Technical Highlights/Administrative Report
for
the Nuclear Regulatory Commission (NRC)
Reactor Safety Research Program

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TECHNICAL HIGHLIGHTS/ADMINISTRATIVE REPORT
FOR
THE NUCLEAR REGULATORY COMMISSION (NRC)
REACTOR SAFETY RESEARCH PROGRAM
April-May 1985

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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, and J. E. Gronager, 6422)

1.1.1 Current Progress and Technical Highlights

1.1.1.1 Test of the SURC Heating Method

An important component of future experimental investigations of core debris interactions with concrete will be the tests of sustained urania melt interactions with concrete (the SURC tests). The sustained heating in these tests is achieved with embedded susceptor rings of refractory metals. Unfortunately, these refractory metals can be corroded both by gases evolved from concrete and by molten concrete. A concern then is that the susceptor rings may be destroyed by the corrosion processes before significant test results can be obtained.

A quick test was run to determine how rapidly the rings might corrode. The test involved about 30 kg of urania heated by induction using embedded susceptor rings. Melt was achieved. The rings, which were freely suspended in the charge, sank in the melt and directly attacked the concrete. Though the rings were somewhat corroded, the damage was not severe. It appears, then, that the success of the SURC test program will not be limited by possible corrosion of the susceptor rings.

1.1.1.2 Heat Transfer to Overlying Water

The test SWISS-2 involved the combined interaction of water, core debris, and concrete. Analysis of data from this sustained test shows that the heat flux from the core debris to the overlying coolant layer was quite low. This observed heat flux is consistent with the minimum film boiling heat flux, rather than critical heat flux (CHF), which has been used in many recent reactor accident analyses. The lower heat fluxes that arise in the tests probably occur because of noncondensible gases passing through the vapor film separating the core debris from the coolant. The lower heat fluxes mean that more heat must pass into the concrete or that mean melt temperatures are higher.

1.1.1.3 Posttest Examination of TURC-3

Posttest dissection of the test fixture used in test TURC-3 has begun. The test involved the transient interaction with concrete of about 50 kg of a urania-zirconia melt to which about 10 w/o metallic zirconia had been added. The posttest examinations showed that the melt had frozen quickly upon contact with the concrete. Little erosion had occurred. What gases evolved from the concrete probably did not sparge through the melt.

1.1.1.4 Code Comparison Exercise

The Containment Systems Research Branch of the USNRC is interested in demonstrating the current state of technology for predicting the nature of core debris interactions with concrete. To do this, they have initiated a code comparison exercise in which developers and users of the various existent models of melt/concrete interactions will predict the results of four tests conducted in the Ex-Vessel Core Debris Interactions program, SWISS-1, SWISS-2, CC-2, and TURC-2.

The first of these tests, SWISS-1, involved the sustained interaction of about 45 kg of stainless steel with limestone/common sand concrete. Near the end of this test a flow of water was established over the melt. Test SWISS-2 was nearly identical to test SWISS-1 except a flow of water was established over the melt interacting with concrete very early in the test. Test CC-2 involved the sustained interaction of 206 kg of molten stainless steel with limestone concrete. Test TURC-2 involved the transient interaction of about 150 kg of molten urania-30 w/o zirconia with limestone/common sand concrete.

A document was prepared that describes these four tests in sufficient detail so the participants in the code comparison exercise could attempt to predict the test results. The document also defined test results that would be useful for demonstrating the accuracy of the model calculations. These results were those that had been directly measured in the tests, such as gas composition, concrete erosion, and melt temperatures.

The planned code comparison exercise is restricted to the prediction of features of the attack on concrete. Comparison of predictions of aerosol release during the interactions is not now a part of the exercise.

1.1.1.5 Planning for the WITCH/GHOST Test Series

The WITCH/GHOST Tests are separate effects experiments intended to support the development of the VANESA model of fission-product release and aerosol generation during core debris/concrete interactions. Attention during the report period was focused on the WITCH tests, which will examine the rate of aerosol generation by mechanical processes when gases sparge through a melt. The current implementation of the VANESA model treats the mechanical generation process based on some literature data for aerosol production by air sparging through water. The predicted aerosol production by this process is small in comparison to production by vaporization. More thorough reviews of the literature for such mechanical aerosol production show that correlations exist aside from that used in the current implementation of VANESA. These correlations suggest, in some cases, that the mechanical aerosol production could be a factor of 1000 greater than currently predicted. No data were found that were directly applicable to melts interacting with concrete.

The WITCH test series will provide mechanical aerosol formation data for gases sparging high temperature melts. The tests will use inert gas (argon or helium) to sparge oxidic melts similar to those produced when concrete melts.

1.1.2 Meetings and Documentation

The principal investigators participated in a mid-year review at the Ex-Vessel Core Debris Interactions program in Silver Spring, MD, May 14-16, 1985. At this review, progress in the SWISS test series to investigate combined core debris/coolant/concrete interactions, the TURC series of transient urania melt/concrete interactions tests, and the HS tests of hot, solid core debris/concrete interactions were described.

Two investigators from the project attended the Annual Meeting of the Fine Particle Society held in Miami, FL, April 22-26, 1985. A session of this meeting was chaired by a principal investigator in the Ex-Vessel Core Debris Interactions program. Two papers based on work in the program were presented at the meeting:

D. A. Powers and J. E. Brockmann, "A Mechanistic Model of Release of Radionuclides and Generation of Aerosols During Reactor Core Melt Interactions with Concrete," and

J. E. Brockmann and J. E. Gronager, "Experimental Measurement of Material Release During Melt/Concrete Interaction."

A principal investigator from the program participated in a meeting between the NRC and the Industry Degraded Core Rulemaking project held April 30, 1985 in Bethesda, MD. The purpose of this meeting was to define and resolve differences in the approach to the analysis of ex-vessel phases of severe reactor accidents.

A document describing boundary conditions for four tests (SWISS-1, SWISS-2, CC-2, and TURC-2) was prepared at the request of the NRC. The purpose of this document was to define input needed to perform computer code predictions of the test results.

The topical report on the TURC-1T and TURC-1SS tests was submitted in draft form for NRC review:

J. E. Gronager et al., TURC-1: Large Scale Metallic Melt-Concrete Interaction Experiments and Analysis, SAND85-0707.

1.1.3 Anticipated Activity

The topical report on the SWISS tests will be available in draft form.

The topical report on the first series of hot, solid core debris/concrete interaction tests will be available.

The WITCH test series will begin.

1.2 High-Pressure Melt Ejection and Direct Containment Heating
(W. Tarbell, 6422; M. Pilch, 6425; J. Brockmann, 6422; I. Cook, 6425; W. Frid, 6425; D. Powers, 6422)

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

The previous HIPS (high-pressure streaming) experiments used a cavity design based upon the geometry of the Zion nuclear plant. The nearly complete dispersal of debris during the four tests suggests that this configuration is highly susceptible to discharge of the material into the containment region. The behavior of the debris in the containment building will then determine the extent of energy transfer that will occur. The highly complex nature of a nuclear plant's geometry and large quantity of in-containment equipment may possibly inhibit the energy transfer processes that normally occur. For this reason, the interaction of the ejected debris with realistic plant configurations is important to understanding the potential threat of this type of accident.

A test was performed incorporating a simulated in-containment structure to investigate the behavior of the dispersed debris. The same cavity geometry as the previous HIPS tests is used: a one-tenth linear scale model of the Zion cavity and instrumentation tube tunnel. The simulated structure is plated at the cavity exit and is modeled after the plant configuration, except that all free flow paths are eliminated. This structure represents the room immediately below the seal table in the Zion plant, with a third vertical wall added to be even more enclosed than that found in the plant. The only opening in the structure is a portal for instrument tube entry located in the wall away from the direction of the debris flow.

In the Zion plant, the angle of the instrument tube shaft will cause the gas stream to be directed into the enclosure beneath the seal table. Because the flow paths through and around the seal table are small, the gas will probably stagnate and deflect away from the structure. The initial conditions for both the accident and test are such that only very small particles (estimated to be less than 50 μm) are predicted to follow the gas streamlines. This means that as the gas stream encounters the obstruction of the wall structure, the bulk of the material will not be diverted and will impact onto the object. The objective of this experiment is to simulate the conditions and geometry of the Zion plant to determine the magnitude of the debris trapped and retained.

The principal measurement in this test is the partitioning of mass between that remaining in the cavity, retained by the wall structure, and dispersed away from the apparatus. This is accomplished by obtaining a posttest mass balance from the various portions described above. To facilitate the measurement, the apparatus is placed into a large, open-ended steel container (5-m long, 3-m wide, and 3-m high) to retain the material dispersed away from the device. The apparatus is oriented so that the cavity exit and wall structure are just outside the open end of the container, so that the normal direction of debris flight would be away from the container. In this manner, any debris deflected from the wall structure will likely be captured for posttest retrieval.

The initial test conditions duplicate those of the HIPS-5C experiment, where the debris is allowed to expand freely out of the cavity. Using low-pressure carbon dioxide (approximately 5 MPa) minimizes the extent of melt fragmentation and dispersion caused by the carrier gas coming out of solution. The standard melt mass of 80 kg is supplemented in this test by an additional 1.5 kg of fission-product mocks. These are used to study the propensity of radionuclides to form aerosols that may present a hazard if the containment building is damaged during the accident. Various aerosol sampling devices are used to collect the material formed during the test.

