

SASA MONTHLY

JANUARY 1984

PROJECT TITLE: Severe Accident Sequence Analysis (SASA)

PROJECT MANAGER: S. A. Hodge

NRC B&R NO./FIN NO.: 60 19 01 3 0/BO452

TECHNICAL HIGHLIGHTS:

Major work in progress during January includes preparation of the draft report for the Anticipated Transient Without Scram (ATWS) accident sequence study, continuing modifications and improvements to MARCH-BWR in preparation for the degraded core calculations for the ATWS sequence, and preparation of the draft report of the fission product transport study for the Loss of Decay Heat Removal (DHR) accident sequence. Unit 1 of the Browns Ferry Nuclear Plant serves as the model plant for all studies. Both draft reports will be issued for peer review comments in early February.

The personnel contributing to the SASA effort at ORNL are divided into three working groups. The individual group reports for progress during January are presented below with a brief initial statement of the purpose of each group.

Group I: (R. M. Harrington) Determines and analyzes the events of the accident sequence that would occur prior to core uncover, using the ORNL-developed simulation program BWR-LACP to study the plant response to operator actions.

January work has been devoted to completion of the draft report for the ATWS accident sequence study. The draft report will be distributed for peer review during the first week of February.

Presentations concerning the material contained in the report were given at the SASA program review meeting on January 10 and at a GE-ORNL-TVA information exchange meeting on January 25.

Group II: Determines and analyzes the events of the accident sequence that would occur following core uncover, including core melt and containment failure.

MARCH Modifications for the In-Vessel Phase of the Browns Ferry ATWS Study (L. J. Ott) New reactor vessel pressure control and calculational routines have been developed and are in the checkout process. Previous calculations for the Loss of DHR and the TQUV (loss of injection) accident sequence with stuck-open relief valve have been repeated using the new routines and an assessment of the differences in the results for primary system and primary containment response are in progress.

MARCH Modifications for Containment Analyses in the ATWS Study (C. R. Hyman) The review of the MACE containment analyses package in the ORNL version of MARCH has been completed. Major items scheduled for accomplishment in time for the ATWS calculations include the replacement of subroutine INTER with CORCON Mod 2 and the provision of models to correctly represent the introduction of T-quencher discharge into the pressure suppression pool after primary containment depressurization. Randy Cole of Sandia National Laboratory has indicated that an NRC-approved pre-release version of CORCON Mod 2 will be made available to the ORNL SASA team during March.

BWR Severe Accident Model Development at RPI (M. Podowski et al)

1. MELRPI Code Development (R. Taleyarkhan). Work has begun on modifying the structure of MELRPI, to introduce an improved model of molten material relocation and related phenomena. Currently the control rods are not allowed to relocate independent of the debris. That is, upon debris formation, the control rod is considered part of the debris.

Ongoing modifications will permit the control rods to melt and relocate, independent of what happens to the rest of the node. The slug model developed earlier for molten cladding relocation will also be used for the molten stainless steel released due to control rod melting. Upon complete melting of stainless steel in a node, the control rod structure above (if any) is assumed to slip down and fill the space left vacant by the molten stainless steel.

2. Combination of MELRPI with the ECCS Subroutine. Minor changes are currently being made in the modeling of the film front motion characteristics, upon rewetting of the channel walls and control blades (due to core-spray and/or overflows). Several runs have been made to assess the impact of ECCS on a hypothetical transient, involving core uncover, heat-up and ECC injection. The final report describing the ECC model and its computer implementation is in preparation.
3. Model development for the Lower Head. Work has recently begun on the development of a model for the reactor vessel lower plenum and lower head. At present, the general framework for the relocation of molten debris and progression along the walls of the CRD and instrument guide tubes in the lower plenum is being set up. A simple mechanistic approach that captures the basic physics involved is envisioned.

The model for heat-up, melting and molten material relocation in the lower plenum and lower head will be based on a two-dimensional multi-node concept, similar to that used previously for the reactor core.

4. Investigation of Radiant and Volumetric Heat Sources in the BWR Steam Separators and Standpipes (J. C. Conklin, Dissertation). Formulation and scaling of the governing conservation equations of mass, momentum, and energy have continued. During the initial phase of a severe transient, the standpipe wall is postulated to be cooler than the steam. This situation will cause the bouyancy force to be in the opposite direction of the inertial force. Another threshold ratio of the Grashof and Reynolds Numbers has been identified for this situation without entrained fission products. This particular threshold ratio is indicative of a zero value of fluid shear and might be of significant importance. The value of this threshold ratio will be modified as necessary to account for the applicable fission product heat source strengths.

Group III: Determines the magnitude and timing of fission product release from the fuel, establishes the various pathways for fission product release to the atmosphere, and performs the fission product transport calculations for each Severe Accident sequence analyzed.

Fission Product Transport Analysis (R. P. Wichner and C. F. Weber). All background and explanatory analyses for the pending draft report concerning fission product transport in the Loss of Decay Heat Removal (LDHR) accident sequence have been completed and the pertinent sections of the report have been written. Fission product transport calculations have been completed and appropriate sections for the report are being written.

Aerosol Production and Transport (A. L. Wright). Chapter 4 and Appendix B of the LDHR accident sequence report have been written; both of these relate to core-concrete aerosol production and transport in the drywell and reactor building. The overall results from the study can be summarized as follows:

1. For the first eight hours following reactor vessel bottom head failure and the beginning of core-concrete interaction, calculations performed with the CORCON-MOD1 and VANESA codes indicate that about 973 kg of core-concrete aerosols would be produced.
2. Aerosol transport calculations performed with the QUICK code for the drywell and reactor building indicate that a pathway for aerosol release to the environment would develop through the Standby Gas Treatment System (SGTS) after the HEPA filters, with flow blockage by aerosols, are predicted to rupture. The calculations indicate that filter rupture would occur about five hours after the start of core-concrete interaction. The predicted aerosol transport to the environment through the ruptured SGTS filters is 87 kg, or about 9.6% of the total core-concrete aerosol predicted to be produced.

Chemical Change Effects (E. C. Beahm). Data from the Containment Systems Experiment (CSE) have been used to derive an equation for

organic iodide removal as a function of time (t) and temperature (T). The removal of organic iodide can be expressed as a first order process as:

$$\frac{dC_o}{dt} = -\gamma C_o$$

where

C_o = the concentration (gmol/cm³) of organic iodide at time t (s),

$\gamma = \gamma(T) = \exp[a-b/T]$ and values of a and b are adopted as:

a = 0.14, b = 4.6×10^3 .

Values of inorganic iodine partition coefficients measured in the large Containment Systems Experiment yield values in the range of 10^3 to 10^4 . In line with these values, an iodine partition coefficient of 5×10^3 has been adopted for use in the analysis of this accident sequence, compared with a value of 5×10^4 employed for our earlier studies.

MEETINGS AND TRIPS:

R. M. Harrington, S. A. Hodge, and R. P. Wichner attended the SASA program review meeting held on January 10 and 11 at Silver Spring, MD, and gave presentations concerning their work.

R. M. Harrington and S. A. Hodge gave presentations to visiting personnel Deborah Hankins and Lowell Claassen of the General Electric Company and Ken Keith of TVA at ORNL on January 25 concerning the ongoing work on BWR ATWS studies.

S. A. Hodge attended a meeting at TVA Headquarters, Knoxville, on January 31 concerning RAMONA calculations for the ATWS accident sequence.

REPORTS, PAPERS AND PUBLICATIONS:

None.

PROBLEM AREAS:

The Model BWR MK II containment plant for future ORNL SASA studies remains to be identified.

A6354: Severe Accident Sequence Analysis Program (SASA)
EG&G Program/Technical Monitors: S. R. Behling/R. C. Gottula
DOE Technical Monitor: D. Majumdar
NRC Technical Monitor: B. Agrawal ✓

The objective of this project is to use deterministic calculational tools to provide detailed analyses of severe accident sequences to support, verify, and modify probabilistic event sequences, to aid in the development of accident recovery strategies, to provide parametric values for experimental programs such as the Power Burst Facility (PBF) Severe Fuel Damage testing, and to point out the need for additional computer code development and experimental data.

1. Summary of Work Performed During Reporting Month *June 1984*

General:

The quarterly SASA technical interchange meeting was attended by S. Behling, R. Gottula, W. Jouse, and E. Holcomb on June 18-20. Presentations were made concerning Idaho National Engineering Laboratory (INEL) SASA work completed since January 1984, RELAP5/MOD1.6 benchmarking efforts with RAMONA-3B, and the status and needs of BWR containment analysis capabilities.

An Inspection and Inforcement (I&E) meeting was attended on June 18 and 19 by S. Behling and R. Gottula at the request of NRC.

Pressurized Water Reactor Studies:

The Bellefonte TMLB' calculation was rerun with RELAP5/MOD2 Version 22 beyond 1500 s. Version 22 corrects an algebraic error in the heat transfer correlation used for free convection from/to vapor (void fraction = 1.0). The revised calculation produced a faster heatup of fuel and slower heatup of vapor and vessel internals. The results of the RELAP5/MOD2 Version 22 calculation were sent to SAI/Sandia for use in their Bellefonte containment analyses.

A copy of the RELAP5/MOD2 Bellefonte deck along with a short steady state run with Version 22 was also sent to Tennessee Valley Authority (TVA).

Iterations between RELAP5/MOD2 and SCDAP for the TMLB' accident sequences for both the Bellefonte and Seabrook plants are temporarily on hold. A problem in calculating a consistent vapor internal energy between the two codes with noncondensable gases is currently being resolved.

Work is continuing to develop a preliminary RELAP5/MOD2 multi-D model of the Bellefonte vessel. Checkout runs will begin about the end of July.

1. Summary of Work Performed During Reporting Month (continued)

A report documenting the Bellefonte SCDAP sensitivity study has received initial review and comments are being incorporated.

An updated SCDAP calculation for the Bellefonte TMLB' sequence was completed using output from the RELAP5/MOD2 Version 22 rerun of the TMLB' sequence. SCDAP input decks have been assembled for three core channels corresponding to the three parallel channels in the RELAP5/MOD2 model and will be used for future calculations.

For the Seabrook plant, a SCDAP calculation for three parallel channels corresponding to the three parallel channels in the RELAP5 model is in progress for the TMLB' sequence.

Also, a SCDAP calculation was completed for the TMLB' sequence where the operator attempts to decrease system pressure by opening power operated relief valves (PORVs). This sequence will be investigated further when the RELAP5/SCDAP linked code is available next fiscal year.

Boiling Water Reactor (BWR) Studies:

The report entitled "Sequence Matrix for the Analysis of an ATWS in a BWR/4; Phenomena, Systems, and Operation of Browns Ferry Nuclear Plant Unit 1" written by W. Jouse was distributed for peer review. The report documents the BWR ATWS analysis methodology and some preliminary results of dominant ATWS sequences. This report will be integrated into a NUREG report including other ATWS studies.

A proposal letter was transmitted to the NRC outlining current BWR containment code development and data requirements. Large uncertainties still exist in predicting the probability, timing, and mode of containment failure during anticipated accident without scram (ATWS) transient. Models of important phenomena are either fragmented among numerous codes or do not exist. Data which could indicate the importance of various hydraulic phenomena on containment loading are either proprietary or do not exist.

The CONTEMPT/MOD4, PELE-IC, COBRA-NC codes were procured from BNL, LLNL, and PNL, respectively. Special modeling features of these codes are expected to yield insight to BWR ATWS analyses. They will be implemented on the INEL operating system on a time available basis.

A meeting was held on June 18 between the INEL, BNL, ORNL, and NRC to review comparisons between RELAP5/MOD1.6 and RAMONA-3B. Differences in power level between the two codes still exist for a level and pressure controlled transient (Sequence 439) largely due to differences in downcomer condensation efficiency. RELAP5 predicts 9% power versus 28% power for RAMONA-3B. However, RELAP5 also predicts 28% if no condensation is assumed in the downcomer, similar to the BNL calculation. The correct emergency core coolant (ECC) flow rate (and

1. Summary of Work Performed During Reporting Month (continued)

Boiling Water Reactor (BWR) Studies (Continued)

therefore power level) depends on core hydraulic parameters, downcomer condensation efficiency, and reactivity coefficients. An analysis of natural circulation data from the Full Integral Simulations Test (FIST) facility was begun to quantify the partitioning of core pressure drop. A letter was drafted which requests reactivity data from BNL for the level and pressure controlled transient (Sequence 439).

A pseudo-steady state solution was obtained with the TRAC-BD1/MOD1 model of Browns Ferry Unit 1. Jet pump performance and core pressure drops are being adjusted to available data.

Severe core damage analysis with SCDAP concentrated on calculating the total production of hydrogen for a high pressure boiloff (Sequence 551). A core average model was generated to supply that information, and will supplement the hot channel analysis already performed.

2. Summary of Work to be Performed Next Month

PWR Studies:

A preliminary multi-D RELAP5/MOD2 model of the Bellefonte vessel will be completed and checkout runs initiated.

The report documenting the Bellefonte SCDAP sensitivity analysis will be distributed for peer review.

BWR Studies:

Resolution of uncertainties during level control scenarios will continue. The TRAC-BD1/MOD1 model will be exercised against available plant data.

The significance of hydrogen production during Sequence 551 will be investigated relative to containment integrity. New containment codes will be implemented as warranted.

A SCDAP calculation will be performed to determine the possibility of cladding oxidation and ballooning during the predicted power spikes seen in Sequence 483 (automatic transient).

3. Problems and Potential Problems

None.

4. Cost Breakdown

Individual total cost and variance explanations are provided on the Financial and Progress and Status Summary.

189 No. A6354

<u>Cost Categories</u>	<u>(\$0.0K)</u>	
	<u>Current Month</u>	<u>Year-to-Date</u>
Direct Salaries	\$ 30.3	\$ 221.4
Materials, Services and Other Costs	2.0-	16.3
ADP Support	24.0	118.8
Subcontracts	0.0	0.0
Travel	3.4	17.0
Indirect Labor Costs	39.5	281.2
General and Administrative	15.7	105.1
Capital Equipment	0.0	0.0
TOTALS	<u>\$ 110.9</u>	<u>\$ 759.8</u>

REP MILESTONES 1983-1984
THRU JUNE 24, 1984

PAGE 00001

A NUMBER STATUS NO	NODE STATUS	PROJECT LABEL	PERSON	DUE DATE	REP-D-CR	NEW DATE	COMP DATE	COMP REF
6354 50	21-01 TASK COMPLETED	COMP BROWNS FERRY NUREG REPORT	SRB	022884		043084	041584	NUREG/CR 339
6354 50	21-02 TASK COMPLETED	COMP SCDAP SENSITIVITY STUDY	SRB	043084			043084	NO DOC REC
6354 50	21-03 TASK COMPLETED	COM BELL TMLB & S2D TREES & DRFT RP	SRB	043084		051184	051884	LPL 178 B1
6354 3	21-04 A/S, ANALYSIS UNDERWAY	COM BR FR DOM SEQ CALC & DRAFT RPT	SRB	093084				
6354 3	21-05 A/S, ANALYSIS UNDERWAY	COM BELLEFONTE TMLB CALC & DRFT RPT	SRB	093084				
6354 3	21-06 A/S, ANALYSIS UNDERWAY	COMP SEABROOK TMLB' CALC & DRFT RPT	SRB	093084				
6354 1	21-07 A/S, NOT YET STARTED	COMP BELLEFONTE S2D CALCULATION	SRB	103184	NTPD-00-84	043084		
6354 3	21-08 A/S, ANALYSIS UNDERWAY	COMP SEABROOK S2D CALCULATION	SRB	093084				
6354 3	21-09 A/S, ANALYSIS UNDERWAY	BELLF. MULTI-D R5 CHECKOUT RUNS	SRB	093084				

A6354: Severe Accident Sequence Analysis Program (SASA)
 EG&G Program/Technical Monitors: S. R. Behling/R. C. Gottula
 DOE Technical Monitor: D. Majumdar
 NRC Technical Monitor: B. Agrawal

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1. Scheduled Milestones for May 1984

<u>Description</u>	<u>Due Date</u>	<u>Actual Date</u>
Complete SCDAP sensitivity calculations for the Bellefonte TMLB' Sequence	4/30/84	4/30/84C (No Ref.)
Complete draft Bellefonte sequence event tree report for review	4/30/84	5/18/84 LPL-178-84

2. Summary of Work Performed in May 1984

General:

A containment loads working group subcommittee meeting was attended on May 14th. A presentation was made regarding heatup of Reactor Coolant System (RCS) pressure boundaries for a pressurized water reactor (PWR) during a high pressure core damage accident sequence. Also, the Nuclear Regulatory Commission/Industry Degraded Core Rulemaking (NRC/IDCOR) technical exchange meeting on integral calculations of containment behavior on May 15-17 was attended.

Pressurized Water Reactor Studies:

A more detailed RELAP5/MOD2 model of the Bellefonte core was constructed including three parallel channels with crossflow. The purpose is to perform a more realistic prediction of natural circulation flow patterns in the vessel during high pressure core damage accident sequences and to provide for more accurate core damage predictions using the SCDAP code.

A report documenting the Bellefonte TMLB' SCDAP sensitivity studies is 50 percent complete. The Severe Core Damage Analysis Package (SCDAP) calculations used output from the previously reported RELAP5/MOD1.6 analysis for boundary conditions.

2. Summary of Work Performed in May 1984 (Continued)

A variation of the base case TMLB' sequence was performed for the Seabrook plant which involved an operator opening the two PORVs when the upper plenum (core outlet) indicated 25°F superheat. The purpose of this analysis was to determine if operator actions could reduce the system pressure down to accumulator pressure prior to significant core damage to prevent a high pressure melt through of the lower head. The results indicated that the plant could be depressurized to accumulator pressure prior to significant core melt, however, the feedback of hydrogen produced during cladding oxidation was not included in the analysis.

A more detailed RELAP5/MOD2 model of the Seabrook core was completed including three parallel channels. Also, an input model for the S₂D analysis is being prepared.

FRAPCON-2 calculations were completed for three different power levels corresponding to the three parallel channels in the RELAP5/MOD2 Seabrook model to provide initial conditions for SCDAP calculations. Initial SCDAP calculations were completed for a high powered bundle with control rods and a bundle with plugged guide tubes. Iterations with RELAP5 will begin soon.

A draft report documenting the TMLB', S₂D, and Anticipated Accident Without Scram (ATWS) sequence event trees for the Bellefonte plant was sent out for peer review. The report concludes that further study of ATWS initiated severe accidents is not warranted.

Boiling Water Reactor (BWR) Studies:

Investigation of the differences observed between the RAMONA 3-B and RELAP5 Sequence 439 (level and pressure control) for Browns Ferry is in progress. Data transmitted from Brookhaven National Laboratory (BNL) has been reduced to reactivity functions, and re-simulation of the sequence will be performed. A sensitivity model to parametrically study condensation efficiency, core two-phase losses, bypass reflux ratio, and reactor stagnation conditions has been constructed and run to a steady state. Resolution of these technical uncertainties is expected in June.

A letter was transmitted to LLNL requesting the PELE-IC coupled fluid/structures code. Acquisition of this capability is considered to be important because of the significant uncertainties associated with short-term containment challenge during BWR ATWS. Concern over hydrogen generation during Sequence 551 (high pressure boiloff) has resulted in contact with BNL to procure CONTEMPT-IV/MOD4, which considers hydrogen. This effect is considered to be important during long-term containment challenges with core damage.

A letter to the NRC has been drafted which outlines containment system analysis needs. These needs center on the significant uncertainties that exist in the computed ATWS sequence results. In particular, there exists a need to resolve short term containment challenge concerns. It is an active concern that the torus structure could be breached within minutes after transient initiation. Data requests to the vendor have not resulted in data release.

2. Summary of Work Performed in May 1984 (Continued)

Boiling Water Reactor (BWR) Studies (Continued)

A draft report describing SCDAP results for the high pressure boiloff sequence (#551) received internal peer review. Comments are being incorporated.

3. Scheduled Milestones for June 1984

None.

4. Summary of Work to be Performed in June 1984

PWR Studies:

Iterations between RELAP5/MOD2 and SCDAP will be initiated for the TMLB' analysis for the Bellefonte and Seabrook plants.

The draft report documenting the SCDAP sensitivity studies for the Bellefonte TMLB' sequences will receive internal peer review.

The S₂D analysis for the Seabrook plant will be initiated.

BWR Studies:

Differences between RELAP5 and RAMONA-3B for the Browns Ferry ATWS studies will be resolved. Sensitivity studies concerning condensation efficiency and core pressure loss coefficients will be conducted. Modeling of core loss coefficients will be further evaluated.

Exercising the TRAC-BD1 model of Browns Ferry for ATWS sequences will begin in June. Comparisons with RELAP5 and RAMONA-3B will be made.

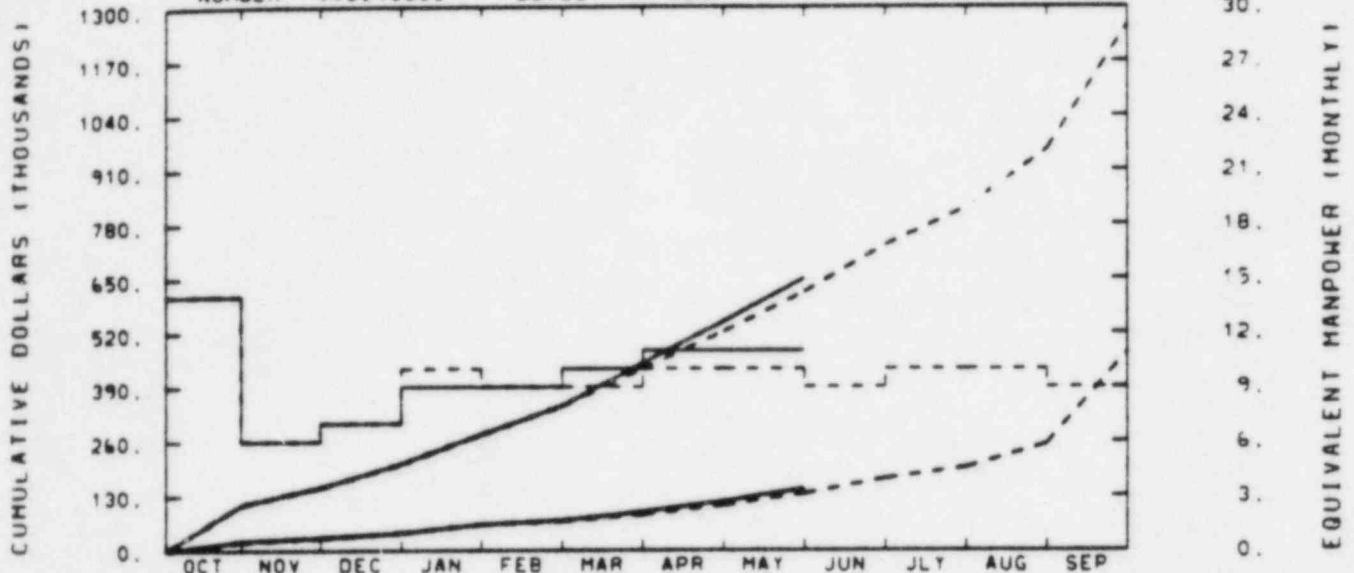
A letter describing BWR containment modeling needs will be transmitted to NRC.

5. Problems and Potential Problems

None.

RESPONSIBLE
MANAGER
L LEACH DAE

EG&G IDAHO INC.
SEVERE ACCIDENT SEQUENCE A6354
NUMBER 443540000 LEVEL 4 WBS



TOTAL PROGRAM												
BUDGET	109	150	207	276	344	434	522	617	729	818	958	1256
ACTUAL	109	152	207	276	345	447	546	649				

MATERIAL												
BUDGET	21	30	42	62	70	86	107	135	171	197	254	472
ACTUAL	22	31	42	62	72	92	117	147				

MANPOWER												
BUDGET	14	6	7	10	9	9	10	10	9	10	10	9
ACTUAL	14	6	7	9	9	10	11	11				

BUDGET

ACTUAL

189 NO. A6354

COST CATEGORIES		----- (\$0.0 K) -----	
		CURRENT MONTH	YEAR-TO-DATE
DIRECT SALARIES		\$ 27.2	\$ 191.1
MATERIALS, SERVICES AND OTHER COSTS		3.6	18.4
ADP SUPPORT		19.1	94.9
SUBCONTRACTS		0.0	0.0
TRAVEL		3.4	13.6
INDIRECT LABOR COSTS		35.5	241.7
GENERAL AND ADMINISTRATIVE		14.2	89.5
CAPITAL EQUIPMENT		0.0	0.0
TOTALS		\$ 103.0	\$ 649.2

See following page for financial comments

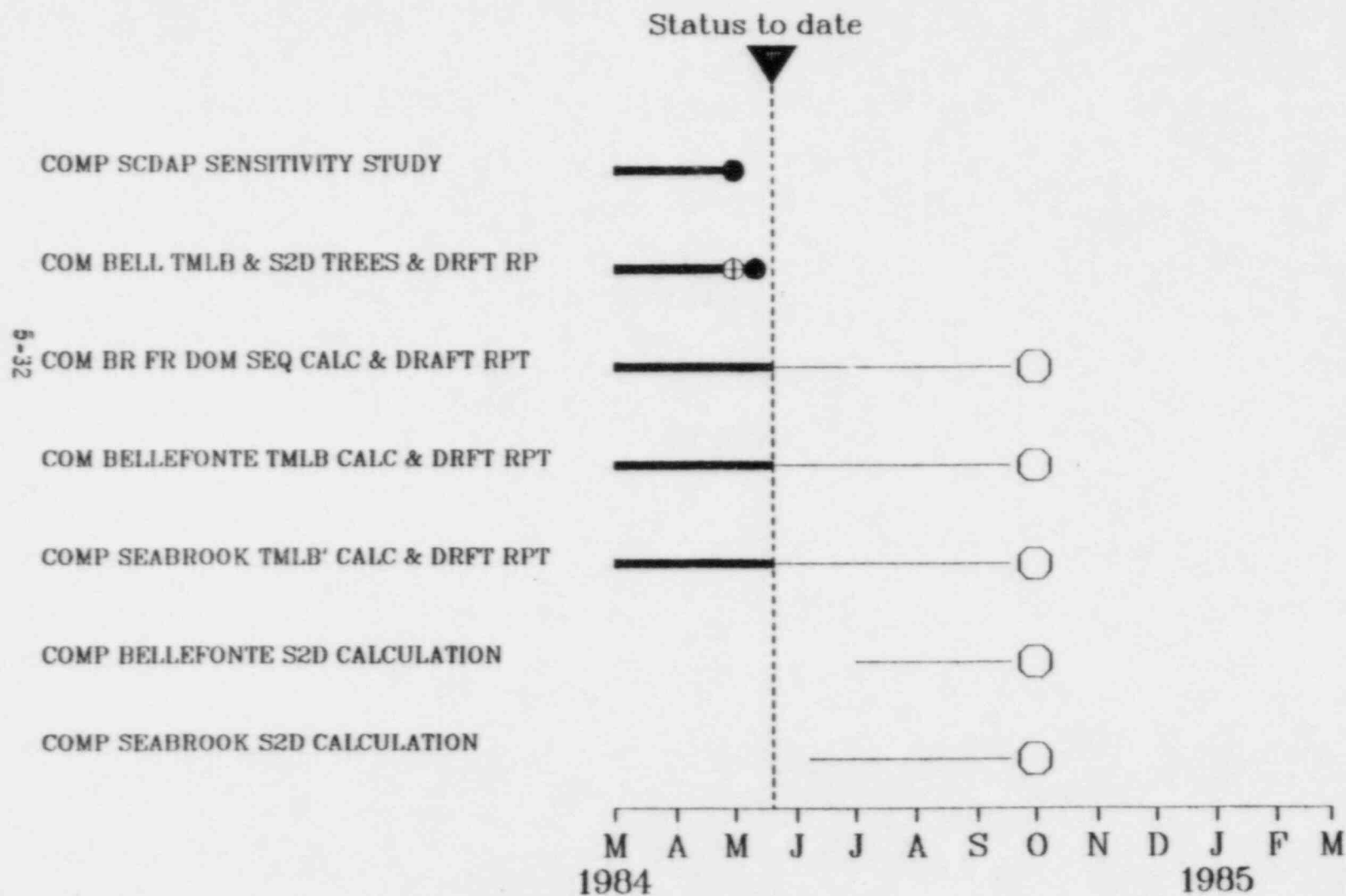
A6354

YTD VARIANCE: <32> (5%)

The variance is due to slightly higher labor cost than anticipated.
The current variance is expected to be corrected within two months.

DIVISION OF ACCIDENT EVALUATION

SEVERE ACCIDENT SEQUENCE ANALYSIS - (A6354)



Date April 25, 1984

To Dr. Bharat Agrawal From R. C. Gottula *RG*
Org. NRC Org. _____
A. _____ Address _____

SUBJECT: INEL SASA PROGRESS REPORT FOR APRIL 1984

PWR STUDIESBellefonte Plant

The draft letter report documenting the previous RELAP5/MOD1.6 station blackout (TMLB') analysis has been completed and will be sent out for peer review. SCDAP calculations have been completed, using the RELAP5/MOD1.6 results as boundary conditions, for low, medium, and high powered bundles. The results indicate that the bundles are steam starved, which limits the cladding oxidation rate, cladding heatup rate and hydrogen generated. Melting of zircaloy begins at about 52 minutes into the transient for the medium and high powered bundles. Sensitivity calculations were performed for core inlet flow rate. The results show that significant differences in cladding oxidation and heatup rate and hydrogen generation exist for higher core inlet flow rates.

The RELAP5/MOD2 steady state and initial TMLB' calculations have been completed. The results were very similar to the earlier RELAP5/MOD1.6 results. Development of a multi-channel core model was initiated. A portion of the TMLB' calculation will be repeated to provide better input to the SCDAP bundle analyses.

Sequence event trees were completed for the TMLB', S₂D, and ATWS accident sequences to provide a logical representation of the systems that are challenged by the accident and potential operator actions that could be taken to mitigate the consequences of the accident. The TMLB' tree indicates that without recovery of offsite or onsite power in less than 45 minutes, the only way to prevent core damage is to provide a built-in mechanism to rapidly decrease the primary pressure to accumulator and LPIS pressure prior to core damage. The S₂D tree indicates about an equal probability of a high pressure or low pressure core damage situation, the high pressure situation being reached with no mitigating actions and the low pressure situation being reached with a manual secondary feed and bleed operator action to LPIS pressure but with failure of the LPIS. Mitigating actions to prevent core damage for the S₂D sequence include a manual secondary feed and bleed operator action to bring the primary down to LPIS pressure or a built-in mechanism to decrease the primary pressure down to LPIS pressure before core damage. For the ATWS with loss of auxiliary feedwater, the probability of core damage is on the order of 10⁻¹² for the Bellefonte plant with an automatic auxiliary feedwater system. For other plants without the automatic feedwater system, the probability of core damage is still on the order of 10⁻⁸. Therefore, it will be recommended that this ATWS sequence not be analyzed by the SASA program for core damage due to the extremely low probability. A draft report documenting these sequence event trees will be sent out for peer review by May 7.

