)

The LWR Degraded Core Coolability (DCC) Program investigates the coolability of damaged core debris in water. The debris is fission heated in the ACRR to simulate the decay heat expected in an LWR severe core-damage accident. The governing phenomenological uncertainties being investigated are pressure effects, deep bed behavior, particle size distributions, stratified beds, bottom coolant feed, and material effects. Each DCC experiment will determine the coolability in three thermal regimes: (1) convection/boiling, (2) dryout, and (3) extended dryout. The staff is using experimental results to confirm and/or modify the present analytical models used to predict degraded-core coolability.

The DCC program activities were highlighted by the completion of the DCC-3 experiment plan and the successful conclusion of the DCC-3 safety meetings.

3.2.1 Current Progress and Technical Highlights

The experiment plan for the DCC-3 experiment has been completed. The published plan includes an equipment description, the design analysis, quality assurance program, operational procedures, safety evaluation, and operational limits. The experiment plan was reviewed in three ACRR Safety Committee meetings in which the hardware safety was reviewed and approved. The Sandia Reactor Safety Committee (SRSC) reviewed the overall safety of the proposed experiment plan. The SRSC approved the plan as presented with ACRR operation at 4 MW considered as a separate modification of the reactor limits requiring Department of Energy (DOE) review and approval. After being briefed on the experiment plan, DOE personnel approved the increase in steady-state power from 2 MW to 4 MW.

The assembly of the hardware was nearly completed during this period of time. Loading of the urania into the DCC-3 crucible resulted in a stratified debris bed of 24.1 kg of fuel. The primary containment has been welded shut and leak checked. Reliability testing of the water injection system is ongoing.

3.2.2 Anticipated Activity

Reliability testing will be completed. Loading of the experiment into the reactor is scheduled for early August with operation of the experiment expected to take about two weeks. G. Hofmann (KfK) will arrive to participate in the experiment.

4. MELT PROGRESSION PHENOMENOLOGY
(J. E. Kelly, M. F. Young, P. J. Maudlin, J. L. Tomkins, P. K. Mast, K. A. Williams, W. D. Sundberg, W. J. Camp, 6425)

4.1 Current Progress and Technical Highlights

4.1.1 Code Development and Improvements

Work continued on preparing the 1-D version of MELPROG (MODO) for limited release. While development of this version has been completed, work continues on debugging and improving the code as necessary.

During the testing, a few substantial problems were discovered. In particular, there were two problems which will have significant impact on the stability and results of the code.

The first is related to the interfacial heat exchange between the gas and liquid fields. It was found that the heat transfer coefficients would not allow the gas field to be either superheated or subcooled to a large extent if water were present. This means that certain flow regimes were not treated properly. This problem has been fixed by incorporating correlations from the TRAC code.

The second is related to the proper treatment of flow area changes. The fluid dynamics method in the code calculates the state variables at a given time step assuming the flow area is constant. If the flow areas do not change, then there is no problem. However, if the areas change, then total mass and energy will be in error unless proper care is taken in solving the equations. The solution to this problem is to include the structure volume fraction in the conservation equations. Basically, this can be done by modifying the volume fraction constraint equation. Current testing of this change is under way as it is important throughout the code.

Development of the 2-D version of MELPROG (MOD1) has also continued. The main developments on this version have been in the new core structures module (CORE), the fission-product release and transport module (VICTORIA), and the coupling of all modules together in the code.

Development work on the heat transfer part of the CORE module has been completed. For initial testing and debugging, the CORE module has been coupled to the 1-D version of MELPROG. Once testing is complete, only a minor effort will be required to implement the CORE module into the 2-D version of the code. Work is also continuing on models to treat candling and to treat BWR type core structures.

The VICTORIA stand-alone model (which will serve as the basis for the VICTORIA module in MELPROG) has been completed and is operational. The model is being used in the design of the ST experiments. Currently, it can be used to calculate fission-product behavior throughout the vessel by using a soft-link with MELPROG. Coupling of the model into MELPROG has been initiated.

Work on coupling all modules to the new 2-D FLUIDS module has entered the final stages of development. The majority of the work has been completed and only a few areas need to be resolved. The main one involves the incorporation of the multicomponent mass conservation equations into the 2-D FLUIDS module. Work on this task is currently being performed.

4.1.2 Code Testing

Testing of the MELPROG-PWR/MOD0 has continued using a simulation of an S1D accident sequence. This sequence exercises all modules and can be run at low cost. The current geometric model includes the entire vessel (i.e., upper and lower plenums and the core region). Boundary conditions for this problem have been obtained from a RELAP-5 calculation. This test case is being used to test and debug the 1-D version of the code (MOD0).

The calculation has been run past the point where the lower core plate failed. Failure of this plate allowed the solid and molten debris to fall into the lower plenum. At that point debris regions formed on the lower plenum and began to heat the vessel head.

This test case is being closely monitored to assure that the code is predicting consistent results. The physical correctness and implications of these results are also being studied.

