NRC Research and/or Technical Assistance Rept

PDR

November 1982
EGG-WRR-6129

MONTHLY REPORT REPRESENTING THE RESEARCH PORTION OF THE WATER REACTOR RESEARCH DEPARTMENT AND THE THERMAL FUELS BEHAVIOR PROGRAM

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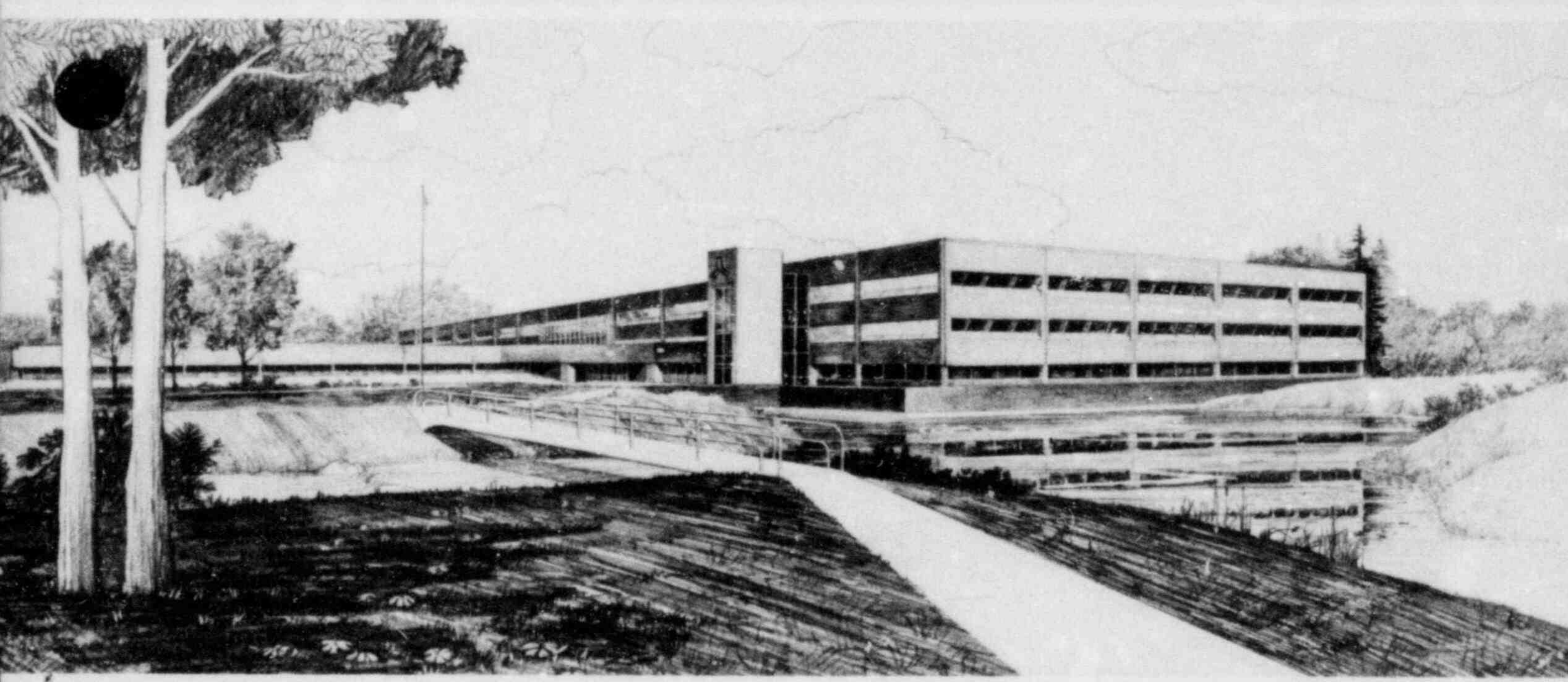
Idaho National Engineering Laboratory
Operated by the U.S. Department of Energy



This is an informal report intended for use as a preliminary or working document

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EG&G Idaho

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ACRONYMS

A/E	Architect Engineer
ACRS	Advisory Committee on Reactor Safety
AECL	Atomic Energy of Canada Limited
AMB	Applied Mechanics Branch
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
B&W	Babcock and Wilcox
BD/ECC	Blowdown/Emergency Core Coolant
BWR	Boiling Water Reactor
CA&AD	Code Assessment and Application Division
CAM	Constant Air Monitor
CC	Component Checkout
CCB	Change Control Board
CCTF	Cylindrical Core Test Facility
CDC	Control Data Corporation
CDD	Code Development Division
CDUM	Code Description and User's Manual
CE	Combustion Engineering
CHF	Critical Heat Flux
CLLMS	Conductivity Liquid Level Measurement System
CM	Corrective Maintenance
CPM	Critical Path Method
CSNI	Committee on Safety for Nuclear Installation
DAPS	Data Acquisition and Processing System
DARS	Data Acquisition and Reduction System
DAS	Data Acquisition System
DDAPS	Digital Data Acquisition and Processing System
DE	Division of Engineering
DER	Data Evaluation Report
DL	Division of Licensing
DOE	Department of Energy
DP	Differential Pressure
DSI	Division of Systems Integration
DSRR	Division of Systems and Reliability Research
DST	Division of Safety Technology
EI	Energy Incorporated
EICS	Electrical Instrumentation and Control System
EDF	Engineering Design File
EDR	Experimental Data Report
EM	Energy Measurements
ENICO	Exxon Nuclear Idaho Company, Incorporated
EOS	Experiment Operating Specifications

EP&A	Experimental Planning and Analysis
EPRI	Electric Power Research Institute
EQDB	Equipment Qualification Data Base
FCF	Facility Change Form
FDG	Fluid Distribution Grid
FIST	Full Integral Simulation Test
FMEA	Failure Mode Effects Analysis
FRG	Federal Republic of Germany
FSAR	Final Safety Analysis Report
GE	General Electric
GRS	Gesellschaft fur Reaktorsicherheit
HDR	Heiss Dampf Reaktor
HLS	Hot Leg Spool Piece
HPIS	High Pressure Injection System
HSST	Heavy Section Steel Technology
I&C	Instrumentation and Controls
IFA	Instrumented Fuel Assemblies
IGSCC	Intergranular Stress Corrosion Cracking
ILSG	Intact Loop Steam Generator
INEL	Idaho National Engineering Laboratory
IOER	Integrated Operational Experience Reporting System
IPT	In-Pile Tube
IREP	Interim Reliability Evaluation Program
ISDMS	Idaho National Engineering Laboratory Scientific Data Management System
ISI	In-Service Inspection
ISP	International Standard Problem
IST	In-Service Testing
JAERI	Japan Atomic Energy Research Institute
KfK	Kernforschungszentrum Karlsruhe
LANSL	Los Alamos National Scientific Laboratory
LER	Licensee Event Report
LLD	Liquid Level Detection
LLL	Lawrence Livermore Laboratory
LOC	Loss-of-Coolant
LOCA	Loss-of-Coolant Accident
LOFT	Loss-of-Fluid Test
LPIS	Low Pressure Injection System
LTSF	LOFT Test Support Facility
LVDT	Linear Variable Differential Transformer
LWR	Light Water Reactor
MFD	Master Facility Drawing
MIT	Massachusetts Institute of Technology
MSLB	Main Steam Line Break

NESC	National Energy Software Center
NPRDS	Nuclear Plant Reliability Data System
NPSH	Net Positive Suction Head
NRL	Naval Radiation Laboratory
NRR	Nuclear Reactor Regulation
NSRDC	Naval Ship Research and Development Center
NSSS	Nuclear Steam Supply System
NTOL	Near-Term Operating License
OPTRAN	Operational Transient
OR	Operating Reactor
ORNL	Oakridge National Laboratory
P&IA	Plant and Instrument Air
P&ID	Process and Instrument Diagram
PAS	Probabilistic Analysis Staff
PBF	Power Burst Facility
PCM	Power Cooling Mismatch
PCP	Primary Coolant Pump
PCS	Primary Cooling System
PIE	Postirradiation Examination
PKL	Primary Coolant Loop
PM	Preventive Maintenance
PMG	Program Management Group
PMIS	Performance Management Information System
PNL	Pacific Northwest Laboratory
PORV	Power Operated Relief Valve
PPS	Plant Protection System
PR	Combination of PCM/RIA
PRAC	Power Reactors Advisory Committee
PWR	Pressurized Water Reactor
QA	Quality Assurance
QDR	Quality Discrepancy Report
QLR	Quick Look Report
QPP	Quality Program Plan
RCCS	Reactor and Canal Cleanup System
RCG	Radioactivity Concentration Guide
RES	Office of Nuclear Regulatory Research
RFQ	Request for Quotes
RIA	Reactivity Initiated Accident
RIL	Research Information Letter
ROSA	Rig of Safety Assessment
RPG	Radiation Protection Guide
RSB	Reactor Systems Branch
SAI	Scientific Applications Incorporated
SASA	Severe Accident Sequence Analysis
SBE	Small Break Experiment
SCDAP	Severe Core Damage Analysis Package
SCTF	Slab Core Test Facility