Notegram
April 25, 1984
To Dr. Bharat Agrawal
From R. C. Gottula
Page 2

Seabrook Plant

The RELAP5/MOD2 steady state and initial TMLB' calculations have been completed. The results will be used for boundary conditions to the SCDAP code. The SCDAP model of the Seabrook Plant has been completed and analysis of the TMLB' sequence has begun. Iterations between RELAP5 and SCDAP will be done to carry the calculation out to completion.

BWR STUDIES

Browns Ferry Plant

The draft report "Potential Effects of several ATWS Sequences on the Browns Ferry Nuclear Plant Unit 1 Containment" was issued for peer review and comment. The report contains results of CONTEMPT calculations that describe the thermal hydraulic response of the containment for various BWR ATWS sequences.

The draft report describing the BWR ATWS analysis methodology, the Browns Ferry RELAP5/MOD1.6 model, and RELAP5 results for several ATWS sequences has received internal review. The report will be issued for peer review in May.

A meeting was attended at BNL to accomplish an initial comparison of RELAP5 and RAMONA-3B results for the level-pressure control operator mitigated ATWS and to compare deck input. Preliminary comparisons indicate significant differences in HPCI flow rate, and therefore, core power as well. Differences in core two phase pressure loss dynamics are suspected as the cause. Further investigation of these differences is in progress.

Sensitivity analyses with CONTEMPT/LT-028 were performed to quantify the uncertainty in the ATWS time to high drywell pressure. Assuming steam breakthrough at about 177°F in the pool results in ~13 MIN to HDWP, compared to ~35 MIN if complete condensation is assumed to TSAT. Mechanistic analysis of this phenomenon is beyond the code capability.

The quality assurance of the TRAC-BD1/MOD1 input deck of Browns Ferry is approximately 80% complete. A steady state solution is being obtained in parallel.

Notegram
April 24, 1984
To Dr. Bharat Agrawal
From R. C. Gottula
Page 3

A SCDAP analysis of the ATWS sequence #551 (high pressure boiloff) was completed. The SCDAP model included the upper plenum. The results are about the same as shown earlier at the SASA review meeting on January 10, 1984. Core damage is predicted with delayed breach of the cladding. A draft letter report describing the SCDAP model and results of this analysis will be issued in May for peer review.

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1. Scheduled Milestones for April 1984

<u>Description</u>	<u>Due Date</u>	<u>Actual Date</u>
Complete Browns Ferry ATWS Sequence Event Tree NUREG Report	4/30/84	4/20/84C NUREG/CR-3596

2. Summary of Work Performed in April 1984

Pressurized Water Reactor Studies:

The draft report documenting the previous RELAP5/MOD1.6 station blackout analysis for the Bellefonte plant was completed and sent out for peer review. SCDAP calculations have been completed, using the RELAP5/MOD1.6 results as boundary conditions for low, medium, and high powered bundles. The results indicate that the bundles are steam starved, which limits the cladding oxidation rate, cladding heatup rate and hydrogen generated. Melting of zircaloy begins at about 52 minutes into the transient for the medium and high powered bundles. Sensitivity calculations were performed for core inlet flow rate. The results show that significant differences in cladding oxidation and heatup rate and hydrogen generation exist for higher core inlet flow rates.

The RELAP5/MOD2 steady state and initial TMLB' calculations for the Bellefonte plant have been completed. The results were very similar to the earlier RELAP5/MOD1.6 results.

2. Summary of Work Performed in April 1984 (Continued)

The RELAP5/MOD2 steady state and initial TMLB' calculations for the Seabrook plant have been completed. The results will be used for boundary conditions to the SCDAP code. The SCDAP model of the Seabrook plant has been completed and analysis of the TMLB' sequence has begun. Iterations between RELAP5 and SCDAP will be done to carry the calculation out to completion.

Sequence event trees were completed for the Bellefonte plant for the TMLB', S₂D, and ATWS accident sequences to provide a logical representation of the systems that are challenged by the accident and potential operator actions that could be taken to mitigate the consequences of the accident. A draft report documenting these sequence event trees will be sent out for peer review in May.

Boiling Water Reactor (BWR) Studies:

The draft report "Potential Effects of several ATWS Sequences on the Browns Ferry Nuclear Plant Unit 1 Containment" was issued for peer review and comment. The report contains results of CONTEMPT calculations that describe the thermal hydraulic response of the containment for various BWR ATWS sequences.

The draft report describing the BWR ATWS analysis methodology, the Browns Ferry RELAP5/MOD1.6 model, and RELAP5 results for two ATWS sequences has received internal review. The report will be issued for peer review in May.

A meeting at BNL was attended to accomplish an initial comparison of RELAP5 and RAMONA-3B results for the level-pressure control operator mitigated ATWS and to compare deck input. Preliminary comparisons indicate significant differences in HPCI flow rate, and therefore, core power as well. Investigation of the reasons for these differences is in progress.

Sensitivity analyses with CONTEMPT/LT-028 were performed to quantify the uncertainty in the ATWS time to high drywell pressure. Assuming steam breakthrough when the pool reaches about 177°F results in ~13 min to high drywell pressure, compared to ~35 min if complete condensation is assumed to 212°F. Mechanistic analysis of this phenomenon is beyond the code capability.

A comparison of CONTEMPT/LT-028 predictions and Marviken Series II Blowdown 18 test data was completed. The results indicated a very good comparison between the code predictions and the test data.

The quality assurance of the TRAC-BD1/MOD1 input deck of Browns Ferry is approximately 80% complete. A steady state solution is being obtained in parallel.

2. Summary of Work Performed in April 1984 (Continued)

Boiling Water Reactor (BWR) Studies (Continued)

A SCDAP analysis of the ATWS sequence #551 (high pressure boiloff) was completed. The SCDAP model included the upper plenum. The results are essentially the same as shown at the SASA review meeting on January 10, 1984. Core damage is predicted with delayed breach of the cladding. A draft letter report describing the SCDAP model and results of this analysis will be issued in May for peer review.

3. Scheduled Milestones for May 1984

None.

4. Summary of Work to be Performed in May 1984

PWR Studies:

Iterations between RELAP5/MOD2 and SCDAP will be initiated for the TMLB' analysis for the Bellefonte and Seabrook plants. Also, a RELAP5/MOD2 TMLB' calculation will be performed on both the Bellefonte and Seabrook plants to investigate potential depressurization rates due to an open PORV or a boiled open safety relief valve.

Documentation of the SCDAP sensitivity studies for the Bellefonte TMLB' analysis will begin. The Bellefonte sequence event tree draft report will be issued for peer review.

BWR Studies:

The draft report describing the Browns Ferry ATWS analysis methodology, the RELAP5/MOD1.6 model, and RELAP5 results of ATWS sequences will be issued for peer review. This information will be part of a NUREG report to be issued at the end of FY-1984.

Investigation of differences between RELAP5 and RAMONA will continue.

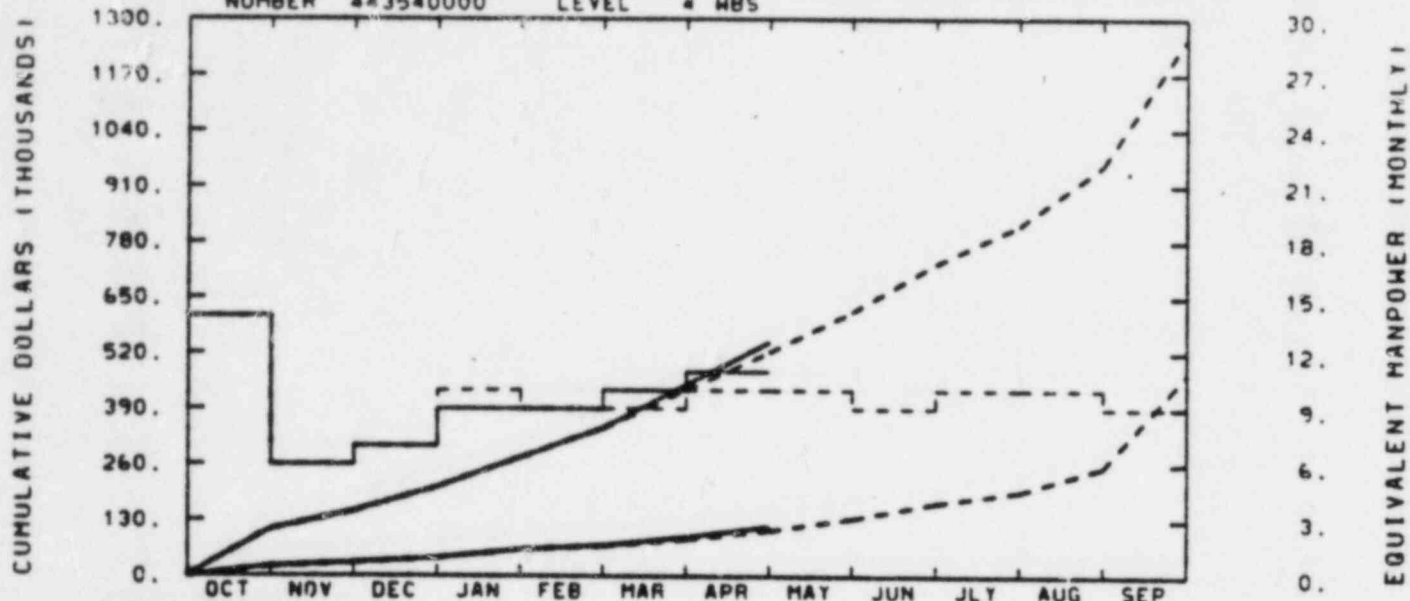
A draft letter report describing SCDAP results for the high pressure boiloff sequence (#551) will be issued for peer review in May. This information will be part of a NUREG report to be issued at the end of FY-1984.

5. Problems and Potential Problems

None.

RESPONSIBLE
MANAGER
L. LEACH DAE

EG&G IDAHO INC.
SEVERE ACCIDENT SEQUENCE A6354
NUMBER 443540000 LEVEL 4 WBS



TOTAL PROGRAM													
BUDGET	109	150	207	276	344	434	522	617	729	818	958	1256	
ACTUAL	109	152	207	276	345	447	546						

MATERIAL													
BUDGET	21	30	42	62	70	86	107	135	171	197	254	472	
ACTUAL	22	31	42	62	72	92	117						

MANPOWER													
BUDGET	14	6	7	10	9	9	10	10	9	10	10	9	
ACTUAL	14	6	7	9	9	10	11						

BUDGET

ACTUAL

189 NC. A6354

CCST CATEGORIES		(\$0.0 K)	
	CURRENT MCNTH	YEAR-TO-DATE	
DIRECT SALARIES	\$ 28.0	\$	163.9
MATERIALS, SERVICES AND OTHER COSTS	5.3		14.8
ADP SUPPORT	14.2		75.8
SUBCONTRACTS	0.0		0.0
TRAVEL	1.7		10.3
INDIRECT LAECR COSTS <i>selected on project common support</i>	36.5		206.2
GENERAL AND ADMINISTRATIVE	13.7		75.3
CAPITAL EQUIPMENT	0.0		0.0
TOTALS	\$ 99.4	\$	546.3

Contract Administration
planning & budget

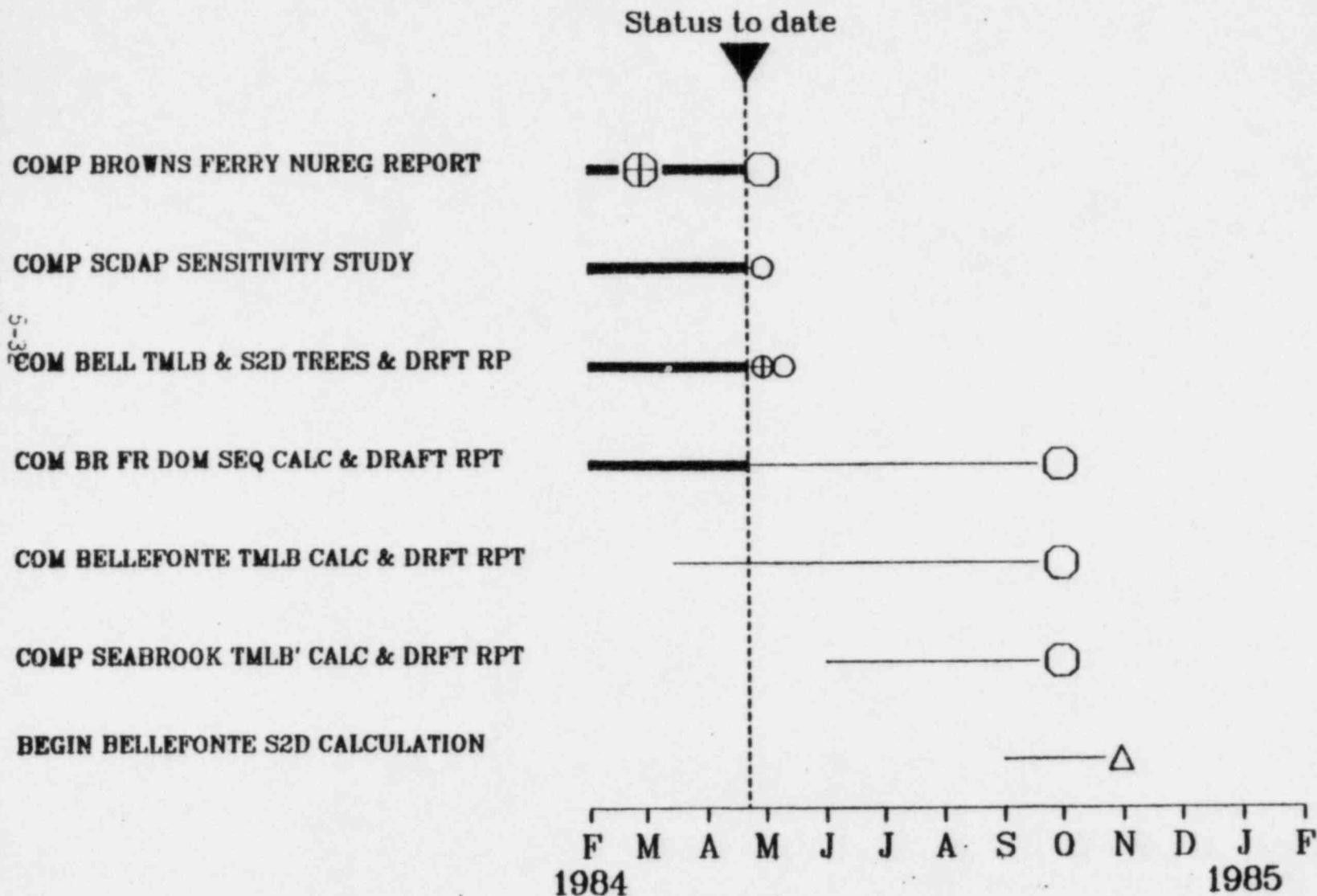
See following page for financial comments

A6354

YTD VARIANCE: <24> (4%)

DIVISION OF ACCIDENT EVALUATION

SEVERE ACCIDENT SEQUENCE ANALYSIS - (A6354)



A6354: Severe Accident Sequence Analysis Program (SASA)
EG&G Program/Technical Monitors: S. R. Behling/R. C. Gottula
DOE Technical Monitor: D. Majumdar
NRC Technical Monitor: [REDACTED]

The objective of this project is to use deterministic calculational tools to provide detailed analyses of severe accident sequences to support, verify, and modify probabilistic event sequences, to aid in the development of accident recovery strategies, to provide parametric values for experimental programs such as the Power Burst Facility (PBF) Severe Fuel Damage testing, and to point out the need for additional computer code development and experimental data.

1. Scheduled Milestones for March 1984

None.

2. Summary of Work Performed in March 1984

Pressurized Water Reactor Studies:

Steady state calculations using RELAP5/MOD2 were completed for the Bellefonte and Seabrook plants. Information needed to begin the TMLB' calculation on the Seabrook plant was identified to the NRC to be obtained from the Seabrook utility. Also, development of a SCDAP model for the Seabrook plant was started.

TVA's comments on the Bellefonte TMLB', S₂D and ATWS sequence event trees were incorporated into the event trees and the draft report.

Boiling Water Reactor (BWR) Studies:

Final comments were incorporated into the Browns Ferry ATWS sequence event tree report. The report will be published as a NUREG report in April.

Internal review of the draft report of CONTEMPT analyses for Browns Ferry ATWS sequences performed during CY 83 was completed. The report will be sent out for peer review in early April. The report describing the corresponding RELAP5 analyses is currently undergoing internal review.

RELAP5 and CONTEMPT analyses have been completed for the following Browns Ferry ATWS sequences:

- (a) Sequence 483: A totally unmitigated transient plus two variations of this sequence.
- (b) Sequence 465: An operator level control mitigated transient.

2. Summary of Work Performed in March 1984 (Continued)

Boiling Water Reactor (BWR) Studies (Continued)

(c) Sequence 439: An operator level and pressure control mitigated transient.

(d) Sequence 551: A high pressure boiloff transient.

Quality assurance of the Browns Ferry TRAC-BD1 deck continued and is about 40% complete.

The ORNL draft report (NUREG/CR-3470) documenting Browns Ferry ATWS sequence analyses was reviewed and comments were sent to the authors.

3. Scheduled Milestones for April 1984

<u>Description</u>	<u>Due Date</u>	<u>Actual Date</u>
Complete Bellefonte TMLB', S ₂ D, and ATWS sequence event trees and draft report and send out for peer review.	4/30/84	

4. Summary of Work to be Performed in April 1984

PWR Studies:

The RELAP5/MOD2 TMLB' analyses for Bellefonte will be completed and iterations with the SCDAP code will be initiated.

Fuel calculations for the Seabrook plant end of cycle 1 using the FRAPCON code will be completed and will provide boundary conditions for a RELAP5/MOD2 steady state calculation that will also be completed in April. An initial TMLB' analysis will be performed on Seabrook using RELAP5/MOD2 and iterations with SCDAP assessment will be initiated.

The Bellefonte sequence event tree draft report will be issued for peer review.

BWR Studies:

The draft report of the CONTEMPT analyses of containment response for Browns Ferry ATWS sequences performed during CY 83 will be issued for peer review.

Additional code assessment for the CONTEMPT code will be performed by comparing CONTEMPT predictions with Marviken blowdown data.

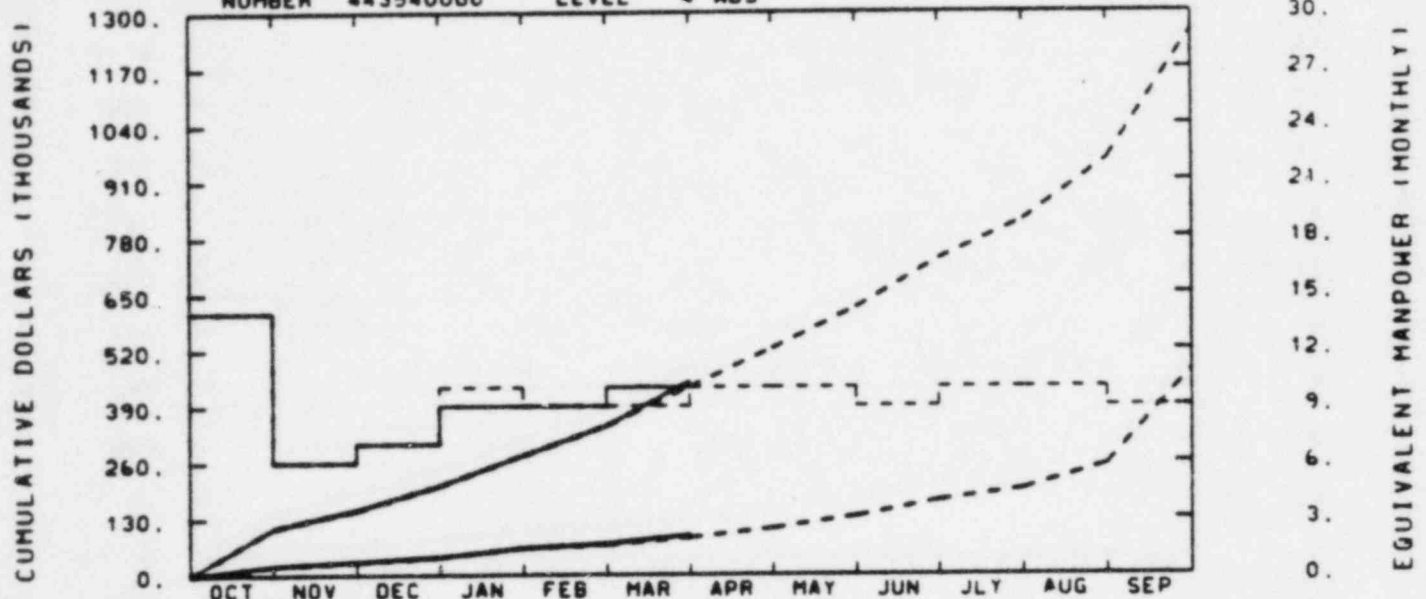
A meeting will be attended at BNL to do a preliminary comparison of RELAP5, RAMONA-3B, and LACP calculations for the Browns Ferry level-pressure control ATWS sequence and to compare deck input.

5. Problems and Potential Problems

None.

RESPONSIBLE
MANAGER
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SEVERE ACCIDENT SEQUENCE A6354
NUMBER 443540000 LEVEL 4 WBS



TOTAL PROGRAM												
BUDGET	109	150	207	276	344	434	522	617	729	818	958	1256
ACTUAL	109	152	207	276	345	447						

MATERIAL												
BUDGET	21	30	42	62	70	86	107	135	171	197	254	472
ACTUAL	22	31	42	62	72	92						

MANPOWER												
BUDGET	14	6	7	10	9	9	10	10	9	10	10	9
ACTUAL	14	6	7	9	9	10						

BUDGET

ACTUAL

189 NC. A6354

CCST CATEGORIES		(\$0.0 K)	
		CURRENT MONTH	YEAR-TO-DATE
DIRECT SALARIES		\$ 30.7	\$ 135.9
MATERIALS, SERVICES AND OTHER CCSTS		1.5	9.6
ADP SUPPORT		14.6	61.6
SUBCONTRACTS		0.0	C.C
TRAVEL		1.2	8.6
INDIRECT LABOR CCSTS		40.0	169.7
GENERAL AND ADMINISTRATIVE		14.1	61.6
CAPITAL EQUIPMENT		0.0	C.C
TOTALS		\$ 102.1	\$ 447.0

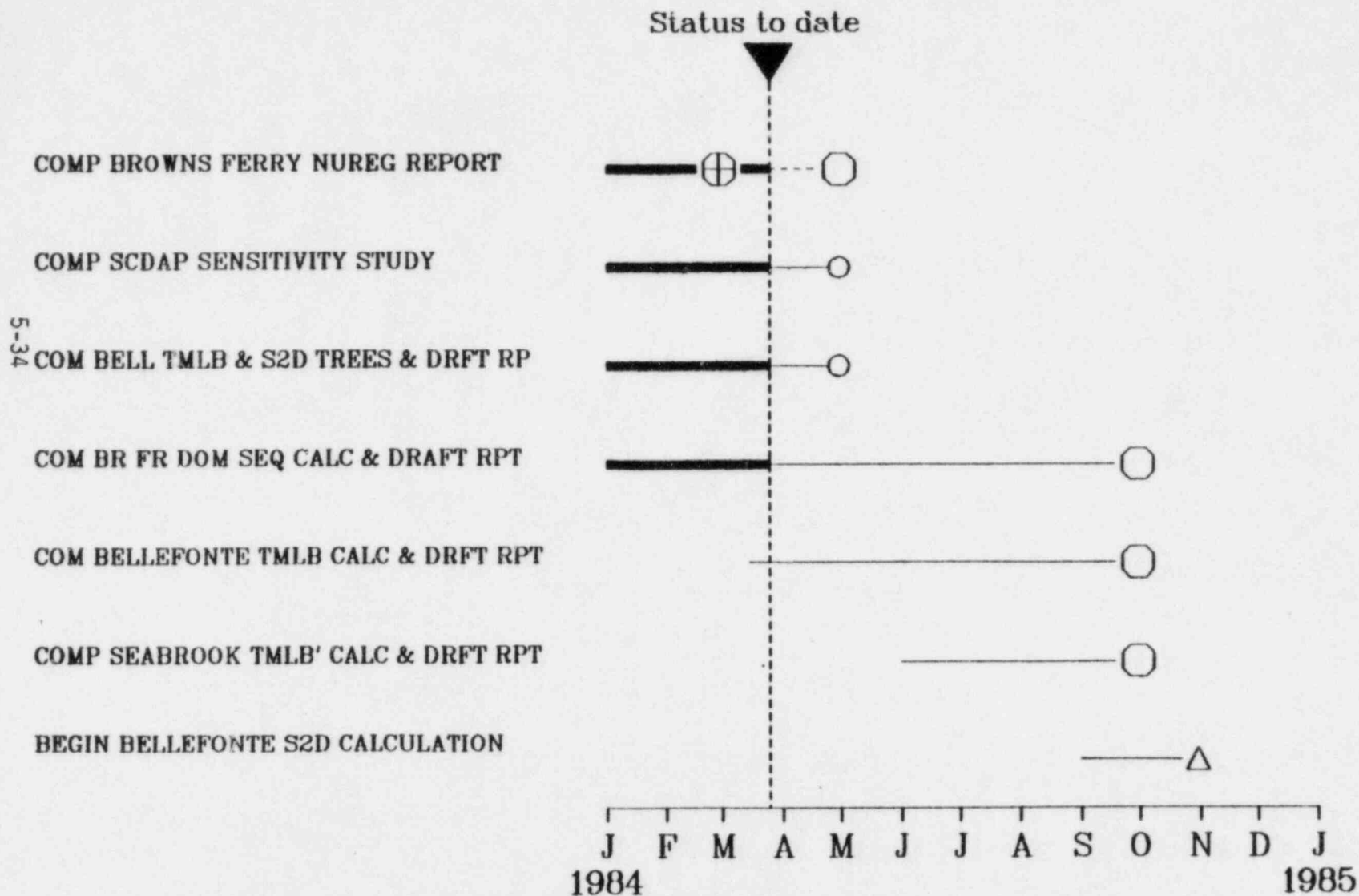
See following page for financial comments

A6354

YTD VARIANCE: <13> (3%)

DIVISION OF ACCIDENT EVALUATION

SEVERE ACCIDENT SEQUENCE ANALYSIS - (A6354)



A6354: Severe Accident Sequence Analysis Program (SASA)
 EG&G Program/Technical Monitor: R. C. Gottula
 DOE Technical Monitor: D. Majumdar
 NRC Technical Monitor: B. Agrawal ←

The objective of this project is to use deterministic calculational tools to provide detailed analyses of severe accident sequences to support, verify, and modify probabilistic event sequences, to aid in the development of accident recovery strategies, to provide parametric values for experimental programs such as the Power Burst Facility (PBF) Severe Fuel Damage testing, and to point out the need for additional computer code development and experimental data.

1. Scheduled Milestones for February 1984

<u>Description</u>	<u>Due Date</u>	<u>Actual Date</u>
Complete Browns Ferry Sequence Event Tree Report	2/28/84	4/15/84

2. Summary of Work Performed in February 1984

Pressurized Water Reactor Studies:

The Bellefonte RELAP5 deck was converted to RELAP5/MOD2 and initialization studies were begun.

The Bellefonte RELAP5/MOD1.6 deck was transmitted to Tennessee Valley Authority (TVA) as requested.

The Seabrook RELAP5/MOD2 and initialization studies were begun.

Quantification of the Bellefonte sequence event tree for the TMLB', S₂D, and anticipated transient without scram (ATWS) sequence were completed. Also, a draft for the report was completed.

Debugging of models in the severe core damage analysis package (SCDAP) code continued in support of the SCDAP predictions for the Bellefonte TMLB' study.

Boiling Water Reactor (BWR) Studies

A draft report of the RELAP5 and CONTEMPT models, performances appraisals, and Browns Ferry ATWS analyses performed during 1983 was completed.

2. Summary of Work Performed in February 1984 (Continued)

Boiling Water Reactor (BWR) Studies (Continued)

Quality assurance (QA) was begun on the TRAC-BD1/MOD1 input deck for Browns Ferry. The deck is being developed for BWR code assessment, and will be used to benchmark uncertainties in the Browns Ferry RELAP5 ATWS calculations.

A detailed calculation matrix of the ATWS dominant sequences was developed, based on the sequence event tree. Sequence 483 (a totally unmitigated ATWS transient) was completed using RELAP5.

Comments on the Browns Ferry ATWS sequence event tree report were received and are being incorporated into the final report.

A SCDAP/MOD1/Version 0 model of the upper plenum for Browns Ferry was developed. The model includes the stand pipes, separator dryers, and steam dome.

3. Scheduled Milestones for March 1984

None.

4. Summary of Work to be Performed in March 1984

PWR Studies:

Steady state calculations with RELAP5/MOD2 will be completed and TMLB' calculations on both the Bellefonte and Seabrook plants will be initiated. An iteration approach between RELAP5 and SCDAP will be used until the RELAP5/SCDAP link is completed.

Upon completion of the the steady state calculations, the Bellefonte RELAP5/MOD2 deck will be sent to TVA.

Internal review of the Bellefonte sequence event tree for the TMLB', S₂D, and ATWS sequence and the draft report will be completed.

SCDAP sensitivity studies for the Bellefonte TMLB' accident sequence will be completed contingent upon satisfactory debugging for certain SCDAP models.

4. Summary of Work to be Performed in March 1984 (Continued)

BWR Studies:

Internal review of the draft reports for RELAP5 and CONTEMPT calculations on Browns Ferry ATWS sequence will be completed.

Final comments will be incorporated into the Browns Ferry ATWS sequence event tree report and the report will be sent to printing.

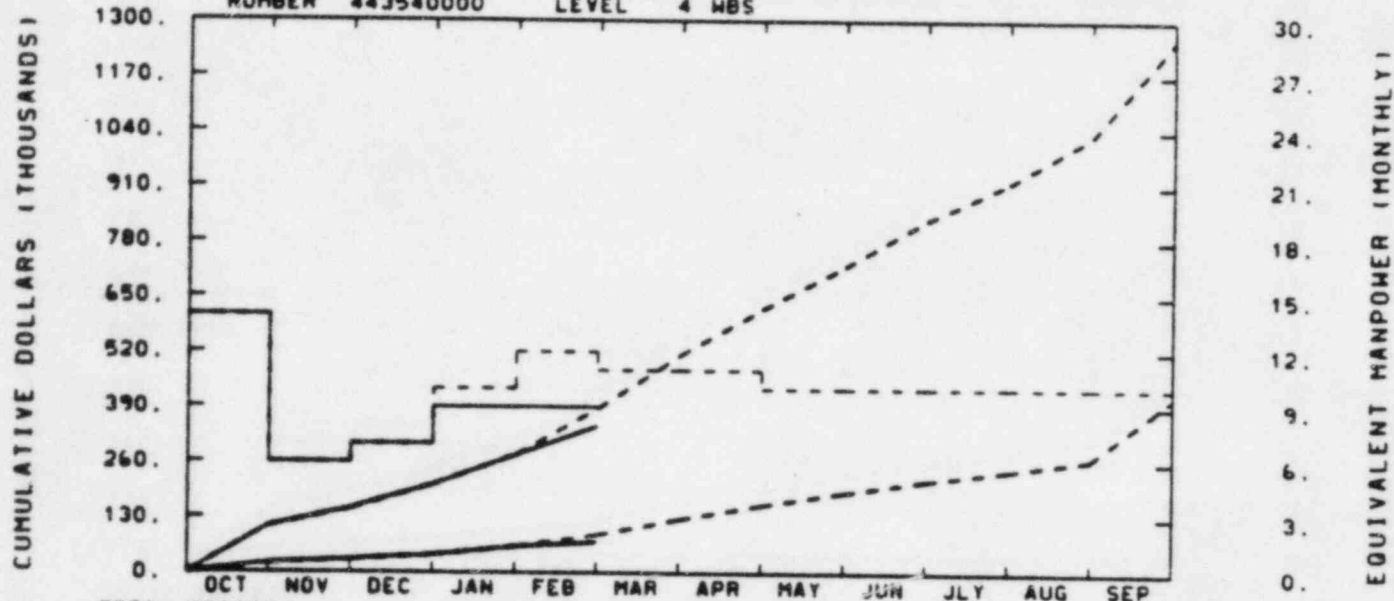
A CONTEMPT calculation for sequence 483 (a totally unmitigated ATWS transient) will be completed. NSSS and containment calculations of other dominant ATWS sequences will continue.