4.2 Anticipated Activity

The MOD0 version of the MELPROG will continue to be tested. The code will also be released to selected users.

The MOD1 version of the code will become operational in the near future. At that time, testing and debugging of the code will commence.

5. ADVANCED REACTOR ACCIDENT ENERGETICS

The Advanced Reactor Accident Energetics Program addresses the key issues in an LMFBR core-disruptive event that determine the progression and severity of the accident. This program involves a series of in-pile experiments and analyses that focus on key phenomena in two general areas:

1. Initiation Phase--Fuel/Clad Dispersal Experiments

2. Transition Phase--Fuel Freezing and Streaming Experiments.

5.1 Initiation Phase

(S. A. Wright, P. S. Pickard, G. Schumaker, 6423)

The Sandia Fuel Dynamics Program provides experimental data and analysis for the initiation phase of an LMFBR core-disruptive accident. The motion of clad and fuel in the initiation phase of an LOF accident is an important consideration in the subsequent progression of the accident. Early fuel dispersal can lead to neutronic termination while limited dispersal and blockage formation continue the accident into the transition phase and the possibility of further neutronic activity.

To obtain data on the important phenomena involved in this phase of an LMFBR accident, the NRC is sponsoring the Sandia Transient Axial Relocation (STAR) experiments in the ACRR test facility. Kernforschungszentrum Karlsruhe (KfK), FRG, and the Power Reactor and Nuclear Fuel Development Corporation (PNC) are cosponsors of this program.

5.1.1 Current Progress and Technical Highlights

The main effort focused on preparation and planning for the upcoming STAR-7 and STAR-8 experiments. All of the hardware for these experiments was ordered, and extensive discussions with PNC were conducted to specify the goals and objectives of the STAR-7 experiment. In addition, a proposal for funding of the STAR-8 experiment was sent to KfK.

Data analysis and posttest calculations for the STAR-5 and STAR-6 experiments were also performed. The results of these efforts will be included in Quick Look reports which will be completed during the next two months.

As a result of the PNC discussions, the STAR-7 experiment will simulate the upper bound for the "best estimate" LOF accident scenario in the MONJU reactor. Extensive SAS3D and SAS4A calculations for this accident were performed and they predict that the power transient rises fro. nominal power to 70-100 times P_o in 200 to 300 ms. This rapid power rise occurs due to a combination of sodium boiling, and positive reactivity-induced fuel motion. (The SAS3D code predicts blow down of the upper pin stubs, driven by plenum gas pressure assuming no slip, while SAS4A predicts limited in-pin fuel motion due to mild fuel vapor pressure.) Fuel dispersal terminates the accident (but leaves the reactor in an uncertain state which is assumed to enter the transition phase) due to fuel vapor pressure in the case of the SAS3D calculations. Steel vapor pressure is responsible for the dispersal in the SAS4A calculations. Thus, the main objectives of the STAR-7 experiment are to provide heating conditions that reproduce those of the SAS calculations to determine which mechanisms are responsible for the fuel dispersal.

The STAR-8 experiment was to simulate a LOF-d-TOP in a partially voided subassembly, and investigate cladding rip size, fuel ejection rate, and potential in-pin fuel motion. A proposal to fund the STAR-8 experiment was sent to KfK, but the proposal was turned down due to lack of funds. As a consequence, STAR-8 will not be performed.

5.1.2 Anticipated Activity

The STAR-7 experiment will be assembled and final preparations for performing this test will be performed. This experiment requires minor changes to the experiment plan due to an increased fuel inventory; thus, a new experiment plan will be submitted to the safety committees. The desired power pulse also requires modifications to the ACRR pneumatic drive system used to insert the transient rods. Thus, these changes will also be included in the experiment plan. These modifications will also have to be approved by DOE.

5.2 Transition Phase
(R. O. Gauntt, P. S. Pickard, A. Furutani, 6423)

Many current heterogeneous core designs are characterized by relatively low-sodium void coefficient and incoherent behavior in the initiation phase. These features generally increase the likelihood of a "transition" or "meltout" phase during a core-disruptive accident. The key questions in the transition phase, highlighted in the CRBR safety review, are whether fuel or clad blockages form, leading to a confined or "bottled" core configuration, and the behavior and reactivity implications of this pool of fuel-steel in the core region if the fuel blockages do lead to this state.

The TRAN program addresses the question of fuel-inventory reduction by penetration into the upper core structure through subassembly (S/A) gaps to the lower core structure. If deep penetrations occur, nonenergetic shutdown is probable while shallow penetrations will lead to a transition phase and the possibility of further energetics. First-of-a-kind in-pile experiments, sponsored jointly by the USNRC and the Japanese Power Reactor and Nuclear Fuel Development Corporation (PNC), are being conducted to provide data to evaluate the various models describing fuel penetration.