The preliminary observations from the test show that nearly all the debris escapes the cavity. The mass retained in the cavity is on the order of 1.5 percent (1.3 kg), principally in the form of a thin crust (2 to 4 mm) on the exposed metal surfaces. As seen in all the previous HIPS cavity tests, debris retention on the exposed concrete surfaces is virtually nonexistent. Very minor concrete attack is found, the most extensive being directly under the melt generator discharge orifice where the melt stream stagnates. Ablation of the generator orifice is typical, enlarging the initial 2.5-cm diameter hole to about 8 cm.

Unlike the appearance of the cavity, a relatively thick layer of debris is found on the surfaces of the wall structure. The overall appearance is that of a fairly smooth layer of frozen melt interspersed frequently by embedded particles. On the sides and back wall, the size of the particles is typically twice as great as the surrounding melt layer. On the horizontal surface (the "ceiling"), however, the melt layer is somewhat thicker so that the particles are less obvious. Most regions of the crust have thicknesses varying from 1 to 4 cm except near the top center of the back wall and ceiling. This location is assumed to be the center of where the debris is directed by the cavity exit. The thickness of the material in this area is on the order of 1 cm, extending for about 10 cm in all directions.

The debris within the wall structure is not tightly adhered to the concrete surface and can be easily removed. The total mass found on the wall structure and that lying loose within its perimeter is 3.3 kg (4 percent of the original mass). Even though the bulk of the debris impacts onto the wall structure, one or more mechanisms subsequently remove the material. Considering that the gas velocities inside the structure are probably low relative to the cavity region, entrainment of the liquid film is unlikely. Preliminary analysis of the phenomena suggests that the momentum of the impinging particles may cause fragmentation on impact, resulting in numerous smaller debris. These smaller particles may rebound from the surface and reenter the gas stream where they are carried away from the apparatus. The possible

1.3.1 Current Progress and Technical Highlights

1.3.1.1 Sensitivity Study of CORCON

A formal sensitivity study of the CORCON mod2 code has been initiated. This study is using the Latin hypercube sampling method. Fifteen input variables are being considered in the analysis. These input variables include parameters used in the CORCON calculations such as heat transfer coefficients, material properties such as surface emissivities, plant characteristics such as concrete type, and initial condition information such as initial melt temperature and the initial steel mass. The sensitivities of the time-dependent outputs from CORCON, such as the core melt temperature and the gas generation rates, to these inputs are being studied. The sensitivities of events in the interaction of core debris with concrete, such as the time required to oxidize all zirconium in the core debris, are also being considered.

The initial study of the sensitivities is based on 50 CORCON runs selected on the Latin hypercube, modified Monte Carlo basis. Preliminary conclusions from these analyses are:

1. The type of concrete is the most important variable.

2. Other important input variables are those that in an accident analysis would be propagated to CORCON from other models used in accident analyses such as the MARCH model of core degradation.

Heat transfer models and models of material properties in CORCON do not seem to affect greatly predictions of quantities considered in the sensitivity study. It is found, however, that significant alterations of the gas film model for melt-to-concrete heat transfer can cause the melt to suddenly solidify.

1.3.1.2 Melt-to-Concrete Heat Transfer Models

The metallic melt/concrete model in CORCON has been a subject of much interest. The model is based on the results of simulant tests such as heat transfer from water to dry ice. Use of CORCON with this heat transfer model yields a poor prediction of concrete erosion observed in recent tests (TURC-1, BETA) of metallic melt/concrete interactions.

Available heat transfer correlations from literature sources all group closely together, except for a model due to Kutateladze. Most of these literature correlations are based on low temperature, low conductivity fluids rather than liquid metals. In most other instances, heat transfer correlations for metallic liquids do not conform with correlations developed for more conventional liquids. The applicability of simulant test data to the analysis of metallic melt/concrete interactions can be questioned.

Examination of heat transfer data for boiling metals seems to suggest a model appropriate for CORCON similar to the Kutateladze model, rather than those developed from the simulant fluids. This correlation of data for boiling-metal liquids, revised to suit the concrete ablation problem, is:

$$Nu = 44(PrRe)^{2/3}(\rho_g/\rho_\ell)^{2/3} \quad .$$

This correlation would yield heat transfer coefficients about an order of magnitude lower than what is now used in CORCON. Replacing the existing model in CORCON with this new correlation would lead to higher predicted melt temperatures for a given rate of concrete erosion. The high melt temperatures might significantly affect the rates of radionuclide release and aerosol production during core debris/concrete interactions.

1.3.1.3 Layer Flip Predicted by CORCON

Use of conventional data on the density of constituents of core debris will lead to the conclusion that core melts emerging from a reactor vessel will consist of a dense oxide phase and a less dense metallic phase. If the melt mixture stratifies according to density in the reactor cavity, the dense oxide layer will reside adjacent to the concrete and the metallic layer will float atop this oxide. As attack on the concrete progresses, low density constituents of concrete will be incorporated into the oxide phase reducing the mean density of the layer. Eventually the oxide density will fall below that of the metal phase and the layers will invert their positions.

The CORCON model performs a conventional density analysis of the core debris constituents. The inversion of the relative densities of the oxide and metallic phases of the melt, the layer flip, is an important event in an accident calculation done with CORCON.

Recent analyses done for the VANESA model suggest that conventional density analyses may be inapplicable to core debris and that layer flip is not a real phenomenon in core debris/concrete interactions. These calculations examined the hypostoichiometry in urania caused by interaction with metallic zirconium. The analyses lead to the conclusion that a significant uranium metallic activity would develop in the fuel. This activity would manifest itself as the appearance of uranium in the metallic phase equilibrated with the fuel in a core melt. The addition of very dense uranium to stainless steel will often be sufficient to cause the mean metal density to exceed that of the oxide. Then the metal phase, not the oxide phase, will be the densest phase in a core melt emerging into the reactor cavity from the reactor vessel. If this melt stratifies according to density in the cavity, then the oxide phase will float on the metal phase. This is, of course, the debris configuration presumed in the VANESA model.

1.3.1.4 VANESA Review and Documentation

A thorough review of the technical basis for the VANESA model of radionuclide release and aerosol generation during core debris/concrete interactions was presented at a program review held in Silver Spring, MD, May 14-16, 1985. This review described:

1. The substantive predictions made by the model, many of which are contrary to predictions made by simpler models such as that devised for the Reactor Safety Study,
2. The treatment of thermodynamics used in the VANESA model,
3. The treatment of vaporization kinetics and how the kinetic processes can limit the rate of approach to the maximum releases predicted considering thermodynamics alone,
4. The modeling in VANESA of mechanical aerosol formation processes,
5. The modeling in VANESA of the nucleation and condensation of vapors evolved from the core melt, and
6. The modeling of the effects a water pool overlying the core melt would have on the magnitude and nature of release of aerosols into the reactor containment.

Documentation of the model is underway. This document will describe what is done in the model and the technical context in which these models were designed.

1.3.2 Meetings and Documentation

Principal investigators from the CORCON and VANESA Development project participated in the program review held in Silver Spring, MD, May 14-16, 1985. A thorough description of the VANESA model was presented at this meeting, as described in Section 1.3.1.4. Also, an update on the status of the CORCON model was presented.

A principal investigator from the project participated in the meeting between NRC and the Industry Degraded Core Rule-making project (IDCOR) held in Bethesda, MD, April 30, 1985.

An investigator from the program participated in the annual meeting of the Fine Particle Society (April 22-26, 1985) where he presented to the society a description of the VANESA model and a comparison of the model predictions to experimental data.

A draft report (D. A. Powers and D. R. Bradley, <u>Some Causes of Uncertainty in Estimates of Ex-Vessel Radionuclide Release During Severe Reactor Accidents</u>, SAND85-1780) on the sensitivity study of the combination of the CORCON and VANESA models was submitted for NRC review.

1.3.3 Anticipated Activity

The draft report on the VANESA model will be submitted for review.

The sensitivity study of the CORCON model will be completed.

1.4 <u>Molten Fuel-Coolant Interactions</u>
(B. W. Marshall, Jr. and 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 light water reactors (LWRs). FCIs can occur in the core region, in the reactor lower plenum, or in the cavity below the vessel. They can occur when melt falls into water or when water falls into melt. The understanding of FCIs achieved in this program should be sufficient to resolve the key reactor safety issues for both terminated and unterminated accidents. Models are being developed to determine:

1. The rates and magnitudes of steam and hydrogen generation due to FCIs;

2. The probability and consequences of direct containment failure by steam explosions;

3. The influence of FCIs on accident progression and the nature of the source term (including fission-product chemistry, release rate, particle size, and dispersal);

4. The consequences of pouring water on the melt in order to terminate an accident;

role of ablating surfaces is being studied to determine if benign materials will react differently.

