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Summary of the SRS Severe Accident Analysis Program, 1987-1992 (U)

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1 INTRODUCTION

The Severe Accident Analysis Program (SAAP) is a program of experimental and analytical studies aimed at characterizing severe accidents that might occur in the Savannah River Site Production Reactors. The goals of the Severe Accident Analysis Program are:

- To develop an understanding of severe accidents in SRS reactors that is adequate to support safety documentation for these reactors, including the Safety Analysis Report (SAR), the Probabilistic Risk Assessment (PRA), and other studies evaluating the safety of reactor operation;

- To provide tools and bases for the evaluation of existing or proposed safety related equipment in the SRS reactors;

- To provide bases for the development of accident management procedures for the SRS reactors;

- To develop and maintain on the site a sufficient body of knowledge, including documents, computer codes, and cognizant engineers and scientists, that can be used to authoritatively resolve questions or issues related to reactor accidents.

The Severe Accident Analysis Program was instituted in 1987 and has already produced a substantial amount of information, and specialized calculational tools. Products of the Severe Accident Analysis Program (listed in Section 9 of this report) have been used in the development of the Safety Analysis Report (SAR) and the Probabilistic Risk Assessment (PRA), and in the development of technical specifications for the SRS reactors. A staff of about seven people is currently involved directly in the program and in providing input on severe accidents to other SRS activities.

Much of the currently anticipated program of experimental and code development studies was completed during FY-1992. For FY-1993 and beyond a continuing program is anticipated, with reduced emphasis on experimental work and an increased emphasis on integrated analyses and applications of the program results to SRS reactor and non-reactor needs. Many of the Severe Accident Analysis Program analytical tools and experimental data bases are being utilized in the Level 2 Probabilistic Risk Assessment which will be issued in early 1993. Another major product of the Severe Accident Analysis Program will be a series of integrated studies of accident progression and accident phenomena, the first of which will be produced in FY-1993.

2 BACKGROUND

2.1 History and Approach

In 1986 a proposal was made that the Savannah River Laboratory (now the Savannah River Technology Center) undertake studies on severe reactor accidents.[1] The goal of the severe accident studies was to provide "a level of protection against the release of radioactivity from severe accidents equal or superior to the commercial nuclear industry". A special task force was formed to consider this proposal and in its report of March 1987 recommended a program of approximately 3.5 years and $12 million.[2] The program was intended to address the severe accident/degraded core issues for the Savannah River Site reactors, with the goal of making the SRS reactor operations comparable or superior to licensed power reactors with respect to severe accident issues. The resulting program, named the Severe Accident Analysis Program, was implemented later in 1987 with the formation of the Severe Accident Group of the Reactor Safety Research Section. Initial guidance for the Severe Accident Analysis Program was obtained from two consulting firms that considered the needs of the SRS reactors in detail.[3,4] The Severe Accident Group was reorganized in 1989 and renamed the Safety Analysis Group (now Safety Analysis and Engineering Services). The function of the group was broadened to include activities related to design basis transients, equipment qualification, functional analysis, and thermal hydraulics. One subgroup within the Safety Analysis Group was dedicated to severe accident issues. Accident management, which was originally envisioned as part of the Severe Accident Analysis Program, was initiated as a separate program.

Soon after the initiation of the Severe Accident Analysis Program in 1987, the Savannah River Site operations were evaluated by a committee for the National Academy of Sciences.[5] Among the recommendations of this group was the following:

> If there is a significant probability that the lives of these reactors will be extended beyond the next few years, the Department of Energy should commit to a significant program of severe accident model development and validation. Data from the program should be applied in a continuing reexamination of the risk of severe accidents and a review of the design and operation of the plants.

The NAS recommendation, together with other material in the text of the report, provided a strong additional incentive for pursuing the activities in the Severe Accident Analysis Program.

Early in the program, a panel of distinguished expert consultants* was assembled. The panel members provided guidance in formulating and executing the elements of the Severe Accident Analysis Program. The program elements were reviewed by the panel members during seven meetings held between 1987 and 1992. Reports have summarized material presented to each meeting of the panel.[6,7,8,9,10,11] Some recommendations from the panel regarding the program were contained in two additional reports.[12,13]

The Severe Accident Analysis Program has been in progress for slightly more than the four years that were originally envisioned. Limited staff and facilities in the Severe Accident Group required most of the program work to be done by subcontractors under SRS direction. Since neither the staffing nor the contracting ever reached the level originally intended, only about 70% of the originally anticipated funding has been spent and the program output has been correspondingly reduced.

Presently, it is envisioned that the Severe Accident Analysis Program will continue beyond FY-1992 with a decreasing level of effort over the next few years. A

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continuing effort in severe accident analysis was strongly recommended by the Severe Accident Advisory Committee inthe fall of 1991.[13]

> Risk from severe accidents is sometimes spoken of, and seemingly regarded, as a rather exotic concern which is not central to a reactor safety program. It is well for managers and decision makers to be reminded that risk from severe accidents, i.e., core melting, is the most important [and very nearly the only] risk there is from the operation of nuclear reactors. The purpose of all of reactor safety efforts is to prevent and mitigate the consequences of severe accidents. Moreover, experience in the commercial field has taught us that in spite of our best efforts at prevention, uncertainty in equipment performance and in human reactions can produce severe accidents. Mitigation, that is procedures and systems for making risks from severe accidents acceptable, is now recognized as an absolutely essential component of safe reactor operation. A better understanding of the nature of severe accidents, as provided by the SAAP, is thus a vital part of the overall safety program for the SRS production reactors.