5. Problems and Potential Problems

The SCDAP code is in a developmental process. Problems have been incurred while trying to use the code for production calculations.

RESPONSIBLE
MANAGER
L LEACH DAE

EG&G IDAHO INC.
SEVERE ACCIDENT SEQUENCE A6354
NUMBER 443540000 LEVEL 4 MBS



TOTAL PROGRAM												
BUDGET	109	150	207	276	385	512	622	722	831	919	1029	1256
ACTUAL	109	152	207	276	345							

MATERIAL												
BUDGET	21	30	42	62	89	126	160	190	218	242	267	413
ACTUAL	22	31	42	62	72							

MANPOWER												
BUDGET	14	6	7	10	12	11	11	10	10	10	10	10
ACTUAL	14	6	7	9	9							

BUDGET

ACTUAL

189 NO. A6354

COST CATEGORIES	
DIRECT SALARIES	
MATERIALS, SERVICES AND OTHER COSTS	
ADP SUPPORT	
SUBCONTRACTS	
TRAVEL	
INDIRECT LABOR COSTS	
GENERAL AND ADMINISTRATIVE	
CAPITAL EQUIPMENT	

----- (\$0.0 K) -----	
CURRENT MONTH	YEAR-TO-DATE
\$ 22.0	\$ 105.3
0.0	8.0
8.6	46.9
0.0	0.0
0.1-	7.4
28.6	129.7
9.5	47.5
0.0	0.0
TOTALS	\$ 344.8

See following page for financial comments

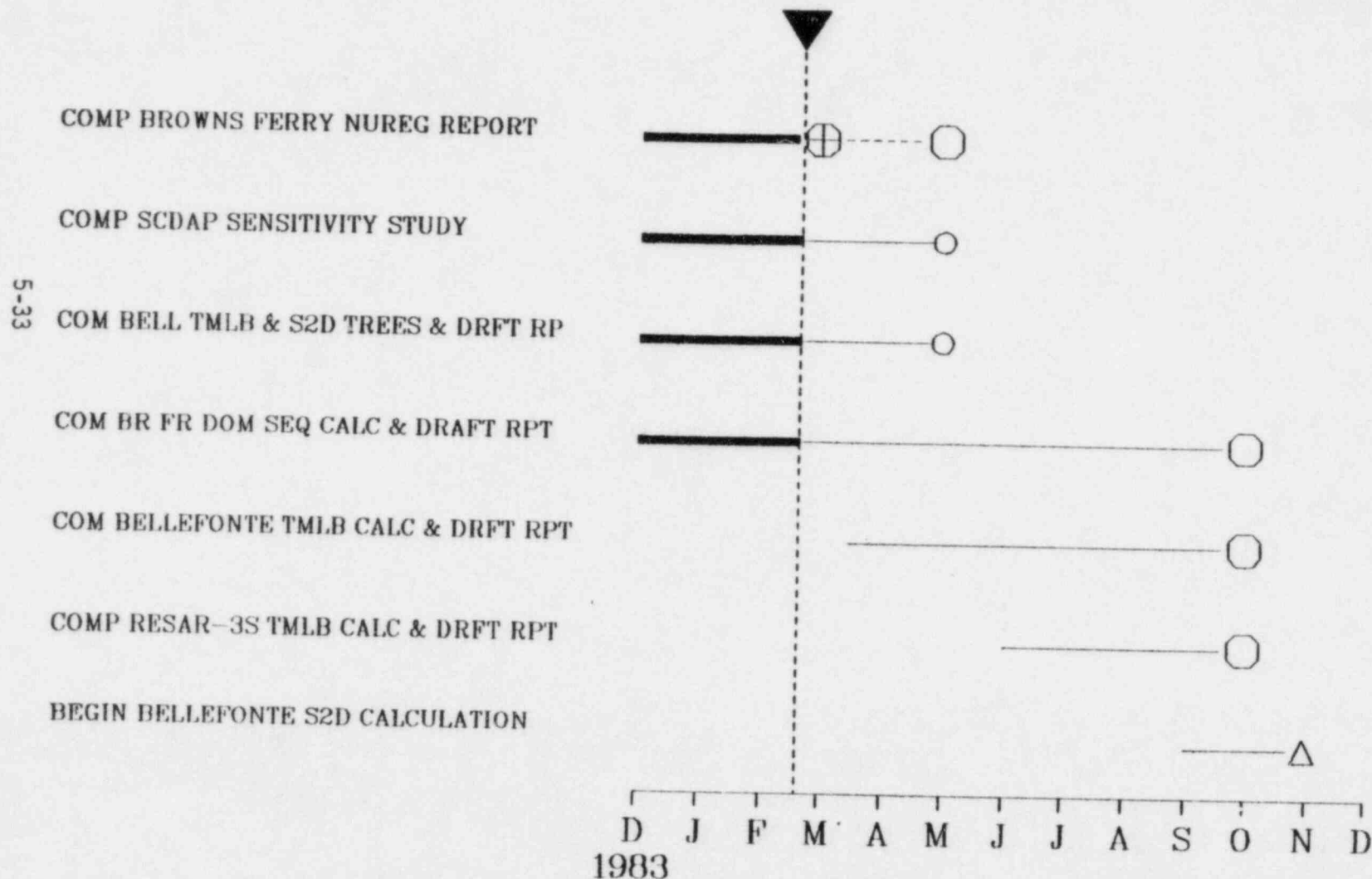
A6354

YTD VARIANCE: 40 (10%)

The \$40K underrun for February was comprised of \$17K non-labor charges and \$23K labor charges. Computer charges were \$13K underrun. There was no travel in February accounting for \$1K underrun and printing of the Browns Ferry sequence event tree report was postponed until April accounting for \$2K underrun. Several people worked part-time on projects other than A6354 accounting for most of the \$23K manpower underrun. Also, there was a three week delay in starting documentation on a TMI small break calculation and one week of effort not used by graphic arts to generate Bellefonte sequence event trees.

DIVISION OF ACCIDENT EVALUATION

SEVERE ACCIDENT SEQUENCE ANALYSIS - (A6354)



A6354: Severe Accident Sequence Analysis Program (SASA)

EG&G Program/Technical Monitor: R. C. Gottula

DOE Technical Monitor: P. E. Litteneker

NRC Technical Monitor: [REDACTED]

The objective of this project is to use deterministic calculational tools to provide detailed analyses of severe accident sequences to support, verify, and modify probabilistic event sequences, to aid in the development of accident recovery strategies, to provide parametric values for experimental programs such as the Power Burst Facility (PBF) Severe Fuel Damage testing, and to point out the need for additional computer code development and experimental data.

1. Scheduled Milestones for January 1984

None.

2. Summary of Work Performed in January 1984Pressurized Water Reactor Studies:

The draft report on "RELAP5 and SCDAP Analyses of a Station Blackout Transient for the Bellefonte Pressurized Water Reactor" was reviewed by management. Due to an error in the cladding ballooning model in SCDAP, the SCDAP calculation is being rerun and will be incorporated into the report during February.

Conversion of the Bellefonte and Seabrook decks from RELAP5/MOD1.6 to RELAP5/MOD2 continued and will be completed by mid-February.

The sensitivity study of the Bellefonte station blackout analyses continued, using the SCDSIMP version of the SCDAP/MODO code. The study investigates the influence of power level, mass, flux, and pressure on the magnitude and timing of source terms. Five runs were completed and analysis of the results began.

The TMLB¹, S₂D, and ATWS sequence event trees were refined and preliminary quantification was completed using representative information from another B&W plant (Crystal River) since data specific to Bellefonte is not available.

Boiling Water Reactor (BWR) Studies

A SCDAP core damage analysis was completed for the Browns Ferry high pressure boiloff transient (Sequence #551). A CONTEMPT analysis of the pressure suppression pool was also completed.

2. Summary of Work Performed in January 1984 (Continued)

Documentation of the RELAP5 and CONTEMPT models, comparisons with plant data, and ATWS calculations performed during 1983 is in progress. A draft report is to be completed by the end of February for review.

General

Two summaries have been written for presentation at the Fifth International Meeting on Thermal Nuclear Reactor Safety.

A presentation was given January 10-11 at the SASA technical interchange meeting on the progress and recent results of INEL SASA studies.

3. Scheduled Milestones for February 1984

<u>Description</u>	<u>Due Date</u>	<u>Actual Date</u>
Publish Browns Ferry NUREG Report	2/28/84	

4. Summary of Work to be Performed in February 1984

PWR Studies:

Conversion of the Bellefonte and Seabrook decks to RELAP5/MOD2 will be completed. Also, a steady state calculation will be completed for the Bellefonte plant. A method for accomplishing the soft link between RELAP5 and SCDAP will be resolved.

A copy of the RELAP5/MOD1.6 deck will be transmitted to the Tennessee Valley Authority.

Revision of the draft report "RELAP5 and SCDAP Analyses of a Station Blackout Water Reactor" will be completed.

The SCDAP sensitivity study will be extended to include a comparison of one fuel bundle versus a core wide model.

Quantification of the Bellefonte sequence event trees will be completed.

BWR Studies:

Documentation of the RELAP5 and CONTEMPT models and results of calculations during 1983 will be completed and ready for review.

Work will begin to QA the Browns Ferry TRAC deck.

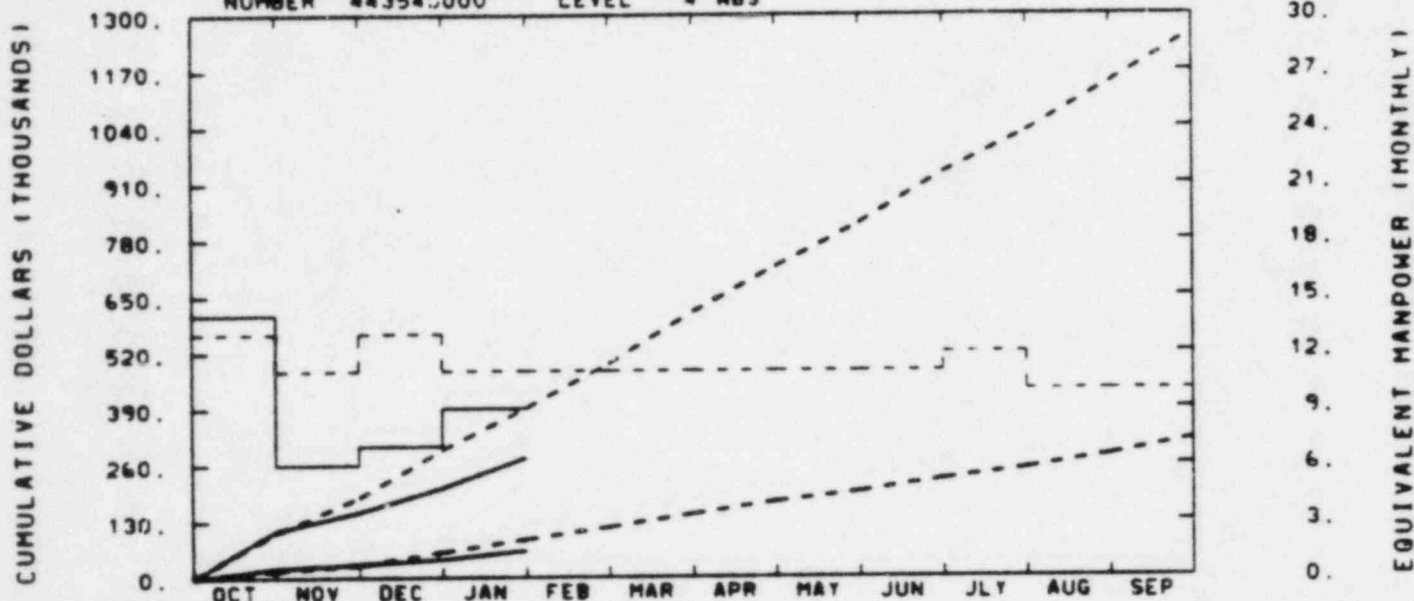
5. Problems and Potential Problems

RELAP5/MOD2 is a new code and may cause some delays in initiating steady state analysis.

The SCDAP code is in a developmental process. Problems have been incurred while trying to use the code for production calculations.

RESPONSIBLE
MANAGER
L LEACH DAE

EG&G IDAHO INC.
SEVERE ACCIDENT SEQUENCE A6354
NUMBER 44354.000 LEVEL 4 HBS



TOTAL PROGRAM												
BUDGET	104	185	295	391	492	612	717	817	934	1028	1141	1256
ACTUAL	109	152	207	276								

MATERIAL												
BUDGET	16	29	59	88	115	145	173	196	225	250	281	317
ACTUAL	22	31	42	62								

MANPOWER												
BUDGET	13	11	13	11	11	11	11	11	11	12	10	10
ACTUAL	14	6	7	9								

BUDGET

ACTUAL

189 NO. A6354

COST CATEGORIES		(\$0.0 K)	
	CURRENT MONTH		YEAR-TO-DATE
DIRECT SALARIES	\$ 18.5	\$	83.3
MATERIALS, SERVICES AND OTHER COSTS	0.9		8.1
ADP SUPPORT	16.0		38.3
SUBCONTRACTS	0.0		0.0
TRAVEL	0.4		7.5
INDIRECT LABOR COSTS	24.0		101.0
GENERAL AND ADMINISTRATIVE	9.5		38.1
CAPITAL EQUIPMENT	0.0		0.0
TOTALS	\$ 69.3	\$	270.3

See following page for financial comments

A6354

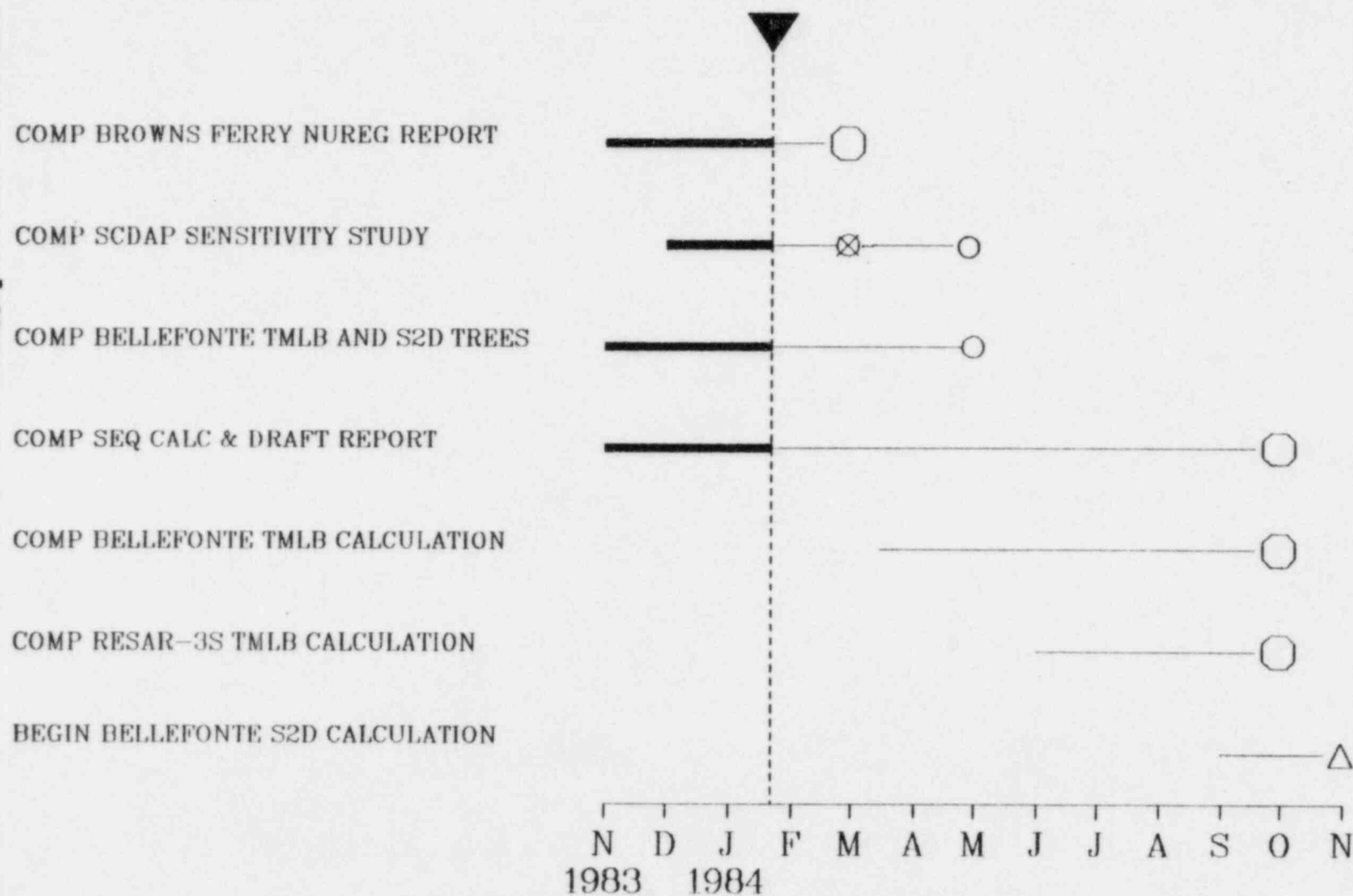
YTD VARIANCE: 115 (29%)

Manpower and computer costs were originally spread fairly even through the year, since November four people have been working part-time on other tasks accounting for 600 hours less-than-anticipated through January. Also, a SCDAP sensitivity study was postponed for two months accounting for 640 hours less-than-anticipated. Computer costs are low thus far due to unavailability of RELAP5/MOD2. Manpower and computer costs will be significantly higher from March through June. A new cost spread has been generated and given to plans and budgets.

DIVISION OF ACCIDENT EVALUATION

SEVERE ACCIDENT SEQUENCE ANALYSIS - (A6354)

5-32



A6354: Severe Accident Sequence Analysis Program (SASA)

EG&G Program/Technical Monitor: T. R. Charlton/R. J. Dallman
 DOE Technical Monitor: D. Majumdar
 NRC Technical Monitor: B. Agrawal

The objectives of this project are to use deterministic calculational tools to provide detailed analyses of severe accident sequences to support, verify, and modify probabilistic event sequences, to aid in the development of accident recovery strategies, to provide parametric values for experimental programs such as the Power Burst Facility (PBF) Severe Fuel Damage testing, to point out the need for additional computer code development and experimental data, and to benchmark risk codes.

1. Summary of Work Performed During April, 1985

SASA personnel at EG&G Idaho attended the Severe Fuel Damage Program Review Meeting, April 16-20, 1985 in Idaho Falls. EG&G Idaho personnel participated in the MELPROG Workshop at Sandia National Laboratory on April 24-26, 1985.

Boiling Water Reactor (BWR) Studies:

A calculation was completed using the SCDAP computer program to benchmark hydrogen generation results from a TQUV sequence. The sequence was simulated for the Grand Gulf plant, and actions were modeled to maximize the hydrogen generation rate. The SCDAP calculation modeled a single 8 x 8 fuel assembly having a power peak to a core average of 1.21. SCDAP predicted a maximum hydrogen generation rate of 0.0047 lbm/s.

Pressurized Water Reactor (PWR) Studies:

The TRAC-PF1 model of the RESAR reactor vessel was renodalized to three core and upper plenum channels. A steady state calculation was begun to duplicate conditions in the Seabrook RELAP5 model. The models will be initialized similarly prior to performing vessel circulation calculations.

Code input models for Seabrook and Bellefonte were prepared for the linked SCDAP/RELAP5/TRAP-MELT computer code. The input models were checked out on the CRAY computer, and are ready to perform steady state calculations. These calculations will provide the starting point for transient analyses of station blackout and small break loss of coolant accident analyses.

The CONTAIN code was made operational on the CYBER, and a checkout model was completed.

#18

2. Summary of Work to be Performed During May, 1985

PWR Studies:

Comparisons of vessel circulation patterns during a high pressure bolloff will be made with RELAP5 and TRAC-PF1.

Results of Bellefonte TMLB' calculations with pump seal leakage will be transmitted to Sandia National Laboratory.

Checkout of CONTAIN 1.01 will continue and assessment of data available for model input decks will begin.

3. Problems and Potential Problems

None.

A6354: Severe Accident Sequence Analysis Program (SASA)

EG&G Program/Technical Monitor: T. R. Charlton/R. C. Gottula

DOE Technical Monitor: D. Majumdar

NRC Technical Monitor: [REDACTED]

The objectives of this project are to use deterministic calculational tools to provide detailed analyses of severe accident sequences to support, verify, and modify probabilistic event sequences, to aid in the development of accident recovery strategies, to provide parametric values for experimental programs such as the Power Burst Facility (PBF) Severe Fuel Damage testing, to point out the need for additional computer code development and experimental data, and to benchmark risk codes.

1. Summary of Work Performed During March, 1985

EG&G Idaho personnel attended the CONTAIN 1.0 workshop at Sandia National Laboratory on March 4-8.

Boiling Water Reactor (BWR) Studies:

The Browns Ferry TRAC-BD1 deck was converted to BFO (Version 23J). Renodalization of the core was accomplished to match input from BNL. A steady state was begun, which will form the basis for incorporating 1-D neutronic cross sections into TRAC-BFO.

An analysis is in progress to determine hydrogen generation rates for a TQV sequence in a BWR6 MARK III plant using the SCDAP code. The results will be compared with IDCOR calculations and NRC calculations using the MARCH 2.0 code.

Pressurized Water Reactor (PWR) Studies:

A 3-channel RELAP5 reactor vessel model for the Seabrook plant was completed. Steady state calculations are being performed. The TRAC model of the RESAR reactor vessel is being converted from 2 to 3 channels in the core and upper plenum, similar to the Seabrook RELAP5 model. Comparisons of natural circulation patterns in the vessel will be made between the two codes.

The auxiliary feedwater system model development for the Bellefonte plant continued.

2. Summary of Work to be Performed During April, 1985

EG&G Idaho personnel will attend a MELPROG workshop in Albuquerque on April 24-26.

2. Summary of Work to be Performed During April, 1985 Continued

BWR Studies:

Variations to the TRAC-BFO reference case will be run to provide the thermal hydraulic basis for fitting of the 1-D neutronic cross section coefficients.

The SCDAP analysis of hydrogen generation rates and comparisons to MARCH 2.0 and IDCOR results will be completed.

PWR Studies:

Renodalization of the RESAR TRAC model will be completed. Comparisons of transient results obtained from this model and the Seabrook RELAP5 model will be made.

The Bellefonte auxiliary feedwater system model will be completed.

3. Problems and Potential Problems

None.

4. Cost Breakdown

Individual total cost and variance explanations are provided on the Financial and Progress and Status Summary.

189 No. A6354

<u>Cost Categories</u>	<u>(\$0.0K)</u>	
	<u>Current Month</u>	<u>Year-to-Date</u>
Direct Salaries	\$ 18.0	\$ 122.9
Materials, Services and Other Costs	1.0	3.8
ADP Support	8.8	102.6
Subcontracts	0.0	0.0
Travel	1.0	11.6
Indirect Labor Costs	26.7	181.5
General and Administrative	9.2	69.6
Capital Equipment	0.0	0.0
TOTALS	<u>\$ 64.7</u>	<u>\$ 492.0</u>

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PER STATUS NO	NODE STATUS	PROJECT LABEL	PERSON	DUE DATE	RSTD-CR	NEW DATE	COMP DATE	COMP REF
6354 50	21-10 TASK COMPLETE	Comp Brns Fry Dft NUREG Rpt	RC6	022885			022885	TRC-24-85
6354 1	21-22 A/S, Not Yet Started	Comp Browns Ferry ATWS NUREG	RC6	092885				
6354 3	21-11 A/S, Analysis Underway	Comp TRAC 1-D ATWS Stdy	RC6	092885				
6354 1	21-23 A/S, Not Yet Started	Brnw Fry Core Dag Anal Using RELAPS/SCDAP/TRAP-MELT	RC6	092885				
6354 50	21-15 TASK COMPLETE	Bellef, Multi-D R5 Ckout Runs MELT Decks/Bellef & Sebr	RC6	021585			021585	TRC-33-85
6354 15	21-24 B/S, Awaiting Add'l Data	Add Aux Feed Sys To Bellefonte Plant	RC6	031585				
6354 15	21-25 B/S, Awaiting Add'l Data	Bellefonte Prel S2D Anal W/RELAPS	RC6	041585				
6354 3	21-26 A/S, Analysis Underway	Compare Seabrook 3 Channel Model W/TRAC Model	RC6	041585				
6354 2	21-14 A/S, Input Data Available	Set Up & Checkout RELAPS/SCDAP TM Decks for Belle & Seabr	RC6	041585				
6354 1	21-16 A/S, Not Yet Started	Belle Base Case TMLB 1 Using RELAP/SCDAP/TRAP-MELT	RC6	073185				
6354 1	21-19 A/S, Not Yet Started	Seabrook Base Case TMLB1 Using RELAP/SCDAP/TRAP-MELT	RC6	073185				
6354 1	21-17 A/S, Not Yet Started	Belle Base Case S2D Using RELAPS/SCDAP/TRAP-MELT	RC6	092885				
6354 1	21-20 A/S, Not Yet Started	Seabrook Base Case S2D Using RELAPS/SCDAP/TRAP-MELT	RC6	092885				

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JER SIRIUS NO	NODE STATUS	PROJECT LABEL	PERSON	DUE DATE	RSTD-CR	NEW DATE	COMP DATE	COMP REF
6354 1	21-18 A/S, Not Yet Started	Bellefonte TMLB & S2D Op Act Analysis	RC6	123185				
6354 1	21-21 A/S, Not Yet Started	Seabrook TMLB & S2D Operator Action Analysis	RC6	123185				
6354 1	21-12 A/S, Not Yet Started	Comp. Bellefonte Drft NUREG/ TMLB1 S2D	RC6	033186				
6354 1	21-13 A/S, Not Yet Started	Comp Seabrook Drft NUREG/ TMLB1 & S2D	RC6	033186				

A6354: Severe Accident Sequence Analysis Program (SASA)

EG&G Program/Technical Monitor: T. R. Charlton/R. C. Gottula
DOE Technical Monitor: D. Majumdar
NRC Technical Monitor: B. Agrawal

The objectives of this project are to use deterministic calculational tools to provide detailed analyses of severe accident sequences to support, verify, and modify probabilistic event sequences, to aid in the development of accident recovery strategies, to provide parametric values for experimental programs such as the Power Burst Facility (PBF) Severe Fuel Damage testing, to point out the need for additional computer code development and experimental data, and to benchmark risk codes.

1. Summary of Work Performed During February, 1985

EG&G Idaho personnel attended the SASA Program Review meeting on February 20-21. Presentations were made on the status and results of the INEL PWR and BWR SASA work. EG&G Idaho presented the capabilities of the SCDAP/RELAP5/TRAP-MELT integrated code and compared models with the MELRPI code.

Pressurized Water Reactor (PWR) Studies:

A TMLB' calculation was performed for the Bellefonte plant, using RELAP5, in which pump seal leakage was modeled. A nominal 475 gpm leak in each pump resulted in an earlier core heatup than with no leakage, but liquid remained in the loop seals at the end of the calculation.

Development of a 3-channel reactor vessel for the Seabrook RELAP5 model began. The Oconee reactor vessel TRAC model obtained from LANL was deemed unsuitable for comparison with the Bellefonte RELAP5 results because it contains only one ring in the core. The RELAP5 results using the Seabrook model will be compared with a 3-D TRAC model of the RESAR reactor vessel.

Development of an auxiliary feedwater system model for the Bellefonte plant, to be used in the S₂D analyses, was initiated.

Preliminary documentation of the Seabrook and Bellefonte pump seal leakage calculations, the Bellefonte 2-D reactor vessel modeling, and the 2-D hot leg modeling was completed.

Any work with the linked SCDAP/RELAP5/TRAP-MELT code has been deferred until fission product transport and deposition models are incorporated.

4. Cost Breakdown

Individual total cost and variance explanations are provided on the Financial and Progress and Status Summary.

189 No. A6354

<u>Cost Categories</u>	<u>(\$0.0K)</u>	
	<u>Current Month</u>	<u>Year-to-Date</u>
Direct Salaries	\$ 23.6	\$ 104.9
Materials, Services and Other Costs	0.4	2.8
ADP Support	24.4	93.7
Subcontracts	0.0	0.0
Travel	3.6	10.6
Indirect Labor Costs	34.9	154.8
General and Administrative	14.3	60.4
Capital Equipment	0.0	0.0
TOTALS	<u>\$ 101.2</u>	<u>\$ 427.2</u>

Reactor Systems Technology Division Monthly Milestone Update 1984 - 1985

Through February 24, 1985

A NUMBER STATUS NO	NODE STATUS	PROJECT LABEL	PERSON	DUE DATE	REPD-CR	NEW DATE	COMP DATE	COMP REF
6354 1	21-20 A/S, Not Yet Started	Seabrook Base Case TMLB1 Using RELAP5/SCDAP/TRAP-Melt	RC6	073185				
6354 1	21-19 A/S, Not Yet Started	Bellefonte TMBB & S2D Op Act Analysis	RC6	123185				
6354 1	21-21 A/S, Not Yet Started	Seabrook TMLB & S2D Operator Action Analysis	RC6	123185				
6354 1	21-12 A/S, Not Yet Started	Comp. Bellefonte Drift NUREG/ TMLB1 S2D	RC6	033186				
6354 1	21-13 A/S, Not Yet Started	Comp Seabrook Drift NUREG/ TMLB1 & S2D	RC6	033186				

A6354: Severe Accident Sequence Analysis Program (SASA)

EG&G Program/Technical Monitor: T. R. Charlton/R. C. Gottula

DOE Technical Monitor: D. Majumdar

NRC Technical Monitor: [REDACTED]

The objectives of this project are to use deterministic calculational tools to provide detailed analyses of severe accident sequences to support, verify, and modify probabilistic event sequences, to aid in the development of accident recovery strategies, to provide parametric values for experimental programs such as the Power Burst Facility (PBF) Severe Fuel Damage testing, to point out the need for additional computer code development and experimental data, and to benchmark risk codes.