The results from the HIPS-7C test suggest that debris is not retained on a structure of simple geometry. Debris trapping is apparently mitigated by mechanisms that do not rely on a high-velocity gas sweeping over the melt surface. Continued analyses of the data may provide a basis for developing a model that can be applied to more realistic situations. Future experiments must consider more complex geometries, such as those found in current nuclear plants in order to ascertain the behavior of debris in the containment building during severe accidents.

The planning for the next experiment in the HIPS test series is proceeding. The test objective is to study the partitioning of ejected debris into the annular gap surrounding the RPV. This region can potentially allow debris to enter directly into the containment dome without passing through a convoluted path. The initial conditions will duplicate the HIPS-5C test except that the cavity will be modified to incorporate a simulated annular gap. One-tenth linear scaling of the cavity and gap dimensions will be used to preserve gas velocities and flow paths. Because the debris size is not scaled, the vertical length will be shortened to avoid extended crust formation and possible plugging of the gap. The width of the gap itself represents an average of the actual geometry with and without the RPV thermal insulation. The apparatus will be installed in the containing structure described above so that the material emerging from the gap region can be collected and measured. Both real-time and flash x-ray techniques will be used to visualize the dispersed debris.

An analytical study of high-pressure accident sequences has revealed a possible serious consequence not previously considered. Fission products may be released during debris ejection and subsequent transit through the containment building by oxidation to a more volatile species. Materials that are susceptible to this type of behavior are tellurium, molybdenum, niobium, and ruthenium. The Reactor Safety Study (Wash-1400) recognizes this type of release during steam explosions. Application of this method for a highpressure accident sequence involving one-half of the core gives a ruthenium release of over 40 percent of the available inventory. The transport of oxygen to the debris in the melt jet, however, is a potentially limiting mechanism not considered in this method. Mechanistic modeling of the oxidation-driven vapor release performed in conjunction with the SURTSEY experimental program is necessary to better estimate the significance of this release mechanism.

1.2.2 Documentation

M. Pilch and W. Tarbell, <u>A Preliminary Study on the Direct Heating of the Containment Atmosphere by Airborne Core Debris</u>, to be published as a Sandia Technical Report.

W. Tarbell et al., <u>HIPS Program Plan</u>, draft report submitted to the USNRC for review.

1.2.3 Meetings

Program personnel participated in the Core Melt Technology and Analysis Review meeting in Silver Spring, MD, May 14-16.

1.2.4 Anticipated Activity

Construction of the HIPS-8C apparatus will be completed and assembly of the experiment begun.

Analysis of the data from HIPS-7C will be continued and documentation initiated.

Designs for the SURTSEY vessel will be received from the supplier and reviewed by Sandia project personnel. The construction of the support pad should be completed.

The final version of the HIPS Program Plan will be made available.

Draft report on the SPIT and HIPS tests involving dry cavities (without water) will be completed.

1.3 <u>CORCON and VANESA Code Development</u>

(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.

5. The characteristics of the debris produced by FCIs, including particle size distributions, porosity, and coolability.

1.4.1 Current Progress and Technical Highlights

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

No experiments were conducted in the FITS (fully instrumented test site) vessel during this period. The FITS-D 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. and 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 FITS-D Experiments
(B. W. Marshall, Jr. and O. P. Seebold, 6427)

Data analysis reports were completed for the FITS-0D and FITS-5D experiments. The FITS-0D experiment exhibited the same general eruption characteristics as noted in the CM series. The FITS-5D experiment was characterized by a double explosion with an elapsed time between explosions of approximately 3 ms. The peak gas phase pressure recorded in the FITS-5D experiment was the largest recorded for a 20-kg experiment in the FITS tank. The peak pressure recorded was 0.625 MPa or 0.542 MPa above the ambient pressure of 0.083 MPa.

1.4.1.4 High Explosive Simulations of a Steam Explosion
(B. W. Marshall, Jr. and J. Fisk, 6427)

In past FCI experiments, we have had some difficulty measuring pressures generated in the water. This is an extremely crucial measurement since codes, such as CSQII, "match" the experimental pressure signature and magnitude and then infer the kinetic energy of the material and the loading on the structures due to the FCI. The accuracy and reliability of these pressure measurements is important when using these codes and the data generated from the analysis of the FITS tests to predict the consequences of FCIs in a reactor vessel.

The high-explosive simulation of a steam explosion to validate the water phase pressure measurements continued. A series of experiments was conducted using a single strand of primacord located in the center of the rigid water chamber (the Straight High Explosive (SHE) series). These experiments addressed the symmetry problems noted in initial HEX experiments.

During the SHE experiments, we noted a sensitivity of the gauge responses to the mounting hardware. As in the HEX experiments, we are able to record consistent and repeatable times-of-arrival and peak-times, but were unable to obtain symmetric peak pressures. The symmetric timing of the gauge responses suggest that the charge is located in the center of the vessel, although the measured peaks were not the same.

Comparison of the gauge responses with CSQII calculations (described in Section 1.4.1.2) have not been favorable. CSQII calculations predict earlier pressure arrivals and peak times, while the magnitude and general pressure signature is noticeably different.

The HEX and SHE data are being analyzed for additional trends, which may aid in diagnosing these problems. We continue to experience difficulty in measuring water phase pressures in close proximity of an explosion. The solutions to these problems will be directly applicable to the FITS-FCI experiments and will yield more accurate pressure results.

1.4.1.5 Upgrade of Experimental Measurements (B. W. Marshall, Jr., 6427)

The mass spectrometer, which will be used in the remainder of the FITS-D series, has arrived and is being installed at the FITS facility. Preliminary check-out and operational tests were performed in hydrogen:air:steam atmospheres. High volumetric concentrations of steam did not present any problems. The calibration and final checkout of the system will be completed before use in the FITS experiments. A computer code is also being written that will interpret the voltage output of the quadrupole as volumetric concentrations (or partial pressures) of hydrogen, oxygen, steam, carbon dioxide, carbon monoxide, nitrogen, and argon as a function of time. Progress in this area will be reported as needed.

1.4.1.6 Modeling of Explosion Propagation and Structural Loading Using CSQII (K. L. Schoenefeld and M. F. Young, 6425)

In past FITS experiments, we have made measurements of the pressure, temperature, debris, and gas samples, and provided photographic coverage of the interaction. We are currently pursuing additional experimental techniques which will provide new and more complete information about the phenomena. This information will then be used to develop more complete and accurate codes to predict the consequences of FCIs in a reactor vessel or containment.

The 2-d hydrodynamic code CSQII was used to model the SHE high explosive tests, which were being conducted to test different pressure gauges and their mountings.

In the HEX series PETN primacord was wrapped in a spiral around a cylinder and placed in a rigid container of water just beneath the top surface. CSQII cannot exactly model this type of burn since it is three-dimensional. To determine whether or not the spiral burn of the PETN may cause azimuthal variations in pressures, a top view of this arrangement was modeled. As a first approximation, both the container and one wrap of the PETN primacord were modeled as squares. Code results indicate that peak pressures in the initial pulses vary by as much as 30 percent for points located 180° apart.

In the SHE series one strip of PETN primacord was suspended vertically in the center of a rigid container of water. A side view of this arrangement was modeled with CSQII in cylindrical coordinates (variations in r and z directions). Code results show an initial pressure pulse of about 110 bars striking the base of the chamber, a pulse of about 200 bars on the lower side wall, and a pulse of about 275 bars on the upper side wall. Pressure waves then reflect off of the walls and "implode," resulting in peak pressures of 275-1100 bars occurring along the base.

In these HEX and SHE tests, pressures were measured with gauges located in offset gauge blocks. To determine the effect of this offset, CSQII was used to model the gauge block geometry with various hole radii and depths. The results indicate that a pressure wave striking the base doubles and then continues down into the gauge block cavity and nearly doubles again when it strikes the bottom of the hole. Variation of hole depth seemed to affect only the timing of the pulse arrival, not its shape or peak. As the diameter of the hole increases, the pulse that continues into the cavity broadens and decreases in magnitude. The magnitude, however, of this pulse was still greater than the original pulse, even with a hole diameter of 6 cm.

1.4.1.7 Modeling of Coarse Fragmentation and Hydrogen and Steam Generation (WISCI) (M. L. Corradini, C. Chu, UW)

During this period we continued to develop benchmark problems to test the TEXAS computer program. The current sample problems are taken to be similar to the EXO-FITS MD series of tests (e.g., MD-19) where a single mass of fuel is dropped into a pool of water. The model developed by Chu for coarse fragmentation of the fuel by relative velocity induced hydrodynamic instabilities has been incorporated into TEXAS for these calculations. The preliminary results indicate that for the conditions of MD-19 the fuel enters into the water pool with an initial diameter of 0.13 m and when it reaches the bottom of the chamber it has broken up due to its relative velocity to a size of 6 mm. The current coarse fragmentation model requires some knowledge of the time of entry into the water pool and its diameter; therefore, we are attempting to make this model more usable without the need for knowledge of its prior history.