The goals of the continuing effort will be to keep abreast of the advances in the pertinent technologies, to maintain the tools and knowledge necessary to support safe operation, and to provide a continuing input into SRS safety programs. At a minimum, the increasing level of knowledge concerning severe accidents in the nuclear industry must be matched by a corresponding sophistication for the production reactors, thus requiring further work on the Severe Accident Analysis Program.

The Severe Accident Analysis Program has included both experimental and analytical activities. Experimental work has concentrated on the peculiar features of the SRS reactors, with particular regard to the behavior of the aluminum-uranium fuel and the aluminum-lithium targets under accident conditions. Analytical work has concentrated on the modification of computer codes developed in the commercial nuclear industry, such as MELCOR and SCDAP/RELAP5, for application to the SRS reactor design. Specialized calculations and work on other related codes, such as CORCON for molten core-concrete interactions and VANESA for the resulting aerosol generation, have also been performed. Each of these programs will be discussed in greater detail in the body of this report.

Information and technology from the Severe Accident Analysis Program has been used to resolve severe accident questions about reactor operations and for the evaluation of proposed new safety equipment. Current applications of Severe Accident Analysis Program results and modeling methods include the Probabilistic Risk Assessment (PRA) for the SRS reactors, accident management, the Safety Analysis Report (SAR) for the SRS reactors, and project evaluations for such facilities as the new confinement system filter compartments. Severe Accident Analysis Program personnel have performed analyses for two major Design Basis Events (DBEs) in Chapter 15 of the K-Reactor SAR, the Loss of Control Rod Cooling (LCRC) accident and the Loss of Coolant Accident (LOCA-Gamma Heating Phase). The analyses addressed control rod cooling, reactivity, steam explosions, and hydrogen combustion.

2.2 Related Programs

Extensive studies of severe accident phenomena have been made for commercial light water reactors (LWRs) during the last two decades, with the support of the U.S. Nuclear Regulatory Commission (NRC), the Electric Power Research Institute (EPRI), and other organizations in the U.S. and other countries. LWR studies, however, are often not directly applicable to the SRS production reactors because of the differences in fuel, construction, and operation of the two reactor types. Commercial power reactors use a Zircaloy-clad uranium oxide fuel containing slightly enriched uranium. The light water coolant is pressurized to generate high temperature steam for driving generators to produce electricity. Commercial reactors have tight containment systems designed to retain major fission product releases during an accident. The SRS reactors are not significantly pressurized and operate below the boiling point (at one atmosphere) of their primary coolant, heavy water, since the purpose of the SRS reactors is to produce special nuclear materials such as tritium and plutonium. The SRS fuel is aluminum-uranium alloy, clad with aluminum, and it is typically enriched to a high concentration of the fissile uranium 235 isotope. The SRS reactor building incorporates a once-through ventilation system designed to contain fission product releases, and thus minimize the hazard to the general public, by filtration and sorption. The sum of these differences makes accident considerations for the SRS reactors quite different from their commercial counterparts. The Severe Accident Analysis Program is directed toward obtaining the information necessary to understand these differences, and to make possible accident analyses of the same quality that is available for the commercial LWRs.

Until recently, the Department of Energy was supporting a separate large research program concerning severe accidents in heavy water production reactors as part of the development program for the new heavy-water production reactor under the direction of the New Production Reactor (NPR) office of the Department of Energy. The NPR-HWR severe accident program is being coordinated by Sandia National Laboratories (SNL). The NPR severe accident program was generally designed so that it did not duplicate the SRS activities. (A study of potential overlap by Argonne National Laboratory and the Savannah River Technology Center concluded that areas of overlap were relatively minor.) Much of the NPR severe accident program was pertinent to the existing reactors, and information obtained from it will also be incorporated into the safety documentation for the SRS reactors. The NPR severe accident program provided an excellent series of literature reviews and compilations of material properties. A major component of this program was in-reactor fuel melting tests using the Annular Core Research Reactor (ACRR) at SNL. The NPR severe accident program was also developing a major severe accident computer code based on the SAS code previously developed for fast reactors.

3 EXPERIMENTAL ACTIVITIES OF THE SEVERE ACCIDENT ANALYSIS PROGRAM

In designing the experimental portion of the severe accident program, emphasis was placed on those features of the SRS reactors that differ from LWRs. The goal was to use the limited resources available to the Severe Accident Analysis Program to investigate the peculiar features of the SRS reactors, and to utilize the available safety studies for other reactors wherever possible. Major features of the SRS reactors requiring special attention in accident analyses include:

- Operation at ambient pressures and sub-boiling temperatures.
- Use of fuels consisting of highly enriched metallic uranium, alloyed with and clad by aluminum.
- Use of a once-through building ventilation system with aerosol filtration and iodine sorption of the effluent air.
- The presence within the core of aluminum-lithium alloy targets for breeding tritium.
- The presence of control rods containing aluminum-lithium alloy.