1. Summary of Work Performed During [REDACTED]Boiling Water Reactor (BWR) Studies:

Browns Ferry ATWS results using the RELAP5, CONTEMPT, SCDAP, and FRAP-T codes were incorporated into the draft NUREG report. Calculations enveloping boron effectiveness indicate a large uncertainty in final PSP temperature with a 50 gpm injection rate. However, when using an 86 gpm SLCS injection rate, final predicted PSP temperatures are well below the 200°F rulemaking violation, even when considering degraded boron mixing efficiency.

Pressurized Water Reactor (PWR) Studies:

A TMLB' calculation was conducted on the Seabrook plant with RELAP5 to investigate the effect of pump seal leakage on loop seal behavior and flow through the steam generator tubes. The results indicated the loop seal would not be broken, even with 475 gpm per pump seal leakage.

Nodalization studies of the Bellefonte vessel using RELAP5 were completed. A 3-channel model was compared to a 5-channel model. The results were nearly the same relative to natural circulation flow patterns. Therefore, a 3-channel model will be used for future Bellefonte and Seabrook analyses.

COBRA-TF calculations of flow patterns in a hot leg pipe were completed. Results of the 2-D model were nonphysical, so no assessment of the RELAP5 model could be performed.

2. Summary of Work to be Performed During February, 1985

BWR Studies:

The Browns Ferry ATWS draft NUREG report will receive internal review and be released for peer review by February 28.

Preparations for a presentation at the February SASA Program Review meeting will be made.

PWR Studies:

Checkout of the RELAP5/SCDAP model of the Bellefonte plant is anticipated to begin on the CRAY computer.

Investigation of reactor vessel modeling of natural circulation phenomena will continue using a TRAC model of the Oconee vessel obtained from LANL. Development of a 3-channel reactor vessel model for Seabrook will begin.

A scoping calculation of a TMLB' transient with reactor coolant pump seal leakage will be performed for the Bellefonte plant using RELAP5.

3. Problems and Potential Problems

PWR TMLB' and S₂D analyses on the Bellefonte and Seabrook plants are awaiting use of the RELAP5/SCDAP linked code on the CRAY computer.

4. Cost Breakdown

Individual total cost and variance explanations are provided on the Financial and Progress and Status Summary.

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<u>Cost Categories</u>	<u>(0.0K)</u>	
	<u>Current Month</u>	<u>Year-to-Date</u>
Direct Salaries	\$ 19.2	\$ 81.2
Materials, Services and Other Costs	0.1	2.4
ADP Support	24.0	69.3
Subcontracts	0.0	0.0
Travel	2.5	7.0
Indirect Labor Costs	28.3	119.9
General and Administrative	12.2	46.1
Capital Equipment	0.0	0.0
	-----	-----
T O T A L S	\$ 86.3	\$ 325.9
	=====	=====

REF SRS-DB NO	MODE STATUS	PROJECT LABEL	PERSON	DUE DATE	REP'D-OR	NEW DATE	COMP DATE	COMP REF
6354 A	21-10 A/S, Not in Preparation Underway	Comp Brns Fm. Dft KUREB Rot	RCB	020885				
6354 C	21-11 A/S, Analysis Underway	Complete TRAC 1-D ATWS Stov	RCB	092885				
6354 I	21-12 A/S, Not Yet Started	Comp. Belle- Drft Wureg/TMLB1 S2D	RCB	092885				
6354 I	21-13 A/S, Not Yet Started	Comp. Belle- Drft Wureg/TMLB1 S2D	RCB	092885				
6354 2	21-14 Input Data Available	Setup & Ckout RELAP5/SCDAP/TRA MELT Decks/Bellef & Sepr	RCB	021885				
6354 *	21-15 A/S, Analysis Underway	Bellef, Multi-D RE Ckout Runs	RCB	021885				
6354 I	21-16 A/S, Not Yet Started	Belle Base Case TMLB Using RELAP/SCDAP	RCB	063185				
6354 I	21-17 A/S, Not Yet Started	Belle Base Case S2D Using RELAP5/SCDAP	RCB	063185				
6354 I	21-18 A/S, Not Yet Started	Belle TMLB & S2D Op Act Anal	RCB	063185				
6354 I	21-19 A/S, Not Yet Started	Seabrook Base Case TMLB Using RELAP5/SCDAP	RCB	063185				
6354 I	21-20 A/S, Not Yet Started	Seabrook Base Case S2D Using RELAP5/SCDAP	RCB	063185				
6354 I	21-21 A/S, Not Yet Started	Seabrook TMLB & S2D Operator Action Analysis	RCB	063185				

A6354: Severe Accident Sequence Analysis Program (SASA)
EG&G Program/Technical Monitor: T. R. Charlton/R. C. Gottula
DOE Technical Monitor: D. Majumdar
NRC Technical Monitor: B. Agrawal

The objectives of this project are to use deterministic calculational tools to provide detailed analyses of severe accident sequences to support, verify, and modify probabilistic event sequences, to aid in the development of accident recovery strategies, to provide parametric values for experimental programs such as the Power Burst Facility (PBF) Severe Fuel Damage testing, to point out the need for additional computer code development and experimental data, and to benchmark risk codes.

1. Summary of Work Performed During [REDACTED]

Pressurized Water Reactor (PWR) Studies:

Modeling schemes using RELAP5 to simulate postulated natural circulation flow in the Bellefonte reactor vessel for a TMLB' sequence, are being investigated. A 5 channel reactor vessel model has been completed to determine nodalization convergence of an earlier 3 channel vessel model. Steady state calculations with the 5 channel model are currently being completed.

For the Seabrook plant, postulated single phase natural circulation flow patterns in the hot leg pipe are being investigated with the COBRA-TF code. A 2-D model has been developed but no results are available yet. Also, a 3-D model of the hot leg pipe is being developed. Results from these models will be compared with Argonne results using the COMMIX code at a future time.

A phone conversation was held with Bharat Agrawal and Rene Audette to discuss their comments on the recent Bellefonte and Seabrook station blackout reports.

Copies of the Bellefonte RELAP5 model workbooks were sent to Dr. James Lime of LANL and Mr. Irby of TVA per NRC request.

Boiling Water Reactor (BWR) Studies:

The final RELAP5 calculations of postulated ATWS transients at Browns Ferry Nuclear Plant Unit (BFNP1) were completed. Analysis of the results is in progress, and sensitivity calculations are planned. The CONTEMPT/LT-028 calculation of Sequence 483/551 is nearly complete, and pertinent information was provided for the RELAP5 analysis. The predicted power spikes during automatic operation were somewhat different than those previously predicted. Thus, a FRAP-T6 input was set up to analyze the possibility of fuel damage during the spikes.

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A NUMBER STATUS NO	NODE STATUS	PROJECT LABEL	PERSON	DUE DATE	REFD-CR	NEW DATE	COMP DATE	COMP REF
6354 4	21-10 A/S, Rpt in Preparation Underway	Comp Brns Fry Dft NUREG Rpt	RCG	022885				
6354 3	21-11 A/S, Analysis Underway	Complete TRAC 1-D ATWS Stdy	RCG	092885				
6354 1	21-12 A/S, Not Yet Started	Comp. Bellef Drft Nureg/TMLB1 S2D	RCG	092885				
6354 1	21-13 A/S, Not Yet Started	Comp Seabr Drft Nureg/TMLB1 & S2D	RCG	092885				
6354 2	21-14 Input Data Available	Setup & Ckout RELAP5/SCDAP/TRA MELT Decks/Bellef & Sebr	RCG	021585				
6354 3	21-15 A/S, Analysis Underway	Bellef, Multi-D R5 Ckout Runs	RCG	021585				
6354 1	21-16 A/S, Not Yet Started	Belle Base Case TMLB Using RELAP/SCDAP	RCG	053185				
6354 1	21-17 A/S, Not Yet Started	Belle Base Case S2D Using RELAP5/SCDAP	RCG	063085				
6354 1	21-18 A/S, Not Yet Started	Belle TMBB & S2D Op Act Anal	RCG	083185				
6354 1	21-19 A/S, Not Yet Started	Seabrook Base Case TMLB Using RELAP5/SCDAP	RCG	053185				
6354 1	21-20 A/S, Not Yet Started	Seabrook Base Case S2D Using REALP5/SCDAP	RCG	063085				
6354 1	21-21 A/S, No Yet Started	Seabrook TMBL & S2D Operator Action Analysis	RCG	083185				

6354-3

A6354: Severe Accident Sequence Analysis Program (SASA)

EG&G Technical Monitor: R. C. Gottula

DOE Technical Monitor: D. Majumdar

NRC Technical Monitor: B. Agrawal

The objectives of this project are to use deterministic calculational tools to provide detailed analyses of severe accident sequences to support, verify, and modify probabilistic event sequences, to aid in the development of accident recovery strategies, to provide parametric values for experimental programs such as the Power Burst Facility (PBF) Severe Fuel Damage testing, to point out the need for additional computer code development and experimental data, and to benchmark risk codes.

1. Summary of Work Performed During November 1984Pressurized Water Reactor (PWR) Studies:

A 3 channel core/3 channel upper plenum RELAP5/MOD2 vessel model for the Bellefonte TMLB' sequence has been developed. Predicted flow patterns in the vessel have been compared with earlier models such as a 3 channel core/single channel upper plenum and single channel core/single channel upper plenum. The 3 channel core/3 channel upper plenum model results appear to be the most reasonable up to cladding temperatures of 1500 K. Circulation between the upper plenum/upper head and the core results in lower core temperatures and higher upper head temperatures than were predicted with the strictly one-dimensional model as well as a slight decrease in the core heatup rate. The differences in the nodalization schemes will be more apparent when the RELAP5/SCDAP linked code is utilized which will include the feedback between oxidation of the cladding and recirculation of steam and hydrogen into the core.

Bellefonte and Seabrook plant decks have been prepared for use on a CRAY machine to checkout the linked RELAP5/SCDAP code, however, the code is not yet running on the CRAY so no checkout runs have been made.

Modeling schemes to simulate postulated single phase natural circulation flow patterns in a hot leg pipe using RELAP5 are being investigated for the TMLB' sequence. Further analysis and investigation of these schemes are required.

Boiling Water Reactor (BWR) Studies:

A FRAP-T6 calculation was completed which simulated fuel behavior during the power spikes predicted for Sequence 483 (plant automatic transient). The power spikes were predicted when low pressure ECC systems flooded the core with cold water. The FRACAS I mechanical interaction package predicted that 73% of the rods would fail. Although the peak cladding temperature was relatively low (1200K), failure was predicted from hoop stresses caused by pellet-cladding interaction. It is felt that the percentage of rod failure predicted by FRACAS I is too large, however, the probability of some fuel damage is high.

1. Summary of Work Performed During November 1984 (continued)

The final Browns Ferry Anticipated Transient Without Scram (ATWS) calculations are in progress using RELAP5. Problems were encountered in getting the MOD2 version to run, so the final calculations will be completed with MOD1.6.

A checkout run for the level controlled ATWS (sequence #465) using a 1-D power profile from the NSAC-70 study resulted in a sustained power level of about 17%. These results will be factored into the NUREG report and recommendations concerning the EPGs.

2. Summary of Work to be Performed During December 1984

PWR Studies:

Checkout runs with the RELAP5/SCDAP linked code will begin provided the code is running on the CRAY machine.

BWR Studies:

The final Browns Ferry ATWS calculations using RELAP5 will be completed. Other final calculations using CONTEMPT or SCDAP will be completed by early January so that the results can be incorporated into the final draft NUREG report due for release on February 28.

3. Problems and Potential Problems

The schedule for the PWR accident mitigation studies for the TMLB¹ and S₂D sequences is contingent on the length of time required to debug the RELAP5/SCDAP linked code on the CRAY machine.

A6354: Severe Accident Sequence Analysis Program (SASA)

EG&G Technical Monitor: R. C. Gottula

DOE Technical Monitor: D. Majumdar

NRC Technical Monitor: B. Agrawal

The objective of this project is to use deterministic calculational tools to provide detailed analyses of severe accident sequences to support, verify, and modify probabilistic event sequences, to aid in the development of accident recovery strategies, to provide parametric values for experimental programs such as the Power Burst Facility (PBF) Severe Fuel Damage testing, to point out the need for additional computer code development and experimental data, and to benchmark risk codes.

1. Summary of Work Performed During October 1984Pressurized Water Reactor (PWR) Studies:

The informal technical report Analysis of a Station Blackout Transient at the Seabrook Nuclear Power Plant, EGG-NTP-6700, was completed and issued. The report documents the base case TMLB' accident sequence from transient initiation through severe core damage using the RELAP5 and SCDAP codes (soft link). The results indicated steam starvation occurs with limited oxidation of zircaloy cladding (4%). These results are considered preliminary since the effects of molten core materials falling into the lower plenum producing steam, and potentially more hydrogen were not considered in the analysis. These effects along with potential natural circulation flows in the vessel and hot leg piping will be investigated further with the RELAP5/SCDAP/TRAP-MELT code during FY-1985.

A paper entitled "PWR Station Blackout Accident Results for the Bellefonte and Seabrook Plants" was presented by P. Bayless at the Twelfth Water Reactor Safety Information Meeting.

Nodalization studies are in progress with RELAP5 to determine the optimum way to model natural circulation flow paths in a PWR vessel during a TMLB' accident sequence.

Boiling Water Reactor (BWR) Studies:

A paper entitled "BWR ATWS Simulations for Browns Ferry Nuclear Plant Unit 1" was presented at the Water Reactor Safety Research Information Meeting (WRSRIM). While at the WRSRIM, a side meeting was attended where the status of INEL BWR SASA work was presented and future work was discussed. Based on published results, the best estimate value of power during a level controlled ATWS transient is bounded between ~10-20%. The groupings of predicted power levels allude to a sensitivity to kinetics formulations. This sensitivity will be evaluated by exercising the same code with point kinetics and dimensional neutronics.

The Browns Ferry RELAP5 core model based on FIST core has been incorporated into the system model. Conversion from RELAP5/MOD1.6 to RELAP5/MOD2 is in progress.

1. Summary of Work Performed During October 1984 (Continued)

A FRAP-T6 analysis of fuel rod behavior for Sequence 483 (plant automatic with ADS) is in progress for the time period including a LPCI-induced power spike. Of particular interest is the possibility of cladding rupture due to pellet-clad interactions. Convergence problems were encountered with FRAP-T6 when cladding temperatures increased above 1000 K. Code modifications have been accomplished which should allow the successful calculation of the sequence.

2. Summary of Work to be Performed During November 1984

PWR Studies:

Checkout runs will begin to exercise the RELAP5/SCDAP linked codes on a CRAY computer.

RELAP5 nodalization studies on the Bellefonte vessel will continue to determine the best way to model natural circulation flow paths during severe accidents.

BWR Studies:

The FRAP-T6 analysis evaluating the possibility of cladding failure during the unmitigated ATWS sequence for Browns Ferry will be completed.

The final ATWS runs for the Browns Ferry ATWS study will begin.

3. Problems and Potential Problems

None.

A6354: Severe Accident Sequence Analysis Program (SASA)

EG&G Technical Monitor: R. C. Gottula

DOE Technical Monitor: D. Majumdar

NRC Technical Monitor: B. Agrawal

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1. Summary of Work Performed During September 1984Pressurized Water Reactor (PWR) Studies:

C. Dobbe attended the fifth International meeting on Thermal Nuclear Reactor Safety at Karlsruhe and presented a paper entitled "Thermal-Hydraulic and Core Damage Analysis of the Station Blackout Transient in Pressurized Water Reactors".

A summary on "PWR Station Blackout Accident Results for the Bellefonte and Seabrook Plants" was prepared and submitted for the Water Reactor Safety Research Information Meeting.

The informal technical report Analysis of a Station Blackout Transient at the Seabrook Nuclear Power Plant, EGG-NTP-6700, was completed and issued.

The informal technical report documenting the Bellefonte base case TMLB¹ transient received internal review. Final comments are being incorporated and the report will be issued early in October.

Boiling Water Reactor (BWR) Studies:

Last month it was reported that the total H₂ released during the Browns Ferry ATWS Sequence 551 (high pressure boiloff) was 500 kg for a high powered bundle and 440 kg for an average powered bundle. Those numbers should have been 500 g and 440 g, respectively.

Sensitivity studies for the Browns Ferry ATWS sequence with downcomer level control were completed using a simplified network model. The preliminary results of the model, which was qualified against the FIST data, indicate that a 17% power level is to be expected with downcomer level at TAF and about 1°F core inlet subcooling. This result agrees well with both NSAC-69 and NSAC-70. With a core inlet subcooling of 50°F, the power level is expected to be about 22%. Variation of the void reactivity coefficient by 50% resulted in no appreciable variation in the expected power. Increasing downcomer subcooling by 50°F, did affect

1. Summary of Work Performed During September 1984 (Continued)

power and reactor steaming. The magnitude of this effect decreases with downcomer hydrostatic head and level. Confirmation of these results with RELAP5 is in progress. The reduced Full Integral Simulations Test (FIST) data used for qualification were transmitted to Nuclear Regulatory Commission (NRC), Oak Ridge National Laboratory (ORNL), and Brookhaven National Laboratory (BNL). It is recommended that codes used to predict natural circulation conditions be qualified using this data base.

TRAC-BD1/MOD1 calculations were compared to data from a recirculation pump trip and a generator load rejection test at the Browns Ferry plant. Results are encouraging. Some difference between the actual and modeled pressure controllers results in a slightly overpredicted response.

A FRAP-T6 input model was completed for use in analyzing the power spikes predicted in Sequence 483 (plant automatic with ADS). Boundary conditions from RELAP5 and FRAPCON were put into a format compatible with FRAP-T6 input.

A preliminary draft of a NUREG report documenting the Idaho National Engineering Laboratory (INEL) Browns Ferry Anticipated Transient Without Scram (ATWS) work was completed. Publication of the report is scheduled for the end of February 1985 so that final RELAP5 results can be included in the report.

2. Summary of Work to be Performed During October 1984

PWR Studies:

Preparation for and presentation of the base case TMLB¹ results for the Bellefonte and Seabrook plants at the Water Reactor Safety Meeting will be completed.

Work will continue on the development and checkout of the multi-channel RELAP5 vessel model for the Bellefonte plant.

BWR Studies:

The Browns Ferry RELAP5 core model based on FIST data will be incorporated into the system model. The system model will be converted from RELAP5/MOD1.6 to RELAP5/MOD2.

A FRAP-T6 analysis of fuel rod behavior for Sequence 483 (plant automatic with ADS) will be conducted for the time period including a LPCI-induced power spike. Of particular interest is the possibility of cladding rupture due to pellet-clad interactions.

A scoping study will begin to analyze the effects of a hydrogen blowdown to the Mark I containment.

3. Problems and Potential Problems

Quality assured 1-D neutron cross sections for the Browns Ferry plant are needed in order to perform TRAC-BD1 ATWS calculations. Funding to collapse 3-D cross sections being generated at BNL for Browns Ferry EOC5 to 1-D cross sections has not been identified.

Constants to be used in the TRAC-BWR enhanced boron mixing model are required in order to perform boron mixing studies for certain ATWS sequences in the Browns Ferry plant. NRC assistance in obtaining proprietary data from General Electric (GE) is needed.

4. Cost Breakdown

Individual total cost and variance explanations are provided on the Financial and Progress and Status Summary.

189 No. A6354

<u>Cost Categories</u>	<u>(\$0.0K)</u>	
	<u>Current Month</u>	<u>Year-to-Date</u>
Direct Salaries	\$ 27.9	\$ 307.2
Materials, Services and Other Costs	8.9	28.2
ADP Support	11.9	184.6
Subcontracts	0.0	0.1
Travel	1.7	20.1
Indirect Labor Costs	36.4	393.3
General and Administrative	13.0	148.8
Capital Equipment	0.0	0.0
TOTALS	<u>\$ 99.8</u>	<u>\$ 1082.3</u>

A6354: Severe Accident Sequence Analysis Program (SASA)
EG&G Program/Technical Monitors: S. R. Behling/R. C. Gottula
DOE Technical Monitor: D. Majumdar
NRC Technical Monitor: B. Agrawal ✓

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1. Summary of Work Performed in August 1984

Pressurized Water Reactor (PWR) Studies:

Basically, the entire month of August was devoted to report writing. Two separate draft NUREG reports documenting the Bellefonte and Seabrook station blackout (TMLB') analyses during FY-1984 are being written. Each report will contain the most recent NSSS results using RELAP5/MOD2 and core damage results using the SCDAP code. The Seabrook draft report has been completed for initial internal review. The Bellefonte report will be ready for internal review by September 7th.

A draft report documenting the Bellefonte TMLB' SCDAP sensitivity study was sent out for peer review. The title of the report is "Analysis of Core Behavior During a Station Blackout Transient (TMLB') for the Bellefonte Pressurized Water Reactor" and was written by Rosanna Chambers. Best estimate calculations indicate steam starvation in the core resulting in limited cladding oxidation, hydrogen production, and fission product release. The results are quite sensitive to any core inlet flow, however.

Preparation of a paper entitled "Thermal-Hydraulic and Core Damage Analysis of the Station Blackout Transient in Pressurized Water Reactors" was completed. The paper will be presented in Karlsruhe in September at the International Meeting on Thermal Nuclear Reactor Safety.

1. Summary of Work Performed in August 1984 (Continued)

Boiling Water Reactor (BWR) Studies:

Sequence 483 (unmitigated ATWS transient) was rerun out to 2200 s with RELAP5/MOD1.6 and CONTEMPT/LT-028 assuming that the ADS valves remained open after actuation. Results were compared to earlier calculations which modeled ADS cycling. A slightly lower mean power was predicted, however, LPCI cycling was more frequent when the ADS valves remained open. Predicted PSP temperature at 2200 s was approximately 280°F, and the drywell was pressurized to 50 psia. The SCDAP hot bundle analysis of Sequence 483 (with ADS cycling) indicated small amounts of cladding oxidation, but no prediction of fuel or cladding failure during the LPCI induced power spikes. The possibility of using the FRAP-T code to predict fuel rod behavior during the power spikes due to the rapidly varying thermal-hydraulic conditions is being investigated.

A SCDAP calculation of Sequence 551 (high pressure boiloff) was completed for a core average bundle. Comparisons between that calculation and the hot bundle calculation performed earlier indicate the following. The total H_2 released was 500 kg for a high powered bundle versus 440 kg for an average powered bundle. Rod failure was predicted at 3200 s in the hot bundle versus 3350 s in the average bundle.

Both a RELAP5/MOD1.6 model and simple analytical model have been qualified against data from the FIST facility. Comparisons with data, which extend over a wide range of bundle power, indicate the importance of the void/quality relation on bundle void, pressure drop and flow partitioning between the various core flow paths. The reduced FIST data is being prepared for transmittal to the Nuclear Regulatory Commission (NRC), Brookhaven National Laboratory (BNL), and Oak Ridge National Laboratory (ORNL).

The TRAC-BD1/MOD1 vessel model of Browns Ferry was renodalized because of adverse interactions between the separator and level tracking models. A new steady state solution was obtained, and decks were set up for plant data comparisons.

A preliminary version of the CONTAIN code was made operational on the Idaho National Engineering Laboratory (INEL) computer.

The draft NUREG report documenting all of the INEL Browns Ferry Anticipated Transient Without Scram (ATWS) work during FY-1984 is being written. The report will be ready for internal review early in September.

2. Summary of Work to be Performed in September 1984

PWR Studies:

The draft NUREG reports documenting the Bellefonte and Seabrook TMLB¹ analyses done during FY-1984 will be completed and sent out for peer review.

2. Summary of Work to be Performed in September 1984 (Continued)

The paper entitled "Thermal-Hydraulic and Core Damage Analysis of the Station Blackout Transient in Pressurized Water Reactors" will be presented at the Karlsruhe meeting.

Work will continue on the Seabrook base case S₂D RELAP5/MOD2 analysis.

BWR Studies:

Reduced FIST pressure drop data will be transmitted to NRC, BNL, and ORNL for use in qualifying Browns Ferry core models.

Sensitivity studies will be conducted to determine the effects of downcomer condensation efficiency and reactivity coefficients on power level for Browns Ferry ATWS mitigation studies.

Comparisons with Browns Ferry plant transient data will be made with TRAC-BD1/MOD1.

The draft NUREG report documenting the Browns Ferry ATWS work during FY-1984 will be completed and sent out for peer review.

Input models will be developed for the CONTAIN code which was recently implemented on the INEL computer.

Use of the FRAP-T code to analyze fuel behavior for the unmitigated ATWS transient (Sequence 483) for Browns Ferry will be investigated and possibly a run made.

3. Problems and Potential Problems

Quality assured 1-D neutron cross sections for the Browns Ferry plant are needed in order to perform TRAC-BD1 ATWS calculations. Funding to collapse 3-D cross sections being generated at BNL for Browns Ferry EOC5 to 1-D cross sections has not been identified.

4. Cost Breakdown

Individual total cost and variance explanations are provided on the Financial and Progress and Status Summary.

189 No. A6354

	(\$0.0K)	
<u>Cost Categories</u>	<u>Current Month</u>	<u>Year-to-Date</u>
Direct Salaries	\$ 31.9	\$ 279.3
Materials, Services and Other Costs	2.1	19.3
ADP Support	33.4	172.8
Subcontracts	0.1	0.1
Travel	1.5	18.4
Indirect Labor Costs	41.7	356.8
General and Administrative	17.7	135.8
Capital Equipment	0.0	0.0
TOTALS	\$ 128.4	\$ 982.5

3. Scheduled Milestones for June 1984

<u>Description</u>	<u>Due Date</u>	<u>Actual Date</u>
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NTPD MILESTONES 1983-1984
THRU AUGUST 26, 1984

PAGE 00001

A NUMBER STATUS NO	NODE STATUS	PROJECT LABEL	PERSON	DUE DATE	REPD-CR	NEW DATE	COMP DATE	COMP REF
6354 50	21-01 TASK COMPLETED	COMP BROWNS FERRY NUREG REPORT	SRB	022884		043084	041384	NUREG/CR3596
6354 50	21-02 TASK COMPLETED	COMP SCDAP SENSITIVITY STUDY	SRB	043084			043084	NO DOC REC
6354 50	21-03 TASK COMPLETED	COM BELL TMLB & S2D TREES & DRFT RP	SRB	043084		051184	051884	LPL 178 84
6354 3	21-04 A/S, ANALYSIS UNDERWAY	COM BR FR DOM SEQ CALC & DRAFT RPT	SRB	093084				
6354 3	21-05 A/S, ANALYSIS UNDERWAY	COM BELLEFONTE TMLB CALC & DRFT RPT	SRB	093084				
6354 3	21-06 A/S, ANALYSIS UNDERWAY	COMP SEABROOK TMLB CALC & DRFT RPT	SRB	093084				
6354 1	21-07 A/S, NOT YET STARTED	COMP BELLEFONTE S2D CALCULATION	SRB	103184	NTPD-05-84	043084		
6354 3	21-08 A/S, ANALYSIS UNDERWAY	COMP SEABROOK S2D CALCULATION	SRB	093084				
6354 3	21-09 A/S, ANALYSIS UNDERWAY	BELLF. MULTI-D R5 CHECKOUT RUNS	SRB	093084				

A6354: Severe Accident Sequence Analysis Program (SASA)
EG&G Program/Technical Monitors: S. R. Behling/R. C. Gottula
DOE Technical Monitor: D. Majumdar
NRC Technical Monitor: [REDACTED]

The objective of this project is to use deterministic calculational tools to provide detailed analyses of severe accident sequences to support, verify, and modify probabilistic event sequences, to aid in the development of accident recovery strategies, to provide parametric values for experimental programs such as the Power Burst Facility (PBF) Severe Fuel Damage testing, and to point out the need for additional computer code development and experimental data.

1. Summary of Work Performed in July 1984

Pressurized Water Reactor (PWR) Studies:

A preliminary multi-D RELAP5/MOD2 model of the Bellefonte vessel including the core and upper plenum was completed and steady state calculations were initiated. Considerable time was required this month to get the RELAP5 model operating on the new computer operating system at the Idaho National Engineering Laboratory (INEL). Three SCDAP decks representing the 3 parallel channels in the RELAP5 model have been developed and boundary conditions from the RELAP5 code are being input to SCDAP. This will provide a more accurate prediction of hydrogen generation and fission product release for the TMLB' accident sequence. Also, SCDAP calculations for 3 parallel channels in the Seabrook core have been completed for the TMLB' sequence.

A RELAP5/MOD2 steady state calculation for the S₂D sequence was completed for the Seabrook plant. An initial S₂D transient assuming a 2 in. break in the cold leg has been completed. The primary system pressure decay rate appears to be very sensitive to the emergency feedwater flow rate. The SCDAP deck for the S₂D transient has been developed.

The report documenting the Bellefonte SCDAP sensitivity study is still in the process of internal review.

Boiling Water Reactor (BWR) Studies:

Full power and natural circulation data from the FIST facility have been reduced to a form amenable to a detailed analysis of core thermal hydraulic phenomena. Early results indicate the importance of channel bypass and leakage path modeling to channel void fraction predictions. The analysis of the FIST data is being documented. The data will be the basis for pressure loss coefficients to be used in the RELAP5 Browns Ferry model.

1. Summary of Work Performed in July 1984 (Continued)

Dominant sequence 483 (plant automatic) was rerun with RELAP5 assuming that the ADS valves remain open after activation. The results will be compared with an earlier calculation which assumed the ADS valves closed on high back pressure.

A paper documenting the status and needs of BWR containment analysis capabilities was submitted to the International Nuclear Power Plant Thermal Hydraulics and Operation Topical Meeting in Taiwan. The paper was revised based on reviewer comments and subsequently accepted.

CONTEMPT/LT-028 was converted to FORTRAN5, to be consistent with steam tables and environmental library routines. The code was also converted to the new operating system, which is being implemented on the INEL computer system.

A preliminary version of the CONTAIN code was received from Sandia National Laboratory (SNL). The code is currently being implemented on the INEL system.

Control system and balance of plant modeling were incorporated into the TRAC-BD1/MOD1 deck of Browns Ferry. Comparisons of results with plant transient data were begun.

The SCDAP average core calculation of Sequence 551 (high pressure boiloff) was completed to cladding melt. Analysis of the results is in progress. A hot rod analysis of the Sequence 483 power excursions was begun.

2. Summary of Work to be Performed in August 1984

PWR Studies:

A SCDAP calculation for the S₂D sequence will be initiated for the Seabrook plant.

Work will begin on writing of the NUREG reports documenting the RELAP5 and SCDAP analyses of the TMLB' sequence for both the Bellefonte and Seabrook plants that were accomplished during FY-1984.

BWR Studies:

CONTEMPT/LT-028 and SCDAP analyses of Sequence 483 will be completed, as will the SCDAP average core analysis of Sequence 551.

Analysis of FIST data will be completed. Work will begin studying the effects of variations in downcomer condensation and reactivity coefficients on level control anticipated transients without scram (ATWS) for the Browns Ferry plant.

2. Summary of Work to be Performed in August 1984 (Continued)

TRAC-BD1/MOD1 comparisons to plant transient data will continue.

3. Problems and Potential Problems

Quality assured 1-D neutron cross sections for the Browns Ferry plant are needed in order to perform TRAC-BD1 ATWS calculations. Funding to collapse 3-D cross sections being generated at BNL for Browns Ferry EOC5 to 1-D cross sections has not been identified.