1.4.1.8 Modeling of the Steam Explosion Phase (TEXAS) (M. L. Corradini, M. Oh, UW)

In this period we have begun to use the nonequilibrium model to analyze the scaling behavior of vapor explosions at scales larger than the current FITS tests including postulated full scale accident situations. This effort is part of the final analysis to be done before the computer model is given to the Sandia researchers for use in modeling the FITS tests. In these calculations we consider two situations for the full scale severe accident case, based on discussions with M. Berman and B. Marshall. In the current calculations, we consider the pressurized water reactor geometry; the boiling water reactor geometry is probably a subset of these considerations, however, we have not been able to get detailed drawings to check this point. The first situation is a vapor explosion which occurs in the lower plenum before it reaches the large internal structure (i.e., the lower support forging). The second situation is a vapor explosion which occurs below this structure and is influenced by it. In the first case the explosion expansion is back up through the core region, while in the second case it is up the vessel downcomer. We intend to determine the effect of the mass of coolant involved in the explosion on the overall conversion ratio.

1.4.1.9 Modeling of Film Collapse and Fuel Fragmentation (M. L. Corradini, B. J. Kim, UW)

Based on our discussions with Sandia personnel on this topic, we have developed a complete model for the fine fragmentation of the fuel during the single droplet explosions as performed by Nelson. This model seems consistent with the qualitative observations of Nelson's tests and is considered to be the logical model for fine fragmentation for the large-scale explosion tests. We are now performing the final calculations using this model to investigate the effect of the change in the ambient pressure before we transmit it to the Sandia researchers.

1.4.2 Presentations, Visits, and Meetings

On April 22, B. Marshall and M. Berman gave a review of the FCI program and a tour of the FITS facility to J. Haeggblom and L. Hammar of Sweden. Topics of discussion included technical aspects of steam explosions, the current status of the program, and the proposed and expected future research plans.

1.4.3 Documentation

On May 21, M. Berman sent a memorandum to J. Telford of the USNRC reviewing the history and current status of the FCI modeling efforts. Each of the code's capabilities and limitations were discussed in addition to the expected contribution of each to the NRC's needs.

A memorandum, written by O. Seebold, was issued to the steam explosion staff reviewing the Hicks-Menzies computer code that has been implemented. The memorandum included four sections: (1) a short discussion of the Hicks-Menzies thermodynamic process, (2) an example of the usefulness of the code as an experimental evaluation tool, (3) a set of runs for the SEALS facility, and (4) a series of runs for a PWR reactor.

1.4.4 Anticipated Activity

The high-explosive evaluation of the pressure gauges is expected to continue. The importance of the coupling media will be addressed by conducting a few experiments identical to the SHE experiments in air.

1.5 <u>Hydrogen Behavior</u>
(J. T. Hitchcock and 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.

We continued to work on the development of a diffusion flame model for the HECTR code. Two approaches have been programmed and are being evaluated. The first is the empirical approach, which utilizes empirical correlations to characterize the diffusion flame process and the flow induced by the flame. This model is being assessed using data from the continuous injection tests performed at NTS. Comparisons include the global temperature and pressure increases. The second model uses a lumped parameter approach, which involves defining one or more flame compartments and having HECTR calculate the gas properties within these compartments, such as the average pressure and temperature rises, assuming complete hydrogen combustion. We are currently debugging this model.

The assessment of HECTR against the large-scale hydrogen combustion experiment performed at the Nevada Test Site continued. We developed a multicompartment model of the spherical dewar for assessing the burn model and the flame propagation model in HECTR. The first test that we analyzed was the base-case premixed test, TEST POO. The resulting sequential burn gave better agreement with the experimental data than previously found for single compartment models.

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

FLAME 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. FLAME was built to be roughly a half scale model of the ice condenser upper plenum region. At this time, the first series of twenty tests has been run and the analysis of the tests is partly complete. In these tests the channel had no obstacles except small mixing fans at each end, and thermocouple instrumentation rakes. A second series of tests with obstacles is being prepared.

The first tests of the second series will be done with no top venting and simple obstructions, one-foot-wide panels symmetrically placed on each wall and extending from floor to ceiling. These obstructions have a greater blockage ratio than in the ice condenser upper plenum, but there are fewer turbulence generating surfaces. These simple obstacles were chosen so that they could be modeled with the CONCHAS SPRAY code at Sandia-Livermore and be compared to experimental data of other researchers.

In order to more accurately model the ice condenser upper plenum, a trip to the Catawba nuclear power plant of the Duke Power Company was arranged for April 25. Al Sudduth and Peter LeRoy of Duke were very helpful in our tour. We spent over an hour in the ice condenser region, using several rolls of film. We are reviewing photographs and measurements to make design decisions on the obstacle configuration in FLAME to best represent the actual ice condenser upon plenum. We have requested additional information from Duke Power on the material properties and moments of inertia of the intermediate and upper deck doors to better understand the degree of venting expected in various accidents.

1.5.1.3 Modeling of Flame Acceleration in Tubes With Obstacles
(K. D. Marx, 8363)

A numerical simulation of a class of flame acceleration experiments has been carried out. The experiments considered are those performed at McGill University involving premixed hydrogen-air flames in tubes with obstacles. The major emphasis of the numerical work is on a comparison of the Magnussen-Hjertager combustion model and a one-step chemistry model with Arrhenius kinetics. The flow is highly turbulent; a one-equation model is included to describe turbulence effects. Experimental data at one concentration are used to fix certain constants in the combustion models. The resulting models are then employed in computations at different hydrogen concentrations. The results are compared with the experimental parametric behavior.

The purpose of this investigation has been to obtain some insight into the way the models describe physical processes in flame acceleration. Models as simple as these cannot simulate such complex phenomena from first principles. They depend on appropriate adjustment of certain parameters. What is required of the models is that once the modeling constants are fixed, one should be able to reproduce parametric behavior of the experimentally measured quantities. Using the

Magnussen-Hjertager model, this has been accomplished reasonably well in the case of flame velocity, less so for pressure. In contrast, the one-step model has not proven to work as well.

Future work will include further improvements in the combustion models, introduction of a transport equation for the rate of dissipation of kinetic energy (k-epsilon model), and consideration of large-scale experiments such as FLAME. We also intend to seek ways to reduce the amount of computer time required for the calculations.

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

The purpose of the Heated Detonation Tube (HDT) program is to develop an experimental data base on H_2-air-steam detonability. These data can be used to develop models that 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 heated detonation tube, 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.

Efforts during this reporting period involved reviewing the uncertainty analysis of the HDT data, documentation,, and test preparation.

The measured variables in the HDT are the detonation cell size, the detonation velocity, and the initial thermodynamic state. Measurement of detonation cell size has always contained an element of observer bias because of the difficulty in interpreting the irregular structure of the cellular pattern on the smoked foil. We are developing digital image processing techniques to eliminate the need to interpret the cellular foils. Digital image processing requires that the smoked foil data be photographed and digitized. All the HDT test results have been photographed and prints are currently being made from the negatives. Within a month these results will be digitized.

Five test results have already been digitized and these results are being used to develop the image processing techniques. This work is being conducted with J. Shepherd, Division 1512 and has application to all detonation cell measurements, not just those in the HDT. Several image processing techniques have been applied; the results of processing are promising but none of the techniques have yielded completely satisfactory results yet.

The detonation velocity measurements in the HDT are established from linear regression of the time of arrival data from the pressure transducers. I. Hall of Division 7223 has finished reviewing the data and has established the uncertainty in velocity as well as the distance along the axis of the tube where the detonation has reached steady state. The uncertainty in the initial thermodynamic state has been established by the sum of square error technique from the uncertainty in the raw measurements of temperature and pressure.

The draft HDT topical report on the facility and testing is being revised. The work focused on tabulation of the test results and uncertainty estimates in a manner compatible with existing graphics routines to provide plots with both horizontal and vertical uncertainty bars on journal quality plots.

Preparation is continuing for the next series of tests. As reported in the NRC Mid-Year review, two areas require more test results. First, to support the local detonation work, more H_2-air-H_2O data is necessary for initial conditions that may occur near a break. These tests will be conducted with saturated steam conditions for a range of temperatures, air densities, and H_2 mole fractions. The test series will be a scoping study and will consist of 12 to 15 tests.

The second area of need is in additional data to help validate the kinetics code, which has shown promise in predicting detonation cell size. Testing in this area includes validating a prediction that the detonation cell size does not substantially decrease with respect to increased pressure for pressures above 1 atm. For most detonable mixtures, the detonation cell size is inversely proportional to the pressure raised to a power greater than one. To validate the effect of diluents, tests need to be run with a direct comparison of CO_2 and steam at the same initial thermodynamic state. Higher temperature tests are needed to validate the temperature effect up to 150°C. Fifteen to twenty tests should provide a sufficient range of data to compare the test results with the model. The preparation for these tests is being slowed somewhat by an internal safety review that has required an assessment of the pressure capability of the HDT.

1.5.2 Presentations, Meetings, and Documentation

The Mid-Year Review of the Hydrogen Behavior program was held in Washington, DC, on April 15-16. Presentations were given by M. Berman, J. T. Hitchcock, C. C. Wong, M. P. Sherman, K. D. Marx, and S. R. Tieszen.

Dr. Manfred Stock from Battelle-Frankfurt visited SNL on May 20 to discuss our work on gas explosions and DDT.