The experimental portion of the Severe Accident Analysis Program is divided into five principle sections, which are discussed individually below:

- Mobilization of radioactivity released from reactor fuel.
- Accident progression phenomena.
- Energetic phenomena associated with severe accidents.
- Response of the reactor confinement system to accident phenomena.
- Supporting studies, including materials studies.

Within each of these categories are several individual studies. A listing of both completed and currently active experimental studies is given in Table 1 (Section 7). The remaining portion of the Severe Accident Analysis Program, analytical and calculational studies, is described in Section 4 and listed in Table 2.

3.1 Mobilization of Radioactivity Released from Core Components.

Details of the fission product release rate were investigated by Woodley[14,15] in a series of studies at Hanford Engineering Development Laboratory (HEDL). The HEDL studies were sponsored by the Savannah River Technology Center (Table 1, Item E-1) and were the precursor to the Severe Accident Analysis Program. The results of Woodley were summarized by Whitkop[16,17] in two documents. Additional interpretation of Woodley's results was provided by Lorenz[18] of Oak Ridge National Laboratory (ORNL), who also considered some earlier ORNL studies. A very recent study by Taleyarkhan[19] of ORNL provides additional insights.

The studies referenced above lead to the following conclusions. Volatile fission products in SRS reactor fuels are released to the surrounding atmosphere when the fuel melts, in the vicinity of 650° C. The noble gas fission products, which are not contained by the reactor confinement system and were not subjects of these studies, are quantitatively released very quickly at this temperature; iodine is released more slowly, and other, less volatile fission products, such as cesium (Cs) and Tellurium (Te), are released more slowly still. Release rates increase with increasing temperature.

The HEDL and ORNL studies do not adequately address the question of the rate of evolution for the fission product isotopes from melts of appreciable depth. Program item E-2 (Table 1) was intended to answer this question. The evolution of simulated fission products from aluminum-uranium melts was to be studied as a function of temperature and geometry. The experiment utilized a set of thermal gradient tubes that could be sequentially valved into position, allowing samples to be collected as a function of time and temperature. The work was conducted under the direction of A. Sharon and L. Baker at the Argonne National Laboratory (ANL). Due to unexpected difficulties in construction of the test facility, contract funding was exhausted shortly after the experimental program began. As a result, only a small number of the planned experiments were run and limited data were obtained.[20]

3.2 Accident Progression Phenomena

Although studies of fuel melting have been made several times over the history of the Savannah River Site, it is still difficult to predict the behavior of molten fuel under accident conditions. Studies were undertaken to acquire additional information to assist in developing the best possible model of accident progression. Such modeling is essential for safety analyses in support of the SAR and the PRA as well as for informed response to accidents.

3.2.1 Fuel melt phenomena

Fuel melt phenomena have been partly characterized by a number of studies conducted at SRS and elsewhere including studies by Morin and Hyder,[21] in-pile transient tests in the SPERT-I reactor,[22] and studies of fuel that reached unusually high temperatures in the high-flux charge.[23] Metallurgical studies were also made of fuel behavior near the melting point.[24] When combined with studies of aluminum-uranium fuels at other laboratories,[25] the above studies give rise to the following conclusions:

1. Localized swelling and blistering may occur in highly irradiated fuel heated above about 450° C.

2. Expansion of the fuel occurs near the melting point, resulting in increased potential for fuel-target contact. The cladding is likely to crack open.

3. The uranium-aluminum core begins to melt before the aluminum cladding, and as a consequence, cladding failure may release a substantial amount of core before the cladding is completely melted.

4. Molten aluminum or core tends to flow in rivulets rather than in films.

5. Melt droplets quenched in water have an expanded "popcorn" consistency, and are readily transported in flowing water.

It has been postulated that because the fission gas pressure is very high at the melting point, irradiated fuel may form an expanded foam upon melting. Fuel foaming has been substantiated by unpublished studies made at Argonne National Laboratory (ANL) and at the Savannah River Technology Center (SRTC). A study by Cronenberg (Table 1, Item E-3) made under this program has predicted that fuel foaming could cause good fuel-target contact and melting of the target material.[26] Program item E-4 (Table 1) is a study, recently conducted at SRTC, that involves melting coupons of irradiated fuel and observing their foaming and swelling behavior. The melting study has provided information on fuel swelling and foam collapse in highly irradiated fuel.

The melt process is more complex than the above experiments would indicate. Neither of the above unpublished experiments properly accounted for the mechanical stresses on an actual fuel assembly. The type of failure in a severe accident and the resulting configuration of relocated fuel and target materials are critical parameters in the recriticality studies discussed below and, prior to cancellation of the NPR program by DOE, an experimental melting and failure program was planned in conjunction with the NPR program.