SDD	System Design Description
SEP	Systematic Evaluation Program
SER	Safety Evaluation Report
SHB	Single Heated Bundle
SO	Systems Operations
SOW	Statement of Work
SPERT	Special Power Excursion Reactor Test
SQRT	Seismic Qualification Review Team
SRP	Standard Review Plan
SRV	Safety Relief Valve
SSE	Safe Shutdown Earthquake
SSRT	Senior Seismic Research Team
SSTF	Steam Sector Test Facility
STP	Standard Temperature and Pressure
SWR	Site Work Release
TAN	Test Area North
TC	Thermocouple
TDP	Technical Development Program
TER	Technical Evaluation Report
TFBP	Thermal Fuels Behavior Program
TFCF	Transient Flow Calibration Facility
THTF	Thermal Hydraulic Test Facility
TLTA	Two Loop Test Apparatus
TMI	Three Mile Island
TRR	Test Results Report
TVA	Tennessee Valley Authority
UHI	Upper Head Injection
UIC	Unique Identification Code
USSP	United States Standard Problem
UPTF	Upper Plenum Test Facility
WBS	Work Breakdown Structure
WRRD	Water Reactor Research Department
WRRTF	Water Reactor Research Test Facilities

MONTHLY REPORT FOR

NOVEMBER 1982

John A Dearien

J. A. Dearien, Manager

B. E. Williams

E. E. Williams
Plans and Budget Branch

MONTHLY REPORT FOR

NOVEMBER 1982

WATER REACTOR RESEARCH TEST FACILITIES DIVISION

Gary W Johnsen for

P. North, Manager

John P. Crouch

J. P. Crouch
Plans and Budget Representative

WATER REACTOR RESEARCH TEST FACILITIES DIVISION

COST SUMMARY AND COMMENTS

WATER REACTOR RESEARCH TEST FACILITIES DIVISION

189	Title	(1) Obligational Authority Carried Over from FY-1982	(2) New FY-1983 Obligational Authority	(3) Total Obligational Authority	(4) November FY-1983 YTD Costs	(5) (3)-(4) Balance	(6) Outstanding Commitments November FY-1983	(7) (5)-(6) Balance
A6038	Semiscale Program	$1,076.0K[1/]	$0.0K	$1,076.0K	$ 894.6K	$181.4K	$41.1K	$140.3K
A6043[2/]	Thermal Hydraulic Experiment Facility	130.0K[3/]	0.0K	130.0K	204.4K[4/]	<74.4K>	16.7K	<91.1K>
	WRRTF TOTALS	$1,206.0K	$0.0K	$1,206.0K	$1,099.0K	$107.0K	$57.8K	$ 49.2K

1/ Includes $170K GSO.

2/ Thermal Hydraulic Experiment Facility Portion.

3/ Includes $52K from LOFT CCB 83-01.

4/ Year-to-date actuals through November are overstated by $67.2K due to an error in billing. The error is being corrected by the Accounting Division and once corrected, the year-to-date actuals for November will be $137.2K.

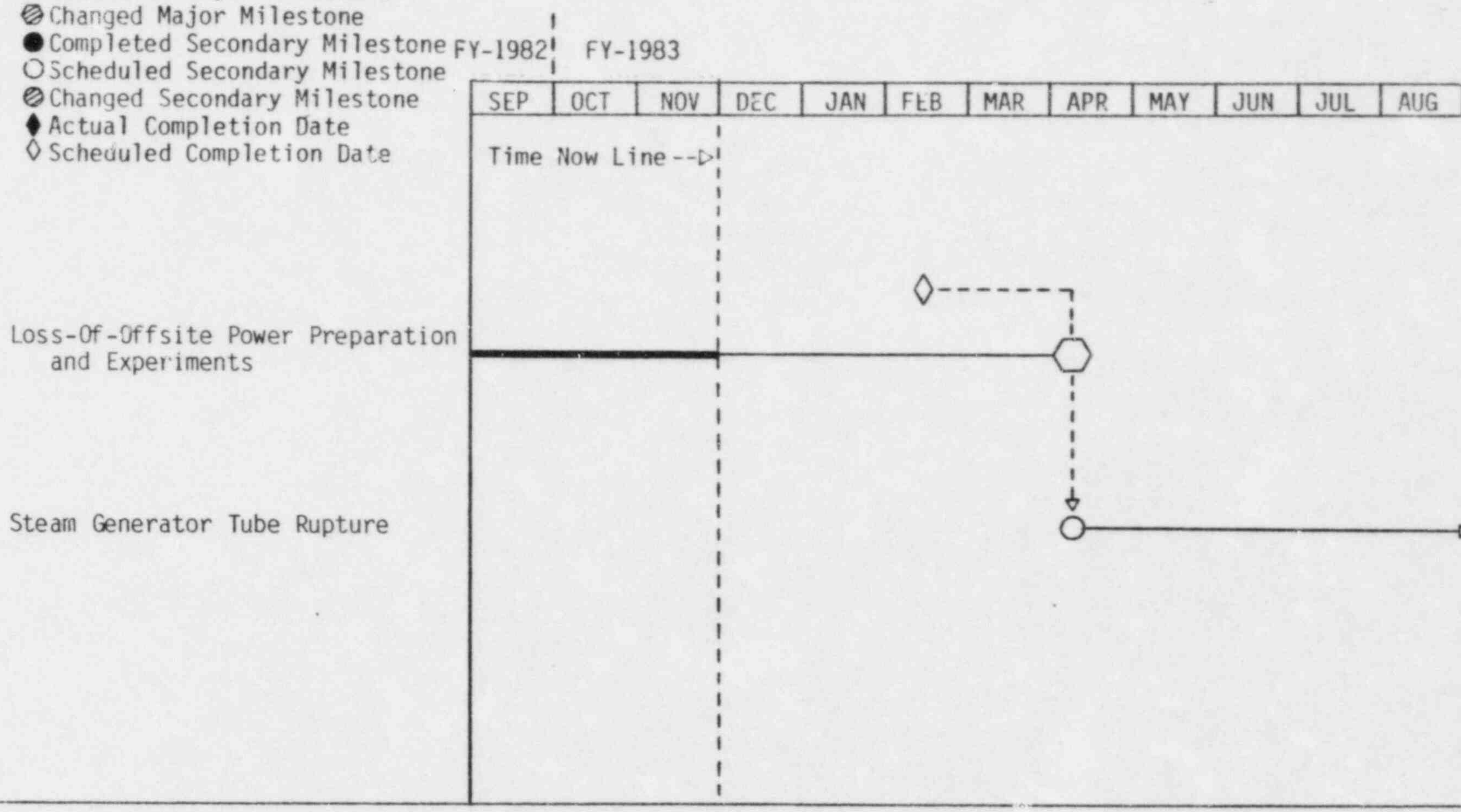
WATER REACTOR RESEARCH TEST FACILITIES DIVISION
MILESTONE SCHEDULE

LEGEND

- ⬢ Completed Major Milestone
- ⬡ Scheduled Major Milestone
- Changed Major Milestone
- ● Completed Secondary Milestone
- ○ Scheduled Secondary Milestone
- Changed Secondary Milestone
- ♦ Actual Completion Date
- ◊ Scheduled Completion Date

WATER REACTOR RESEARCH TEST FACILITIES DIVISION — November 1982

Semiscale Program (A6038)



NOTES:

WATER REACTOR RESEARCH TEST FACILITIES DIVISION

TECHNICAL REVIEW AND SUMMARY

PROGRAM MANAGER'S

SUMMARY AND HIGHLIGHTS

The first test in the Loss-of-Offsite Power Test Series was conducted at month's end, after overcoming several problems connected with the operation of new system hardware (i.e., intact loop pump, pressurizer). Plans to accelerate the conduct of the remaining tests were formulated to enable completion of the series close to the baseline date.

Design activities progressed toward the definition of modifications for the Steam Generator Tube Rupture Test Series, scheduled to begin in April.

A6038 Water Reactor Research Test Facilities Division

EG&G Idaho Technical Monitor: P. North
DOE-ID Technical Monitor: W. R. Young
NRC Technical Monitor: R. R. Landry

The purpose of the 189 is to acquire and interpret thermal hydraulic experimental data to assist in the resolution of light water reactor safety issues.

1. Scheduled Milestones for November 1982

Description	Proposed Date	Actual Date
Publish EDR: SF-4 & 5	11-01-82	10-27-82
Publish EDR: SR-1 and 2	11-01-82	10-27-82
Complete Mod-2B CC,SO,SC Testing	11-05-82	11-19-82 (Note 1)
Run Test PL-1	11-03-82	11-30-82 (Note 2)

NOTE 1: Completion of Mod-2B CC, SO, SC Testing and Test HOT-2B were delayed due to mechanical and electrical problems encountered

NOTE 2: Test PL-1 was delayed due to delays in Mod-2B CC, SO, SC testing. Test PL-1A was run 11-23-82, but was not declared a good test due to a leak which developed in the vessel lower head to core section graylock.