4. Cost Breakdown

Individual total cost and variance explanations are provided on the Financial and Progress and Status Summary.

189 No. A6354

<u>Cost Categories</u>	<u>(\$0.0K)</u>	
	<u>Current Month</u>	<u>Year-to-Date</u>
Direct Salaries	\$ 26.0	\$ 247.3
Materials, Services and Other Costs	0.9	17.2
ADP Support	20.5	139.4
Subcontracts	0.0	0.0
Travel	0.0	17.0
Indirect Labor Costs	33.9	315.2
General and Administrative	13.0	118.1
Capital Equipment	0.0	0.0
TOTALS	<u>\$ 94.3</u>	<u>\$ 854.2</u>

Severe Accident Research Program

Quarterly Report
November, 1984



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QUARTERLY REPORT

on

SEVERE ACCIDENT RESEARCH PROGRAM

November, 1984

Compiled by Battelle's Columbus Laboratories for
U.S. Nuclear Regulatory Commission

NOTICE

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QUARTERLY REPORT
on
SEVERE ACCIDENT RESEARCH PROGRAM

November, 1984

INTRODUCTION

This is the seventh quarterly report on the Severe Accident Research Program. The Severe Accident Research Program (SARP) consists of approximately 60 separate research programs organized into 13 programmatic elements. In order to assist NRC management and staff in following the progress of these diverse but related programs, Battelle's Columbus Laboratories is compiling progress reports for all of the SARP programs based on the monthly reports of the contractors. These reports are issued on a quarterly basis but do not necessarily cover the previous three months because of differences in the timing of receipt of contractor reports. The most recent results are those provided in reports for September, 1984. In some cases no monthly reports were available, and summaries of the quarter's progress were provided by the NRC program monitors.

The purpose of the Quarterly Report is to provide the NRC staff with a convenient summary of progress on SARP programs. Program progress and results are presented within the context of the 13 program elements so that progress can be interpreted in terms of the overall objectives of the SARP program. This report does not serve as a mechanism for presenting SARP program results to the public. Indeed, many of the results presented in the monthly reports of the contractors are preliminary and have not received internal approvals.

The intent of the SARP program as presented in NUREG-0900 is "to establish a sound technical basis on which an evaluation of the need for changes in nuclear power plant design and operation can be made". In Appendix A of this report, each of the SARP programs is identified. This Appendix provides basic management data for each program such as the FIN Number and NRC project monitor. The Appendix also identifies the project reports that are summarized in this quarterly report.

The next section of this report discusses some of the more significant accomplishments of the S/PD program in the reporting period. The last section of the report is a summary of progress for projects in each of the program elements.

GENERAL OVERVIEW

Staff met with the Commissioners on October 9, 1984, to discuss the proposed Severe Accident Policy Statement and the accompanying report, "NRC Policy on Future Reactor Designs: Decisions on Severe Accident Issues in Nuclear Power Plant Regulations", NUREG-1070. Commission action is anticipated in November.

Representatives of the IDCOR program met with staff and their contractors on August 28 and 29, 1984, to discuss IDCOR methods and results for integrated analyses of severe accidents. This meeting was the fourth in a series of technical exchanges aimed at understanding the bases for the IDCOR methodology and comparing the NRC and IDCOR technical positions. The most significant change in IDCOR methodology described at this meeting involved the use of an empirical correlation to describe aerosol behavior in the reactor coolant system and containment, rather than the use of mechanistic computer codes. The results presented also accounted for the reevolution of fission products from RCS surfaces. The next meeting in the series, which will include the assessment of risk profiles and cost-benefit tradeoffs for plant modifications, is currently scheduled for December, 1984.

From September 17 to 19, 1984, the annual FRG/US Core Melt Information Exchange Meeting was held in Obrigheim, West Germany. Early results from the BETA facility indicated much greater vertical penetration and less horizontal penetration of concrete by molten core debris than has been predicted by the state-of-the-art core-concrete models; CORCON, WECHSL, and KAVERN. Results of dry aerosol experiments were presented from tests in the DEMONA facility. Agreement with computer code predictions was good.

Major Accomplishments in the Reporting Period

- The DCC-2 experiment (core debris coolability) was conducted. In 91 hours of fission heating 73 dryouts were achieved.
- SFD 1-3 was conducted August 3 in the PBF. In this test a 32-rod, one-meter long fuel bundle was heated up to 2400 K under core uncover conditions similar to those in the TMI-2 accident.

- Tests F-13 and F-14 were conducted in the FLAME facility on May 31. The tests were conducted with no top venting and no obstructions. In test F-14 flame acceleration occurred resulting in damage to the facility.
- Results of HECTR calculations were compared to experimental results from NTS and FLAME tests. In general, agreement between analytical and experimental results was good.
- Three core melt-coolant interaction experiments were conducted in the FITS facility at Sandia. Only one of the three produced a steam explosion.
- TURC-2, the first test of molten UO_2 interactions with concrete was conducted. Results of the test indicate that the interaction was quite mild.
- CORCON-MOD2 was released to selected users.
- Fission product release tests HS-2 and HS-3 using German-fabricated simulant fuel were conducted. In these tests the specimens were heated to 2400 C and 2000 C, respectively.
- The last of the planned aerosol resuspension tests was completed.

TECHNICAL PROGRESSPROGRAM ELEMENT 1: ACCIDENT LIKELIHOOD REEVALUATION

Program Element 1 is comprised of three programs: (1) Accident Sequence Evaluation Program, conducted at SNL and INEL, (2) Accident Sequence Precursors, conducted at ORNL, and (3) Pressurized Thermal Shock, also conducted at ORNL. Highlights of this quarter's work in the three programs are presented below.

Accident Sequence Evaluation
Program (Activities at INEL)

- Recovery analysis for the sequences being considered in the Interim Accident Sequence Evaluation (IASE) was completed.
- The staff completed the analyses of the isolation condenser boiling water reactor (BWR) plants, a part of the earlier deductive plant modeling effort.
- The ASEP catalog report was completed and published as a draft NUREG/CR (instead of an EG&G report as originally planned).

Accident Sequence Evaluation
Program (Activities at SNL)

- Work on both PWR and BWR ATWS sequences continued. Comments on the ASEP treatment of those sequences were received and incorporated as model development proceeded.
- The ASEP PWR and BWR event trees were developed further. A meeting was held with the Accident Sequence Precursor (ASP) staff; the ASP and ASEP event tree needs were compared.
- Work on the sensitivity analysis and the presentation of results began.

LWR Accident Sequence Precursor Study

- A letter report detailing the assessment of 11 events from an ASP standpoint was completed and sent to the NRC for comment.
- Work is proceeding to define the level of modeling considered necessary for LWR operational events. ASP and ASEP are cooperating in the development of an event tree philosophy document and draft plant class event trees.
- The ASP data base was reviewed. Exploratory trends analyses concerned with the impact of plant age and calendar date on precursor frequency have been initiated.
- Containment system designs and their models both in PRAs and in the University of Maryland event trees were reviewed. Work proceeded on the identification of containment-related events.

The Pressurized Thermal Shock Program focused on three plants: Oconee-1, Calvert Cliffs-1, and H. B. Robinson-2. All work related to Oconee-Unit 1 is complete. The report is under review and will be published as NUREG/CR-3770. The work related to Calvert Cliffs-Unit 1 is complete. Most of the documentation is also complete. Tasks related to H. B. Robinson-Unit 2 are complete except for the fracture mechanics analysis of vessel response which has begun.

PROGRAM ELEMENT 2: SEVERE ACCIDENT SEQUENCE ANALYSIS

Work on Severe Accident Sequence Analysis is being conducted at five laboratories, SNL, INEL, LANL, ORNL, and BNL. A brief description of work done at each laboratory is given below.

Sandia National Laboratories

- Work continued on the Thermal-Hydraulic Analysis for large dry PWR containments (Bellefonte). RELAP5 results from the INEL TMLB' calculations were incorporated into a MARCH 2.0 deck and the initial TMLB' calculation was performed. A MARCON input deck was completed. Work continued on preparations for CONTAIN and HECTR calculations.
- Thermal-hydraulic analysis activities for PWR ice condenser containments included revision of a draft report describing the containment pressure/temperature response to a variety of accident sequences.
- The MARCON code was modified, and documentation for the code was drafted.
- The LTAS code was transmitted to Sandia from ORNL and made compatible with the Sandia computer system.
- Work continued on the preparation of links from MARCON to HECTR and CONTAIN and on the linking of the MEDICI reactor cavity models to HECTR.

Idaho National Engineering Laboratory

- A preliminary RELAP5/MOD2 model of the Bellefonte vessel, including the core and upper plenum, was completed and steady state calculations were begun.
- SCDAP calculations for three parallel channels in the Seabrook core were completed for the TMLB' sequence.

- RELAP5/MOD 2 was used to do both a steady state calculation for an S2D sequence and an initial S2D transient assuming a 2-inch break in the cold leg.
- A draft report on TMLB' analysis for Seabrook was completed.
- A draft report documenting the Bellefonte TMLB' SCDAP sensitivity study was released for peer review.
- Data from the FIST facility were analyzed, and the analysis was documented. Both a RELAP5/MOD1.6 model and a simple analytical model were qualified against FIST data.
- Dominant sequence 483 (plant automatic) was rerun with RELAP5 assuming that the ADS valves remain open after activation.
- A preliminary version of CONTAIN was received from Sandia and implemented on the INEL computer system.
- Control system and balance of plant modeling were incorporated into the TRAC-BD1/MOD1 deck of Browns Ferry.
- The SCDAP average core calculation of Sequence 551 (high pressure boiloff) was completed to cladding melt.

Los Alamos National Laboratory

- A revised draft of the boron dilution document was submitted to the NRC in July.
- Dominant accident sequences identified in a probabilistic risk assessment conducted at Sandia were reviewed, and the ATWS sequences were found to comprise a significant fraction of the overall core damage probability for sequences considered by Sandia. The ATWS sequence selected for study at INEL is the T2K2 (MTC) sequence for Calvert Cliffs.
- Staff members began work on the first TRAC/MELPROG model of Oconee-1 in preparation for running the TMLB' transient.

- In support of TAP A-45 an evaluation of the atmospheric steam dump system as a vehicle for post-LOSP cooldown and depressurization was begun. The first study (Case 1) evaluated the feasibility of operator-initiated atmospheric dump valve (ADV) control 720 seconds after a postulated loss-of-offsite power event at Calvert Cliffs Unit 1.

Oak Ridge National Laboratory

- BWR-LACP was converted to Fortran and renamed BWR-LTAS. During the reporting period LWR-LTAS was modified. In July the draft of the BWR-LTAS user's manual was completed.
- BWR-LACP calculations were performed for the case of the MSIV-closure-initiated ATWS without operator action.
- The ZRWATR routine was adopted for use with MARCH-BWR to determine the zirconium-water reaction rate under accident conditions.
- A MARCH analysis of the no-operator-action case for the MSIV-closure initiated ATWS was completed and compared with BWR-LACP calculations. The MARCH results compared favorably prior to the power/pressure spikes. Work continued on modifications of MARCH to allow it to model the accident accurately beyond the time of the spikes.
- Efforts continued to incorporate ORNL-BWR models into MARCON (MARCH 2.0 with the INTER package replaced by CORCON MOD2).
- CORCON MOD2 was made operational on the ORNL computer system.
- Computer coding to solve for the transient temperature distribution in standpipes as a function of axial and radial position was written. Results of the coding were compared with analytical solutions for two cases, and agreement was good.
- Modifications were made in MELRPI MOD2 including (1) restructuring of the input file so that any ECC system can be turned on without restarting the code, (2) development of an improved model

for coolant redistribution under overflow conditions in a partially rubblized zone, and (3) correcting some models for two-phase flow and heat transfer in the submerged region of the core. The final version of the code was extensively tested.

- Work continued on the modification, testing, and verification of LPFRPI, the subroutine of MELPRI which models lower plenum/head failure.
- ORIGEN2 calculations of nuclide inventories in the Browns Ferry Cycle 6 core were completed. Results will be used to determine fission product inventories that should be used in ATWS studies.
- Work continued on upgrading the procedure for calculating the fission product species distribution in the reactor vessel. The code SOLGASMIX is being modified and used.
- CORCON-MOD1 and MOD2 were used to perform a calculation for the "base case" LDHR code input parameters. The two versions gave different results in the calculation of gas evolution rates, layer temperatures, and Zircaloy oxidation rates.

Brookhaven National Laboratory

- Arrangement of the feedwater spargers used in the new RAMONA-3B HPCI/RCIC jet condensation model was revised based upon information from the Browns Ferry reactor.
- CASMO was used to calculate macroscopic cross sections for the five major fuel types identified in Cycle 5 at Browns Ferry. The cross sections were calculated over a typical range of exposures and void histories.

PROGRAM ELEMENT 3: ACCIDENT MANAGEMENT

The two major projects under this program element are Human Factors Review for Severe Accident Sequence Analysis which is being conducted concurrently with the SASA BWR ATWS project at ORNL and the Accident Management Study at BCL.

Human Factors Review

During this reporting period work continued on Tasks 2 and 3 with development of the high-level operator severe accident management model. The functional classification was improved. A format for assessing operator actions included in the unconventional operator response event tree was drafted with four major categories: (1) alarms and cues per operator/system failure mode, (2) operator decision criteria for possible responses, (3) expected performance for the selected response, and (4) consequences of actions to the plant. End states of the event tree were assessed qualitatively. Operator actions for the ATWS scenario which has been chosen for the initial demonstration of the model were assessed linking the functional classification and certain unconventional operator responses included in the event tree. These responses are being assessed according to the format identified above.

In July Task 2 was completed and work on the final report was begun. This report will provide additional information on the front-end human factors analysis of ATWS which was discussed in Appendix C of the SASA ATWS report (NUREG/CR-3470).

Accident Management

- Two white papers entitled "A Systems Approach to Accident Management" and "Definition of Accident Management" were drafted and submitted to the NRC for review.
- Two other white papers on the review and evaluation of French "U" procedures and on industry upgrades for accident management were drafted.

- SERs for the NSSS vendors' emergency procedure guidelines were obtained and reviewed.
- BWR Emergency Guidelines were reviewed and the limits of their applicability established.

PROGRAM ELEMENT 4: BEHAVIOR OF DAMAGED FUEL

Program Element 4 is the Severe Fuel Damage Program, which has substantial international participation. The work is being carried out primarily in four laboratories. INEL is conducting a series of integral severe fuel damage and fission-product release and transport experiments in the Power Burst Facility (PBF) and developing SCDAP, the Severe Core Damage Analysis Package Code. In addition, EG&G personnel serving as NRC technical representatives participate in and report on fuel behavior studies being conducted at KfK in Karlsruhe, Germany. SNL is conducting experiments in the ACRR test reactor on LWR debris formation and relocation, and on LWR core debris coolability. SNL is also analyzing accident sensitivities and developing the MELPROG melt progression code. PNL is performing experiments to determine high temperature properties of fuel and cladding and is preparing for four severe fuel damage experiments in NRU with full length fuel bundles. Finally, LANL is working on the development of TRAC/MELPROG. A summary of this quarter's work under each program is presented below.

LWR Debris Formation and Relocation (SNL)

- Assembly of the DF-2 capsule began.
- A preliminary analysis of the DF-1 experiment with the DFRMOD thermal/hydraulics code was completed. Fuel/clad temperatures at the upper end of the fuel bundle predicted by DFRMOD agree well with experimental values, but agreement between code predictions and experimental results at the middle of the bundle are only fair.
- Work began on the improvement and verification of COPOX, a computer code which was written to infer H₂-generation rates from H₂-getter temperature measurements.
- Radiometric temperature estimates were calculated for the DF-1 experiment.

- The fuel bundle from the DF-1 experiment was radiographed. The radiograph showed that much of the fuel pellet stack was intact but distorted. A large mass accumulated near the location of the prototypical grid spacer (about 17 in. from the top of the fissile region) indicating that molten materials accumulated and refroze at the grid spacer. This interpretation was consistent with the videotape, which showed a filling of the space between the fuel rods.
- An out-of-pile CuO test apparatus has been constructed to collect data necessary to develop and verify the COPOX code. It will also be used to determine the steam dilution requirements to avoid excessive H₂-getter temperatures during the DF-2 experiment.

LWR Core Debris Coolability (SNL)

- The DCC-2 experiment was conducted April 2-13, 1984. In this experiment 73 dryouts were achieved in 91 hours of fission heating.

LWR Accident Sensitivities and Melt Progression (SNL)

- Checkout of the two-dimensional species conservation model for the VICTORIA fission product module was completed.
- Work continued on the aerosol behavior model which is the MAEROS portion of CONTAIN.
- The STRSET routine in the STRUCT module, which is responsible for calculating flow areas and hydraulic diameters for use in heat transfer and fluid flow, was completely rewritten.
- In connection with the revision of STRSET, some recoding was done in PINZ to ensure consistency between the changing flow area when fuel rods fail and the amount of mass released.

- Additional mass transfer variables were added to PINZ and to FLUIDS. This will allow tracking of the individual corium components in more detail.
- In interfacing MELPROG to TRAC-PF1, a series of test cases were run in preparation for its use in running the TMLB' and S2D sequences for the Surry PWR plant.

Severe Fuel Damage Program (INEL)

Severe Fuel Damage test 1-3 was successfully performed August 3 in the Power Burst Facility (PBF). Test 1-3 is the first of the final two PBF Severe Fuel Damage (SFD) tests using fuel that has been pre-irradiated for build up of a long-lived fission-product inventory.

In the SFD 1-3 test, the 32-rod, 1-meter long bundle of test fuel underwent a heating transient up to a planned 2400K maximum temperature under core-uncovery conditions similar to those of the TMI-2 accident.

Severe Fuel Damage Model Development

- The SCDAP fuel behavior models were merged into RELAP5/MOD2, replacing the conventional RELAP5 heat structure description in the core region.
- A calculation of the PBF SFD 1-1 experiment was performed to test this combined code and the results compared favorably with the experimental data.
- The design report on fission product release, transport, and deposition modeling for the coupled code was completed and issued.
- A driver code which simulates RELAP5 has been created to test the models prior to incorporation into the RELAP5 structure. The models, primarily adapted from TRAP-MELT (including agglomeration, thermophoresis, impaction, condensation, and settling) have been coded and testing has begun. Once added, these models will

provide the basis of a complete fission product transport and deposition analysis for the reactor coolant system.

- RELAP5 was modified to include transport fields for multi-specie soluble fission products (Cs, I, etc.), and noncondensibles (H₂, Xe, Kr, etc.). Time-varying geometry of the core region was added to account for ballooning and relocation.
- SCDAP improvements were made in UO₂ dissolution thereby improving calculations both of flow and freezing of liquefied material and of fission product release.
- A SCDAP workshop involving participants from seven national laboratories and five foreign nations was held during July. Participants were provided with information on SCDAP code architecture, solution schemes, models, and future plans and were given an opportunity for hands-on experience.
- Work began on pretest analysis of the OCED-LOFT FP-2 experiment.

TRAC/MELPROG (LANL)

- Work was completed and reports were issued on flow patterns in the Surry vessel. The two sequences studied were loss of feed-water with failure of the emergency core cooling system (TMLB') and hot leg break with failure of the emergency core cooling system (AB).
- Testing of the two-dimensional MELPROG fluid-dynamics module continued. The Surry upper vessel circulation calculations were run with the stand-alone two-dimensional fluids module.
- Work began on the draft manual for TRAC/MELPROG (Mod 0).

Resident Scientist at KfK

- Preparations were begun for a series of 9-rod bundle tests with control materials in the NIELS facility.

- Work continued on reduction of data from ESA-4 and ESA-5.
- A series of tests was run in which small Zircaloy samples were reacted in flowing gas containing specified amounts of hydrogen, steam, and argon. The specimens were heated to 800 C. Argon is used to maintain a sufficient gas flow rate when the concentrations of H_2 and steam are low. It does not affect the reaction.

PROGRAM ELEMENT 5: HYDROGEN GENERATION AND CONTROL

All tasks under Program Element 5 are being performed at Sandia. This quarter's progress in each major task is summarized below.

Operability of Hydrogen Igniters in a Water Spray Environment

- The General Motors AC 7G diesel glow plug igniter was tested in three experimental sequences. The igniter was operated without a spray shield at 10.4, 12.0, and 14.0 VAC. The igniter did not maintain satisfactory ignition temperatures above fluxes 13 to 17, 25 to 29, and 38 to 42 $\text{J/m}^2\cdot\text{min}$, respectively.
- Helical and cylindrical igniters were exposed to horizontal gas flows in the absence of water sprays. These flows spanned and exceeded those anticipated in the actual reactor containment situation. The cylindrical igniter maintained a higher surface temperature in all tests.
- Three-dimensional flows induced by air entrained by downward fall of water drops in a reactor containment were modeled using the CONCHAS-SPRAY code. Horizontal gas flows at the base of the containment were shown to be the largest, with return flow up the containment walls diminishing with height.

Aerosols and the Combustion of Hydrogen

- Equipment needed for experiments involving the combustion of hydrogen in the presence of aerosols (core materials) was installed in the VGES chamber.

HECTR Analysis and Code Development

- Results of HECTR calculations were compared to experimental results from NTS and FLAME tests. In general HECTR accurately

modeled the NTS tests. However, it slightly overpredicted the peak pressure and gas temperature. This discrepancy is due to the difference between the combustion completeness observed in the experiment and that calculated by HECTR.

- A fan cooler model has been developed for HECTR to analyze the Three Mile Island (TMI) accident.
- The radiative heat transfer package in HECTR was modified to account for CO and CO₂ gases.
- At the request of the NRC, calculations were performed to evaluate the percentage of hydrogen generated in several typical nuclear power plants with emphasis on the large dry containment.

FLAME Facility

- Tests F-13 and F-14 were conducted in the FLAME facility on May 31. The tests were conducted with no top venting and no obstructions. Test F-13 was conducted at 14 percent hydrogen with the fans running. In Test F-14, which was conducted with stoichiometric hydrogen-air mixture, there was a transition to detonation. The facility suffered significant damage. Repairs are being planned.

Heated Detonation Tube

- Thirteen tests were performed in the heated detonation tube with the temperature fixed at 100 C while the steam mole fraction and hydrogen mole fraction varied. Results indicate that increased temperature increases the likelihood of a detonation and that the mixture is detonable even at steam-saturated conditions.

Air Currents Driven by Sprays in Reactor Containment Buildings

- Air flow fields caused by droplets of constant mass in a simple geometry have been predicted using an analytical model developed for this purpose and a modified version of CONCHAS-SPRAY.

Vortex Dynamics Modeling of Flame Acceleration

- Three types of experiments were simulated: flame propagation in (1) a channel that contains a single baffle, (2) a channel with a few baffles near the ignition source and no other obstructions, and (3) a channel with sidewall vents. In case (2), calculations indicate that the flame accelerates in the unobstructed section of the channel. Results of calculations for case (3) indicate that side venting slows flame acceleration.

FITS Facility

- Work resumed on data analysis for test series 3. Almost 200 tests were conducted to investigate deflagration behavior in hydrogen-air-steam mixtures. Data were gathered on flammability limits, pressure and temperature histories, and flame speeds.

Catalytic Mitigation Evaluation Program

- Experiments are designed to measure the rate of water production when hydrogen flows through a channel packed with a noble metal catalyst, palladium in this case. The hydrogen and oxygen adsorbed on the palladium react to form water. The rate of water production increases with increased temperature.

Diffusion Flame Research

- Simple models of diffusion flames in closed vessels were applied to several NTS tests. The models predict pressure histories during the experiments fairly well, but they require specification of spatial variation of the heat transfer coefficient in order to accurately predict wall temperatures.

Data Reduction and Analysis

- The SMOKE computer codes were modified.
- Computer codes were assembled for processing NTS data. Analysis of those tests began using SMOKE and the equipment survival algorithm, HYBER.

Detonation Modeling

- Calculations were done to determine the effects of (1) initial mixture temperature and pressure and (2) water dilution on the detonation cell size. Initial temperature was found to have a strong effect on reaction zone length. Predictions of the effects of water dilution agree with the results of earlier, independent calculations.

Hydrogen Burn Survival Program

- A test plan and schedule were written for a series of hydrogen burn simulation tests to be performed at the Central Receiver Test Facility (CRTF). Plans include (1) simulation of the NTS tests, (2) simulation of hydrogen burns in large dry PWR containments to evaluate aged and unaged equipment survival, and (3) fragility tests on aged and unaged equipment to determine failure thresholds.

- Hydrogen burn tests were conducted with aerorols present. It is expected that results will show that aerosols absorb some of the radiated heat, thus reducing peak temperatures experienced by the equipment.

PROGRAM ELEMENT 6: FUEL-STRUCTURE INTERACTION

Program Element 6 encompasses three projects, all of which are being conducted at Sandia. The three programs are Core Melt/Coolant Interactions, Core Melt Technology, and Containment Load Mitigation. Highlights of this quarter's progress are presented below.

Core Melt-Coolant Interactions

- FITS 5D, a core melt-coolant interaction (CMCI) experiment, was conducted using cold water (66 cm deep) in a lucite chamber. The ambient pressure was 83 kPa (atmospheric). An energetic steam explosion occurred.
- In test FITS 2D initial conditions included an ambient pressure of 1103 kPa and a water depth of 66 cm. No spontaneous explosion occurred. The melt fragmented as it fell through the water and reagglomerated on the bottom of the chamber.
- A third CMCI test was conducted with the melt at atmospheric pressure, a water depth of 15 cm and a water temperature of 92 C. There was no spontaneous explosion.
- Work continued on the computer modeling of fuel-coolant mixing, and of triggering in single droplet tests.
- The final report was issued on modeling of the PWR cavity FAC phenomena for low pressures and BWR Mark II steam explosion dynamics.
- Sensitivity analyses were completed on the models of (1) triggering in single droplet tests and (2) non-equilibrium explosions. Results indicate that fuel fragment sizes and mixing mass ratios are the most important initial conditions affecting the explosion conversion ratio.

Core Melt Technology

- TURC-2 the first test of molten UO_2 interactions with concrete, was conducted. About 190 kg of molten UO_2 --25 w/o ZrO_2 was prepared in the Large Melt Facility (LMF) using IRIS melting techniques. Temperature of the melt was about 2550 C.

Results of the test indicated that interaction of the melt with concrete was quite mild. The prolonged, vigorous attack on concrete, which had been observed with steel, did not occur. Early examination of instrumental data suggest that a crust formed between the melt and the concrete. Heat transfer through the crust, which would lead to concrete attack, was slow.

Heat prevented by UO_2 crusts from going into the concrete may raise the melt temperature and cause aerosol evolution or will be rejected into the containment, adding to containment loads.

- CORCON-MOD1 was used to model three TURC experiments: the interactions of a thermite melt, a stainless steel melt, and a UO_2/ZrO_2 melt with concrete. The calculation for thermite agreed well with the experimental results, but agreement in the cases of steel and UO_2/ZrO_2 was not good. The steel ablated the concrete much faster than predicted, and the ablation caused by UO_2/ZrO_2 was much slower than expected.
- CORCON-MOD2 was released to selected users, and work continued on an accompanying user's manual.
- Test HIPS 4W, a high-pressure melt ejection into a scale model reactor cavity filled with water, was conducted. The objective of the test was to determine whether the presence of water affects the expulsion of core debris from the reactor cavity. The experiment did not provide the desired information because pressurization due to the core melt-coolant interaction ruptured the model reactor cavity. Note that the model was not designed to accurately represent a reactor cavity's strength.

Containment Load Mitigation

- Preparations were completed for the first series of sustained tests of combined core debris/concrete/coolant interactions. The tests will provide (1) baseline data on core debris/concrete interactions without water and (2) effect of water addition after a significant molten concrete layer has developed over the core debris.

PROGRAM ELEMENT 7: CONTAINMENT ANALYSIS

Work under Program Element 7 is being conducted at Sandia and Brookhaven National Laboratories. Programs entitled Molten Fuel-Concrete Interactions and Containment Analysis are underway at Sandia while Brookhaven is working on Thermal-Hydraulic Reactor Safety Experiments.

The Thermal Hydraulic Reactor Safety Experiments Program is divided into three major areas. Highlights of this quarter's progress in each area are presented below.

Heat Transfer in Core-Concrete Interactions

- Analysis of the single phase and two phase liquid-liquid film boiling experimental data continued.
- Analysis of the attack of molten core debris on the drywell containment liner in a Mark I BWR was completed.
- A series of liquid-liquid film boiling experiments with water as the boiling fluid were begun (Series 300 tests).

Ex-Vessel Quenching

- Reduction of data from the top-flood packed bed quenching experiments continued. Results of calculations made with models developed under this program were compared with the data.
- The experimental program to study the thermal interaction between falling particulates and water continued.
- The superheated packed bed quench data analysis continued. The data are being compared with bed quench models that include the effect of superheating the vapor.

In-Vessel Quenching

- A topical report entitled "Debris Bed Quenching Under Bottom-Flood Conditions (In-Vessel Degraded Core Cooling Phenomenology)" was completed.
- Work continued on improvement of the transient debris bed quench model and on preparation of the experimental apparatus for the next series of experiments.

PROGRAM ELEMENT 8: CONTAINMENT FAILURE MODE

The report on technical progress under Program Element 8 focuses on the Containment Safety Margins Program, the Integrity of Containment Penetrations Under Severe Accident Loads Study and the Electrical Penetration Assemblies (EPA) Program.

Containment Safety Margins

- Preparations for the October test of the large steel model (LSM) were completed including the application of transducers and gages, software for the data acquisition system, hook-up and checkout of the pressure controller, and construction of the theodolite stations.
- A draft of the Safe Operating Procedures for large model tests was completed and distributed for review.
- Analysis of the LSM using computer models continued.
- The request for proposal for the 1/6-scale model concrete containment was mailed to prospective contractors on August 2.
- Analysis of concrete containers continued. Liner behavior across cracks in reinforced concrete containments was studied using ABAQUS.

Integrity of Containment Penetrations Under Severe Accident Loads

- A draft of the overall program plan utilizing data from Argonne's survey of 19 plants was completed and submitted to the NRC.
- Argonne National Laboratory (ANL) has completed a survey of 23 additional plants and has issued a draft report.
- A draft task plan for the testing of seal and gasket materials has been prepared and submitted to the NRC.
- The following penetrations were tentatively selected for testing

- (i) BWR drywell head
 - (ii) equipment hatch with pressure seating and pressure unseating type seals for steel and concrete containments
 - (iii) bellows connections
 - (iv) airlocks with pressure seating and inflatable seals.
- Efforts continued to determine availability of penetrations from cancelled plants.
 - A survey of existing test facilities to carry out the proposed large scale penetration assembly testing was completed by ANL.

Electrical Penetration Assemblies

- The test plan has been submitted to the NRC.
- Staff members met with NRC staff on April 30 to discuss issues related to fault-current tests under severe accident conditions.
- Requests for quotes for electrical penetration assemblies (EPAs) were mailed to prospective suppliers in March. Some orders for EPAs were placed in May.
- The Area I steam system was completed and tested in Sandia's large test chamber with a "dummy" EPA inside.
- Several thermal aging simulations were conducted using the new forced air system.
- Installation of the transient superheat system in Area V began.
- Some equipment needed for leak-rate testing was delivered.