On April 22, M. Berman gave a presentation on the hydrogen program to Dr. Hans A. Haeggblom, a specialist in aerosol physics at Studsvik Engergiteknik, Nykoeping, Sweden and Dr. Lennart H. Hammar, Director of Research at the Swedish Nuclear Power Inspectorate, Stockholm. A tour was held the following day at the FLAME facility, heated detonation tube, and other experimental facilities.

Three papers were accepted for presentation at the 23rd ASME/AIChE/ANS National Heat Transfer Conference, Denver, CO, August 6-9, 1985: M. Berman, "A Critical Review of Recent Large-Scale Experiments on Hydrogen-Air Detonations"; J. E. Shepherd, "Stagnation-Point Heat Transfer From Jet Flames"; and A. C. Ratzel and J. E. Shepherd, "Heat Transfer Resulting from Premixed Combustion."

K. D. Marx, J. H. S. Lee, and J. C. Cummings, "Modeling of Flame Acceleration in Tubes with Obstacles," was accepted for presentation at the 11th World Congress of the International Association for Mathematics and Computers in Simulation, Oslo, Norway, August 5-9, 1985.

Four papers were accepted for presentation at the 10th International Colloquium on Dynamics of Explosions and Reactive Systems, Berkeley, CA, August 4-9, 1985: J. E. Shepherd, "Chemical Kinetics and Hydrogen-Air-Diluent Detonations"; M. P. Sherman et al., "The Effect of Transverse Venting on Flame Acceleration and Transition to Detonation in a Large Channel"; S. R. Tieszen et al., "Detonation Cell Size Measurements in H_2-Air-H_2O Mixtures"; and W. B. Benedick et al., "Critical Charge for the Direct Initiation of Detonation in Fuel-Air Mixtures."

M. P. Sherman et al., The FLAME Facility--Test Series 1: The Effect of Transverse Venting on Flame Acceleration and Transition to Detonation, SAND85-1264, Draft Topical Report.

S. R. Tieszen et al., Detonability of H_2-Air-Diluent Mixtures, SAND85-1263, Draft Topical Report.

1.5.3 Anticipated Activity

We will continue the development and assessment of the diffusion flame model in HECTR. Test Series 2 in FLAME with simple obstacles will be initiated. Detonation research will focus on refining the digital image processing technique to reduce the uncertainty in detonation cell size data.

1.6 Hydrogen Mitigative and Preventive Schemes
(L. S. Nelson and 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 only with lean mixtures of hydrogen in air (6-8 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) the 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) the effects of oxidic and metallic aerosols on the combustion of lean mixtures, (5) the 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 Presence of Water Sprays and Gas Flows
(L. S. Nelson and 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 Reactor Containment Buildings
(K. D. Marx, 8363)

The analytical portion of this work has been completed. A draft report entitled "Air Currents Driven by Sprays in Reactor Containment Buildings" has been revised per NRC comments, and is ready for SNL management review.

Several new calculations requested by the NRC are being performed and will be included in the final version of the report.

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

We further tested the rotating disc drop generators in the 5.6 m^3 FITS chamber. The parameters studied were (1) drop diameter distribution, (2) downward water fluxes, (3) suspended water density, and (4) uniformity of dispersion throughout the chamber as a function of generator operating conditions. We developed an infrared backscattering technique for these measurements; droplet measurements made this way are being analyzed.

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 will be minimized with new gas handling hardware now being installed. Moreover, real-time on-site mass spectrometric gas analyses will be available for the new series of experiments to further reduce these uncertainties.

1.6.1.4 The Effects of Aerosols on Hydrogen Combustion
(L. S. Nelson and G. D. Valdez, 6427)

We are analyzing all data from the VGES experiments with aerosols present during lean hydrogen-air burns. The aerosols are both oxidic and metallic simulants of high-temperature condensates and mists produced during nuclear reactor core degradation and core melting accidents. For simplicity, the simulants used were metallic iron and ferric oxide, Fe_2O_3, aerosols. While the oxide produced little effect on the hydrogen-air burns, the metallic aerosols increased peak pressures several fold, attributed to partial combustion of the iron.

A series of smaller scale experiments with the same iron metal was completed in the 0.2 m^3 chamber at McGill University. The effects of both hydrogen and aerosol concentrations were studied. Although the lean ignition limit of hydrogen in air (≈4 percent) did not change with the aerosols present, both impulses (dP/dt) and peak pressures were increased substantially at all hydrogen concentrations studied (up to 6.5 percent). Peak pressures were not as high as in a similar situation in the VGES experiments, suggesting a possible effect of scale on the burns.

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

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.

Chemical analyses continued to determine (a) the final state of the Cs, and (b) the time behavior of the molecular I_2 during and after the burns.

Theoretical modeling of the consequences of results such as these in an accident situation were continued with the Containment Modeling Division (6449).

1.6.1.6 Nonpowered Hydrogen Igniters
(L. R. Thorne and J. V. Volpani, 8353)

A platinum igniter has been prepared for evaluation in several static dry hydrogen-air mixtures in a 5 m^3 combustion chamber. We are preparing the chamber and associated hardware for these tests.

1.6.2 Presentations, Meetings, and Documentation

The entire Hydrogen Mitigative and Preventive Schemes program was reviewed for the USNRC in Silver Spring, MD, April 16, 1985.

A meeting of the LACE Technical Advisory Committee was attended on April 20 and 21, 1985. A presentation on the effects of hydrogen burns on CsI-containing aerosols was made.

A meeting was held at Sandia National Laboratories with Professor Ron Lee of the University of Kansas, Manhattan and Lawrence Livermore National Laboratories on May 16, 1985. The topic of discussion was hybrid dust explosions as they relate to laser isotope separation, grain elevator explosions, and metal dust explosions.

Draft reports on the effects of water sprays on igniters, air currents driven by water sprays, and nonpowered igniters have been prepared for NRC review.

1.6.3 Anticipated Activity

We will continue 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 experiments performed between December 1984 and March 1985. More burns are planned at 6.5 percent hydrogen in air with and without sprays.

We will do more diagnostics on the drop generation and their densities in the FITS chamber with optical techniques, emphasizing particularly radial uniformity.

Several more VGES experiments will be performed with metallic iron aerosols in lean hydrogen-air mixtures. The purpose of these experiments is to resolve certain differences between the smaller scale experiments performed at McGill University and those performed in the VGES chamber under apparently identical conditions.

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

The purpose of the Fission-Product Chemistry and Transport 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

2.1.1.1 Effects of Radiation on Iodine Chemistry

An important development in the understanding of the behavior of fission products during severe reactor accidents has occurred during the report period. It has been long-recognized that the chemistry of fission products released from reactor fuel during an accident is affected by temperature, pressure, and the ambient atmosphere composition. This variability in fission-product chemistry and its effects on radionuclide transport processes can be studied in laboratory experiments such as the experiments done in the Fission-Product Reaction Facility as part of this program. There is, however, a unique feature of reactor accident environments that can affect the chemistry of radionuclides that is not easily simulated in the laboratory. Severe reactor accidents involve intense radiation fields. It has been suspected by some that this radiation can affect the chemistry of fission products. Definitive evidence of the effects of radiation on chemistry has now been obtained.

The previous monthly report for the High-Temperature Fission-Product Chemistry program described a series of experiments investigating CsI chemistry in an intense gamma radiation field. Results of the experiments have now been partially analyzed and indicate that CsI chemistry was radically altered by the radiation field. The radiation induced the CsI to ionize and dissociate. The ionized cesium reacted with structural steels in the experimental apparatus. The freed iodine passed through the reaction system and was collected when steam condensed upon emerging from the reaction zone. Analyses of the cesium and iodine ratios in the condensate suggest more than 90 percent of the cesium iodide injected into the reaction system was dissociated. Even more complete alteration occurred when silver iodide rather than CsI was injected into the reaction system.

In the absence of an irradiation field CsI will deposit on, but not react extensively with, oxidized stainless steel. Microprobe examination of stainless steel coupons exposed to CsI and steam in a gamma irradiation environment showed that cesium had penetrated deeply into the oxide coating on the stainless steel and may have reacted with constituents of the oxide. The behavior of cesium from CsI in the tests was similar to the behavior of CsOH observed in previous tests without a radiation field present.

The radiation-induced alteration of CsI chemistry proved remarkably efficient. Such efficiency cannot be explained if direct interaction of a photon with CsI is needed to cause an ionization:

$$CsI + h\nu \rightarrow CsI^+ + e^- \rightarrow Cs^+ + I + e^- \quad .$$

It is suspected that steam ionization, which would be relatively efficient at the high steam densities of the tests, leads to CsI ionization:

$$H_2O + h\nu \rightarrow H_2O^+ + e^-$$

$$H_2O^+ + CsI \rightarrow CsI^+ + H_2O \rightarrow Cs^+ + I + H_2O \quad .$$

This ion transfer may explain the rather complete CsI dissociation observed in the tests.

The radiation-induced dissociation of CsI is significant and was not expected. The availability of a trap for cesium ions (the oxide coating on stainless steel) and the presence of steam, both of which are available in reactor accidents, may be important features of the process.