Test PL-1B was run 11-30-82, and the initial data review is currently being held.

2. Summary of Work Performed in November 1982

A. 41B000000 - SR-1 and -2 Carryover

1. 41B118100 - SR-1 and -2 EDR Preparation

The EDR to report Tests S-SR-1 and S-SR-2 was published 10-27-82.

2. Summary of Work Performed in November 1982 (continued)

B. 41C000000 - IB Series Carryover

1. 41C119310 - IB Series Posttest Analysis

The S-IB Series Topical Report was submitted for Branch and Division review. Comments were received and the report returned for resolution and submittal to technical editing.

C. 412100000 - Special Projects

1. 412111200 - Semiscale Configuration Reporting System

Completed interviews with internal (EG&G) users of Semiscale configuration information. Prepared and transmitted a questionnaire to external (non-EG&G) users of Semiscale configuration information for use in developing a proposal for development of a configuration reporting system. Summarized information with respect to objectives for the proposed system, and alternatives for meeting those objectives. The summary information will be used in proposal development following response to the questionnaire from external users.

2. 412123100 - Special Projects--Engineering

A draft revision of Dwg. 417260 on the pump seal cavity cooling installation was completed and an engineering review was performed. The drawing is now ready to be finalized and released. Requisitions to procure materials have been prepared and issued.

Drawings 409570 and 408489 were released to incorporate dual springs for bearing preload in the high speed pump. Several other pump drawings were revised to correct minor errors.

A backup seal was designed for potential future use in the event of an irrepairable leak of the flexitallic seal at the lower flange of the steam generator.

A work package was completed and detail design initiated for the external heater overtemperature protection upgrade.

Engineering support was provided to resolve a leak problem at the vessel lower head flange (Grayloc) joint. Design work was initiated on a backup seal weld configuration to use in the event of future leak problems at this location to preclude excessive down time for seal replacement.

2C. Summary of Work Performed in November 1982 (continued)

Started design of a modification to enlarge the working area around the steam generator platform. The purpose of this task is to eliminate safety concerns and improve working conditions.

An engineering evaluation of ways to minimize the programmatic impact of a steam generator U-tube failure was initiated.

3. 412148100 - Semiscale Measurements Improvement

   a. Phase 1 (Review of Measurements) of the work package was started November 1. A detailed scope of work for the task was developed and approved. Semiscale data handling procedures at the test facility and in town were reviewed and documentation (including recommendations) has been started. Low energy densitometer measurements have been reviewed, with particular emphasis on data correction techniques. A preliminary set of low-energy densitometer bench tests was proposed, and equipment requirements were submitted to MSE for acquisition and assembly.

   b. Minor effort commenced in familiarization with the set up and operation of the low energy density measurements. Test equipment is currently being collected to start bench testing this type of measurements on December 10.

D. 413100000 - Feedwater/Steamline Break Test Series

1. 413133100 - Feedwater/Streamline EDR Preperation

The EDR to report Tests S-SF-4 and S-SF-5 was published 10-27-82.

E. 414110000 - Level of Effort

1. 414119100 - EP&A Supervisor, Training, Report Preparation

Prepared work package status summary information for October status review. Provided section level review for the following reports: IB TRR, SF-4/5 RELAP Report, SF-4/5 QLR, SG-RDD, SG-DRD, PL-2 EOS Appendix, PL-3 EOS Appendix, PL-1 PTP.

2E. Summary of Work Performed in November 1982 (continued)

A review of vendor upper plenum and core exit thermocouple installations was performed. The applicability of these TC's as indicators of inadequate core cooling has recently come under discussion. A letter will be written documenting the information uncovered and will include suggested installations in the Semiscale system which could be made next time access is obtained to the vessel.

Numerous inquiries have been received in this past month on the SR and SF test series. EP&A personnel have provided information and performed short analyses to answer the questions posed.

2. 414119300 - NRC/DOE Support

An outline for the Small Break RIL was prepared and submitted for branch review. The work scope was delineated and divided between two authors. A bibliography of the existing work on SBLOCA's which has been performed in-house has been prepared. A table including all of the SBLOCA experiments conducted by the Semiscale program and summarizing important features from each test was compiled. The literature review is underway and work has begun on writing the report.

An uncertainty analysis was performed for inclusion in the film boiling heat transfer study. Branch level review comments were incorporated and the paper was approved. The paper has been sent to Technical Publications for final editing. It will be submitted to "Nuclear Technology" for publication.

3. 414123100 - Engineering Level of Effort

Work was started on the revision of ES-70052, intact loop pump assembly and disassembly procedure.

Piping physicals drawings were released.

The CAD system P&ID drawings for the PL no-break configuration were released.

An outline plan for the Semiscale configuration documentation task was completed. Assignments were made for the preparation of design descriptions of the new hardware (pressurizer, hot water makeup, intact loop pump and external heaters).

A6038

2E. Summary of Work Performed in November 1982 (continued)

4. 414148100 - Measurement Engineering - LOE

a. Installed and checked out the new Hewlett-Packard 1000 computer mainframe and operating system. (The new HP-1000 system will hereafter be referred to as System 2. The existing operational HP-1000 will be designated as System 1.) System 2 still lacks several pieces of hardware installation and several weeks of software effort before it will be capable of data acquisition. However, it is planned to have the system capable of data reduction by February 1, 1983. At that time it will be determined how much more effort will be required in preparation for data acquisition. A tentative plan calls for System 2 readiness prior to the start of the Steam Generator Tube Rupture Test Series.

b. The HP2100 computer system was moved from TAN 645 to TAN 641, Room 103. This will allow the continued processing of old Semiscale data until software can be generated on the new System 2.

c. Work commenced on selecting the method of instrumentation to be used to acquire data for the Steam Generator Tube Rupture Tests. A preliminary proposal will be presented to the Experimental Planning and Analysis Branch on or before November 30.

d. Both the high and low power bus core current shunts were recalibrated. Some effort was expended on the WRRTF Catastrophic Failure Analysis. A study was completed by November 30 which utilized the Expected Monetary Value (EMV) analysis technique for determining courses of action.

e. Effort continued on evaluating pressure transducer recalibrations to determine further useability of these instruments. The rough draft of the temperature uncertainty is 90% complete.

f. Processed special data requests for tests S-NC-8, S-NC-8B, and S-NC-9. Data was retrieved from Records Storage, processed, and tapes sent to EP&A.

2. Summary of Work Performed in November 1982 (continued)

F. 415100000 - Feedline/Steamline Break Analysis

1. 415119100 - Pre Feedline/Steamline Break Analysis

Research to clarify and develop issues to be addressed in the test series to be conducted in FY-84 was not initiated during November, but will be initiated in December.

G. 416100000 - Loss-of-Offsite Power Test Series

1. 416119910 - S-PL Test Support, Section B

Completed support of hot SO and SC testing required prior to S-PL-1. Transmitted the EOS Appendix for S-PL-2. Prepared and provided the EOS Appendix for S-PL-3 for Branch review. Prepared the first draft of the EOS Appendix for S-PL-4. Transmitted measurement requirements for S-PL-2, 3, and 4. Initiated preparation of the EOS Appendices for S-PL-5 and 7. Provided test support and data review support for Experiments S-PL-1A and repeat S-PL-1B.

2. 416119930 - S-PL Test Support, ECS

Completed RELAP5 pretest calculations for Test S-PL-1. Letter Report discussing the results was sent out November 19, 1982. The RELAP5 pretest calculation for Tests S-PL-2 and S-PL-3 were begun.

Several questions were received from the RELAP5/MOD1 Assessment Group at Sandia Laboratory concerning the modeling of the broken loop steam generator. Investigation of the RELAP5 model revealed some minor arithmetic errors and a typographical error in the RELAP5 Standard Model Description for the Semiscale Mod-2A System (EGG-SEMI-5692) document. These errors are being corrected and the broken loop steam generator model reflecting the error corrections will be included in the next RELAP5 model document.

3. 416123700 - Loss-of-Offsite Power--Hardware Mods

PL shutdown SO testing was completed. To summarize, all systems worked as designed. Several systems required some expected "fine tuning" during the final checkout, and the pressurizer spray system will require some additional modification to achieve the required thermal/hydraulic system response. Status of specific SO testing is as follows:

2G. Summary of Work Performed in November 1982 (continued)

a. The pump peripherals checkout test SO-2B-16 was successfully completed and final approval obtained on the test results.

b. Review of data from the intact loop pump locked rotor resistance test was completed and a letter report issued.

c. The hot water makeup system (SO-2B-17) and steam generator pressure relief system (SO-2B-18) were completed and the test results approved.

d. The pressurizer external heater test SO-2B-15 was completed and approved.

e. The pressurizer test SO-2B-14 was completed. A letter report, summarizing the test results, is in preparation. Some minor control system design modifications were required to achieve the specified heatup rate, and some modifications will be incorporated into the spray system to achieve the specified pressure response.