PROGRAM ELEMENT 9: FISSION PRODUCT RELEASE AND TRANSPORT

Program Element 9 encompasses fifteen programs conducted at six different laboratories. Accordingly, the technical progress summary is organized with one section for each laboratory.

Six projects under Program Element 9 are being carried out at ORNL. The project titles and a summary of the quarter's progress in each project are given below.

LWR Aerosol Release and Transport

- Nuclear Safety Pilot Plant (NSPP)
The NSPP facility was repaired. New equipment was added and tested. Preparations were made for testing of the plasma torch aerosol generators.
- Aerosol-Moisture Interaction Tests
Work began on development of an Aerosol-Moisture Interaction Test (AMIT) Facility in which the effects of various levels of humidity on physical characteristics of aerosols will be studied.
- Analytical
The latest version of MAEROS was received from Sandia and implemented on the ORNL computer system.

In response to an NRC request, some NAUA calculations were made for accident situations in both a PWR and a BWR.

Fission Product Release from LWR Fuel-High Temperature Tests

- Fuel Procurement and Characterization
Additional ORIGEN calculations of the fission product inventories in Ocone and Monticello fuel were made.

Fission Product Release Tests and Results

- The second test of the German-fabricated simulant fuel, HS-2, was performed in June. The sample was placed in a ThO₂ container and heated 10 min at 2000 C and 10 min at 2400 C. After the test, the fuel and container were fused. Extensive melting and oxidation of the Zircaloy cladding were observed, and some dissolution of UO₂ in the melt was apparent.
- Test HS-3 was conducted in July. In this test the specimen was heated at 2000 C for 21 min. For one minute, collection train #1 was used, and train #2 was used for the remaining time.

Iodine and Tellurium Chemistry

- Modeling - Iodine Transport Chemistry
The treatment of organic iodine in the model obtained from the Federal Republic of Germany (FRG) has been modified to make it consistent with the treatment used in the SASA program at ORNL.
- Iodine Volatility Measurements
Studies were done to investigate changes in iodine volatility with concentration of iodine in the sample and with pH.
- Iodine Chemistry Studies
Irradiation experiments continued in the all glass flow system. KIO₃ solutions with various pH factors were irradiated and the percent of IO₃⁻ lost and the percent of iodine volatilized were measured. There was an increasing amount of iodate lost with increasing pH.

TRAP-MELT Validation Tests

- Management and Analysis
TRAP-MELT 2 was used to model Aerosol Transport Tests A103 and A104. For both cases, the code significantly underpredicted the amount of aerosol deposited in the test pipe. Test A103 was also

modeled using the QUICK code whose results were very similar to those of TRAP-MELT.

- Aerosol Transport Tests

Aerosol deposit samples from test A104 (iron oxide aerosol) were collected and preparations for test A105 began.

- Aerosol Resuspension Tests

The last of the planned set of aerosol resuspension tests was completed during this reporting period. The tests involved tin-oxide aerosols and nickel and tungsten powders.

Fission Product Interaction with Aerosols

- Static Tests

Static tests were conducted and some data were analyzed. Results of the test indicate that CsOH vapor reacts with Cr_2O_3 to form Cs_2CrO_4 in a carrier gas of He, but that only weak surface interactions occur in a carrier gas of He-0.5 percent H_2 .

- Dynamic Tests

Silver and iron aerosols were successfully produced in the aerosol generator.

An operation manual has been written for the plasma torch system (aerosol generator).

In Pile Fission Product Release and Transport

- Fission Product Release from Fuel in PBF Tests

Noble gas releases during the heatup phase of SFD-ST and SFD1-1 were more than a factor of 10 lower than predicted by NUREG-0772 release rates. This was attributed to the low burnup of the PBF test fuel.

- Analysis of the Flow History in the PBF Fission Product Sampling Line

Hydrogen concentration versus time data for tests SFD-ST and SFD1-1 were analyzed. Total hydrogen generation in SFD-ST and SFD1-1 was calculated to be 297 g and 65 g, respectively.

All tasks under Program Element 9 being conducted at PNL have been consolidated under a single program, Effectiveness of LWR ESF Systems Under Severe Accident Conditions. Work performed under this program during the past quarter is briefly described below.

- An algorithm was developed for the ICEDF code to apply appropriate thermodynamic limits to particle growth.
- The modified code was used to study the effects of inlet steam concentration, initial particle concentration and initial particle size on the icebed decontamination factor.
- A model describing particle growth restrictions arising from the Kelvin effect was incorporated into ICEDF.
- In August further studies with ICEDF showed that particle retention in the icebed is greatest when the inlet steam velocity and the particle concentration are low.
- Staff members reviewed data from previous pool scrubbing experiments to determine whether they are adequate for SPARC validation.
- Efforts continued to determine the cost of refurbishing the Waltz Mill Ice Condenser Blowdown Facility for use in ICEDF code validation studies.
- A camera-ready copy of the report "Effectiveness of Engineered Safety Features System in Retaining Fission Products: Background Information" (NUREG/CR-3787) was sent to the NRC in June.

The Source Term Reassessment Program is being conducted at BCL. Efforts this quarter focused on writing Volume VII of BMI-2104, "Response to Comments and Questions".

The major Program Element 9 projects being conducted at Sandia are High Temperature Fission Product Chemistry and Quantitative Uncertainty Estimation for the Source Term (QUEST). This quarter's progress under these programs is described below.

High Temperature Fission Product Chemistry

- Reports were completed on (1) CsOH interactions with steel and Inconel in steam and (2) CsOH vapor pressure.
- Chemical interactions of CsI with steel are being documented.
- Responses were prepared to questions from the American Physical Society's committee for reviewing the source term assessment.
- Quantitative analyses of the gas phase mass transport limitations to release were undertaken to investigate whether this additional limitation might explain the differences between release results obtained in recent PBF tests and those from the out-of-pile tests conducted at ORNL.
- Chapters on in- and ex-vessel release of radionuclides for NUREG 1053 were drafted.

QUEST

- Work continued on the S2D accident for Surry, and the analysis of the TC accident in Grand Gulf was begun. A base case calculation for Grand Gulf was made using the ORNL version of MARCH which has been modified for BWRs.
- Interface coding to link the output of the ORNL-modified MARCH to MERGE and NAUA was developed.
- An assessment of the impact of a high-pressure melt ejection on suppression pool scrubbing was made for the Grand Gulf plant. Results indicate that the melt ejection could cause a flow of superheated steam through the saturated suppression pool strong enough to disperse the pool into droplets, thus temporarily eliminating the pool's capability to retain fission products.

Work is being done at Argonne under the Transient Fuel Response and Fission Product Release Program and the Post-Test Fuel Examination Program. Highlights of this quarter's work are presented below.

- A tellurium release model has been incorporated into FASTGRASS.
- Plans being made for 1985 include incorporation of fuel liquefaction release model into FASTGRASS and an update of the steam oxidation model.

PROGRAM ELEMENT 10: RISK CODES

Risk Code Development (MELCOR) is the only project under Program Element 10. It is being conducted at Sandia. Its goal is to develop an integrated set of risk codes, which perform both best estimate and quantitative uncertainty analyses, to replace MARCH/MATADOR/CRAC in the long term.

Work on MELCOR is focused on four primary categories: Offsite Consequences, Uncertainty/Sensitivity Methods, Coding, and Fission Product Behavior. A summary of the quarter's activities in each of these areas is presented below.

Offsite Consequences

- Emergency response models and supporting data were reviewed.
- The revised final draft of the Early Health Effects chapter of the Radiological Health Effects report was received.
- Recent papers on dose-response relationships for early health effects were reviewed, and dose-response relationships recommended in the Harvard study of radiological health effects were updated to reflect recent results.

Uncertainty/Sensitivity Methods

- The first draft of a report summarizing the comparison calculations made using various techniques for sensitivity/uncertainty studies was released for external review.
- Two papers describing the MAEROS sensitivity study were completed.
- A sensitivity study of the CRAC2 code was performed, and work began on the report describing the study.
- Performance of sensitivity calculations using the specially modified version of MARCH continued.

Coding

- The Control Volume Hydrodynamics package was modified.
- Time step control was improved by making time step length sensitive to the accuracy of the solution for the previous time step.
- The MELCOR code (executive structure plus appropriate modules) successfully ran a set of core degradation problems in May. Hydrogen burn test problems were also run using a different set of modules.
- Coding of the lower head module that models the relocation of degraded core material into the lower plenum, debris formation in the lower-plenum, and failure modes of the lower head was completed. The module was debugged and interfaced to MELCOR.
- The Control Function Package was debugged, interfaced to the MELCOR executive structure and successfully tested.
- Development of a core-concrete interactions module by reduction from CORCON continued.
- Coding of the emergency response, exposure pathways and dosimetry modules was completed. The modules were debugged and successfully tested.
- The fission product intercell convective transport module was coded, and implemented and successfully tested in MELCOR.
- MAEROS was implemented in MELCOR.
- The module to calculate decay heats for degraded core materials and for fission products released from those materials was coded, debugged, and tested.

Fission Product Behavior

- Work continued on selection of fission product behavior models to be used in MELCOR. Models used in MAEROS are being considered and appear to be compatible with other MELCOR modules.

PROGRAM ELEMENT 11: ACCIDENT CONSEQUENCE AND RISK EVALUATION

and

PROGRAM ELEMENT 12: RISK REDUCTION AND COST ANALYSIS

Program Elements 11 and 12 are both being conducted at Sandia under four programs: the Severe Accident Risk Reduction Program (SARRP); the Development and Analysis of Vent-Filtered Containment Conceptual Designs Program; the Consequence Modeling and Risk Analysis Program; and the Localized Deposition from Wet Plumes Program.

Severe Accident Risk Reduction Program

- The parent source term event trees were supplemented with sub-event trees which will simplify the parent trees and subdivide the problem into tractable pieces for frequency determination purposes.
- As a result of meetings with people from the NRC Accident Source Term Project Office and members of the QUEST team, the events on the containment event tree were reformulated and the quantification method was modified. The containment event trees for Surry sequences S2D and AB were completed, and estimates of outcome frequencies were calculated under various assumptions.
- The initial frequency rebaselining for NUREG-0956 was completed. A draft of "Containment Event Analysis and Estimation of Source Term Frequencies" was submitted to the NRC for review in July.
- Under the design and cost evaluation task several accident mitigation systems were investigated. The cavity water management system was deemed unfeasible, and the containment flooding option was discarded because the threats it could mitigate could be handled by a simpler system. Burns and Roe continued work on the integrated design of the hydrogen control, alternate containment spray, and passive condenser systems for the large dry, ice condenser and Mark III containments.

- Work began on the Rebaseline SARRP Data Bank.

Development and Analysis of Vent-Filtered Containment Conceptual Designs

- Rand Corporation's second revision of the Peach Bottom report was reviewed.

Consequence Modeling and Risk Analysis

- A trial uncertainty/sensitivity analysis was performed for the CRAC2 computer code using the techniques and tools developed at Sandia for the MELCOR program. The parameters in CRAC2 which potentially contribute significantly to uncertainty in estimated consequences were identified, and subjective estimates of the ranges, distributions, and correlations for those parameters were made.
- A draft of the full report on radiological health effects models, written at Harvard University, was completed and distributed for review.
- A draft report from Harvard entitled "Expedient Methods of Respiratory Protection: III. Submicron Particle Tests and Summary of Quality Factors" was reviewed.

Localized Deposition From Wet Plumes

- A literature search was conducted to discover the best way to incorporate modeling of turbulent agglomeration into the AERFLOW code. Work on the models continued throughout the reporting period.

PROGRAM ELEMENT 13: REGULATORY ANALYSIS AND
STANDARDS DEVELOPMENT

Pacific Northwest Laboratories (PNL) is doing most of the contracted work on Program Element 13. Work done this quarter under the project entitled Improved Methods for Incorporating Risk into Decision Making is summarized below.

- In the Shift Scheduling Case Study the risk evaluation underwent an internal review.
- In the USI A-47 Case Study, the risk assessment for the first sequence identified by INEL was completed and reviewed internally.
- Work in August on the Human Factors Case Study focused on studying existing regulations and regulatory initiatives in the human factors area and comparing them to the regulatory implications of the proposed human factors general design criterion.
- A value-impact analysis in the quality assurance area was begun.
- A draft working paper on graphical display techniques was submitted for the White Paper Working Group meeting.
- The program for the Statistics Symposium on National Energy Issues was completed. The program and registration papers were mailed to potential participants in August.

APPENDIX A

LISTING OF PROJECTS UNDER EACH PROGRAM ELEMENT

PROGRAM ELEMENT (1)		ACCIDENT LIKELIHOOD REEVALUATION		MONTHLY REPORTS RECEIVED				
FIN NO.	TITLE	NRC MONITOR	LABORATORY	APR	MAY	JUN	JUL	AUG
A1228	SNL Accident Sequence Evaluation	C. C. Eng	SNL		X	X	X	
A6301	INEL Accident Sequence Evaluation	C. C. Eng	INEL		X			X
B0435	Accident Sequence Precursors	F. Manning	ORNL			X	X	
B0468	Pressurized Thermal Shock	C. Johnson	ORNL			X	X	

PROGRAM ELEMENT (2)		SEVERE ACCIDENT SEQUENCE ANALYSIS		MONTHLY REPORTS RECEIVED				
FIN NO.	TITLE	NRC MONITOR	LABORATORY	MAY	JUN	JUL	AUG	SEP
A1258	Severe Accident Sequence Analysis-SNL	T. J. Walker	SNL			X	X	
A3273	Applications of RAMONA to BWR ATWS	B. Agrawal	BNL		X			
A6354	Severe Accident Sequence Analysis-INEL	B. Agrawal	INEL			X	X	
A7228	Severe Accident Sequence Analysis-LANL	B. Agrawal	LANL			X		X
B0452	Severe Accident Sequence Analysis-ORNL	T. J. Walker	ORNL			X	X	X

PROGRAM ELEMENT (3)		ACCIDENT MANAGEMENT		MONTHLY REPORTS RECEIVED				
FIN NO.	TITLE	NRC MONITOR	LABORATORY	APR	MAY	JUN	JUL	AUG
B0826	Human Factors Review for Severe Accident Sequence Analysis	C. Overby	ORNL			X	X	
B7499	SARP Support-Accident Management	R. Meyer	BCL			X	X	X

PROGRAM ELEMENT (4)		BEHAVIOR OF DAMAGED FUEL		MONTHLY REPORTS RECEIVED				
FIN NO.	TITLE	NRC MONITOR	LABORATORY	APR	MAY	JUN	JUL	AUG
A1335	LWR Debris Formation and Relocation	R. W. Wright	SNL	X	X			
A1340	LWR Core Debris Coolability	R. W. Wright	SNL	X	X			
A1342	LWR Accident Sensitivities and Melt Progression	J. Han	SNL	X	X			
A2220	Examination of TMI-2 Fuel Specimens	R. Foulds	ANL					
A6305	TFBP Severe Fuel Damage Studies	R. W. Wright	INEL					
A6352	Resident Scientist at KFK Karlsruhe, FRG	H. Scott	INEL		X	X		
A6360	Severe Fuel Damage Model Development	J. Han	INEL					
A7303	TRAC/MELPROG	J. Han	LANL			X	X	X

PROGRAM ELEMENT (4)		BEHAVIOR OF DAMAGED FUEL		MONTHLY REPORTS RECEIVED				
FIN NO.	TITLE	NRC MONITOR	LABORATORY	NOV	DEC	JAN	FEB	MAR
B2084	Severe Core Damage Subassembly Procurement	R. Van Houten	PNL	program completed				
B2277	NRU Coolant Boilaway and Damage Progression Tests	R. Van Houten	PNL					
B2455	Severe Core Damage Materials Property Tests	R. Van Houten	PNL					

PROGRAM ELEMENT (5)		HYDROGEN GENERATOR AND CONTROL		MONTHLY REPORTS RECEIVED				
FIN NO.	TITLE	NRC MONITOR	LABORATORY	APR	MAY	JUN	JUL	AUG
A1246	Hydrogen Behavior Program	J. T. Larkins	SNL	X	X			
A1255	Combustible Gas in Containment	J. T. Larkins	SNL	X	X			
A1270	Hydrogen Burn Survival Experiment	W. S. Farmer	SNL		X	X		
A1336	Hydrogen Combustion Preventative and Mitigative Schemes	P. Worthington	SNL	X	X			
A7247	Hydrogen Migration and Mixing	J. T. Larkins	LANL					
B7232	Charcoal Performance Under Accident Conditions	J. T. Larkins	NRL	complete except final report				
B8208	Hydrogen Combustion and Control Demonstration Experiments	J. T. Larkins	N00	program completed				

PROGRAM ELEMENT (6)		FUEL-STRUCTURE INTERACTION		MONTHLY REPORTS RECEIVED				
<u>FIN NO.</u>	<u>TITLE</u>	<u>NRC MONITOR</u>	<u>LABORATORY</u>	<u>APR</u>	<u>MAY</u>	<u>JUN</u>	<u>JUL</u>	<u>AUG</u>
A1030	Core Melt/Coolant Interactions	J. L. Telford	SNL	X	X			
A1218	Core Melt Technology	T. Lee	SNL	X	X			
A1247	Containment Load Mitigation	T. Lee	SNL	X	X			

PROGRAM ELEMENT (7)		CONTAINMENT ANALYSIS		MONTHLY REPORTS RECEIVED				
<u>FIN NO.</u>	<u>TITLE</u>	<u>NRC MONITOR</u>	<u>LABORATORY</u>	<u>APR</u>	<u>MAY</u>	<u>JUN</u>	<u>JUL</u>	<u>AUG</u>
A1019	Molten Fuel-Concrete Interactions	S. B. Burson	SNL					
A1198	Containment Analysis	S. B. Burson	SNL					
A3024	Thermal-Hydraulic Reactor Safety Experiments	S. B. Burson	BNL			X	X	X
A7301	Pressure Threats to Containment	J. L. Telford	LANL					

<u>PROGRAM ELEMENT (8)</u>		<u>CONTAINMENT FAILURE MODE</u>		<u>MONTHLY REPORTS RECEIVED</u>				
<u>FIN NO.</u>	<u>TITLE</u>	<u>NRC MONITOR</u>	<u>LABORATORY</u>	<u>APR</u>	<u>MAY</u>	<u>JUN</u>	<u>JUL</u>	<u>AUG</u>
A1249	Containment Safety Margins	J. Costello	SNL			X	X	X
A1364	Electrical Penetration Assemblies	W. S. Farmer	SNL	X	X			
A1375	Containment Penetrations	H. Ashar	SNL			X	X	X
A6322	Equipment Qualification (Valves)	W. E. Campbell	INEL					

<u>PROGRAM ELEMENT (9)</u>		<u>FISSION PRODUCT RELEASE AND TRANSPORT</u>		<u>MONTHLY REPORTS RECEIVED</u>				
<u>FIN NO.</u>	<u>TITLE</u>	<u>NRC MONITOR</u>	<u>LABORATORY</u>	<u>APR</u>	<u>MAY</u>	<u>JUN</u>	<u>JUL</u>	<u>AUG</u>
A1227-A	High Temperature Fission Product Chemistry	L. Chan	SNL	X	X			
A1227-B	Sensitivities	M. Jankowski	SNL	X	X			
A2016	Transient Fuel Response and Fission Product Release	L. Chan	ANL					
A2232	Post-Test Fuel Examination (ORNL Fission Product Release Test Specimens)	L. Chan	ANL					
A6321	In-Pile Fission Product Behavior Studies	P. Reed	INEL					
A6829	PBF Fission Product Measurement	P. Reed	INEL					

<u>PROGRAM ELEMENT (9)</u>		<u>CONTAINMENT FAILURE MODE</u>		<u>MONTHLY REPORTS RECEIVED</u>				
<u>FIN NO.</u>	<u>TITLE</u>	<u>NRC MONITOR</u>	<u>LABORATORY</u>	<u>APR</u>	<u>MAY</u>	<u>JUN</u>	<u>JUL</u>	<u>AUG</u>
B0121	LWR Aerosol Release and Transport	J. Telford	ORNL			X	X	
B0127	Fission Product Release from LWR Fuel-High Temperature Tests	L. Chan	ORNL			X	X	
B0453	Post-Accident Fission Product Chemistry	L. Chan	ORNL			X	X	
B0488	Trap-Melt Verification Tests	M. Jankowski	ORNL			X	X	
B0815	Fission Product Deposition on Aerosols	M. Jankowski	ORNL			X	X	
B0827	PBF/SFD Coordination	M. Jankowski	ORNL			X		
B0831	Marviken Support	M. Jankowski	ORNL					
B2444	Effectiveness of LWR ESF Systems Under Severe Accident Conditions	C. Nilsen	PNL			X	X	X
B6747	Fission Product Transport	M. Jankowski	BCL					
B7580	Marviken Program	M. Jankowski	Swe.					

<u>PROGRAM ELEMENT (10)</u>		<u>RISK CODES</u>		<u>MONTHLY REPORTS RECEIVED</u>				
<u>FIN NO.</u>	<u>TITLE</u>	<u>NRC MONITOR</u>	<u>LABORATORY</u>	<u>APR</u>	<u>MAY</u>	<u>JUN</u>	<u>JUL</u>	<u>AUG</u>
A1339	Development of Improved Physical Process Computer Codes for Risk Assessment (MELCOR)	M. A. Cunningham	SNL		X	X	X	

PROGRAM ELEMENT (11) and (12)		ACCIDENT CONSEQUENCE AND RISK EVALUATION and RISK REDUCTION AND COST ANALYSES		MONTHLY REPORTS RECEIVED				
<u>FIN NO.</u>	<u>TITLE</u>	<u>NRC MONITOR</u>	<u>LABORATORY</u>	<u>APR</u>	<u>MAY</u>	<u>JUN</u>	<u>JUL</u>	<u>AUG</u>
A1042	Consequence Modeling/Risk Analysis	J. A. Martin, Jr.	SNL		X	X	X	
A1322	Severe Accident Risk Reduction Program (SARRP)	M. A. Cunningham	SNL		X	X	X	
A1220	Development & Analysis of Vent-Filtered Containment Conceptual Designs	M. A. Cunningham	SNL		X	X	X	
A1369	Localized Deposition from Wet Plumes	J. A. Martin, Jr.	SNL		X	X	X	

PROGRAM ELEMENT (13)		REGULATORY ANALYSIS AND STANDARDS DEVELOPMENT		MONTHLY REPORTS RECEIVED				
<u>FIN NO.</u>	<u>TITLE</u>	<u>NRC MONITOR</u>	<u>LABORATORY</u>	<u>APR</u>	<u>MAY</u>	<u>JUN</u>	<u>JUL</u>	<u>AUG</u>
A1334	Regulatory Decision-Making and Policy Analysis	M. Fleischman	SNL					
B2386	Improved Methods for Considering Risk in Decision-Making	A. J. DiPalo	PNL				X	X

AUGUST 1984 *Agnewal*

Severe Accident Research Program

Quarterly Report
August, 1984



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The time required to reach dryout appeared not to increase significantly as the percentage of power above the dryout power was reduced from 10 percent to 5 percent. On the other hand, some dryout predictions verified by DCC-1 were that a rapid increase in power can pressurize and temporarily fluidize the bed and that after a change in power to a power significantly above the dryout power (a factor of 5 or more), the bed will dry out (or fluidize) quickly.

- Preparations for DCC-2 continued. Components of the experimental system were assembled and tested. The urania fuel for the test was received. Hardware for the data acquisition system was assembled and work was begun on the software.

LWR Accident Sensitivities and Melt Progression (SNL)

- A report entitled "Considerations in the Behavior of LWR Fuel and Coolant Systems During Severe Accidents" was completed. (This report was noted last quarter, but its title has been changed.) No further activities are planned under the Core Damage Sensitivities Studies portion of this program.
- The final draft of the report entitled "Identification of Severe Accident Uncertainties" was completed.
- MELPROG was run to model an S₁D loss-of-coolant accident in the Zion PWR. Computer calculations continued through vessel failure which occurred in the bottom vessel head at 5900 seconds.
- The Sandia version of MELPROG was merged with the Los Alamos version of MIMAS.
- Work continued on several modules within MELPROG. The DEBRIS module for heatup, oxidation, and melt progression of the debris bed was developed. Improvement of the modeling of mass and energy transfer between the FLUIDS and DEBRIS modules required particular attention. Design of an improved structural loading model was completed, and coding was begun.

Severe Fuel Damage Program (INEL)

- The test train for the SFD 1-3 experiment was completed and delivered to the PBF on April 12, 1984, and was subsequently installed in the in-pile tube. A mockup of the test train was fabricated and used to develop procedures for test train installation and removal from the in-pile tube. The upper plenum heater control system installation was completed, as was checkout of the deposition rod handling and transport equipment. Additional instrumentation to provide pressure, temperature, and level indications was installed on the sample system and calibrated.
- An integrated Systems Operational Test was performed to provide a final checkout of the facility modifications installed for Test SFD 1-3.
- The Experiment Safety Analysis (ESA) for Test SFD 1-3 was completed.
- Thermal and stress analyses were performed to model the inward bulging behavior that occurred during Test SFD 1-1, and an Engineering Design File was written to summarize the results.
- The SFD 1-4 test train pressure sensors have been installed and the control rod is ready to be pressurized. The sample line routing and location of the main floor gamma spectrometer were established, and conceptual design of the aerosol monitor continued. The test train radiation shield, closure head, and steam tube assembly were completed and the irradiated rods were installed in the fuel bundle.
- Metallography of the cross sections from the SFD Scoping Test fuel bundle was completed. Release rate constants for several fission product volatility groups have been determined, and a report on the fission product analysis of the Scoping Test is being prepared. Preliminary results of the Scoping Test bundle behavior analysis indicate that bundle power and coolant flow

were 10 percent greater than measured, and that complete zircaloy oxidation prevented liquefaction from occurring.

- The report on the Test SFD 1-1 liquid levels was received from the University of Washington and will be incorporated as an appendix to the Test Results Report. The Test SFD 1-1 sample analysis report was drafted. Fallback barrier samples were collected and sent to the counting lab.

Severe Fuel Damage Model Development (INEL)

- The major accomplishment was the creation of SCDAP/MOD1/VO which expanded the code capabilities from treatment of bundle heatup, disruption, and debris formation and behavior to include modeling of a full reactor vessel including simplified upper and lower plenum models for thermal-hydraulics and structures. These models allow for feedback between the plena and the core region during severe disruption. This version was assessed using the PBF SFD 1-2 test data and using a TMI-2 analysis. The results of the assessment, a code description and user information were documented in report number IS-SAAM-84-002 (January 1984).
- INEL personnel prepared position papers for the NRC on core heatup and hydrogen generation phenomenology. These were done in response to IDCOR reports on the two subjects.
- A draft report on the fission product release, transport, and deposition models for the linked SCDAP/RELAP5/TRAP-MELT code was prepared. This report documents the models to be included directly from TRAP-MELT, the models which are based on, but different from the TRAP-MELT models, and models which are needed to supplement the TRAP-MELT models. Phenomena to be treated include agglomeration, condensation, impaction, thermophoresis, and settling. Transport will be treated by RELAP5 and release by SCDAP.

TRAC/MELPROG (LANL)

- The TRAC calculation of the flow pattern in the Surry vessel during a TMLB sequence (loss of feedwater with failure of the emergency core cooling system) was completed. The calculation indicates that a circulation pattern that has very little flow going to the upper head is set up in the upper plenum. The major energy removal mechanism is flow out the hot-leg to the relief valves. Flow from the core provided by MARCH may be underestimated because it does not take into account recirculation to the core region. Including the core region in the calculation is, therefore, important for determining energy partition.
- In February the fourth fluid-dynamics field was added to the standalone two-dimensional fluids module which is being incorporated into MELPROG. This module can handle vapor/gas, liquid water, debris and molten debris. In April the hydrogen component of the vapor field was added and tested.
- The combined MIMAS/MELPROG code has been renamed MELPROG.
- Work continued on the testing and debugging of the one-dimensional version of TRAC/MELPROG. A one-dimensional model of the Oconee-1 Plant was developed for this test effort.

Resident Scientist at KfK

- Three severe fuel damage tests, ESA-3, 4, and 5, were run in the NIELS Facility during this reporting period. In these tests, single rods were heated to 1700°C, 1900°C, and 2200°C, respectively. After the tests the fuel rods were examined visually. Some of the Zircaloy cladding melted in all tests, and in test ESA-5 some UO₂ melted also.
- The BETA facility which will be used to study core melt-concrete interactions was completed and three preliminary tests were run.
- Tests were run in the SASCHA Facility to study the transport behavior of gaseous I₂ and HI. In these tests, the amount of

iodine plated out on the effluent system transport line, fiberglass filters, and iodine filter beds was measured. When the amount plated out was compared to the initial inventory, between 84 and 93 percent of the initial inventory was accounted for.

PROGRAM ELEMENT 5: HYDROGEN GENERATION AND CONTROL

The majority of the tasks under Program Element 5 are being performed at Sandia. One project which was conducted at the Nevada Test Site was completed last quarter. This quarter's progress in each major task is summarized below.

FLAME Facility

In this time period, the first test series was completed. In this series, the hydrogen concentration was varied between 12 percent and 28 percent. The top of FLAME was half covered with steel plates. There were no obstructions in the channel except for the four thermocouple rakes and two small fans. The results of the test series were:

- (1) No detonations were seen.
- (2) No flame acceleration was seen.
- (3) The flame speed varied greatly with hydrogen concentration, from about 6 m/s at 12 percent to 107 m/s at 28 percent.
- (4) The flame front was steeply inclined with the top leading.
- (5) The peak overpressures seen at stoichiometric were small, 30 to 40 kPa (2 to 3 psi). At lower hydrogen concentrations, the overpressures were negligible.

Six tests were conducted in a second test series without top venting, from 6.9 percent hydrogen to near stoichiometric. The results were considerably different from the first series with 50 percent top venting. Some tentative results of this test series were:

- (1) The overpressures were much higher.
- (2) The flame speeds were higher.
- (3) There was some flame acceleration.
- (4) There appeared to have been a transition to detonation in the near-stoichiometric test just near the channel exit.
- (5) For the 6.9 percent hydrogen test, only a thin layer of hydrogen just under the top plates, of thickness between 3 and 15 cm, burned.

Hydrogen Burn Survival Program

- Two NEMA-4 boxes were installed in the FLAME facility and subjected to two burns to investigate the effect of this type of burn on equipment. The first burn contained 8 percent by volume hydrogen, and had no visible effect on the boxes. The second burn was a stoichiometric mix and caused considerable damage both to the boxes and the facility. The boxes were substantially deformed, indicating that there were severe pressure differentials between the inside and outside of the boxes.
- Using HECTR, a prediction of equipment surface temperatures was made for various accident sequences in an ice condenser containment. A draft report of the study was provided to the NRC in February.
- Work continued on HYBER, the Hydrogen Burn Equipment Response computer code. Final modifications to the routines in HYBER were completed. A users' guide and reference manual were delivered to the NRC.
- Modifications were made in the subroutines of the computer code SMOKE. This code will be used in analysis of NTS, FITS, and VGES data.
- The configuration of HECTR to run the TMI-2 accident sequence was begun.
- The fan cooler model for HECTR was completed in April.