The behavior of iodine following ionization is of interest. In the tests the freed iodine passed through the system, apparently without further reaction. Some have suggested iodine may have an opportunity to react in the more complex environments of a reactor accident. The behavior observed in the tests with AgI is then of interest. AgI has a much higher ionization energy than CsI, yet AgI was observed to dissociate. This suggests that iodine reactions following CsI dissociation may not assure iodine escapes from the reactor coolant system as an iodide.

Previous tests in this program have shown that in the absence of a radiation field CsOH will react with SiO_2 in the oxide coating of stainless steel to form $Cs_2Si_4O_9$. It has been suggested that higher cesium silicates, too, may form at higher CsOH concentrations. Definitive evidence that cesium freed by ionization engages in such reactions was not obtained in the irradiated tests. The cesium was found in the layer of the duplex oxide on steel that contains SiO_2. A close correlation of the spatial locations of SiO_2 inclusions and the cesium could not be ascertained from microprobe analyses of the oxide coatings.

2.1.1.2 Revaporization of Deposited Cesium

It has been established largely as a result of research in the High-Temperature Fission-Product Chemistry program that cesium released from fuel during core degradation can deposit and react with structures found in the reactor coolant system. Interest has arisen about the possibility that such deposited cesium will revaporize as the structures in the reactor coolant system heat in the course of a severe accident. Analyses of this revaporization indicate the process could lead to cesium being injected into the reactor containment at a time when there is little to mitigate the subsequent release from the reactor containment building.

The previous analyses have been based on the deposited cesium not transforming to a less volatile material, such as $Cs_2Si_4O_9$. An attempt has been made to develop a model of the revaporization process that includes the effects of chemical transformation. The analysis involves development of thermodynamic models of the cesium vapor pressures for the chemical species produced by deposited CsOH. It also involves analysis of the revaporization kinetics. Processes that could retard the revaporization of deposited CsOH include mass transfer to the surface and mass transfer away from the surface.

Preliminary analyses with the model indicate that the rate of cesium revaporization can be reduced by six orders of magnitude as a result of chemical transformations of the deposited material. While this result does not preclude the possibility that there will be extensive cesium revaporization, it does suggest that revaporization will take a longer time and may require that structures reach higher temperatures than had been supposed previously.

2.1.2 Meetings and Documentation

Two investigators from the High-Temperature Fission-Product Chemistry program participated in the Severe Fuel Damage partners meeting held in Idaho Falls, ID, April 16-20, 1985. At this meeting these investigators described to the partners preliminary results from the tests of CsI chemistry in a radiation field, analyses of cadmium-silver-indium alloy vaporization, and analyses of cesium revaporization that include chemical transformation of deposited cesium.

Two topical reports were submitted for NRC review during the report period:

1. R. A. Sallach, R. M. Elrick, S. C. Douglas, and A. L. Ouellette, Reaction Between Some Cesium-Iodine Compounds and the Reactor Materials 304 Stainless Steel, Inconel 600, and Silver, Volume II Cesium Iodide Reactions, SAND83-0395, NUREG/CR-3197 (2 of 3).

2. R. M. Elrick, Effect of Ionizing Radiation on the Chemistry of the CsI-Steam-Hydrogen-304 Stainless Steel System --A Preliminary Report.

2.1.3 Anticipated Activity

A topical report on revaporization thermodynamics and kinetics will be completed.

A definition of a test of SnTe(g) deposition will be formulated.

Several tests of pressure drops and temperatures in the Fission-Product Reaction Facility will be run in an effort to improve the material balances for experiments run in the facility.

2.2 ACRR Source Term Tests

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

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. This program is related to out-of-pile programs at ORNL and BCL.

2.2.1 Current Progress and Technical Highlights

During this period, efforts continued in all major areas of source term experiment planning. With the first operational version of the VICTORIA fission-product release module of the MELPROG code nearing completion, a reevaluation of the ST program test matrix has been initiated. The testing of ST filter samplers continued with additional tests of Cs species on SiO_2 and I on metallic substrates. The ST experiment package hardware final layout drawings have been initiated and design work for major hot cell modifications is now underway. Tests on the sensitivity of Ion Chromatography for Ba and Sr detection in the presence of expected contaminants have also been performed.

2.2.1.1 MELPROG/VICTORIA Applications to ST Tests

Sensitivity analyses will be performed with MELPROG/VICTORIA to refine the definition of experimental data needs and the ST Test Matrix. Eventually, the analysis of the source term experiments will serve as verification tests for fission-product release models in MELPROG. The VICTORIA module of this code is designed to deal with the fission-product release and chemistry aspects of the tests. This module handles release, equilibrium chemistry, and aerosol physics/mass transport for the fission products. The aerosol physics/mass transport packages are in place and tested. The fission-product release model is in place and data base assembly for testing is being carried out at this time. The equilibrium chemistry model is being tested for a variety of scenarios.

A new data base is currently being constructed to expand its use in terms of number of elements and chemical species and temperature ranges. This model is now being used to estimate the effects of H_2/H_2O ratio on the release of the semivolatile fission products, in particular, barium and strontium. Their release should be very dependent on the oxygen potential of the system.

The complete VICTORIA module will very soon be in use for predicting fission-product behavior in the ST tests. These predictions will be used to both refine the proposed ST test matrix in order to better address the data needs for source term estimates as well as provide code validation data.

2.2.1.2 ST Filter Sampler Tests

Additional tests were conducted this period for the sampler train that is being developed for the source term experiments. Several stages in the train have been defined to selectively capture and isolate fission-product vapors (tellurium on nickel, cesium species on SiO_2, and iodine and HI on silver) and capture particles with a high degree of efficiency in fiber filters designed from a filter model.

Initial tests on tellurium, cesium iodide, and iodine have been completed and reported on in the two previous bimonthlies. However, two additional demonstration tests were completed this period to measure the reaction rates of cesium hydroxide with SiO_2 and of iodine vapor with copper and nickel.

The reaction rate of CsOH with SiO_2 was large enough to react essentially all of the cesium under the following test conditions. At a system temperature of 1170 K the hydroxide vapor with steam and hydrogen flowed at ~10 cm/s through an array of parallel SiO_2 plates placed 0.125 cm apart; the plates were 7.5-cm long in the flow direction. This is a considerably greater reaction rate than that observed between CsI and SiO_2. In order to react an appreciable amount of cesium hydroxide at much faster flow velocities, the surface area of the SiO_2 could be increased if it were in the form of quartz wool or quartz frits. These forms will be tried as ways to enhance cesium collection by reaction.

The purpose of the iodine experiment was to find materials other than silver that might react rapidly with iodine. There is some concern that the photosensitivity of AgI may cause this reaction product to be unstable in the γ-field of the ACRR. A sampler was made of slats of Ag, Cu, and Ni in the geometry described above. The iodine vapor with steam and hydrogen at 770 K flowed between the slats. Assuming the discoloration on the leading edges of the slats is due to the reacting iodine, then the iodine reaction with copper is at least as fast as with silver, but with nickel is somewhat slower. These slats are being examined by microprobe for reaction products.

2.2.1.3 ST Experiment Capsule Design

The final layout drawings of the ST experiment capsule were begun. This process, which involves detailed integration of all the components into the assembly, will define final piece part drawings. This final design process will take approximately six weeks to complete.

The hardware feasibility questions associated with low pressure, hydrogen atmosphere experiments were previously addressed and resolved. The remaining question involves use of the bellows sealed compressor at high line pressures (up to 30 atm). A prototype system to permit use of the compressor at high pressure has been designed and is being fabricated. This system involves surrounding the compressor with an environment pressurized to and maintained at line pressure to minimize the pressure difference across the seal bellows.

2.2.1.4 Hot Cell Modifications

Several modifications of the Area V Hot Cell are required for the assembly and disassembly of the ST experiments. The most significant of these is the completion of a vertical entrance into the cell and development of a transfer cask to accept the ST package from the hot cell for transport to the ACRR. The new penetration and cask provide the required handling height for assembly of the package as well as providing for insertion and removal of the long experiment assembly from the hot cell. Designs of the facility modifications and the transfer cask have been initiated.

2.2.1.5 Multicomponent Filter Tests
(R. Elrick, 6422; Harlan Stockman, 1543)

To aid design of filters and analytical techniques (filter processing and leaching), filter tests in an H_2/N_2 (10 w/o H_2) atmosphere using a variety of fission-product simulants are being performed. These tests are quick and are oriented toward identification of "problem" interactions which would lessen filter efficiencies; they are run in parallel with more carefully controlled tests to quantify efficiency parameters.

Initial tests involved SnTe vapor and aerosol intercepting a Ni filter at :700°C. Te was effectively removed by the Ni filter, despite initial combination as SnTe. Subsequent tests involved Sn_3Te, CsI, BaO, La_2O_3, and CeO_2 heated at 1400°C in an H_2/N_2 stream containing Cs vaporized at 650°C, and were primarily intended to determine if condensation of Cs on the filters will lessen efficiencies. Tests at higher temperatures are scheduled for late June.

2.2.2 Meetings and Presentations

A review of the current status of the ACRR Separate Effects Source Term Experiments was presented at the Severe Fuel Damage and Source Term Research Program Review Meeting in Idaho Falls, ID, on April 18, 1985.