Pressurizer internal heater overtemperature protection (Trip T/C's) system was installed and tested.

All prerequisite SWR's, CC and SO tests, and QDR's were closed out (thru Quality Division Review) to support initiation of PL testing.

The revision of MFD Purification System Dwg. 404729 is being reviewed by operations prior to final check and release. Other MFD's being reviewed by operations for potential revision are 414754, 4?0455, 404728, 404731, and 406772.

Engineering and drafting work was completed to "as build" system drawings and update the Master Facility Drawing List. This effort involved incorporation of approximately 50 electrical interim drawings involving the new intact loop pump power and controls, annunciator panel, heat loss makeup panel, pressurizer and makeup display chassis and process control calibration panel.

A work release was issued to the CFA fabrication shop to make metal photo labels for as-building (relabeling) the chassis in the control room.

2G. Summary of Work Performed in November 1982 (continued)

During system testing, several leakage problems were "dispatched." This effort required much work and many different organizations. The design fixes were rapidly implemented and the entire effort was commendable. The specific problems were:

a. Pressurizer leak problems in the lower head at the penetrations for internal heaters were resolved by incorporating changes as follows: tightening tolerances on the heater rod collar thickness, and V-seal hole depth and flatness in the lower head; and use of a different bolt material (with smaller coefficient of thermal expansion than that of 316 S.S.) to hold the heater rod retainer plate in place. These changes are being incorporated in the pressurizer drawings.

b. A seal failure occurred on the intact loop pump during system SO testing, as evidenced by excessive water leakage. Subsequently, a full rebuild was performed on the pump, including the following: replacement of the failed seal; replacement of the primary shaft bearings; and installation of the dual springs for bearing loadings.

c. Leakage problems in the steam generator secondary shell flange connection and lower head Grayloc connection were resolved. "Backup" seals will be manufactured for each of these locations to preclude a lengthly downtime in event of a recurrence of these leaks.

The turbine meter spool piece drawing was revised to alleviate a misalignment problem when assembling the spool piece.

The overheating problem experienced on the intact loop pump inverter was corrected. A circuit was designed by the ME&DS Branch, verified by EG&G and vendor engineering, to disable the overheating SCR in the inverter. The vendor representative also assisted in correcting an erratic, pulsating voltage in the rectifier. Following these modifications, an overhaul of the intact loop pump resulted in increased starting torque, and it was necessary to raise the electrical control system trip circuitry until the break-in sequence was completed.

2G. Summary of Work Performed in November 1982 (continued)

A drawing, SWR package, and integrated planning were issued to install external heaters on the pump suction break assembly.

Drawing revisions and an SWR package were issued to fabricate and install the "dead head" no-break configuration of the pump suction break.

An SWR package was issued to relocate a portion of the compact condensing system to the pressurizer PORV manifold area.

A letter was prepared describing the design concept for a second generation pressurizer thermal liner using the preformed Min-K 2000 insulation. This conceptual design was transmitted to EP&A branch for analysis.

Work was initiated on the writing of a pressurizer system description to include in an updated Semiscale Facility description, currently in preparation.

4. 416136500 - Mechanical Instrumentation for Power Loss Test Series

Work consisted of providing support to MOD-2B CC, SO, and SC testing and Test HOT-2B for the PL-Series Shutdown Modification. Support was also provided for Test S-PL-1. All the Drag Device Transducers were flushed and cleaned.

5. 416136600 - Test Engineering for Power Loss Test Series

Work consisted of support for the PL-Series Shutdown/Modification. Coordination of Test and Data Review accomplished, and Data provided to EP&A and Technical Support.

CC and SC Testing of the Primary Volume Remnant System was completed with the performance of Test PL-1B. A letter report of the results of this testing will be issued by 12-17-82.

The Test Plans for Tests S-PL-1A and S-PL-1B were delivered to Operations and Pretest/Posttest activities for these tests were provided.

Work was begun on the Test Plan for Test S-PL-2.

2G. Summary of Work Performed in November 1982 (continued)

6. 416136700 - Operation Support for Power Loss Test Series

Test HOT-2B was run from November 1 to November 19th. Mechanical and Electrical problems were encountered. (Intact Loop pump and pressurizer variable heaters for example) causing this extended testing period. Test HOT-2B provided Data to complete Mod-2B CC, SO, and SC testing. This Data was accepted on November 22, 1982. The Test Procedure for Test S-PL-1A was prepared and the test performed November 23, 1982. The results were not accepted as an unacceptable leak developed in the Grayloc connecting the Lower Vessel Head to the Core Section.

The Test Procedure for Test S-PL-1B was prepared and the Test performed November 30, 1982. The initial review will be held December 1,1982, to determine if the test can be accepted.

7. 416148600 - Loss of Power Test Series Data Acquisition

a. Continued work on checkout of new data acquisition computer system and associated software.

b. Completed providing support on troubleshooting assistance to accomplishment of Mod-2B SO and SC tests.

c. Completed semiannual process control system preventive maintenance checks on the core power system crowbar circuits and the pressurizer heater control system.

d. Intact loop inverter was repaired by replacing shorted power diodes and Silicone Controlled Rectifiers (SCR). The severe chopper overheating, that occurred only during the idle state, was circumvented by inhibiting the drive on the 1052 control card to one chopper power SCR, utilizing a "no load demand signal" from the voltage cut-in op-amp on the 1050 control card. This same signal was used to drive two solid state relays to provide a pump "running" & pump "off" control room indication.

The DC rectifier's sporadic power kicking was eliminated by correcting a vendor wiring error that had tied a relay coil to a DC voltage instead of common.

2. Summary of Work Performed in November 1982 (continued)

H. 417119100 - Steam Generator Series Pretest Analysis

1. 417119100 - Steam Generator Series Pretest Analysis, Section A

The RDD for the SG test series has received branch level review and comments were incorporated. The document was distributed November 30. The DRD was completed and has been distributed, although continued efforts are underway to identify further system hardware and instrumentation modifications. RELAP5 scoping calculations are in progress as is preparation of a series EOS.

2. 417119103 - SG Series Pretest Analysis, ECS

Pretest RELAP5 scoping analyses for the S-SG Test Series to investigate the influence of tube rupture elevation on transient signature are in progress. A RELAP5 calculation with a sigle tube rupture at the top of the BL steam generator tube sheet and a calculation with a single tube rupture at the top of the "U-tube" bend are being run to further this investigation. The calculations are 80% complete. Results of these and other scoping calculations will be used in preparing the Experiment Operating Specification for the test series.

3. 417123100 - Tube Rupture Hardware Mods

Preliminary design requirements were established in a meeting of November 15, 1982 with Technical Support Engineering, EP&A and Operations personnel in attendance. Work associated with the steam generator secondary system was identified and initiated.

A meeting was held with EP&A and Measurements Engineering on November 30, 1982 to discuss proposed designs for making low flow bi-directional measurements. Two measurement techniques were discussed as follows: ΔP measurements across a venturi and across the break orifice; and a positive displacement piston-type flow rate transducer. These two types of measurement techniques can be used simultaneously and in series. The incorporation of a video probe and gamma densitometer into the break assembly was also discussed, as well as details of the break orifice.

2. Summary of Work Performed in November 1982 (continued)

I. 419519601 - EP&A Posttest Analysis (NC, UT)

1. 419519601 - S-NC RELAP5 Posttest Analysis

The assessment of the capability of RELAP5/MOD1.5 to accurately calculate the phenomena associated with single-phase, two-phase, and reflux natural circulation continued. Single-phase calculations showed good agreement with data. Initial two-phase calculations indicated good agreement with most measured system parameters, but further analysis of calculated mass flow rates is required.

Two-phase calculations under very slight loop voiding conditions identified an error in the RELAP5/MOD1.5 vapor generation/condensation calculation. The error, which caused large, undamped oscillations, was discussed with code development and fixed. The temporary fix now being used will be incorporated into RELAP5/MOD1.5 in the next released cycle.

2. 419519602 - UT Series TRR

No work was performed on this document.

3. 419519603 - S-SF-4 and 5 RELAP5 Posttest Analysis

The report entitled "Posttest RELAP5 Simulations of the Semiscale S-SF-4 and 5 Steam Line Break Experiments" (EGG-SEMI-6106) was completed and published.

4. 419519604 - Test S-SR-2 RELAP5 Analysis

A RELAP5/MOD1 calculation of Test S-SR-2 (Feed and Bleed), in which sensitive parameters such as HPIS flow rates would be closely matched to data, was initiated. Verification of input parameters to ensure the accuracy of modeled heat loss and corresponding core power has been completed.

J. 9D0800000 - Semiscale Equipment

1. 9D0820600 - Intact Loop Pump

Completed engineering review and comments on several manufacturing plans submitted by Associated Machine for the spare high speed intact loop pump. Expected delivery for the pump is early January 1983. A change requisition was issued to incorporate the latest drawing revisions which are minor and not expected to affect delivery schedule.