3.2.2 Melt spreading

Melt spreading is an important factor in predicting the course of an accident in which a substantial fraction of the core has melted. If the melt spreads out over a sufficiently large surface, on the floor of the pin room or the pump room of K Reactor, for instance, it may cool and solidify. Depending on the accident progression, water may be present to augment cooling. Program item E-5 (Table 1) consists of an experimental characterization of melt spreading on both wet and dry surfaces. The experiments were conducted at several scales to test the scaling factors derived from theory. Preliminary results indicate that the melt spreading is highly dependant on the melt temperature and, in the case of a wet surface, on the depth of the water pool. The corresponding melt pool geometry can vary from a flat disc shape to a pile of discreet spheres. The studies were performed at Brookhaven National Laboratory (BNL) under the direction of George Greene, and a report on the results is expected in October 1992.

3.2.3 Melt-vessel interaction

In extremely severe accidents in which all cooling water is lost, the molten fuel would accumulate in the reactor vessel and primary cooling system. The chemical interaction of molten core with the reactor vessel and piping may be an important factor in the penetration of the piping. W.C. Mosley of the Savannah River Technology Center made a preliminary study (Table 1, Item E-6) of this effect at temperatures ranging from 800°C to 1050°C.[27] The aluminum in the Al-U alloy attacks the stainless steel at a rate which depends on the temperature of the molten alloy, the rate of mixing in the molten pool, and the condition of the contact surface. Additional studies at SRTC are being made as part of the NPR severe accident program. The NPR studies will provide additional data at higher temperatures, and under conditions more closely resembling those expected in the primary cooling system following a very severe accident.

3.3 Energetic Phenomena Associated with Severe Accidents

The release of radioactivity to the environment may be increased as the result of energetic phenomena that interfere with, or bypass, the normal functioning of the reactor confinement system. In addition, these phenomena can alter the course of the accident. Five such phenomena have been identified and incorporated into the severe accident program.

3.3.1 Molten core-concrete interactions

Molten core-concrete interactions (MCCI) can occur when the uncooled core contacts the concrete floor of the reactor building. Molten core-concrete reactions are potential threats to the confinement system since the exothermic reactions use elements of the concrete basemat as "fuel". Program item E-7 (Table 1), now in progress, has been conducted using small scale experiments at SRS and Rice University, and larger scale experiments at Sandia National Laboratories. Results to date have shown that at high temperatures, above approximately 1400°C, some of the chemical reactions become highly exothermic. Hydrogen and other gases are produced by the oxidation of aluminum. At the high temperatures involved in MCCI, the addition of water, beyond that associated with the concrete, may result in the explosive ignition of the aluminum. Thus, uncooled core material at very high temperature (greater than about 1400°C) on the building floor is a potentially important source of hydrogen.

The core-concrete interactions also generate very large amounts of aerosols, which may threaten the integrity of the filter compartments. It has become clear that the key to this area is quenching the melt (by water) at a temperature low enough to avoid the highly exothermic reactions. Priority has thus been directed toward the cooling aspects of the melt spreading experiments described in section 3.2.2. Some modeling work is also planned, as described in section 4.

3.3.2 Fuel-coolant interactions

Steam explosions may occur when water comes in intimate contact with molten core material. Steam explosions fragment the metal melt and heat transfer to the water phase occurs within milliseconds. Such explosions are a concern because they can be very energetic and damaging, especially when large amounts of superheated melt are formed by a reactivity transient; this happened, for example, in the SL-1 reactor accident, and probably also at Chernobyl. Extensive investigations of steam explosions involving molten aluminum and water have been conducted within the aluminum industry, and have shown that such reactions can be both energetic and somewhat unpredictable. The unpredictability is caused by a layer of steam that forms spontaneously between the melt and the water; no explosion will occur unless the steam layer is disrupted. The trigger may be a shock, or any other effect that brings the materials together. The magnitude of the explosion, once triggered, depends on a complex variety of factors that make prediction and replication of steam explosions very difficult. Oxidation of aluminum, with generation of heat and hydrogen, has also been reported. These factors, taken together, imply that calculations of explosive energy and effects must use conservatively high values, the choice of which is not well established. Post ignition calculations of steam explosions have been attempted by several authors, but this area remains controversial.

The SRS severe accident program does not attempt to solve all the problems associated with steam explosions, but rather is restricted to several specific issues, and has been primarily involved with the behavior of molten control rod material containing lithium.

Item E-8 (Table 1) is a study of mechanisms for repressing steam explosions using water additives, and is being conducted under Prof. S. Abdel-Khalik at the Georgia Institute of Technology (Georgia Tech). The tests are small scale, employing a few grams of molten material in each experiment. The purpose is to determine whether there are any practical methods for suppressing explosions using relatively small amounts of additives. The final report has been delivered[28] and is currently being reviewed.

Item E-9 (Table 1), conducted under the direction of Lloyd Nelson of Sandia National Laboratories, is a continuation of very small scale steam explosions aimed at quantifying the effects of lithium during the quenching of lithium-aluminum melts. The addition of lithium is known to increase the energy of steam explosions,[29] presumably because of the chemical reactivity of lithium. The Sandia experiments measured the presence of lithium indirectly by collection of the hydrogen produced by lithium oxidation.