Members of the Sandia Source Term Experiment team attended the STEP Technical Advisory Committee Meeting at ANL on May 7-8, 1985.

2.2.3 Anticipated Activity

The detailed design of the ST experiment package should be completed in July. Procurement of commercial components and component fabrication for the out-of-pile system test and the ST-1 and ST-2 experiments will be initiated. Design of the hot cell facility modifications will continue. Development of hot cell fixturing and procedures for assembly and disassembly of the ST experiments will begin.

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. Sandia is 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 Damaged 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.

3.1 ACRR Damaged Fuel Relocation and Quench
(A. C. Marshall and 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 primary activity in the DFR program was the assembly of the test section and internal hardware for the DF-3 Ag-In-Cd Control Rod test. The DF-2 package was disassembled and prepared for posttest examination during this period. Analysis activities are currently underway for the DF-4 experiment, which will examine the effect of B_4C control materials on damage progression in BWR-type geometry.

3.1.1.1 DF-3 Assembly

The assembly of the DF-3 package has progressed during this period with the completion of the test section. The fuel/control rod bundle, ThO_2 retainer cup, W/Re and Pt/Rh thermocouples, ZrO_2 shrouds and ZrO_2 fiber insulation have been installed. The thermal limit monitors, H_2 getters and thermocouple trees, test section jacket heaters, and related components have been installed, and proof- and leak-tested. The design has been completed for the DF-3 time-dependent aerosol monitor and an initial version is being fabricated. The test conditions currently planned for DF-3 are compared to DF-1 and DF-2 below:

Parameter	Approximate Values DF-1	DF-2	DF-3
Fuel Rods	9	9	8
Control Rods (Ag-In-Cd)	0	0	1
Steam Pressure	40 psi	220 psi	~100 psi
Fuel Rod Internal Pressure (cold)	<1 psi	15 psi	440 psi
Steam Mass Flow Rate (g/s/rod)	0.05	0.025	0.10
Nuclear Heatup Rate at 1200°C	4°C/s	1°C/s	2°C/s
Maximum Reactor Power	1.5 MW	1.5 MW	1.5 MW

The DF-3 test will examine the effects of:

1. Control rod aerosol and material relocation behavior,
2. High fuel rod internal pressure, and
3. High steam flow rate

on damage progression.

3.1.1.2 DF-2 Posttest Examination

The DF-2 capsule was disassembled and potted during this period in preparation for metallographic examination. Six areas have been designated for detailed examination. The area of partial blockage formation, located at the grid spacer location, the area of partial fuel dissolution just above the grid spacer, and the region of significant interstitial material at mid bundle are the primary areas of interest in DF-2. The PIE will focus on the extent and composition of blockages, the amount of fuel liquefaction, and the extent of in-place Zr oxidation. Results from the first samples will be available for the next report.

3.1.1.3 DF-4 Plans

The DF-4 experiment is currently planned as a B_4C control material test using a section of a BWR-like control blade in a 9 to 12 rod bundle.

The B_4C is of concern for two reasons: (1) it has a strong exothermic reaction with steam, which can add to the rate of core degradation; and (2) its reaction products with steam may alter the chemistry of iodine, leading to iodine release to containment.

Current tentative test design is to use 12 BWR-size fuel rods, separated into groups of 6. The two groups are divided by Zircaloy assembly walls, and a section of a B_4C control blade. Preliminary calculations of experiment performance were made using a simple computer model. A temperature difference of up to 400 K was found to develop between the two fuel groups during the temperature excursion. The B_4C temperature lags the fuel temperature by up to 400 K at first, but the temperature difference drops at higher temperature as radiation heat transfer becomes more effective. Further studies will be performed to investigate the prototypicality of this behavior.

3.1.2 Anticipated Activity

Assembly of DF-3 should be completed. DF-3 is currently scheduled for moving to ACRR in late July with the experiment performance in early August.

3.2 <u>ACRR LWR Degraded Core Coolability</u>
(T. R. Schmidt, 6421; K. Boldt, 6222; A. W. Reed, 6425)

The LWR Degraded Core Coolability (DCC) Program investigated 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, and bottom coolant feed. 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-1 and DCC-2 analysis, and the continued assembly of the DCC-3 experiment.

3.2.1 Current Progress and Technical Highlights

Work is continuing on the assembly and testing of the DCC-3 hardware. The second cooling coil for heat removal from the experiment was received and tested. Fifteen kW was removed at bath temperatures less than 100°C, which should be adequate for the experiment. The thermocouples have been installed. Modifications were required at the inlet to the crucible because of the expected low dryout powers with no flow. Difficulties were encountered with the welds on the components for the bottom flow system. Several components were redesigned and rewelded.

The motor and pump are prepared for welding. The shield plug has been modified for water cooling in place of the helium. The fuel has been received and is being sized and measured for porosity. The experiment plan is nearing completion.

The DCC-1/DCC-2 analysis report has been completed and is currently in review. The major conclusions are:

1. The DCC-1 debris would be uncoolable in a reactor accident as predicted by many of the dryout correlations.

2. Measurements of capillary pressure suggest that the lower pressure dependence in the DCC-1 dryout data may reflect a difference between narrow and broad particle size distributions. Correlations for relative permeability used to date seem to show the correct trend, but the results are inconclusive. Direct measurements of relative permeability of dryout measurements expressly addressing the issue are probably the only way of resolving the question. The pressure dependence does not have a strong impact on the issue of coolability, but does impact the ability of the models to accurately predict heat transfer over a full range of accident parameters.

3. The DCC-2 configuration would be coolable in a reactor accident and was well predicted in magnitude and pressure dependence by several of the models.

4. Based upon dryout data using prototypic materials, the models developed for coolability of LMFBR debris adequately describe the coolability of uniform LWR debris with adiabatic bottom boundary conditions.

Out-of-pile experiments on dryout of an inductively-heated stratified bed are being conducted at the University of New Mexico. Preliminary measurements indicate that dryout occurs with the initiation of boiling, and that the waiting time is several hours. While radial heat losses have been functionally eliminated in the measurement, subcooling of the overlying pool may have affected the results. The measurements are being checked using a saturated overlying pool. Procedures to reduce waiting time are being investigated.

3.2.2 Meetings

The status of the analysis for DCC-1 and DCC-2 and the DCC-3 experiment was given at the SFD meeting in April at Idaho Falls, ID.

3.2.3 Anticipated Activity

The experiment plan for DCC-3 will be finished and reviewed by the safety committees. The experiment will begin in July, assuming there are no major problems. More out-of-pile dry-out measurements will be conducted at the University of New Mexico to provide operational experience.

4. MELT PROGRESSION CODE DEVELOPMENT (MELPROG)

(J. E. Kelly, 6425; M. F. Young, 6425; P. J. Maudlin, 6425; J. L. Tomkins, 6425; P. K. Mast 6425; K. A. Williams, 6425; W. D. Sundberg, 6425; 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 (Mod0) for limited release. This version is essentially complete but some enhancements are being made in the graphics, input, and output sections of the code. Also as the test calculations are run, any errors and inconsistencies found are being eliminated.

A new plotting program was written to display calculated results as a function of time. This program, as well as the 3-D and spatial plotting routines, are all postprocessing routines. As the code runs, a special graphics file is created. After the run is completed, the graphics programs will use the graphics file to display the data.

The input data requirements are also being carefully examined in order to simplify and clarify the input process. Extraneous input variables are being eliminated and those which are ambiguous are being clarified.

Development on the 2-D version of MELPROG (Mod1) also continued. The main development activities on this version have been in the new core structures module (CORE) and the fission-product release and transport module (VICTORIA).

Development work continued on the new CORE module. The data storage architecture was defined. Coding was done for pointer routines for temporary and permanent pointers, defining common blocks, the routine (CORE IN) to read the core input data and allocate permanent storage, and the CORE module driver routine. The heat transfer solution for the new CORE module is being adapted from the one in the structure module. These routines are currently being modified to handle core structures. A new property routine has been added to handle structure and core structure properties in a more efficient and consistent manner. Work on adding a rezoning capability to the heat transfer solution has begun, which will allow for internal melt relocation, external candling, and crust formation.

The species release model using a variable fuel geometry was formulated and implemented into the VICTORIA fission-product behavior module. This model solves the species diffusion equation over multiple material regions in 2-D cylindrical/cartesian coordinates or 1-D cylindrical/spherical/cartesian coordinates. The implementation of this model is complete and it is currently being tested with some ST scoping calculations.

4.1.2 Code Testing

Testing of the code continued by using a simulated S1D accident. This accident allows testing of all modules and, since it is fairly rapid, it can be run at low cost. The current model includes the upper and lower plenums and the core region. The dumping of the accumulator water will also be simulated. Boundary conditions for this problem have been obtained from a RELAP-5 calculation. This test case is being used to shake-down the 1-D version of the code.

The calculation has been run to where 80 percent of the fuel rods failed. The code successfully calculated the fuel rod heat-up and rapid oxidation of the Zr cladding. Debris bed regions formed due to the melting relocation and freezing of the fuel rod material. Also, the quenching of the vessel by the accumulators was initiated.