3. Scheduled Milestones for December 1982

Description	Proposed Date
Run Test S-PL-2	12-09-82
Perform SC-2B-24 (Safety Invection Pumps)	12-15-82
Run Test S-PL-3	12-21-82

4. Summary of Work to be Performed in December 1982

A. 41C100000 - Intermediate Break Test Series

Comments received from Division review of the TRR will be resolved and the report will be submitted to Technical Editing for their review prior to final distribution.

B. 412100000 - Special Projects

1. 412111200 - Semiscale Configuration Reporting System

Response to an inquiry of external users of system configuration information will be factored into the preliminary proposal for development of a reporting system. The preliminary proposal will be reviewed with WRRTFD management, comments resolved, and a final proposal prepared for transmittal to DOE on January 3, 1983.

2. 412123300 - Special Projects--Engineering

Issue revised Dwg. 417260 on the pump seal cavity cooling installation. Prepare and issue SWR package to install seal cavity cooling pump.

Issue Dwg. 417279 for the steam generator lower flange backup seal design, and issue the fabrication SWR.

Complete design and drafting on a modification of the broken loop steam generator work platform to provide additional platform space.

Develop and evaluate contingency plans for use in the event of steam generator U-tube failure during PL testing.

Complete the design for external heater overtemperature protection system, hold a final design review and initiate SWR package.

4B. Summary of Work to be Performed in December 1982 (continued)

Complete design of the vessel lower head backup seal and issue SWR to fabricate the hardware (seal and handling/support fixtures).

3. 412148100 - Semiscale Measurements Improvement

Documentation and recommendations resulting from Phase 1 will be completed. Phase 2, primarily the densitometer bench testing, will be started.

a. Continue support of data system set up and operation during Power Loss testing.

b. Continue installation of HP-1000 System 2 as equipment arrives.

c. Comnlete rough draft of the temperature section of the uncertainty document.

d. Finalize instrumentation to be used and commence procurement/fabrication in preparation for Steam Generator Tube Rupture Tests.

e. Continue transducer calibration support for Power Loss testing.

f. Commence bench on low energy densitometer measurements. Target of 15% completion by December 31.

C. 414110000 - Level of Effort

1. 414119100 - EP&A Supervision, Training, Report Preparation

Prepare work package status information for November status review. Provide section review for the following reports: IB-TRR (with resolution of review comments), PL-3 and 4 EOS Appendices, PL-1 QLR, PL-2 QLR, PL-2 and 3 PTP's, and WRVLIS Reports from NC and IB Series. Assist in preparation of summary information for the Management Meeting scheduled for December 8 and 9, 1982.

2. 414119300 - NRC/DOE Support

Work will continue on preparation of a Small Break RIL scheduled for completion in February.

4C. Summary of Work to be Performed in December 1982 (continued)

3. 414123100 - Engineering Level of Effort

Continue work on the revision of ES-70052, intact loop pump assembly and disassembly procedure, and release preliminary changes.

D. 415100000 - Feedline/Steamline Break Analysis

1. 415119100 - Pre-Feedline/Steamline Break Analysis

Pre-series research analysis to develop issues to be addressed in a future test series will be initiated.

E. 416119900 - S-PL EP&A Test Support

1. 416119910 - S-PL Test Support, Section B

EOS Appendices for S-PL-3 and S-PL-4 will be transmitted. The QLR for S-PL-1 will be transmitted. Experiment support for S-PL-2 and S-PL-3 will be provided, and QLR's initiated upon successful execution of those tests. Preparation of EOS Appendices for S-PL-5, 6, and 7 will be continued. First draft review for S-PL-5 and 7 EOS Appendices will be performed.

Core power control techniques to account for reactivity effect of voiding expected in S-PL-7 will be assessed using data and a hybrid computer simulation.

2. 416119930 - S-PL Test Support, ECS

RELAP5 pretest analyses of Tests S-PL-2 and 3 will be completed and the results documented in a single pretest prediction report before the S-PL-2 test date.

3. 416123700 - Loss-of-Offsite Power--Hardware Mods

Complete Operations and engineering review of various MFD and other drawings for as-building, incorporate comments for change as appropriate, and subsequently release drawings.

Obtain additional ΔP data on the pump turbine meter during future Semiscale systems testing to substantiate results of previous R' testing.

Complete the writeup of pressurizer system description.

4E. Summary of Work to be Performed in December 1982 (continued)

Revise design of the pressurizer spray system to achieve the specified system pressure response upon activation of the spray system.

Install a smaller orifice in the pressurizer relief line to obtain relief flow desired for PL test conditions.

Complete revision of pressurizer drawings to incorporate recent modifications to eliminate leakage at heater penetrations in the lower head.

Prepare SO test procedure of rupture disc pressurization system for PL-4.

Document the changes in electrical control circuitry on the intact loop pump inverter.

Provide electrical engineering support for the installation of external heaters on the pump suction break assembly.

Verify control room chassis drawings to determine as built conditions.

4. 416136500 - Mechanical Instrumentation for PL

Work will consist of providing instrumentation support for Tests S-PL-2, SC-2B-24, and S-PL-3.

5. 416136600 - Test Engineering for PL

Assuming acceptance of Test S-PL-1B, the Data will be qualified and Data Tapes prepared and sent to Data Processing in TSB. Work will then continue on the preparation of an EDR to report the results of Tests S-PL-1B, S-PL-2, and S-PL-3.

A Test Plan will be completed for Test S-PL-2 and provided to Operations. Pretest and Posttest activities will be completed to perform Test S-PL-2. The results will be included in the EDR with Test S-Pl-1B.

Upon receipt of the EOS Appendix for Test S-PL-3, a Test Plan will be completed and provided to Operations. Pretest and Posttest activities will be performed to perform Test S-PL-3.

4E. Summary of Work to be Performed in December 1982 (continued)

Coordination efforts for Measurement Instrumentation for Test S-PL-4 (Pump Suction Break) will be provided, especially in regard to Break Flow Measurements.

6. 416136700 - Operation Support Power Loss

Continued operational support will be provided for the PL Series. Test Procedures will be written for Tests S-PL-2 and S-PL-3 as the corresponding Test Plans are received.

Performing of SC-2B-24, Safety Injection Pumps, will be provided.

F. 417119100 - SG Series Pretest Analysis

1. 417119100 - SG Series Pretest Analysis, Section A

An EOS outline will be approved and work will begin on writing the draft EOS.

2. 417119103 - SG Series Pretest Analysis, ECS Support

RELAP5 scoping analyses for the S-SG Test Series will be continued. Analyses to investigate the influence of tube rupture elevation on transient signature will be completed. Studies will be started to investigate the influence of "U-tube" inlet versus outlet rupture locations and also to determine the sensitivity of the number of tube ruptures on transient response.

3. 417123100 - Tube Rupture--Hardware Mods

Obtain a further refinement of design criteria for the SG tube rupture and conduct a preliminary design review. Identify long lead procurement items and prepare ordering data as required.

G. 419519600 - Posttest Analysis

1. 419519602 - UT-Series TRR

Work will resume on the UT Series TRR.

2. 419519604 - Test S-SR-2 RELAP5 Analysis

RELAP5/MOD1 analysis of Test S-SR-2 (Feed and Bleed) will continue. Calculations including the effects of steam generator heat loss and measured HPIS mass flowrates will be completed, and compared with data.

4. Summary of Work to be Performed in December 1982 (continued)

H. 9D0800000 - Semiscale Equipment

1. 9D0820600 - Intact Loop Pump

Continue engineering review of vendor data submittal from Associated Machine on the spare high speed intact loop pump.

Ship the Welco motor stator, special tools and other remaining parts to Associated Machine to complete final assembly of the spare intact loop pump. A full inventory is being conducted on the pump tools, and a SWR package will be issued to fabricate replacements for those that are missing.

Visit the vendor to verify status of work in progress and plan for the necessary EG&G engineering assistance during final assembly operations.

5. Problems and Potential Problems

None

A6043 LOFT Test Support Facility

EG&G Idaho Technical Monitor: P. North
DOE-ID Technical Monitor: W. R. Young
NRC Technical Monitor: R. R. Landry

The purpose of this 189 is to make available a separate effects test facility for the purpose of running future experiments to acquire fundamental data relating to two-phase flow and heat transfer.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

A. 481100000 - FY-82 Carryover

1. 481100310 - Two-Phase Test Reports

The analysis report for characterization tests conducted in the Two-Phase Flow Loop was reviewed and returned for resolution of comments. No work was performed on the 2D/3D instrumentation EDR.

2. 481202010 - THEF Engineering

Completed and closed out all on-going drawing (as-built) revisions and documentation as appropriate to support current (standby) shutdown of THEF. This item will be dropped from future reports.

3. 48199AA00 - Nine-Rod Bundle Quench Report

No work was performed on the analysis report. The report was submitted for final review in October, and will be submitted to technical editing following branch review of resolution of comments.

4. 48199AP00 - L5-1 Drag Disk Rake EDR

The report was submitted for initial review in October. No work was performed in November.