Items E-10 and E-11 (Table 1) are experiments conducted at Brookhaven and Argonne National Laboratories, respectively, on the interaction of melted control rod material with water. The intent was to determine whether steam explosions occur under conditions prototypic of a blocked septifoil in an SRS reactor and what potential damage could occur. The control rod material is aluminum-3% lithium alloy. The experiments were performed inside septifoil housings in order to simulate exactly the geometric effects. The amount of melt involved ranged up to a kilogram in the Argonne experiments, and about 100 grams in the Brookhaven experiments. This permits fully prototypical experiments, inasmuch as the maximum amount of melt that is anticipated in the septifoils is probably a few hundred grams at most. Potential triggers, including temperature, agitation, and mechanical triggering by falling rods, were investigated in the Brookhaven studies. No steam explosions occurred in any of the 20 experiments.

Larger experiments, involving several kg of aluminum, and experiments with aluminum-uranium alloy, were being conducted in the NPR experimental program. Close coordination will be maintained with this program to obtain and use the results. Experiments with small amounts of Al-U alloy, originally planned for this program at Sandia, were canceled because of problems in handling the radioactive debris.

3.3.3 Recriticality

As indicated in Section 3.2.1 above, the relocation of fuel may bring it into a critical configuration. In extreme cases, recriticality can produce a large, nearly instantaneous, release of radiation and heat, possibly causing a correspondingly large steam or chemical explosion. (A chemical explosion would be caused, for example, by the reaction of vaporized aluminum with steam or air.)

Given the geometric configuration of fissile material, reliable methods exist to calculate the nuclear reactivity (see section 4.4.1.5). The primary questions concerning recriticality are the rate at which criticality is approached, the process by which the nuclear reaction is terminated, and other parameters which determine the energy released by the recriticality. The above parameters depend strongly on the details of the fuel and target relocation during a core melt. Recriticality was a major focus of the NPR severe accident program, and much information was anticipated from the NPR studies.

3.3.4 Hydrogen explosions

Hydrogen deflagrations and explosions may take place if a sufficient concentration of hydrogen mixes with air. Conditions required for hydrogen to burn or explode have been characterized in the LWR severe accident program, and are not being investigated experimentally in the Severe Accident Analysis Program. Preliminary studies indicate that in the well ventilated reactor buildings hydrogen burning would not be a problem, unless the hydrogen is generated by a rapid process or isolated gas pockets with high hydrogen concentration can form. Rapid, in this context, means a process that operates on a scale of a few minutes or less. Two such processes are molten core-concrete reactions, and steam explosions. Hydrogen reactions will be considered in the analyses of potential MCCI and FCI events.

3.4 Response of the Reactor Confinement System to Accident Phenomena

The response of the confinement system to challenges has been extensively studied, especially with regard to the confinement of steam and iodine.

3.4.1 Steam flow and humid environments

The response of the moisture separators and HEPA filters to steam flow was studied and reported by Peters.[30] In addition, program item E-12 (Table 1) found that the ambient relative humidity may have a significant effect on filter performance due to the presence of hygroscopic aerosol particles during severe accidents. A final report has been received and reviewed.[31] No additional work in this area is planned as part of the Severe Accident Analysis Program.

3.4.2 Filter response

The response of the moisture separators and HEPA filters has been determined in a study by Novick, et. al. at Argonne National Laboratory, program item E-12 (Table 1). The study is aimed at determining the potential for plugging the filter system by aerosol loading, which would restrict the ventilation of the building, prevent cooling the filter compartments by the normal air flow, and possibly fail the filters. The work is complete and has been reviewed by Long and Monson.[31,32] The results provide quantitative information on filter blockage as a function of mass loading and particle size which is being applied to the MELCOR/SR filter models.

3.4.3 Iodine deposition

Deposition of iodine within the building would reduce the amount of iodine challenging the filter compartments. A study of the effectiveness of sprays in removing iodine and other particulate has been performed and published.[33,34] A study of particulate deposition within the building was proposed for incorporation into this program, but was abandoned due to high cost.

3.4.4 Iodine retention

A long series of reports on iodine was summarized by Evans,[35] and Hyder has published a report summarizing current knowledge.[36] The effects of radiation, temperature, filter age, and humidity on the ability of the carbon bed filters to retain iodine have been studied. Experimental data on iodine desorption has been analyzed by Severe Accident Analysis Program personnel with respect to the current MELCOR/SR models.[37] No additional work is planned in this area under the Severe Accident Analysis Program.

3.4.5 Aerosol Transport in the Reactor Building

A full scale experimental study of aerosol transport in the K reactor building was initiated to test the building Decontamination Factor (DF). After discussing preliminary work with the expert panel the experiment was canceled. In addition to strong concerns regarding the repeatability of the experiment, the expected DF for the building did not justify the expected high cost of the experiment. Thus, it was decided that the funds would be more effectively utilized in other program areas.

3.5 Supporting Studies

Studies are being conducted to obtain other supporting information for use in interpreting the results of the work described above and for validating computer codes. Most of the supporting studies were halted due to budget constraints.

3.5.1 Reaction rate of steam and aluminum-uranium

A study of the rate of reaction of aluminum-uranium alloy with steam was proposed but was abandoned after a few preliminary tests because sufficient data was found from literature sources. No work is presently planned in this area.