During this exercise, several code errors were corrected and phenomenological modeling improvements were made. The most notable model improvements were to the steam-water interfacial mass and heat transfer processes. These improvements were needed to handle the condensation of steam by subcooled water when the accumulators dump occurred. The code now calculates interfacial heat transfer coefficients for both the gas and liquid fields based upon flow regime dependent correlations. This new procedure is similar to that used in TRAC.

4.2 Meetings and Presentations

A workshop for selected users was held at Sandia April 23-25, 1985. At this workshop the various modules were discussed and the functional details of these modules were given. Sample problems were described and the participants, together with the staff, ran these calculations with the code.

A review of MELPROG and VICTORIA modeling as well as the VICTORIA calculated DF-1 aerosol results were presented at the SFD partners meeting in Idaho Falls, ID, on April 19, 1985.

4.3 Anticipated Activity

The Mod0 version of the code will be released to selected users in early July. Work on the Mod1 version will also continue.

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, 6423; P. S. Pickard, 6423; G. Schumaker, 6423)

The Sandia Fuel Dynamics Program provides needed 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 continues 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 USNRC is sponsoring the Sandia Transient Axial Relocation (STAR) experiments in the ACRR test facility. Kernforschungszentrum Karlsruhe (KfK), FRG is cosponsor of this program.

5.1.1 Current Progress and Technical Highlights

The STAR program's main focus during this report period was in two areas: (1) performing the two experiments STAR-5 and STAR-6, and (2) presenting the results of the first four STAR experiments at the International Fast Reactor Safety Conference in Knoxville, TN.

The STAR-5 and -6 experiments were the first to use preirradiated multipin bundles. Each experiment used four preirradiated fuel pins and one fresh fuel pin. The experiments were designed to produce fuel disruption and fuel and clad relocation behavior under conditions that reproduce loss of flow (LOF) accidents in low and high sodium void worth LMFBRs. These were the same general goals as the two-pin STAR-3 and -4 experiments, and thus the power transients are nearly identical. The main difference, however, was that only a single preirradiated pin was used in STAR-3 and -4, compared to multipins in STAR-5 and -6.

The STAR-5 experiment used a power transient that caused clad melting at nominal power (P_0) about 1.5 seconds before a mild power increase that raised the power level 4 to 6 times P_0. This power transient simulated the power rise due to sodium boiling in a reactor having a mild sodium void coefficient. During the increased power portion of the reactor transient fuel melting and frothing occurred. Shortly after the onset of clad melting, some mild upwards clad motion was observed, but the major portion of the cladding (at least 75 percent) flooded. This is consistent with the earlier STAR-2 multipin experiment where significant early flooding was observed. Some "crumbling" and sweep out of the fuel coincided with the clad motion in the pre-irradiated fuel pins; however, the amount of fuel involved was only a few percent of the total fuel inventory. During the clad motion phase of the power transient the denuded fuel pins remained relatively intact, but had strong tendencies to bow and agglomerate, thus effectively forming a single fat fuel pin. During the higher power level portion of the transient, some fuel frothing due to fission gas was observed. Some of this fuel (a piece several centimeters long) moved upwards, but did not leave the heated section of the pin. Even though significant fuel melting was observed, no slumping occurred, probably due to the ability of the fission gas bubbles to prevent the foam of molten fuel to collapse.

In the STAR-6 experiment the power transient rose to 15 x P_0 on a 100-ms period, until clad melting and fuel disruption occurred. These heating conditions simulated the power rise due to sodium boiling in a reactor having a moderate-to-high sodium void coefficient. The initial fuel disruption was the same solid state fuel cracking as seen in the STAR-4 experiment and in the Fuel Disruption experiment FD4.3. The axial rip was about 1-cm long and did not propagate. Instead, other rips developed in other pins and the development of these rips progressed axially up the fuel bundle. Solid state fuel ejection through each of the rips was observed. Fuel was ejected through each of these rips at 10 to 20 m/s and the amount of fuel ejection increased with the later rips. Late in the power transient when fuel melting occurred the entire coolant channel was blocked, probably due to volume expansion caused by fission gas induced fuel foaming. Shortly after the formation of this blockage, fuel removal by the sweep out process stopped completely. No significant fuel vapor pressures were generated to further disperse the fuel.

These observations point to two general conclusions about pin failure under these type heating conditions. First, because of the size of the rip and the lack of propagation, the rip is triggered by fission gas effects in the fuel which caused the solid-state fuel ejection to be limited in its extent. Second, the early ejection of this fuel into the coolant channels provides a mechanism for some early fuel ejection prior to the main fuel dispersion due to fuel melting and vaporization.

5.1.2 Presentations and Meetings

The results of the first four STAR experiments were presented at the International Fast Reactor Safety Conference in Knoxville, TN. The presentation focused on visually illustrating the fuel disruption processes and mechanisms that are active over a wide range of heating conditions, and how these disruption processes might lead to fuel dispersal. These fuel disruption ideas were illustrated by a poster summarizing the earlier FD program, and the fuel dispersal mechanisms were illustrated by video movies of the first four STAR experiments.

5.1.3 Anticipated Activity

During the next two months, the STAR-5 and -6 experiments will be analyzed with the SANDPIN/CMOT code. The results of these calculations will be included in quick-look reports. In addition, an effort will be made to document the first four experiments in a final report.

5.2 Transition Phase

(R. O. Gauntt, 6423; P. S. Pickard, 6423; 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 implication of this pool of fuel-steel in the core region of the fuel blockages do lead to this state.

The TRAN program addresses the question of fuel-inventory reduction by penetration into 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

Postexperiment analysis on the B-4 multipin experiment used the gamma scan and radiographic information obtained after performing the experiment. The analyses include both PLUGM and SIMMER-II calculations. Qualitatively, the multipin experiment, B-4, has been found to be quite consistent with the conclusions drawn from the earlier simple geometry experiments, with fuel penetration characteristic of conduction limited crust growth mechanisms. These calculations will be refined as more data become available from the post-irridiation examination (PIE) being performed in the Hot Cell Facility. The PIE is nearly complete at this time, with only polishing of the cross-sectional cuts remaining.

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 that approximates the intercan wall gaps in the lower blanket structure of an LMFBR, is currently under construction. The major assembly steps will be carried out in June with a completion date set for July 1, 1985. Precalculations using SIMMER and PLUGM are currently underway. Without major problems, the GAP-2 experiment will be carried out in July.

5.2.2 Documentation

A review of the TRAN program including the immediate results and analyses pertaining to the recent B-4 experiment were presented at the ANS topical meeting in Knoxville on fast reactor safety. A second paper is under preparation for the Fast Reactor Safety meeting in Guernsey, U.K., set for May, 1986. A topical report for the B-4 experiment is also under preparation.

5.2.3 Anticipated Activity

The PIE for the B-4 experiment will be completed and the GAP experiment, G-2, will be performed.

6. POSTACCIDENT HEAT REMOVAL

6.1 Debris Bed Coolability

(C. A. Ottinger, 6421; T. R. Schmidt, 6421; H. Meister, 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 USNRC, EURATOM (JRC, Ispra), and the PNC (Japan).

6.1.1 Current Progress and Technical Highlights

Activities this period involved deposition of the D-13 experiment and reduction of the data. Data tapes were provided to the staff of the cosponsors for their analysis. The draft experiment report was begun. The paper for the Knoxville Fast Reactor Safety meeting was completed and presented. Also, an abstract summarizing the program was submitted for the Fast Reactor Safety Conference at Guernsey, U.K.

6.1.2 Anticipated Activity

The analysis of the D13 experiment 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 postirridiation examination of the DC1 and DC2 experiment continued. Much of the analysis is delayed, since the new SEM is not yet operating. M. Gambini, Joint Research Centre, Ispra, EURATOM, visited the lab and reviewed the DC1 and DC2 results.

Also, our earlier furnace tests with urania and steel were discussed. He presented results of their furnace tests showing very interesting metallic and intermetallic structures in the oxide debris. In the experiments with molten steel, the urania becomes slightly hypostoichiometric, leading to metallic formations on cooldown, which include uranium metal as a constituent. The change in stoichiometry is also considered to influence the meltability of the urania by the molten steel.

The SPARTA (Sandia PostAccident Rubble Transient Analysis) model, developed to perform detailed debris calculations to interpret the in-pile Dry Capsule (DC) experiments, has now progressed to be a "stand-alone" model that analyzes the important phenomena related to debris heating and melting. The model has been incorporated into MELPROG to form the basis of the debris analysis module in that code.

SPARTA predicts debris heat-up, melting, crust formation, crust remelt and all heat transfer processes within the debris bed. The model satisfactorily reproduces all the major features of the DC-1 molten configuration.

6.2.2 Documentation

The results of the Melt Progression experiments were presented to the SFD Review meeting in April at Idaho Falls, ID. A paper was prepared on SPARTA for presentation at the summer ANS meeting.

6.2.3 Anticipated Activity

The microscopic examinations should be completed.

7. DISCLAIMER NOTICE

This informal document contains information of a preliminary nature and was prepared primarily for interim use in reactor safety programs in the United States. Thus, it is subject to revision or correction, does not constitute a final report, and should not be cited as a reference in publications.

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