2. Summary of Work Performed in November 1982 (continued)

B. 487248100 - THEF Operations

Work continued to put the Blowdown Facility and Two-Phase Flow Loop in a Ready Stand-by condition. Both loops have been drained of water, both main coolant and industrial cooling water. The loops were purged with nitrogen and have nitrogen purges established on them. Facility spare parts have been moved to one location and are undergoing classification and labeling. Operations files are being updated and put in order. Material from the Bone Yard and surrounding outside areas has been loaded and excessed or sent out as salvage.

C. 5J1223100 - Post CHF Analysis and Report

All data on modcomp tapes intended to be used for analysis was converted to CDC format. Formulation of the data reduction program was delayed due to redirection of the analyst, but was continuing at month end. Analysis of steady--state test data were performed in support of a technical paper to be presented at a future ASME meeting. The final paper was submitted and reviewed, and was scheduled for transmittal in early December.

4. Summary of Work to be Performed in December 1982

A. 481100000 - FY-82 Carryover

1. 481100310 - Two-Phase Test Reports

The Two-Phase Flow Loop Characterization report will be transmitted. Completion of the 2D/3D instrumentation EDR will be postponed into January, 1983.

2. 48199AA00 - Nine-Rod Bundle Quench Report

The analysis report will be submitted to Technical Editing for final preparation. Transmittal will be postponed into January, 1983. Slides for an ANS presentation will be completed.

4. 48199AP00 - L5-1 Drag Disk Rake EDR

Final preparation of the EDR will be postponed into January. Transmittal will be rescheduled for February, 1983.

4. Summary of Work to be Performed in December 1982 (continued)

B. 487248100 THEF Operations

Work will continue on placing the Blowdown Facility and Two-Phase Loop in a Ready Stand-By condition.

C. 5J1233100 - Post CHF Analysis and Report

The data reduction program for the transient data will be completed. Preparation of the data report will be initiated. Preparation of ASME presentation slides will be initiated.

5. Problems and Potential Problems

None

MONTHLY REPORT FOR

NOVEMBER 1982

THERMAL FUELS BEHAVIOR PROGRAM

W. A. Spencer, Manager

T. A. Olsen
Plans and Budget Representative

THERMAL FUELS BEHAVIOR PROGRAM

COST SUMMARY AND COMMENTS

THERMAL FUELS BEHAVIOR PROGRAM

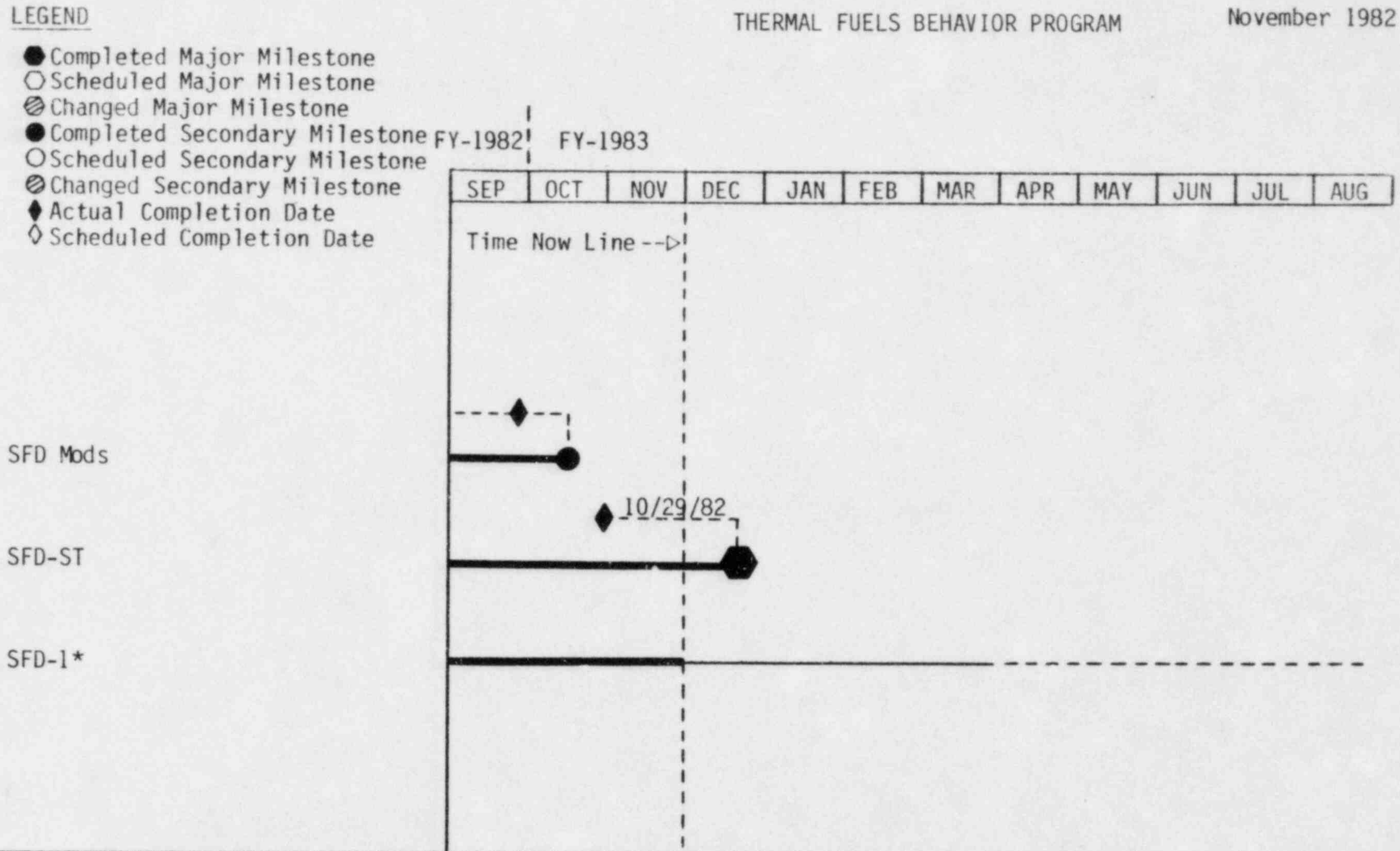
189	Title	(1) Obligational Authority Carried Over From FY-1982	(2) New FY-1983 Obligational Authority	(3) Total Obligational Authority	(4) November FY-1983 YTD Costs	(5) (3)-(4) Balance	(6) Outstanding Commitments November FY-1983	(7) (5)-(6) Balance
A6041	TFBP Experiment Design and Analysis	$225.2K	$ 0.0K	$ 225.2K	$ 20.3K	$ 204.9K	$ 0.0K	$ 204.9K
A6044	PBF Engineering	44.7K	469.0K	513.7K	305.2K	208.5K	37.2K	171.3K
A6057	PBF Operations	229.9K	1,221.0K	1,450.9K	928.2K	522.7K	80.0K	442.7K
A6305	Severe Fuel Damage	45.7K	1,674.0K	1,719.7K	1,105.0K	614.7K	234.9K	379.8K
A6321	In-Pile Fission Product Behavior	0.0K	309.0K	309.0K	208.5K	100.5K	47.9K	52.6K
A6351	Core Melt Mitigation	9.6K	0.0K	9.6K	4.2K	5.4K	0.0K	5.4K
A6352	NRC Rep to KFK	34.0K	9.0K	43.0K	20.2K	22.8K	0.0K	22.8K
A6372	Fission Product - Past Accidents	20.0K	0.0K	20.0K	0.6K	19.4K	0.0K	19.4K
	TFBP TOTALS	$609.1K	$3,682.0K	$4,291.1K	$2,592.2K	$1,698.9K	$400.0K	$1,298.9K

THERMAL FUELS BEHAVIOR PROGRAM

MILESTONE SCHEDULE

THERMAL FUELS BEHAVIOR PROGRAM

November 1982



2-05

NOTES: * TFBP is currently in the process of establishing an FY-1983 baseline. New test completion commitment dates will be contingent on final funding resolution.

THERMAL FUELS BEHAVIOR PROGRAM

MANAGEMENT SUMMARY SCHEDULE

The Thermal Fuels Behavior Program is currently being rebaselined, therefore, the monthly Management Summary Schedule is being revised. As soon as the rebaselining effort is complete, a new Management Summary Schedule will be issued.

THERMAL FUELS BEHAVIOR PROGRAM

TECHNICAL REVIEW AND SUMMARY

PROGRAM MANAGER'S

SUMMARY AND HIGHLIGHTS

Analyses of the results from the Severe Fuel Damage (SFD) Scoping Test, conducted during the last reporting period, are continuing. The Quick Look Report, to be issued in December, presents the preliminary findings, some of which are summarized here.

Fission product and hydrogen release were successfully measured with the new, state-of-the-art collection system, which proved to be an excellent tool for investigating the time-dependent release of those materials. Radioactive isotopes of krypton, rubidium, xenon, and cesium were among those identified in the gaseous effluent, and these four isotopes, together with tellerium and iodine, were identified in the liquid effluent.