3.5.2 Fuel viscosity

The viscosity is important in considering melt spreading and cooling, and also for estimating the rate of assembly in recriticality events. This work involves the construction and application of a falling needle viscometer for use at the temperature of molten aluminum. Because of reduced funding, the fuel viscosity measurements had to be abandoned before completion. In addition, viscosity data for aluminum melts with impurities has recently appeared in the literature.[38]

4 ANALYTICAL ACTIVITIES OF THE SEVERE ACCIDENT ANALYSIS PROGRAM

4.1 Development of Severe Accident Computer Codes

The principle code development activities of the Severe Accident Analysis Program are contained in Table 2 (Section 7) and are described in this section. Other available calculational tools for which developmental funds are not currently available are also described here.

Additional development of codes pertinent to severe accidents has been done under other SRS programs. In particular, the TRAC, CONTAIN, and MELCOR codes have been modified for use in the development of the SRS Probabilistic Risk Assessment (PRA).[39] RELAP5 is also being used for success criteria determination. A code modeling steam explosion phenomena has also been developed for use in the PRA.

4.1.1 Steam explosion codes

Item A-1, which is being conducted by Prof. Abdel-Khalik at Georgia Tech, is a calculational program pertinent to steam explosions. The task involves developing a computer code, based on the K-FIX and FLEX codes, specifically for predicting the effects of a large ex-vessel steam explosion of a given magnitude on the reactor building and fixtures.[40,41,42,43] The code development is complete and trial calculations have been performed on the effects of steam explosions within the K-reactor confinement building.[44]

4.1.2 MELCOR/SR

The MELCOR severe accident code,[45] originally developed by Sandia National Laboratories for light water reactors, has been modified (Table 2, Item A-2) for use with SRS reactors by Science Applications International Corporation (SAIC). Several new models were developed for this work, notably including the core, the ventilation, and the fission product release from molten pool models, and the material properties package was modified for SRS reactors. The initial modeling was completed in 1990 with further model improvements (Melcor/SR Mod 4) halted in mid 1992 for lack of funding. MELCOR/SR Mod 4 was delivered in April 1992 and the required additional modeling to incorporate new information from the experimental parts of the Severe Accident Analysis Program will be performed by Severe Accident Analysis Program personnel. Some improvements, such as CORCON Mod 3 and a VANESA upgrade for the SRS chemistry, will soon be available as stand alone modules and will require implementation into MELCOR/SR. Extensive work on and with MELCOR/SR is expected during 1992 with continuation to future years.

4.1.3 SAS-HWR

Separate from the Severe Accident Analysis Program, the NPR-Heavy Water Reactor program modified SAS,[46] a coupled thermal hydraulics/neutronics code developed for the liquid metal fast breeder reactor program, for the metal alloy fuel and target tubes used in the SRS reactors. SAS-HWR uses 3-D space time kinetics, rather than point kinetics, for neutronics predictions, and thus should provide a more rigorous theoretical basis for predicting reactivity as fuel and target relocate during a meltdown event. The severe accident group has obtained an interim version of the SAS-HWR code being developed for the Heavy Water NPR and is currently investigating the possibility of modifying the code for use with K reactor.

4.1.4 SCDAP/RELAP5

The SCDAP/RELAP5 code was developed at Idaho National Engineering Laboratory (INEL) to predict the thermal-hydraulic response, core damage progression, and fission product release and transport within the reactor coolant system during light water severe accidents. SCDAP/RELAP5 is a mechanistic code for in-vessel phenomena as compared to the more parametric MELCOR code. INEL has modified SCDAP/RELAP5 for SRS reactors by adding models for melt progression, relocation of liquified material within the reactor cooling system, and fission product release from fuel and molten pools. The new models have been incorporated into the code but the documentation is not complete and only limited testing of the code had been performed when the program was canceled due to lack of funds. A report summarizing the work performed is in progress.[47]

4.1.5 MCNP

MCNP[48] is a three dimensional Monte Carlo transport code capable of performing neutron, photon, and electron transport calculations as well as coupled neutron/photon and electron/photon transport. Among the advantages of using MCNP are the heterogeneous geometry modelling capability, the intense neutron cross section structure, and the lack of approximations in the representations of particle physics interactions. The primary use of MCNP for the Severe Accident Analysis Program has been to analyze the reactivity of melt relocation configurations to define recriticality conditions and the associated power excursion response.

4.1.6 CORCON

CORCON[49] is a program to model the interaction of molten metal and a concrete basemat. MELCOR/SR Mod 4 uses CORCON Mod 2 for the CAV package. CORCON Mod 2 has been modified to include SRS materials but is coupled to a version of VANESA (see Section 4.1.7) which does not reflect the chemistry of SRS materials. The next update of CORCON (Mod 3), which should be released in the near future, will include models such as transient concrete heat conduction, interlayer mixing, and non-ideal chemistry modifications. In addition, CORCON and VANESA will be integrated into a single code. CORCON Mod 3 will be evaluated as a stand alone code prior to coupling to MELCOR/SR.