5.2.1 Current Progress and Technical Highlights

5.2.1.1 B-Series Experiments

The postirradiation examination (PIE) for the TRAN B-4 multi-pin fuel freezing experiment has been completed. A total of seven cross-sectional cuts and one longitudinal cut were made through the channel. The cuts made near the entrance of the channel revealed extensive cladding ablation. The character and thickness of the UO_2 crust varied considerably over the first 20 cm of channel length, showing evidence of a two-stage deposition of crust material. Cladding initially present in the first 2 cm of the channel showed evidence of having been quickly ablated into the flow leaving behind only bare UO_2 pins coated with a thin uniform layer of UO_2 crust. Further into the channel, the steel clad also has been removed but apparently at late times following crust solidification. This is apparent from the shape of the crust which has remained in place even though the clad did melt and drain away. This feature continues through all of the fuel penetration length of about 60 cm.

A SIMMER analysis is currently being carried out on this experiment and is expected to be completed in the next reporting period. A topical report is also being prepared.

5.2.1.2 The GAP Experiment, G-2

The GAP-2 experiment package, which is designed to investigate the downward injection of kilogram quantities of molten fuel into a channel structure which approximates the inter-can wall gaps in the lower blanket structure of an LMFBR, continues to be assembled. Precalculations using SIMMER and PLUGM are being pursued. The safety analysis for GAP-2 has been reviewed and approved by both the ACRR Safety Committee and the Sandia Reactor Safety Review Committee. Fuel has been received and will be installed in the package in August followed shortly by the remainder of the package assembly. The design pulse for melting the fuel load is being analyzed at this time. Plans are being made to conduct the calorimetry preexperiment which is used to measure the experiment-ACRR energy coupling factor. The experiment is planned for early September.

6. POSTACCIDENT HEAT REMOVAL

6.1 Debris Bed Coolability
(C. A. Ottinger, 6322; T. R. Schmidt, 6421)

The objective of the Debris Bed Coolability Program is to develop experimentally validated models for the behavior of LMFBR core debris after a severe accident. The primary tools in pursuing this objective are coolability experiments using conditions as prototypic as possible so as to determine all the important phenomena needed to develop models to predict coolability limits. Fission heating of UO_2 is the only heating method currently available that can provide an adequate simulation of decay heat for many of the expected debris and coolant configurations. Fission-heated coolability experiments provide the foundation of the debris bed coolability research at Sandia. The program is cosponsored by the NRC, EURATOM (JRC, Ispra), and the PNC (Japan).

6.1.1 Current Progress and Technical Highlights

Activities involved reduction of the data from the D13 experiment, which is the last in the experimental program. Preparations were made to x-ray the experiment and place it in long-term storage. A criticality assessment was performed for the experiment in storage and submitted to the Radiological and Criticality Safety Committee.

The channeled dryout data in the D10 experiment was not well predicted by the models. The capillary pressure was measured on the D10 broad particle size distribution in the laboratory and showed a marked difference from the previously used Leverett function. Using this capillary pressure data, the results of the channeled dryouts are bounded by the uncertainties in the channel model. The principal unknown is the cohesion of the particulate which influences channel depth.

The report on compatibility of refractory metals with sodium in the presence of oxygen and UO_2 at high temperatures was published. The report on the D9 experiment (a shallow stratified bed) was published.

6.1.2 Anticipated Activity

The analysis of D13 will continue and the experiment will be x-rayed.

6.2 Dry Debris Melt Progression
(T. R. Schmidt, 6421; J. T. Hitchcock, 6427; J. E. Kelly, 6425)

In order to establish the release time frame and quantity of radioactive materials following a severe accident, it is necessary to determine the fuel melt dynamics and the characteristics of melting attack by molten fuel on reactor structure and containment barriers should postaccident debris heat removal not be adequate. Simulation of those portions of debris beds undergoing extended dryout and melting is necessary to support modeling activities. This study is aimed at providing such data through in-core experimentation with typical reactor material undergoing sustained, intrinsic heating at temperatures of interest. Models developed are provided to the LWR severe accident code MELPROG as well as to the direct analyses of LMFBR debris coolability. This work is cosponsored by the NRC, the Japanese PNC, and EURATOM (JRC, Ispra).

6.2.1 Current Progress and Technical Highlights

The continuation of the postirradiation examination has been delayed pending receipt of the hot stage for the new SEM. The photomicroscopic examination has been completed. The large temperature gradients in the DC1 experiment, which resulted in containment of the molten zone, produced unique conditions for lenticular pore formation and migration toward the hot zone. There were large amounts of sintering and urania vapor transport and condensation in certain regions of the experiment. In DC2, the steel (1) wetted the urania which inhibited gross transport, and (2) tended to agglomerate locally with lesser amounts in the hottest zone.

The experiment report for the DC1 and DC2 experiments was published.

6.2.2 Anticipated Activity

The SEM analysis will be performed as soon as the needed equipment arrives.

7. DISCLAIMER NOTICE

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