The inner zircaloy liner of the shroud failed as planned about 184 min after the power ramp was begun. The cladding inside surface temperatures at various elevations continued to rise at about the same rate (0.16 to 0.18 K/s) after liner failure as before, until about 197 min into the transient. Catastrophic oxidation of the zircaloy began at the 0.7-m elevation at 197 min, at the 0.5-m elevation at about 200 min, and at the 0.35-m elevation at about 203 min. The flame front was apparently propagating down the bundle starting at about 197 min. The reactor was scrammed after the cladding temperatures reached 2400 K.

The coolant flow rate had to be set slightly higher than planned, 0.02 L/s instead of 0.016 L/s, in order to keep the bundle check valve closed so that all test effluent flowed into the SFD coolant collection system loop. To eliminate this problem at the much lower flow rates planned for subsequent SFD tests, a positive displacement pumping system is being designed to provide positive control of the bundle inlet flow rate.

A very large portion of the bundle instrumentation performed satisfactorily, and it was determined that thermocouples with a zircaloy sheath containing a tantalum liner were clearly superior to those constructed of pure zircaloy or zircaloy with an oxidized inner liner.

Other activities during the reporting period were primarily directed toward SFD-ST posttest efforts, as well as plant preparations for the upcoming SFD 1-1 modifications.

Portions of the experiment cooling lines have been removed, and two of the four sample casks have been shipped to the hot cells for analysis and gamma scanning. Sectioning of the Fission Product Detection System (FPDS) steam sample line was initiated; analysis of the samples will begin in December.

A baseline budget and schedule was submitted and was reviewed by DOE-ID and NRC at a meeting on 18 November. A revised baseline will be submitted in December reflecting the guidance from this meeting.

A6041: Experiment Design & Analysis

EG&G Program/Technical Monitor: W. A. Spencer/P. E. MacDonald
DOE Technical Monitor: D. L. Rose
NRC Technical Monitor: G. P. Marino

The objective of this program is to complete the reporting of the original Thermal Fuels Behavior Program's 40-test program. This program is an integrated experimental and analytical program designed to provide information on the behavior of reactor fuels under normal, off-normal, and accident conditions. The remaining tasks include completing examinations of materials from the Operational Transient tests, and reporting of tests from the Reactivity Initiated, Loss-of-Coolant, and Operational Transient Test Series.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

a. Operational Transient (OPTRAN) Test Series

Sectioning diagrams have not yet been generated. Mr. S. A. Ploger, along with R. VanHouten (NRC), visited several nuclear facilities in Europe to determine, among other things, what methods are used by other labs to identify pellet-cladding interaction defects. Two zircaloy tubes with defects were scanned with the Pulsed Eddy Current (PEC) equipment as part of an international study on Eddy Current capabilities. All of the defects were found with the EG&G Idaho PEC machine.

b. Power-Cooling-Mismatch Test Series

No effort was expended on the Test PCM-7 Fuel Rod Materials Behavior Report.

c. Reactivity Initiated Accident Test Series

Review of the Test RIA 1-4 Fuel Behavior Report has been delayed.

d. Data Processing Management Methods

Data from the Severe Fuel Damage Scoping Test were processed, which included 27 tapes for a total of 90.1 x 10^6 data points. Approximately 135 MAPPER slides were prepared for the Quick Look Report and posttest presentations, and one movie was generated. Data were sent to the University of Washington.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

a. Operational Transient (OPTRAN) Test Series

A report of Mr. Ploger's European trip will be written. The visual examination, PEC, and gamma scan data from the OPT 1-1 and OPT 1-2 rods will be reviewed to determine locations for fuel rod sectioning and further destructive examinations.

b. Power-Cooling-Mismatch Test Series

The Test PCM-7 Fuel Rod Materials Behavior Report will be revised as time permits.

c. Reactivity Initiated Accident Test Series

Review of the Test RIA 1-4 Fuel Behavior Report will occur as time permits.

d. Data Processing Management Methods

Organization of Severe Fuel Damage Scoping Test data on the computer will be completed, the Quick Look Report figures will be finalized, and the data will be plotted as requested.

5. Problems and Potential Problems

None.

A6044: PBF Engineering

EG&G Idaho Technical Monitor: W. A. Spencer/J. P. Kester
DOE-ID Technical Monitor: J. R. Sanders
NRC Technical Monitor: H. H. Scott

The objective of this program is to provide engineering support to safely maintain Power Burst Facility (PBF). Included in this activity are safety analyses and the design and installation of modifications required to assure safe conduct of the coordinated test program assigned to the PBF, currently the Severe Fuel Damage Test Series.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

a. PBF Spare Loop Pump Repair

A scheduled trial assembly of the rotor liner was performed. The technique was verified by successfully completing the assembly, although the liner was damaged due to electrical arcing of the tooling in the heating process. On the basis of these results, there is confidence that the general approach used - electrically heating the can while suspended in a vertical position and lowering the rotor into it - will be successful. The tooling has been revised to operate at higher temperatures without arcing and another liner has been prepared for a second trial assembly.

The stator has been set up for liner installation, but absence of specific experienced personnel has resulted in deferral of the actual liner insertion and welding.

b. Contamination Control Equipment for Test Train Transfer

The containment sleeve to be used in SFD-ST test train transfer operations after the test has been fabricated and received from the vendor. Tooling to be used with the sleeve is being fabricated.

c. Contamination Control System for the PBF Canal

Fabrication is in progress on the new ventilation system components, the tent framework, and the hoist rails on the canal parapet. The system will be used to minimize spread of contamination in the facility during disassembly of the SFD-ST test train fuel bundle.

d. SFD-1 Facility Modifications

Requirements were outlined and design efforts started on the following modifications to support Test SFD-1:

2. Summary of Work Performed in November 1982 (continued)

(1) Heat tracing the sample system steam line to prevent condensation of effluent before it passes the sample bombs.

(2) Addition of a demineralized water injection system to dilute the fission product concentration and reduce delay times downstream of the sample system condenser.

(3) Rerouting of the sample system steam line, in the main bay of the reactor building, to provide improved access and disposal capabilities following the transient. This change is required due to high radiation fields from the steam line that are not reduced sufficiently during posttest flushing operations.

(4) Addition of hydraulic resistance in the experiment cooling line to improve flow and pressure control during early parts of the transient.

e. Technical Specifications

A draft, including the comments received from the annual review of the Technical Specifications, was completed.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

a. PBF Spare Loop Pump Repair

The liners for the rotor and stator are expected to be installed and welded in place.

b. Contamination Control Equipment for Test Train Transfer

Fabrication of the handling tools will be completed and the containment sleeve will be used to control airborne contamination during the SFD-ST test train removal from the in-pile tube and transfer to the PBF canal.

c. Contamination Control System for the PBF Canal

The ventilation system modifications and the canal tent structure fabrication will be completed. Installation of the canal hoist and the tent are scheduled to be started.

4. Summary of Work to be Performed in December 1982 (continued)

   d. SFD-1 Facility Modifications

      (1) Design will be started on a shielding structure in the cubicle containing most of the SFD sample system. Radiation levels are excessively high following a test, thus restricting access for sample bomb removal.

      (2) A final design review will be held to cover the bundle low flow injection system.

      (3) Preliminary design reviews are scheduled for the sample system steam line rerouting in the cubicle, for the experiment cooling line modification, and for the sample system dilution injection modification.

   e. Technical Specifications

      Efforts to complete the revisions resulting from the annual review will continue.

5. Problems and Potential Probems

   None.

A6057: PBF Operations

EG&G Program/Technical Monitor: W. A. Spencer/C. O. Doucette, Jr.
DOE Technical Monitor: L. E. Montoya
NRC Technical Monitor: H. H. Scott

The objective of this program is to operate the Power Burst Facility (PBF) reactor to perform the Thermal Fuels Behavior Program (TFBP) Severe Fuel Damage (SFD) test series for the Nuclear Regulatory Commission (NRC). The data produced during the performance of the SFD tests is qualified and provided to personnel conducting the TFBP SFD (A6305) and the In-Pile Fission Product Behavior (A6321) studies for their analysis work.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

a. PBF Plant Operations

The work performed during this reporting period was primarily directed toward completing the Severe Fuel Damage Scoping Test (SFD-ST) and preparation of the plant for the upcoming SFD 1-1 modifications.

Portions of the experimental cooling lines have been removed and two of the four sample casks have been shipped to the Hot Cells for analysis and gamma scanning.

The Instrument and Data Section completed all data reduction requests and started the Reactor Building Radiation Activity Monitor modification and the replacement of the Closed Circuit Television Channel.

b. PBF Operations Support

Preventive Maintenance (PM) examinations for September are 95% complete, October examinations are 90% complete, and the November examinations are 50% complete.

Corrective Maintenance efforts include beginning the installation of the new gasket-type silver zeolite housing, the correction of plant deficiencies, and support work for cleanup and shielding.