4.1.7 VANESA

The current VANESA[50] aerosol chemistry model in MELCOR/SR does not incorporate the proper species for the SRS reactors. SNL is modifying VANESA for use with the SRS reactors (Table 2, A-3) and will provide a stand alone version of the code in 1992. Once the updated VANESA code is available, a study is planned to evaluate the technical benefits of the stand alone code. Validation and verification of the updated VANESA code followed by incorporation into CORCON and MELCOR/SR will be performed by the severe accident group at SRTC provided the initial studies indicate this task to be technically beneficial and cost effective.

4.2 Integrated Analyses and Reports

A major product of the Severe Accident Analysis Program will be a series of studies and reports (Table 2, A-4) summarizing the results of the program in specific areas of interest and concern. The summary reports will be integrated analyses, incorporating all available information from within the program, along with data from other sources, and utilizing all the experimental results and calculational tools that are appropriate. A schedule for the reports will be developed during the last quarter of FY-1992 with the first reports being issued in 1993.

5 THE FUTURE OF THE SEVERE ACCIDENT ANALYSIS PROGRAM

The first four years of the Severe Accident Analysis Program emphasized experimental work. The experiments increased the understanding of the basic phenomena and provided a foundation upon which to build and test the models used in the computer simulations. With the delivery of several computer programs and final reports from experimental programs in mid-1992, the severe accident group is currently involved in a comprehensive validation and verification efforts for the SRS specific modifications to MELCOR 1.8.1. Once the codes are approved for use, a number of integrated analyses are planned to investigate reactor response to severe accidents. MELCOR/SR is currently being used in support of the Level II PRA effort. The severe accident group is currently in a transition phase, closing out many of the existing experimental programs and assimilating the data into the computational programs.

5.1 Experimental Work

The severe accident group is currently closing out all of the existing experimental work and is expecting numerous reports to arrive in 1992. Many issues have been dealt with sufficiently but a few outstanding issues remain. The remaining questions were detailed and prioritized by the Severe Accident Advisory Committee in November 1991[12] and comprise the bulk of the desired experimental program.

5.1.1 MCCI Chemistry

The chemistry involved in MCCI is still an unknown although the primary reactions appear to be between molten metal and silicon dioxide in the concrete. The aerosol generation aspects of MCCI are equally unknown. The next version of VANESA, revised to reflect the SRS reactor materials, is expected to aid in the study of this phenomenon. The aerosols are a concern because of the presence of pyrophoric materials on the filters in the SRL experiments conducted at Sandia. Chemical analysis may still be possible to determine the nature of the pyrophoric compound on the filters.

5.1.2 Steam explosions

The current understanding of steam explosions is comparable to that of the commercial nuclear industry and thus is somewhat qualitative. In particular, the nature of triggering in steam explosions is still in dispute. The advisory committee recommends further work in this area.[12,13] The MELCOR steam explosion model is heavily input dependant with little expectation of improvement until the phenomena are better understood. A report has been written summarizing the current state of knowledge of steam explosions.[51]

5.1.3 Fission product release

Studies of the fission product release mechanisms from deep melt pools are not currently well understood. The planned fission product release studies at ANL exhausted the available funds while still in the preliminary phase and only limited data was received. While the release from molten pools is primarily governed by the conditions outside the pool, additional studies are required to characterize the effects of crust formation on the pool surface, heat transfer, and natural circulation on pool releases.[13] The severe accident group is currently following the progress of the literature and the NPR efforts.

5.1.4 Realistic accident progression model

The PRA assumed a number of accident sequences based on either simple assumptions or limited scoping parametric studies. Event probabilities were estimated for various events, but experimental or analytical evidence is often lacking. This is especially true with regard to core melt scenarios and recriticality events. Assessments of recently completed fuel melt studies (Section 3.2.1) indicate the importance of thermal-mechanical stresses on the course of a melt which could significantly alter the core melt scenario. Since the PRA consequences depend on a series of actions and events, it is not clear that altering the core melt scenario will effect the PRA results. Mark 22 assembly melt studies planned for the NPR-HWR would have reduced the uncertainty of the melt progression but were canceled along with the NPR program.

5.2 Analytical Work

5.2.1 MELCOR/SR V & V

The large body of experimental work performed in the first phase of the Severe Accident Analysis Program has laid the foundation for future computational and analytical work. The latest version of MELCOR must be Verified and Validated (V & V'd), which will draw heavily on the experimental database. Deficient models will need to be identified and revised. In particular, CORCON Mod 3 and VANESA must be V & V'd as necessary and incorporated into MELCOR/SR, either as input or as updated modules. Once V & V'd, MELCOR/SR will be used to make integral analyses of the reactor during accident situations and address open issues on recriticality, melting, and fuel-coolant interaction.

5.2.2 SCDAP/RELAP5 and SAS-HWR

In addition to MELCOR/SR, SCDAP/RELAP5 (modified and enhanced for SRS) and SAS-HWR are available for computational work. Both SCDAP/RELAP5 and SAS-HWR are more mechanistic in some areas than MELCOR and could be more accurate in their assessments of in-vessel phenomena. For example, RELAP5 should give better predictions of water inventory in the reactor coolant system because the heat transfer packages are more detailed than MELCOR. Therefore, the code may be more important in accident management applications than in PRA applications. Documentation of SCDAP/RELAP5 still needs to be completed as well as V & V work but no funding currently exists for development of SCDAP/RELAP5.