HECTR Analysis and Code Development

- Work continued on the HECTR users' manual and on preparation of the code for release.
- Two projects were done in support of the Containment Loads Working Group: (1) the PWR ice-condenser standard problem was run using MARCH coupled with CORCON to generate source terms for HECTR for a TMLB' accident scenario. HECTR was then executed

separately to do the containment analysis. (2) The BWR Mark III standard problem was run.

- The report on the MARCH-HECTR analysis for an ice condenser plant was revised and will be released next quarter.
- The major HECTR assessment effort was begun.
- Performance comparison calculations for the Nevada Test Site (NTS) tests were begun.
- Two models were developed for incorporation into a postrelease version of HECTR: (1) the CO/CO₂ model for radiation heat transfer and (2) a fan cooler model.

FITS Facility

- Preparations continued for future hydrogen combustion experiments in the FITS tank. Two groups of tests are planned. In the first group about 25 hydrogen:air:steam combustion experiments will be used to assess the burn time and burn completeness correlations used in the HECTR code. The second group consists of hydrogen:air:water fog experiments for the Hydrogen Mitigation Program.

CONCHAS-SPRAY Calculations

- A series of calculations of hydrogen combustion in closed or vented vessels was completed to review the performances of CONCHAS-SPRAY and HECTR in computing vessel overpressures. Results of the CONCHAS-SPRAY and HECTR calculations agreed quite well.

Vortex Dynamics Modeling of Flame Acceleration

- Efforts were focused on numerical simulation of premixed flame propagation through a single vorticity region formed when gas flows past the exit of a channel. Overall agreement between experiments and calculations was good. Both experiments and

calculations showed that the flame front jets through the center of the channel and enhanced burning takes place behind each of the obstacles. However, the calculated rate of acceleration of the gas was too low--perhaps by a factor of 2.

- A report entitled "Simulation of Flame Propagation Through Vorticity Regions Using the Discrete Vortex Method" (NUREG/CR-3835), was published.

Hydrogen-Steam Jet-Flame

- Report SAND84-0060, Hydrogen-Steam Jet-Flame Facility Experiments, was revised and distributed for review.
- The small-scale experimental facility was dismantled.

Diffusion Flame and Heat Transfer Modeling

- A simple model of combustion and heat transfer was developed and used to model several of the NTS continuous-injection tests. Agreement between experimental results and calculations was good.

Heated Detonation Tube

- During this reporting period, thirteen tests were conducted completing the planned ambient temperature tests. Results of the tests indicate that increased temperatures cause a decreased detonation cell size and an increased mixture detonation sensitivity.

Operability of Igniters in a Water Spray Environment

- A draft report entitled "Operability of a General Motors AC 7G Glowplug Igniter in a Water Spray Environment" was submitted to the NRC.

- Horizontal gas flows were observed during the tests on operability of igniters in water spray environments. Experiments are being developed to determine whether the horizontal flow is strong enough in a severe accident to reduce the protection offered by the spray shield.

Aerosol Studies

- Work continued on reduction of thermocouple data recorded during eight oxidic aerosol hydrogen combustion experiments performed in the VGES tank.

PROGRAM ELEMENT 6: FUEL-STRUCTURE INTERACTION

Program Element 6 encompasses three projects, all of which are being conducted at Sandia. The three programs are Core Melt/Coolant Interactions, Core Melt Technology, and Containment Load Mitigation. Highlights of this quarter's progress are presented below.

Core Melt Technology

- SPIT 18 and 19 were conducted. In SPIT 18 a 10.3-kg thermally generated melt pressurized to 2250 psi was ejected into a 1:20-scale reactor cavity. In SPIT 19 a 10.3-kg melt pressurized to 1852 psi was ejected into a 1:20-scale reactor cavity. Temperature and pressure readings were taken in the chamber. Particle size distribution was determined. Preliminary results of SPIT 18 indicate that about 1 percent of the ejected material was converted to particles of diameter $<15\text{ }\mu\text{m}$. About 60 percent of the melt mass was ejected in the 1:20-scale test. About 95 percent was ejected in previous 1:10-scale tests. Only about 30 percent was ejected in an earlier 1:30 scale test at Argonne. A preliminary conclusion is that nearly 100 percent melt ejection can be expected in a full-scale accident. Data analysis continued.
- All components required for the Molten UO_2 /Concrete Interactions Tests (TURC series) were received during the quarter. The first two tests were conducted.
- HSS-1, hot solid scoping test, was conducted using hot solid stainless steel as a surrogate for UO_2 . It was a preliminary test in a series designed to characterize hot solid UO_2 debris/concrete interactions. In the preliminary test gamma-ray imaging of core debris interacting with concrete was successfully employed.
- HIPS-1J and HIPS-2C were performed. In both tests an 80-kg iron/alumina thermite was pressurized to about 1500 psi and

ejected from a 1:10-scale reactor cavity. High speed photographs were taken. In test HIPS-2C about 96 percent of the melt was ejected and was lofted to an elevation of about 125 ft and a distance of 275 ft from the reactor cavity mock-up.

- SNL staff have undertaken some projects for the Containment Loads Working Group this quarter. They have worked on estimates of the possibility of prompt failure of the diaphragm floor below the reactor vessel in a Mark II BWR containment. They have also calculated potential effects of direct containment heating.
- VANESA calculations are being made for comparison with DECOMP results. DECOMP is IDCOR's model of core debris interactions.

Containment Load Mitigation

- Transient Water Tests TWT-1A and TWT-1B were conducted. The objective of these tests was to determine whether water can mitigate core debris interactions with concrete. In these tests a 45-kg melt was prepared in concrete crucibles by metallothemic reaction, and water was streamed onto the melt. Decay heating of the melt was simulated by induction. Temperature, pressure, aerosol formation and strain on the concrete were measured. In test TWT-1A pressure inside the concrete crucible rose quickly, and some of the melt was expelled, damaging the water quench actuation system. Test TWT-1B was successful. It resulted in a benign quench with no steam explosion or melt fragmentation. Hydrogen generation during TWT-1B was significant. During formation of the thermitic melt, some instruments were damaged in both TWT-1A and TWT-1B. The problem can be solved by using an inductive melt formation system which has been developed for future tests in this series.
- Reduction of data from large-scale UO_2 interactions tests with MgO and Al_2O_3 particle beds continued.

PROGRAM ELEMENT 7: CONTAINMENT ANALYSIS

Work under Program Element 7 is being conducted at Sandia and Brookhaven National Laboratories. Programs entitled Molten Fuel-Concrete Interactions and Containment Analysis are underway at Sandia while Brookhaven is working on Thermal-Hydraulic Reactor Safety Experiments.

The Containment Analysis Program involves development of an interactive generic system code, CONTAIN, for the analysis of abnormal loads imposed on reactor containment systems under severe accidents. Activities for the quarter included:

- The semi-implicit pool boiling and atmospheric flow model developed for MEDICI was incorporated into CONTAIN. This model allows for interactive coupling between the pool to atmosphere heat and mass transfer and atmospheric thermodynamics and flow.
- Refinement and testing of the Engineered Safety System (ESS) models for CONTAIN continued.
- Efforts in November focused on the accumulation of information and then a coding effort to define heat transfer coefficients for various conditions. Some of those conditions include heat transfer between the various layers in the lower cell and heat transfer to the walls within the lower cell, to particles associated with sprays, and to aerosols.
- It was decided that HECTR will be the hydrogen burn module in CONTAIN.
- The model which treats aerosol washout by containment sprays was upgraded.
- A realistic M1-M2 integration test case was run with MEDICI.
- In October CONTAIN analyses of the Surry AB-γ and AB-δ accident sequences were completed.
- CONTAIN was used to calculate fission product intercell transport and fission product heating of structures. Hand calculations

were made also and compared with the CONTAIN results. There was good agreement.

The Thermal-Hydraulic Reactor Safety Experiments Program is divided into three major areas. Highlights of this quarter's progress in each area are presented below.

Heat Transfer in Core-Concrete Interactions

- Eleven tests of liquid-liquid film boiling with gas injection were performed in March. The test apparatus was destroyed in the last test.
- The liquid-liquid boiling apparatus was rebuilt in April, and analysis of the boiling heat transfer data within gas bubbling from below was begun. Seven heat balance experiments were performed to assess several trends discovered in the analysis of earlier data.
- Construction of a visualization apparatus for liquid-liquid film boiling with gas bubbling was initiated. High-speed photography will be utilized to examine the film boiling interface with and without non-condensable gas bubbling from below.
- Analysis of the liquid-liquid film boiling data, both with and without noncondensable gas bubbling from below, continued.

Ex-Vessel Quenching

- Preliminary experiments were performed with 3-mm diameter spheres, preheated to 1300 F. Five- and ten-kilogram masses of this shot have been dropped into pools of saturated water of 0.5 and 1.0 meter depths, contained in a pipe of 100-mm diameter. Steam generation rate was measured giving an integral measure of the interaction process, and high-speed motion pictures have been taken. The results thus far indicate that the initial interaction of water and particles is quite efficient.

- Particle quench experiments in one-dimensional geometry using 5-15 kg masses of 6-mm stainless steel spheres were completed.
- Preparation of a final report on the debris bed quenching experiments was initiated. The analytical model characterizing the bed quench process was modified to include the effect of superheating of the steam. A program to perform final data reduction was written and debugged.

In-Vessel Quenching

- A transient debris bed quench model, developed earlier to predict the time history of void fraction and solid temperature at various axial locations within the debris bed during the quenching process, was extended to include the fluid momentum equations explicitly. The Lipinski model for the solid-fluid interfacial drag terms is being used.
- Modification of the test apparatus for the next series of experiments was begun. In those experiments, the debris bed height will be doubled.

PROGRAM ELEMENT 8: CONTAINMENT FAILURE MODE

The report on technical progress under Program Element 8 focuses on the Containment Safety Margins Program and the Integrity of Containment Penetrations Under Severe Accident Loads Study being conducted at Sandia. Highlights of this quarter's activities are listed below.

Containment Safety Margins

- Work continued on preparation for testing of the large steel model (LSM). Interior and exterior strain gages have been attached. A plan for instrumenting the hatches with displacement transducers was completed and implemented.
- Material properties tests for the steel in the LSM were completed.
- Finite element analyses of the personnel lock representation and the surrounding shell were completed. Results indicate that the strains in the cylinder near the thickened section around the penetration increase more rapidly than in the free-field membrane region.
- A data report for the SC-3 test was distributed to NRC and to contractors who had predicted the behavior of the models during the test.
- The purchase requisition for a one-sixth size model of a reinforced concrete containment has been approved by Sandia's line management.

Integrity of Containment Penetrations Under Severe Accident Loads

- A draft of the overall program plan utilizing the data from the survey of 19 plants was submitted to the NRC for review and comments in May.
- Work continued on development of a conceptual design for the test fixture for testing seals and gaskets in a severe accident

environment. The fixture will utilize the test set-up that is being built and tested under the Electrical Penetration Assemblies Program.

- A draft test plan for seal and gasket materials was submitted to the NRC for review.
- Preliminary temperature and pressure profiles for BWRs and PWRs in severe accident environments were developed.
- Analyses of an equipment hatch and a BWR Mark II drywell head under severe accident conditions to evaluate their structural behavior and susceptibility to leakage was completed at ANL. A report on the work was sent to SNL for review. ANL also completed a scoping evaluation of a typical piping bellow connection to determine the critical parameters that affect its structural behavior and has concluded that large deflections are more likely than high pressure to cause failure.

PROGRAM ELEMENT 9: FISSION PRODUCT RELEASE AND TRANSPORT

Program Element 9 encompasses fifteen programs conducted at six different laboratories. Accordingly, the technical progress summary is organized with one section for each laboratory.

Six projects under Program Element 9 are being carried out at ORNL. The project titles and a summary of the quarter's progress in each project are given below.

LWR Aerosol Release and Transport

- Nuclear Safety Pilot Plant

The sixth U_3O_8 aerosol test in a quasi-steady state steam-air environment (NSPP 407) was conducted in February.

Test number 522, the second limestone-aggregate concrete aerosol test in a quasi-steady state steam-air environment, was conducted in April. Preliminary results indicate that the aerosol mass concentration reduction rates for tests 522 and 521 (identical to 522) were similar. Test 531, differing from 521 and 522 only in that dry air was used, produced similar results. It appeared that condensing steam had little effect on the mass concentration reduction rate for concrete aerosols even though it changed the shape of the aerosol from chain agglomerate to partially spherical.

- LWR Core-Melt Studies

Successful operation of the 250-kW rf induction furnace was demonstrated by heating a simulated large fuel bundle (126 empty, stainless steel tubes, 12 in. in length and weighing a total of 14.2 kg) to almost melting ($\sim 1450^\circ\text{C}$) in 30 seconds.

- Marviken Technical Support

The final recalibration of the Lovelace small spiral aerosol particle centrifuge was completed to a maximum particle diameter of $8.13\text{ }\mu\text{m}$ AMMD.

- Analytical

Data from various tests in NSPP were analyzed.

As part of the aerosol code validation project, work continued on the various models that comprise the NAUA code.

Fission Product Release from LWR Fuel--High Temperature Tests

- Fuel Procurement and Characterization

A shipment of simulated high burnup LWR fuel was received from Karlsruhe, Germany, in mid-April.

- Fission Product Release Tests and Results

The first test using simulated high burnup LWR fuel from Germany was conducted in May. The 128-gram specimen was heated 15 min. at $\sim 1600^{\circ}\text{C}$ and 30 min at $\sim 1900^{\circ}\text{C}$ in a steam/helium atmosphere. After the test, the specimen was swollen, distorted and thoroughly oxidized but almost completely intact. Very high release fractions of the simulated volatile fission products were observed.

- Species Identification by Laser-Induced Fluorescence Spectrometry

Preliminary tests were done to evaluate the feasibility of this technique for fission product identification and measurement.

Iodine and Tellurium Chemistry

- Modeling--Iodine Transport and Chemistry

A German model for iodine transport was installed in the ORNL computer system. A program to convert the output from this model to graphical form was written. The German model was tested for sensitivity to different values of the iodine partition coefficient and to different CsI and I_2 ratios.

- Iodine Volatility Measurements

Work continued on studies of the effects of radiation on iodine volatility. Results show that the presence of silver powder significantly decreases iodine volatility during irradiation.

- Iodine Chemistry Studies

Irradiation experiments in the all-glass flow system continued. Some specific goals of this quarter's work included eliminating some experimental uncertainties and obtaining a mass balance of iodine after irradiation.

TRAP-MELT Verification Tests

- Aerosol Transport Tests

Test A103, an iron-oxide test in which aerosols were generated for 9 min, was conducted in March. Analysis of samples began in April. Test A104 was conducted in May.

- Aerosol Resuspension Tests

Dozens of tests using tungsten, manganese, and nickel powder and iron-oxide aerosols were performed during this reporting period. Results indicate that resuspension rates for the different materials vary significantly even though flow conditions are similar.

- Analysis

Efforts continued to make the computer code TRAP-MELT 2 operational on the ORNL computer.

Fission Product Interaction with Aerosols

- Static Tests

Through February, the static test apparatus was used to evaluate the seven candidate aerosol materials.

During the rest of the reporting period efforts were focused on preparation of the test apparatus and the samples.

- Dynamic Tests

Preparation of the equipment required for the tests continued.

- Planning and Analysis

Final revisions were made on two reports: "Quantity and Nature of LWR Aerosols Produced in the Pressure Vessel During Core Heatup Accidents--A Chemical Equilibrium Estimate", NUREG/CR-3181 and "Fission Product Deposition on Aerosols: Program Plan".

Reactions between CsI or CsOH and metallic or oxidic aerosols were evaluated thermodynamically.

A draft of "Data Summary Report for CsI Vapor Interaction with Several Aerosol Materials and Thermodynamic Evaluation of the CsOH-Fe-Fe₂O₃ Interactions" was submitted for final approval.

In-Pile Fission Product Release and Transport

- Fission Product Release from Fuel
in PBF Tests

Fission product releases for PBF tests ST and SFD1-1 were estimated based on reported fuel temperature histories and NUREG-0772 release rates. Calculated and measured releases agreed reasonably well. However, the NUREG-0772 correlation overpredicted early releases and underpredicted cooldown releases. Work began on development of an empirical release model.

- Analysis of Flow History in PBF
Fission Product Sampling Line

A computer program was written to analyze hydrogen concentration vs. time data from SFD-ST and SFD1-1 tests.

- Analysis of Fission Product Distributions
in the PBF Sampling Line

Work continued on interpretation of reported iodine partition coefficients for the SFD-ST streamline samples.

All tasks under Program Element 9 being conducted at PNL, have been consolidated under a single program, Effectiveness of LWR ESF Systems Under Severe Accident Conditions. Work performed under this program during the past quarter is briefly described below.

- A condensational particle growth model was developed for incorporation into SPARC and ICEDF. Work continued on debugging the model.
- Additional sensitivity studies were performed using the ICEDF code. The studies show that the most important factors in determining particle retention are particle size, ice surface availability and inlet air humidity.
- A computer program was written to estimate the aerosol deposition in air cooling coils such as typically used in containment air coolers. This program, called KOILDF, includes the contribution of the following deposition mechanisms:
 - Thermophoretically augmented diffusion
 - Diffusiophoresis
 - Impaction
 - Interception.

The program will predict penetration factors, decontamination factors, and mass removal efficiencies for each of 20 particle grades described in the program input. In its present form KOILDF is applicable only to atmospheric pressure situations.

- A code needed to compute the "puff" release resulting from a bottom head failure was completed. The code system being assembled to evaluate ESF effectiveness is not yet complete. VANESA is not yet available.
- Two reports were published
 - "Plutonium Recycle Test Reactor (PRTR) Accident: A Final Report on the Investigation of Fission Product Chemical Forms", NUREG/CR-3669 (PNL-5003).
 - "Fission Product Removal in Engineering Safety Feature (ESF) Systems: Data Base Assessment and Suggested Experimental Program", NUREG/CR-3727 (PNL-5050).

The Source Term Reassessment Program is being conducted at BCL. Efforts this quarter were focused on the revision of BMI-2104 and on the combination of MERGE and TRAP-MELT. Work on BMI-2104 included:

- Source terms for Peach Bottom and Grand Gulf were recalculated using the corrected SPARC code.
- Thermal hydraulic calculations were made for the variable containment leakage rate cases for Zion.
- Source term calculations were repeated for representative sequences for each plant to provide release fractions for the WASH-1400 fission product groups. This work is in response to a Peer Review Group request that more fission product species be included.
- Source term calculations for Zion were completed for the TMLB' sequence with two variable leak rate cases and for an isolation failure case.
- Calculations for Volume V were completed in May.

The major Program Element 9 projects being conducted at Sandia are High Temperature Fission Product Chemistry and QUEST. This quarter's progress under these programs is described below.

High Temperature Fission Product Chemistry

- An experiment to react tellurium vapor with 304 stainless steel in a steam and hydrogen environment was successfully completed. It is hypothesized that earlier attempts to produce this reaction were unsuccessful because of the formation of an oxide layer on the stainless steel. In the most recent test, steps were taken to reduce the rate of stainless steel oxidation.
- Two reports were completed. One dealt with the interactions of CsOH with stainless steel and Inconel. The other contained calculations of the thermodynamics of the Cs-O-H-B-I vapor phase system.

QUEST

- A draft report on the TMLB' analyses for Surry was completed. Sources of uncertainty in calculations of radionuclide releases from containment were identified, and the uncertainty ranges were estimated.
- Work continued on analysis of the S2D accident in Surry.

Work is being done at Argonne under the Transient Fuel Reponse and Fission Product Release Program and the Post-Test Fuel Examination Program. Highlights of this quarter's work are presented below.

- The effect of grain growth on fission product release due to a temperature increase was modeled and incorporated into FASTGRASS.
- Grain growth due to steam oxidation of UO_2 was modeled empirically and incorporated into FASTGRASS.
- Prediction calculations for PBF SFD 1-3 were completed.

Two projects are being conducted at INEL. The programs are identified and progress in each one is briefly outlined below.

In-Pile Fission Product Behavior Studies

- In the area of fission product measurements for Test SFD 1-4, the final design of the new gamma spectrometer installation continued and an evaluation of the reactor building floor load-bearing capability was started. Included in this task is a revision to the steamline routing to reduce aerosol and fission product deposition between the gamma spectrometer and the filtered steam samples. In addition, a simplified Laser Aerosol Monitoring System concept that would provide an on-line measurement of effluent turbidity (particle concentration) was developed and preliminary design of the "Optical Fiber Turbidity Meter" was initiated.

Modification to Fission Product
Measurement Systems for Severe
Fuel Damage Tests (INEL)

- This is a new program whose objective is to improve measurements and analyses of samples taken during Tests SFD 1-3 and 1-4. The modifications to be made to the PBF system under this program include the addition of six filtered gas samples on the PBF main floor, just downstream of a new gamma spectrometer, which will be used to measure the release of fission products from the fuel before they travel 30 m to the existing measurement location. The posttest analysis improvements provided by this program include an array of advanced techniques for determining fission product species and their behavior under a variety of conditions.

PROGRAM ELEMENT 10: RISK CODES

Risk Code Development (MELCOR) is the only project under Program Element 10. It is being conducted at Sandia. Its goal is to develop an integrated set of risk codes, which perform both best estimate and quantitative uncertainty analyses, to replace MARCH/MATADOR/CRAC in the long term.

Work on MELCOR is focused on four primary categories: Offsite Consequences, Uncertainty/Sensitivity Methods, Coding, and Thermal Hydraulic/Fission Product Behavior. A summary of the quarter's activities in each of these areas is presented below.

Offsite Consequences

- Final drafts were received for all chapters in the Radiological Health Effects Report except the chapter on early health effects.
- A review of emergency response models and supporting data was initiated.

Uncertainty/Sensitivity Methods

- The first draft of the report summarizing all of the comparison calculations made using the Factorial Design/Response Surface, Latin Hypercube Sampling, and Differential Sensitivity techniques for the performance of sensitivity uncertainty studies was completed in March.
- In the version of MARCH used for the sensitivity/uncertainty analyses, the INTER subroutine was replaced by CORCON. Sensitivity calculations were begun using this modified version of MARCH.

Coding

- Coding of the implicit method for treating fluid flow between control volumes was completed, the module was implemented within the MELCOR hydrodynamics package, and several multi-volume test

problems were run successfully and yielded physically reasonable results.

- Debugging of the full core degradation package by performance of test core heatup calculations was completed, and the package was interfaced to the MELCOR executive structure.
- Coding of models that treat hydrogen combustion events was completed, and the module was debugged and interfaced to MELCOR's executive structure.
- The structure of the data base for the ex-plant portions of the MELCOR code system was defined and coded.
- Development of models that treat the relocation of degraded core material into the lower plenum, formation of debris beds within the lower plenum, and failure modes of the lower head continued.
- Coding of the subroutines required to interface the containment heat transfer package to the overall MELCOR code structure continued.
- Coding of the Exposure Pathways module continued.
- Development of a core-concrete interactions module by reduction from CORCON was begun.
- Coding of the Dosimetry and Emergency Response modules began.

Thermal Hydraulic/Fission Product Behavior

- A first draft was completed of a report that describes (a) how to construct decay heat data (decay heat per unit mass) for the 13 chemical element classes that will be used to specify chemical compositions in MELCOR and (b) the ORIGEN calculations performed to illustrate the use of the method.
- Models for removal of fission products from a gas stream by pool scrubbing were reviewed and the Fuchs' model was recommended for implementation of MELCOR.

- Models for removal of fission products by containment sprays were reviewed.

PROGRAM ELEMENT 11: ACCIDENT CONSEQUENCE AND RISK EVALUATION

and

PROGRAM ELEMENT 12: RISK REDUCTION AND COST ANALYSIS

Program Elements 11 and 12 are both being conducted at Sandia under four programs: the Severe Accident Risk Reduction Program (SARRP); the Development and Analysis of Vent-Filtered Containment Conceptual Designs Program; the Consequence Modeling and Risk Analysis Program; and the Localized Deposition from Wet Plumes Program.

Severe Accident Risk
Reduction Program

- A source term event tree which will be used to calculate the source term frequencies, with uncertainties, for Zion, Surry, Peach Bottom, and Grand Gulf was completed. The resulting calculations will be used in the pending source term document, NUREG-0956.
- Burns and Roe completed the Severe Accident Mitigation Systems Study (Phase I Report) and submitted it to SNL for review.
- Following is a brief summary of significant findings made in March: (1) the cavity water management scheme may not be feasible for PWRs because of cavity size constraints; (2) the use of existing fan cooler units in a passive condenser mode is not straightforward because of their placement; (3) the containment flooding system does not appear to be feasible for Babcock and Wilcox plants; (4) the "smart" deliberate ignition system may not be workable because of a dichotomy between the time constants for generation and detection; and (5) the alternate containment spray option appears feasible, but a recirculating system appears more practicable than a once-through system for some designs.
- At the NRC's request SNL staff members provided information for the PRA Reference Document and for the Phenomenological Issues Prioritization.

Development and Analysis of
Vent-Filtered Containment
Conceptual Designs

- Containment venting procedures proposed by the BWR Owners' Group were reviewed.
- The Rand Corporation's revision of the SNL Peach Bottom report was reviewed.

Consequence Modeling and
Risk Analysis

- Drafts of three chapters of the report by the Harvard University Health Effects Working Group were received. The chapters describe the models for latent health effects, generic effects, and thyroid effects.

Localized Deposition from
Wet Plumes

- Parameter studies on the jet and plume were performed, including studies of (1) the effects of ambient and containment conditions or condensation in a jet and (2) droplet dispersion and deposition.
- Releases from large-scale containment failure (a "Puff") and cracks were modeled to compare to the jet case.
- The jet model was completed.
- Results from the aerosol code indicate that the water aerosols in the jet grow to be a uniform size.

PROGRAM ELEMENT 13: REGULATORY ANALYSIS AND
STANDARDS DEVELOPMENT

Pacific Northwest Laboratories (PNL) and its subcontractor, BCL, are performing a significant part of the contracted work on Program Element 13. The NRC staff is also devoting a great deal of effort to this program element. Work performed this quarter under the project entitled Improved Methods for Incorporating Risk into Decision Making is summarized below.

- A cooperative agreement has been reached to perform a regulatory analysis of shift scheduling options for reactor operators. This regulatory analysis will be one of several value-impact case studies performed as part of this project, "Improved Methods for Considering Risk in Decision Making," (FIN B2386). The regulatory analysis will be in support of the Shift Scheduling Project being performed at PNL under the sponsorship of the Licensee Qualifications Branch/Division of Human Factors Safety/NRR. The regulatory analysis will evaluate shift scheduling issues and policies, and develop a set of recommendations for the NRC to consider in regulating shift work and overtime in the nuclear industry.
- A review meeting was held in June 1984 on USI A-47, The Safety Implications of Non-Safety Grade Control Systems. A work plan and schedule have been agreed to between PNL and NRR for this case study using the methods in NUREG/CR-3568, "A Handbook for Value-Impact Assessment". Work began on the event tree analysis to quantify the risk associated with the various overfill and overcooling scenarios identified by INEL.
- Work began on preliminary development of a software package to assist in performing and documenting value-impact assessments.
- Work continued on the development of a White Paper on Severe Accident Information Needs. PNL staff reviewed the recent literature on graphical display techniques and identified a set of representative applications of graphical displays in various regulatory analyses and value-impact analyses. Material has been

drafted for the revised White Paper, specifically for the chapter on Information Communication.

- Preparation for the Statistics Symposium on National Energy Issues is continuing. Some abstracts for contributed papers have been received and are being reviewed.