SAS-HWR is a coupled thermal hydraulic-neutronics code with fuel/target relocation modeling capable of properly describing reactivity driven power excursion events leading to severe accident phenomena. The fuel relocation models are more mechanistic than parametric since the SAS code was originally intended as a tool for mitigating reactors accidents through design. Thus, input deck models can easily be altered to investigate design changes for avoiding onset of significant voiding (OSV) and fuel melting. The severe accident group is currently evaluating this technology for use in the Severe Accident Analysis Program.

5.2.3 Recriticality

Recriticality concerns have been voiced by the Severe Accident Advisory Committee as well as the Defense Nuclear Facility Safety Board (DNFSB). A parametric assessment of recriticality concerns for K reactor Bottom End Fitting was made for the Level II PRA.[52] A summary report on the state of recriticality knowledge, including an interpretation of the results of recent SRS fuel melt experiments, is scheduled for late 1993. Future work on recriticality will be determined by this report.

6 ADDITIONAL APPLICATIONS

Under funding outside of that provided by the K reactor restart program, the Severe Accident Analysis Program has also provided technical support to the Heavy Water New Production Reactor (NPR) program. Severe Accident Analysis Program staff have served as reviewers and consultants to NPR in the development of the Deterministic Severe Accident Criteria and provided other technical support as requested. Prior to cancellation of the NPR program in September 1992, NPR support was planned for FY-1993 with respect to continued fuel melt experiments and the effect of melt progression on recriticality issues.

Much of the technology developed in executing the Severe Accident Analysis Program is also applicable to non-reactor areas. For instance, a report was issued by Ellison[53] on the magnitude of possible steam explosions in the Defense Waste Processing Facility (DWPF) glass melter. The Severe Accident Analysis Group is currently exploring new avenues for transferring technological capabilities developed under the reactor restart program to other applications in the DOE and industrial complex.

7 TABLES

Table 1. Severe Accident Analysis Program Experimental Elements

Item	Activity	Investigator	Status	Nature of Work
E-1	Fission product release rate	Woodley, HEDL	Completed	Two fission product release rate studies which are summarized by Whitkop in DPST-86-406 and DPST-87-412.
E-2	Fission product release rate	Sharon and Baker, ANL	Completed	Study of fission product release rate from deep pools considering the effects of both temperature and geometry. Limited data.
E-3	Fuel foaming	Cronenberg, ANL	Completed	Al-U fuel foaming and recriticality considerations. Summarized in WSRC-RP-89-1422.
E-4	Fuel melting and foaming	Monson, Long, Dewald, SRTC	In progress, completion in 1992	Hot cell studies of melting and foaming of irradiated SRS fuel.
E-5	Melt spreading	Green, BNL	In progress, completion in 1992	Studies of Al melts on wet and dry surfaces. Includes scaling and pour rate effects.
E-6	Melt-vessel interaction	Mosley, SRTC	Completed	Preliminary studies of chemical interaction of molten fuel with reactor vessel and piping.

Table 1. Severe Accident Analysis Program Experimental Elements (cont.)

Item	Activity	Investigator	Status	Nature of Work
E-7	Molten core-concrete interactions	Randolph, SRTC Margrave, Rice Copus, et.al., SNL	In progress, completion in 1992	Study of molten Al and fission product interactions with concrete at high temperatures with particular attention to chemical reactivity and aerosol generation.
E-8	Steam explosion studies	Abdel-Khalik, Georgia Tech	Complete	Studies of steam explosion inhibition by varying the fluid properties.
E-9	Lithium effects in steam explosions	Nelson, SNL	In progress, completion in 1992	Studies of steam explosions using small amounts of melt (Al and structural alloys) falling through water.
E-10	Li-Al steam explosion studies	Greene, BNL	Complete	Triggering and interaction of 100g Li-Al melts with water inside a septifoil housing.
E-11	Li-Al steam explosion studies	Cho, ANL	Complete	Similar to E-10 but with 1kg of Li-Al.
E-12	Filter response	Novick, ANL	Complete	Behavior of demister and HEPA filter components in response to aerosols of varying size and hygroscopicity.
E-13	Iodine removal by CHRS	Hyder, SRTC	Complete	Removal of iodine by confinement sprays. Reported in *Nuclear Technology*, 94:80 (April, 1991).

Table 2. Severe Accident Analysis Program Analytical Elements

Item	Activity	Investigator	Status	Nature of Work
A-1	Steam explosion effects	Abdel-Khalik, Georgia Tech	Complete	Determining material response to steam explosions.
A-2	MELCOR/SR	SAIC, SRTC	Stopped at interim phase 4 version due to lack of funds	Adaptation of MELCOR to the SRS Heavy Water Reactors. Work included modifications of core and ventilation modules and the material properties package.
A-3	VANESA	Powers, SNL	In progress, completion in 1992	Update of VANESA aerosol generation package to include chemistry relevant to SRS reactors. Ties in with CORCON package from MELCOR.
A-4	Integrated studies	SRTC	In progress	Topical reports combining experimental and calculational results of the SAAP.

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