APPENDIX A

LISTING OF PROJECTS UNDER EACH PROGRAM ELEMENT

PROGRAM ELEMENT (1)		ACCIDENT LIKELIHOOD REEVALUATION		MONTHLY REPORTS RECEIVED				
FIN NO.	TITLE	NRC MONITOR	LABORATORY	JAN	FEB	MAR	APR	MAY
A1228	SNL Accident Sequence Evaluation	C. C. Eng	SNL			X	X	
A6301	INEL Accident Sequence Evaluation	C. C. Eng	INEL			X	X	
B0435	Accident Sequence Precursors	F. Manning	ORNL		X	X	X	X
B0468	Pressurized Thermal Shock	C. Johnson	ORNL		X	X	X	X

PROGRAM ELEMENT (2)		SEVERE ACCIDENT SEQUENCE ANALYSIS		MONTHLY REPORTS RECEIVED				
FIN NO.	TITLE	NRC MONITOR	LABORATORY	FEB	MAR	APR	MAY	JUN
A1258	Severe Accident Sequence Analysis-SNL	T. J. Walker	SNL			X	X	X
A3273	Applications of RAMONA to BWR ATWS	B. Agrawal	BNL			X	X	
A6354	Severe Accident Sequence Analysis-INEL	B. Agrawal	INEL		X	X	X	
A7228	Severe Accident Sequence Analysis-LANL	B. Agrawal	LANL			X	X	X
B0452	Severe Accident Sequence Analysis-ORNL	T. J. Walker	ORNL			X	X	X

PROGRAM ELEMENT (3)		ACCIDENT MANAGEMENT		MONTHLY REPORTS RECEIVED				
FIN NO.	TITLE	NRC MONITOR	LABORATORY	JAN	FEB	MAR	APR	MAY
B0826	Human Factors Review for Severe Accident Sequence Analysis	C. Overby	ORNL		X	X	X	

PROGRAM ELEMENT (4)		BEHAVIOR OF DAMAGED FUEL		MONTHLY REPORTS RECEIVED					
FIN NO.	TITLE	NRC MONITOR	LABORATORY	JAN	FEB	MAR	APR	MAY	JUN
A1335	LWR Debris Formation and Relocation	R. W. Wright	SNL	X	X	X			
A1340	LWR Core Debris Coolability	R. W. Wright	SNL	X	X	X			
A1342	LWR Accident Sensitivities and Melt Progression	J. Han	SNL	X	X	X			
A2220	Examination of TMI-2 Fuel Specimens	R. Foulds	ANL						
A6305	TFBP Severe Fuel Damage Studies	R. W. Wright	INEL				X	X	X
A6352	Resident Scientist at KFK Karlsruhe, FRG	H. Scott	INEL	X	X	X	X		
A6360	Severe Fuel Damage Model Development	J. Han	INEL	X	X	X			
A7303	TRAC/MELPROG	J. Han	LANL		X	X	X	X	

PROGRAM ELEMENT (4)		BEHAVIOR OF DAMAGED FUEL		MONTHLY REPORTS RECEIVED				
FIN NO.	TITLE	NRC MONITOR	LABORATORY	NOV	DEC	JAN	FEB	MAR
B2084	Severe Core Damage Subassembly Procurement	R. Van Houten	PNL	program completed				
B2277	NRU Coolant Boilaway and Damage Progression Tests	R. Van Houten	PNL					
B2455	Severe Core Damage Materials Property Tests	R. Van Houten	PNL					

PROGRAM ELEMENT (5)		HYDROGEN GENERATOR AND CONTROL		MONTHLY REPORTS RECEIVED				
FIN NO.	TITLE	NRC MONITOR	LABORATORY	DEC	JAN	FEB	MAR	APR
A1246	Hydrogen Behavior Program	J. T. Larkins	SNL			X	X	
A1255	Combustible Gas in Containment	J. T. Larkins	SNL			X	X	
A1270	Hydrogen Burn Survival Experiment	W. S. Farmer	SNL	X	X	X	X	X
A1336	Hydrogen Combustion Preventative and Mitigative Schemes	P. Worthington	SNL			X	X	
A7247	Hydrogen Migration and Mixing	J. T. Larkins	LANL					
B7232	Charcoal Performance Under Accident Conditions	J. T. Larkins	NRL	complete except final report				
B8208	Hydrogen Combustion and Control Demonstration Experiments	J. T. Larkins	NOO	program completed				

PROGRAM ELEMENT (6)		FUEL-STRUCTURE INTERACTION		MONTHLY REPORTS RECEIVED				
<u>FIN NO.</u>	<u>TITLE</u>	<u>NRC MONITOR</u>	<u>LABORATORY</u>	<u>NOV</u>	<u>DEC</u>	<u>JAN</u>	<u>FEB</u>	<u>MAR</u>
A1030	Core Melt/Coolant Interactions	J. L. Telford	SNL					
A1218	Core Melt Technology	T. Lee	SNL	X	X	X	X	X
A1247	Containment Load Mitigation	T. Lee	SNL	X	X	X	X	X

PROGRAM ELEMENT (7)		CONTAINMENT ANALYSIS		MONTHLY REPORTS RECEIVED				
<u>FIN NO.</u>	<u>TITLE</u>	<u>NRC MONITOR</u>	<u>LABORATORY</u>	<u>OCT</u>	<u>NOV</u>	<u>MAR</u>	<u>APR</u>	<u>MAY</u>
A1019	Molten Fuel-Concrete Interactions	S. B. Burson	SNL					
A1198	Containment Analysis	S. B. Burson	SNL	X	X			
A3024	Thermal-Hydraulic Reactor Safety Experiments	S. B. Burson	BNL			X	X	X
A7301	Pressure Threats to Containment	J. L. Telford	LANL					

PROGRAM ELEMENT (8)		CONTAINMENT FAILURE MODE		MONTHLY REPORTS RECEIVED				
FIN NO.	TITLE	NRC MONITOR	LABORATORY	JAN	FEB	MAR	APR	MAY
A1249	Containment Safety Margins	J. Costello	SNL			X	X	X
A1364	Electrical Penetration Assemblies	W. S. Farmer	SNL					
A1375	Containment Penetrations	H. Ashar	SNL			X	X	X
A6322	Equipment Qualification (Valves)	W. E. Campbell	INEL					

PROGRAM ELEMENT (9)		FISSION PRODUCT RELEASE AND TRANSPORT		MONTHLY REPORTS RECEIVED					
FIN NO.	TITLE	NRC MONITOR	LABORATORY	JAN	FEB	MAR	APR	MAY	JUN
A1227-A	High Temperature Fission Product Chemistry	L. Chan	SNL	X	X	X			
A1227-B	Sensitivities	M. Jankowski	SNL	X	X	X			
A2016	Transient Fuel Response and Fission Product Release	L. Chan	ANL						
A2232	Post-Test Fuel Examination (ORNL Fission Product Release Test Specimens)	L. Chan	ANL						
A6321	In-Pile Fission Product Behavior Studies	P. Reed	INEL				X	X	X
A6829	PBF Fission Product Measurement	P. Reed	INEL				X	X	X

PROGRAM ELEMENT (9)		CONTAINMENT FAILURE MODE		MONTHLY REPORTS RECEIVED				
FIN NO.	TITLE	NRC MONITOR	LABORATORY	JAN	FEB	MAR	APR	MAY
B0121	LWR Aerosol Release and Transport	J. Telford	ORNL		X	X	X	X
B0127	Fission Product Release from LWR Fuel-High Temperature Tests	L. Chan	ORNL		X	X	X	X
B0453	Post-Accident Fission Product Chemistry	L. Chan	ORNL		X	X	X	X
B0488	Trap-Melt Verification Tests	M. Jankowski	ORNL		X	X	X	X
B0815	Fission Product Deposition on Aerosols	M. Jankowski	ORNL		X	X	X	X
B0827	PBF/SFD Coordination	M. Jankowski	ORNL		X	X	X	X
B0831	Marviken Support	M. Jankowski	ORNL					
B2444	Effectiveness of LWR ESF Systems Under Severe Accident Conditions	C. Nilsen	PNL			X	X	X
B6747	Fission Product Transport	M. Jankowski	BCL			X	X	X
B7580	Marviken Program	M. Jankowski	Swe.					

PROGRAM ELEMENT (10)		RISK CODES		MONTHLY REPORTS RECEIVED				
FIN NO.	TITLE	NRC MONITOR	LABORATORY	JAN	FEB	MAR	APR	MAY
A1339	Development of Improved Physical Process Computer Codes for Risk Assessment (MELCOR)	M. A. Cunningham	SNL			X	X	

<u>PROGRAM ELEMENT (11) and (12)</u>		<u>ACCIDENT CONSEQUENCE AND RISK EVALUATION and RISK REDUCTION AND COST ANALYSES</u>		<u>MONTHLY REPORTS RECEIVED</u>				
<u>FIN NO.</u>	<u>TITLE</u>	<u>NRC MONITOR</u>	<u>LABORATORY</u>	<u>JAN</u>	<u>FEB</u>	<u>MAR</u>	<u>APR</u>	<u>MAY</u>
A1042	Consequence Modeling/Risk Analysis	J. A. Martin, Jr.	SNL			X	X	
A1322	Severe Accident Risk Reduction Program (SARRP)	M. A. Cunningham	SNL			X	X	
A1220	Development & Analysis of Vent-Filtered Containment Conceptual Designs	M. A. Cunningham	SNL			X	X	
A1369	Localized Deposition from Wet Plumes	J. A. Martin, Jr.	SNL			X	X	

<u>PROGRAM ELEMENT (13)</u>		<u>REGULATORY ANALYSIS AND STANDARDS DEVELOPMENT</u>		<u>MONTHLY REPORTS RECEIVED</u>				
<u>FIN NO.</u>	<u>TITLE</u>	<u>NRC MONITOR</u>	<u>LABORATORY</u>	<u>FEB</u>	<u>MAR</u>	<u>APR</u>	<u>MAY</u>	<u>JUN</u>
A1334	Regulatory Decision-Making and Policy Analysis	M. Fleischman	SNL					
B2386	Improved Methods for Considering Risk in Decision-Making	A. J. DiPalo	PNL			X	X	X

QUARTERLY REPORT

on

SEVERE ACCIDENT RESEARCH PROGRAM

August, 1984

Compiled by Battelle's Columbus Laboratories for
U.S. Nuclear Regulatory Commission

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APPENDIX A. LISTING OF PROJECTS UNDER EACH PROGRAM ELEMENT

QUARTERLY REPORT
on
SEVERE ACCIDENT RESEARCH PROGRAM

August, 1984

INTRODUCTION

This is the sixth quarterly report on the Severe Accident Research Program. The Severe Accident Research Program (SARP) consists of approximately 60 separate research programs organized into 13 programmatic elements. In order to assist NRC management and staff in following the progress of these diverse but related programs, Battelle's Columbus Laboratories is compiling progress reports for all of the SARP programs based on the monthly reports of the contractors. These reports are issued on a quarterly basis but do not necessarily cover the previous three months because of differences in the timing of receipt of contractor reports. The most recent results are those provided in reports for June, 1984. In some cases no monthly reports were available, and summaries of the quarter's progress were provided by the NRC program monitors.

The purpose of the Quarterly Report is to provide the NRC staff with a convenient summary of progress on SARP programs. Program progress and results are presented within the context of the 13 program elements so that progress can be interpreted in terms of the overall objectives of the SARP program. This report does not serve as a mechanism for presenting SARP program results to the public. Indeed, many of the results presented in the monthly reports of the contractors are preliminary and have not received internal approvals.

The intent of the SARP program as presented in NUREG-0900 is "to establish a sound technical basis on which an evaluation of the need for changes in nuclear power plant design and operation can be made". In Appendix A of this report, each of the SARP programs is identified. This Appendix provides basic management data for each program such as the FIN

Number and NRC project monitor. The Appendix also identifies the project reports that are summarized in this quarterly report.

The next section of this report discusses some of the more significant accomplishments of the SARP program in the reporting period. The last section of the report is a summary of progress for projects in each of the program elements.

GENERAL OVERVIEW

Status of Severe Accident Policy Statement (NUREG-1070)

On July 18, 1984, the Advisory Committee on Reactor Safeguards sent a letter to the Chairman providing comments on the April 18, 1984, draft version of NUREG-1070. For plants now in operation or under construction the letter concludes "taking into account the results of programs now in progress, and assuming a systematic examination of each plant, the proposed policy provides an acceptable basis for dealing with the severe accident issue". Several recommendations were made concerning the proposed policy for dealing with new plants including:

- (1) A statement that new plants should produce less risk
- (2) That guidance will be developed for the scope of PRA to accompany proposed designs
- (3) Consideration of the balance between prevention and mitigation and clarification of expectations for containment performance
- (4) Greater attention to human performance
- (5) Greater attention to sabotage protection.

Plan for Severe Accident Research and Regulatory Implementation

A number of RES and NRR staff have contributed to an integrated plan for research and regulatory activities related to severe accident and source term behavior. It is expected that after extensive review in August, the plan will be issued as a NUREG report which will provide guidance to Phase II of the Severe Accident Research Program in the same manner that NUREG-0900 guided Phase I research. Figures 1 and 2 illustrate the relationship between the major products of the Phase I and Phase II research program and the severe accident and source term regulatory implementation plans. The Phase II products include not only revisions to the SARRP report and NUREG-0956 report but also an update on the status of outstanding issues in the summer of 1986.

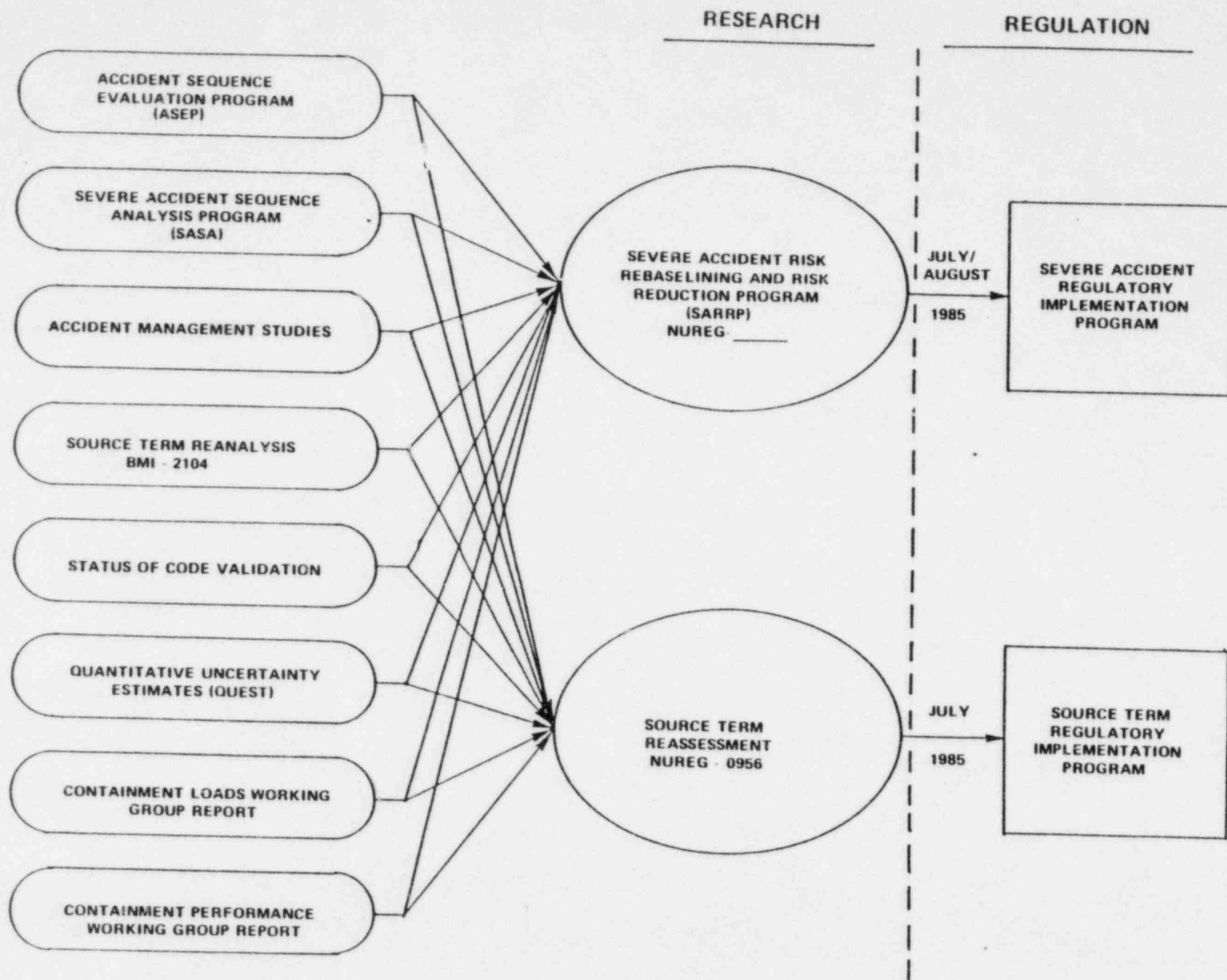


FIGURE 1. RELATIONSHIP BETWEEN PHASE I STUDIES AND REGULATORY ACTIVITIES

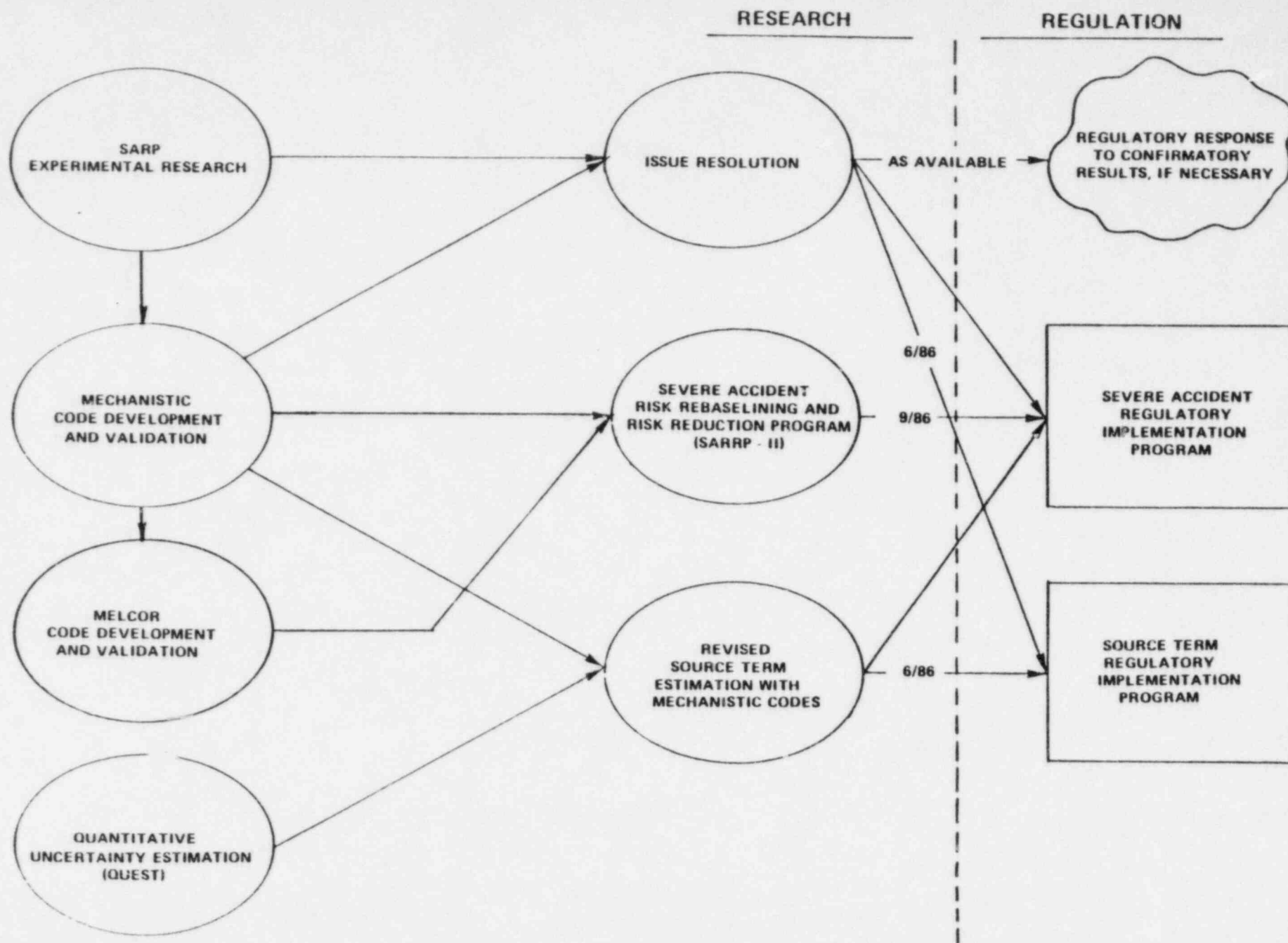


FIGURE 2. RELATIONSHIP BETWEEN PHASE II STUDIES AND REGULATORY ACTIVITIES

Major Accomplishments in the
Reporting Period

- DF-1 was conducted, and melting of the fuel rods was observed.
- Data from DCC-1 were analyzed and major results reported regarding (1) variation of the dryout heat flux with pressure, (2) quench rate, and (3) dryout behavior.
- SPIT 18 and 19 were conducted. In these tests melts were ejected into 1:20-scale reactor cavities. One preliminary conclusion is that nearly 100 percent melt ejection can be expected in a full-scale accident.
- Two series of tests were completed in the FLAME facility--one with and one without top venting. Various hydrogen concentrations were used.
- The first test using simulated high burnup LWR fuel from Germany was conducted in May. The 128-gram specimen was heated 15 min. at 1600°C and 30 min. at 1900°C in a steam/helium atmosphere. After the test, the specimen was swollen, distorted and thoroughly oxidized but almost completely intact. Very high release fractions of the fission product simulants were observed.
- Transient Water Tests TWT-1A and TWT-1B were conducted. The objective of these tests was to determine whether water can mitigate core debris interactions with concrete.
- HIPS-1J and HIPS-2C were performed. In both tests an 80-kg iron/alumina thermite was pressurized to about 1500 psi and ejected from a 1:10-scale reactor cavity. high speed photographs were taken.
- HSS-1, hot solid scoping test, was conducted using hot solid stainless steel as a surrogate for UO₂. It was a preliminary test in a series designed to characterize hot solid UO₂ debris/concrete interactions. In the preliminary test X-ray imaging of core debris interacting with concrete was successfully employed.

- Preparations for SFD 1-3 were completed.
- The Sandia version of MELPROG was merged with the Los Alamos version of MIMAS. The new code will be called MELPROG.
- MARCON, the MARCH-CORCON link, has been completed and sent to ORNL where improved BWR models will be incorporated.
- The final version of the ATWS accident sequence analysis report was completed in May.
- At the end of this quarter the report documenting the Watts Bar, Maine Yankee, and Bellefonte structural analyses was ready for publication.
- Final drafts were received for all chapters in the Radiological Health Effects Report except the chapter on early health effects.
- NUREG/CR-3592, ASP Computer Code: A Program for Use in the Accident Sequence Precursor Study", was completed.

TECHNICAL PROGRESSPROGRAM ELEMENT 1: ACCIDENT LIKELIHOOD REEVALUATION

Program Element 1 is comprised of three programs: (1) Accident Sequence Evaluation Program, conducted at SNL and INEL, (2) Accident Sequence Precursors, conducted at ORNL, and (3) Pressurized Thermal Shock, also conducted at ORNL. Highlights of this quarter's work in the three programs are presented below.

Accident Sequence Evaluation
Program (Activities at INEL)

- Quantification of the Interim Accident Sequence Evaluation (IASE) accident sequences for both PWRs and BWRs was completed in March. Documentation of the quantification methodology was completed in April.
- Work continued on fault tree modeling for the isolation condenser BWR plants.
- Recovery analyses were being done for the sequences under consideration.
- The ASEP catalog was modified in preparation for publication of an INEL report to be published in August 1984.

Accident Sequence Evaluation
Program (Activities at SNL)

- Work on BWR and PWR ATWS sequences continued.
- The methodology for applying recovery factors to the ASEP sequences was developed. Work began on application of recovery factors to PWR plant classes. The recovery process was later mechanized so that plant classes could be processed quickly.

- Sequence likelihood information required for NUREG-0956 was gathered.
- Event trees representing BWRs were developed at the functional and systemic levels.

LWR Accident Sequence Precursor Study

- The final draft of NUREG/CR-3591, "Precursors to Potential Severe Core Damage Accidents: 1980-1981" was revised in accordance with comments from the NRC and the ASP Review Committee.
- NUREG/CR-3592, "ASP Computer Code: A Program for Use in the Accident Sequence Precursor Study", was completed.
- Evaluation of safeguards incidents was begun.
- A study was begun on the applicability of empirical Bayesian methods to the existing ASP data base.

The Pressurized Thermal Shock Program focuses on three plants. This quarter's progress on work related to each plant is discussed below.

Oconee-Unit 1

- Technical work on this task is complete. The report will be published as NUREG/CR-3770.

Calvert Cliffs-Unit 1

- Documentation of the quantification of event trees was completed.
- Work continued on fracture mechanics analysis of vessel response. As of the end of May, this task was about 90 percent complete.

HB Robinson-Unit 2

- Quantification of event trees was completed.
- Estimates of $P(t)$, $T(t)$, and $h(t)$ were made and documented for all transients not explicitly calculated by INEL.
- Fracture mechanics analyses of vessel response had been completed for all cases received from INEL as of the end of May.

PROGRAM ELEMENT 2: SEVERE ACCIDENT SEQUENCE ANALYSIS

Work on Severe Accident Sequence Analysis is being conducted at five laboratories: SNL, INEL, LANL, ORNL, and BNL. A brief description of work done at each laboratory is given below.

Sandia National Laboratories

- Preparations continue for MARCON, CONTAIN, and HECTR calculations for Bellefonte (large dry containment PWR).
- Several sets of calculations were done for the Containment Loads Working Group. The types of containment investigated were PWR ice condenser and BWR Mark I, II, and III.
- MARCON, the MARCH-CORCON link, has been completed and sent to ORNL where improved BWR models will be incorporated.
- At the end of this quarter the report documenting the Watts Bar, Maine Yankee, and Bellefonte structural analyses was ready for publication.

Idaho National Engineering Laboratory

- The RELAP5/MOD2 steady state and initial TMLB' calculations for Bellefonte and Seabrook were completed in April. In May a more detailed RELAP5/MOD2 core model was developed for both plants.
- Sequence event trees were completed for the TMLB', S₂D, and ATWS accident sequences in Bellefonte to provide a logical representation of the systems that are challenged by the accident and potential operator actions that could be taken to mitigate the consequences of the accident.
- Work continued on a report documenting the Bellefonte TMLB' SCDAP sensitivities studies.
- RELAP5 and CONTEMPT analyses were completed for the following Browns Ferry ATWS sequences:

- (a) Sequence 483: A totally unmitigated transient plus two variations of this sequence.
- (b) Sequence 465: An operator level control mitigated transient.
- (c) Sequence 439: An operator level and pressure control mitigated transient.
- (d) Sequence 551: A high pressure boiloff transient.

A SCDAP analysis of Sequence 551 was also completed.

- Results of RELAP5 and RAMONA 3-B calculations for Sequence 439 were compared and found to differ significantly. Efforts to identify reasons for the differences continue.
- A draft report entitled "Potential Effects of Several ATWS Sequences on the Browns Ferry Nuclear Plant Unit 1 Containment" was issued for peer review and comment.

Los Alamos National Laboratory

- The draft NUREG document entitled "Dominant Accident Sequences in the Oconee-1 Pressurized Water Reactor" was submitted for peer review.
- Under the "Boron Dilution for CE, B&W, and Westinghouse Plants" task, steady-state calculations for Oconee (Mode 4) with the source term included were completed.
- Information was collected on dominant ATWS events in Calvert Cliffs-1.
- A base case calculation of the Oconee-1 loss-of-offsite-power transient (TMLB') has been run to 8400 seconds using the TRAC/MIMAS code. This is the first application of the hard linkage configuration of TRAC/MIMAS.

Oak Ridge National Laboratory

- The final version of the ATWS accident sequence analysis report was completed in May.

- Conversion of the CSMP-based BWR-LACP code to its Fortran version BWR-LTAS (Boiling Water Reactor--Long-Term Accident Simulation) was nearly complete in June. A preliminary draft of the BWR-LTAS users' guide was sent to Sandia where the code will be installed and operated.
- In June a 90-minute training presentation concerning important aspects of BWR reactor plant construction and containment and the results of ORNL SASA BWR severe accident studies was delivered at three training seminars for NRC Office of Inspection and Enforcement Personnel.
- Several modifications were made in the MARCH code used for BWR analyses. Separate models for fuel, gap, and cladding were completed. A subroutine was developed to calculate axial heat conduction in the fuel and axial conduction in the cladding separately. A new solution technique for the fuel, gap, and cladding models was implemented. The new technique allows larger time steps, making the computational time for the new models about the same as for the old models in which fuel and cladding were treated as a lumped mass. Work began on adapting BCL's ZRWATER routine for use in BWR core models.
- Work continued on the implementation of MARCON at ORNL. MARCON is MARCH 2.0 with INTER replaced by CORCON Mod 2. Efforts are being made to incorporate ORNL-BWR models in MARCON.
- MARCH-BWR was run, with the input used for the BMI-2104, Volume III calculations, for the TC sequence at Grand Gulf.
- In the investigation of radiant and volumetric heat sources in the BWR steam separators and standpipes work continued on formulating and scaling of the governing mass, momentum, and energy conservation equations.
- Some changes were made in the MELRPI code. A new model for the molten mass relocation was added to a modified SLUG subroutine and tested. This new model calculates the amount of molten/refrozen zirconium (from cladding and channel walls) and

stainless steel (from control blades) during the relocation of these materials to the bottom of the core. In addition, an improved rubble bed model has been developed in which rubble bed formation criteria and propagation phenomena are treated separately for the control blades and canisters.

- The modeling of all the basic components/phenomena in MELRPI is complete. The overall lower plenum/head model has been coded as a subroutine, LPFRPI.
- Two changes were made in the fission product transport (FPT) code. Minor modifications were made to improve the accuracy of calculations of (1) I transport and retention in containment volumes, and (2) noble gas solubility in water volumes. Models for tellurium transport and retention in containment and reactor building control volumes were developed.
- The SOLGASMIX computer code is being modified to permit rapid calculation of equilibria among fission products, steam, hydrogen, and reactor vessel materials.
- The spray model for absorption of gaseous iodine by water droplets was compared to experimental data. The model fairly consistently predicts a removal rate about one-third of the observed rate. The discrepancy is due to the simplicity of the model.

Brookhaven National Laboratory

- RAMONA-3B was used to calculate Browns Ferry ATWS Sequence 439. This sequence was also calculated at INEL using RELAP5. The results were significantly different, and time was spent identifying the reason for the differences.
- Work began on the calculation of cross sections for the five major fuel types identified in Cycle 5 Browns Ferry fuel. The CASMO computer code is being used.

PROGRAM ELEMENT 3: ACCIDENT MANAGEMENT

The major project under this program element is Human Factors Review for Severe Accident Sequence Analysis which is being conducted concurrently with the SASA BWR ATWS project at ORNL. Accomplishments during this reporting period are listed below.

- The rewriting of Appendix C to the SASA Anticipated Transient Without Scram (ATWS) Report was completed.
- Preliminary development of barrier breach diagrams showing pathways of radiological release and plant protective barriers was completed. This includes preparation of system descriptions of relevant control room instrumentation and operator actions for mitigating radiological release.
- Work continued on the development of the operator severe accident management model. Safety function and control requirements included in the functional classification are being defined. The specific ATWS scenario for demonstrating use of the functional classification model continued to be assessed in order to further study possible operator actions to mitigate the accident.
- An effort was begun to identify potential applications of the functional classification method by individuals representing regulatory, research, or industry interests.

PROGRAM ELEMENT 4: BEHAVIOR OF DAMAGED FUEL

Program Element 4 is the Severe Fuel Damage Program, which has substantial international participation. The work is being carried out primarily in four laboratories. INEL is conducting a series of integral severe fuel damage and fission-product release and transport experiments in the Power Burst Facility (PBF) and developing SCDAP, the Severe Core Damage Analysis Package code. In addition, EG&G personnel serving as NRC technical representatives participate in and report on fuel behavior studies being conducted at KfK in Karlsruhe, Germany. SNL is conducting experiments in the ACRR test reactor on LWR debris formation and relocation, and on LWR core debris coolability. SNL is also analyzing accident sensitivities and developing the MELPROG melt progression code. PNL is performing experiments to determine high temperature properties of fuel and cladding and is preparing for four severe fuel damage experiments in NRU with full length fuel bundles. Finally, LANL is working on the development of TRAC/MELPROG. A summary of this quarter's work under each program is presented below.

LWR Debris Formation and Relocation (SNL)

- Data from the out-of-pile systems test (OPST) were analyzed and compared with pretest predictions made with the DFR-SS code.
- Preparations for DF-1 were completed. These included a series of out-of-pile tests, a series of in-pile tests and a complete safety review of the experiment.
- DF-1 was conducted on March 15, 1984.

Most of the instruments in the system performed well. Preliminary observations reported are quoted below:

"After 2 min. of high-power nuclear operation, rapid oxidation ensued. Cladding/fuel heatup rates ($>25^{\circ}\text{C/s}$) were observed and some cladding movement was observed on the video monitor. After 3 min., fuel rod distortion was observed. Some liquefaction had probably occurred. At 3.5 min., a dark cloud of material appeared to jet out of one of the damaged rods and most of the flow channels appeared to be partially blocked

by relocated fuel/cladding material. The image darkened due to aerosols, becoming almost completely obscured for more than 1 min.

"At 5 min. into full-power operation, one corner of the image became visible and the disrupted bundle image gradually reappeared. At this time, one of the corner rods was observed to be relocated. The upper grid spacer, which supports the upper end fittings, was observed to have fallen or melted from its original location. Thermocouples near the unfueled upper end of the bundle showed a temperature increase, which suggests that the cladding in this region was probably undergoing rapid oxidation. The CuO bed temperatures also reached maximum values at this time due to the exothermic reaction of the CuO with hydrogen. These results indicate that rapid oxidation occurred at this stage."

- Work continued on fabrication of major components for tests DF-2 and DF-3.

LWR Core Debris Coolability (SNL)

- Data from DCC-1 were analyzed. The following major results were reported last quarter and are updated in the second paragraph: (1) The variation of the dryout heat flux with pressure was much weaker than predicted by any currently proposed model. The observed variation over the entire pressure range was less than a factor of 2, whereas a factor of about 6 was predicted. (2) The quench rate of the bed was a factor of 3 to 10 slower than predicted by current quench theories. Liquid did not penetrate the bed in liquid fingers during the quench process, as observed in out-of-pile experiments; instead, the quench front proceeded uniformly through the bed, quenching each elevation only when the entire bed at higher elevations was quenched. (3) The dryout behavior of the bed contradicted some predictions of current time-dependent theories. Dryout almost always occurred at the same elevation.