Data qualification for the Operational Transient (OPT) Tests 1-1 and 1-2 continued. Data processing was completed and data qualification started on the SFD-ST. Uncertainty user's programs were converted from the calculator to the Data Qualification System.

3. Scheduled Milestones for December 1982

Node	Description	Due Date	Actual Date
N/A	Severe Fuel Damage Scoping Test	12-23-82	10-29-82

4. Summary of Work to be Performed in December 1982

   a. Plant modifications for the upcoming Test SFD 1-1 will continue.

   b. All PM examinations for 1982 will be completed.

   c. Installation of the gasket-type silver zeolite housing will continue.

   d. Data qualification of Tests OPT 1-1, OPT 1-2, and SFD-ST will continue.

   e. Preparation of the Test SFD 1-1 Experiment Operating Procedure draft will be initiated.

5. Problems and Potential Problems

   None.

A6305: TFBP Severe Fuel Damage Studies

EG&G Program/Technical Monitor: W. A. Spencer/P. E. MacDonald
DOE Technical Monitor: D. L. Rose
NRC Technical Monitor: G. P. Marino

The objective of this program is to provide the Nuclear Regulatory Commission (NRC) staff with a technical basis for evaluating the consequences of severe core damage accidents. This program will provide integral test data to be used in establishing fission product source terms, developing realistic, probabilistic risk assessments, and evaluating engineered safety features.

1. Scheduled Milestones for November 1982

Node	Description	Due Date	Actual Date
N/A	Publish NUREG Topical Report on Fuel Foaming Potential	12-15-82	11-12-82C

2. Summary of Work Performed in November 1982

a. Severe Fuel Damage (SFD) Test 1-1 Experiment Prediction Analysis

Comments were received on the draft report. Revision of the report has been delayed due to work being done in revising TRAC analysis for the Scoping Test Quick Look Report.

b. Severe Fuel Damage (SFD) Test 1-2 Experiment Prediction Analysis

Analysis continued on insulating the test train cooler line. The reflood analysis was postponed because of the TRAC analysis for the Quick Look Report.

c. Severe Fuel Damage (SFD) Scoping Test Quick Look Report

A draft Quick Look Report was prepared. Revised TRAC and FRAP analyses were performed, since the test inlet flow and bundle powers were significantly different from those issued in the pretest analysis. These efforts are continuing with an updated power history based on the test train fission chamber.

d. Severe Fuel Damage Test 1 (SFD 1-1) Experiment Operating Specification (EOS)

A revised draft of the SFD 1-1 EOS was submitted for management approval.

e. Postirradiation Examination (PIE) and Hot Cell Support

The hot cell equipment was assembled in a mockup area and testing was started, using the written procedures.

2. Summary of Work Performed in November 1982 (Continued)

f. Severe Fuel Damage Analysis

The topical report on the potential for fuel foaming during a severe accident was published to meet the milestone.

g. Instrument Development and Fission Chamber

All the 10-Hz fission chamber data have been processed and sent to the University of Washington (10 tapes); the 100-Hz data have been received from PBF. Preliminary estimates of the steam-water to steam interface levels during the Scoping Test transient were made from strip-chart recordings of five fission chambers.

h. Test Train Assembly Facility (TTAF)

The SFD 1-1 test train assembly continued and is complete to the point of installing a redesigned steam line heat exchanger. The components for the SFD 1-2 test train have been received and preparations for assembly have started. The redesign of the upper structure for Test SFD 1-3/1-4 and the design of Series II test trains continue.

i. Phase II Program Development

The SFD Series II Experiment Specifications Document (ESD) was approved and distributed. Design efforts for Series II were halted temporarily while a major effort is put into a Kepner-Tregoe format review of functional requirements. Principal difficulties continue to exist in the functional requirements for systems which permit the measurement of fission product release and plenum plate-out.

New feedback from NRC has led to the conclusion that:

1. Depressurization capability will not be required.

2. Posttest debris bed studies (in-pile, in situ) will not be required.

3. Dry inlet steam mixed with hydrogen to simulate upper zones of a reactor will not be required.

A contract with Battelle Northwest Laboratories (BNWL) to develop oxidation resistant coatings for thermocouples to be used in Series II has received EG&G Idaho approval and is now ready for DOE approval.

The video probe test model was tested in a furnace to 2300 K. The data from this test were reduced and the conclusions reached are

2. Summary of Work Performed in November 1982 (Continued)

i. Phase II Program Development (Continued)

that the tip design will be adequate, but that the stainless multiple-reflector design must be changed, probably to a ZrO_2 insulation. It was also concluded that water can be used as the coolant.

Efforts at Los Alamos National Laboratory to develop and test thoria forms has slowed, pending new direction from EG&G Idaho. High density (92-95% TD) tiles have been produced successfully and a plan has been developed to test thoria tiles (for shrinkage, creep, and a tendency to liquefy) against other core materials at temperatures up to 3100 K in an inert environment.

j. Safety Analysis - Severe Fuel Damage Test 1-1 (SFD 1-1)

Analyses were completed on (a) the expected particle size distribution for the Scoping Test, (b) the fraction of the particle mass expected to be washed into the loop during quench, and (c) fuel bundle differential pressure measurements. These results will be used to predict the probability of a secondary criticality resulting from particles washed into the loop during Test SFD 1-1.

k. Severe Fuel Damage Test 1-1 (SFD 1-1) Experiment Safety Analysis (ESA)

Draft A of the SFD 1-1 ESA has been completed and distributed for review.

l. Modifications

Fabrication of the replacement sample system filters and sample bombs was started.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

a. Severe Fuel Damage (SFD) Test 1-1 Experiment Prediction Analysis

The Experiment Prediction will be revised to incorporate comments and the report will be issued.

b. Severe Fuel Damage (SFD) Test 1-2 Experiment Prediction Analysis

The steam cooler line analysis will be completed and efforts will begin on the reflood analysis.

4. Summary of Work to be Performed in December 1982 (Continued)

c. Severe Fuel Damage (SFD) Scoping Test Quick Look Report

The Quick Look Report, containing the updated bundle power history and TRAC analysis, will be issued.

d. Severe Fuel Damage Test 1 (SFD 1-1) Experiment Operating Specification (EOS)

Review comments will be incorporated into the EOS.

e. Postirradiation Examination (PIE) and Hot Cell Support

Gamma scanning of the steam pipes will begin. The SFD-ST bundle will be gross gamma scanned in the PBF canal, provided there are no problems removing the test train from the in-pile tube.

f. Severe Fuel Damage Analysis

Work scopes for potential TRAP-MELT analyses and review of the PBF Severe Fuel Damage program will be completed.

g. Instrument Development and Fission Chamber

All the 100-Hz fission chamber data will be processed and sent to the University of Washington. This will conclude the sending of data from the SFD Scoping Test.

h. Test Train Assembly Facility (TTAF)

The redesign of the SFD 1-1 test train heat exchanger will be completed and the parts fabricated. The installation of the new check valve and steam line will be partially completed. The assembly of the SFD 1-2 test train will continue. The design of the SFD 1-3/1-4 test train upper structure and the Series II test train will continue.

i. Phase II Program Development

BNWL will begin development of oxidation resistant thermocouples.

j. Severe Fuel Damage Test 1-1 (SFD 1-1) Experiment Safety Analysis (ESA)

Review comments will be incorporated into the ESA.

k. Modifications

Parts required for the filter and sample bomb replacement following Test SFD 1-1 will be ordered.

5. Problems and Potential Problems

Further delays in starting the SFD 1-2 reflood analysis will impact completion of the experiment prediction analysis.

A6321: In-pile Fission Product Behavior Studies

EG&G Program/Technical Monitor: W. A. Spencer/P. E. MacDonald
DOE Technical Monitor: D. L. Rose
NRC Technical Monitor: R. R. Sherry

The objective of this program is to investigate fission product release and transport during in-pile severe fuel damage tests. The results being sought include isotopic release fractions, release fraction histories and release rate constants to aid assessment of source term models.

Measurements are made using on-line gamma spectrometers, radiation monitors, and effluent grab samples. Posttest analysis is conducted on samples from the fuel, test train, effluent sample line and effluent collection tank. This program is coordinated with and directly dependent upon the PBF SFD test program (A6305).

1. Scheduled Milestones for November 1982

Node	Description	Due Date	Actual Date
N/A	Letter Report - Feasibility of On-line Fuel Condition Monitoring	11-29-82	11-15-82C

2. Summary of Work Performed in November 1982

a. Fission Product Detection System (FPDS) Upgrade Completion

A PDP-11/34 computer was identified for potential reutilization in the Fission Product Detection System to implement the option of dual controlling computers. This will satisfy system functional requirements by eliminating out-dated, unreliable equipment and permit the easy addition of a fourth gamma spectrometer in the Series II tests. Modification of shielding to permit calibration source installation and removal was initiated. The design and work release were completed; installation of the shielding plug is expected December 20. The conceptual design for rerouting the steam line to one germanium detector prior to the steam sample bombs was completed. Additional planning is in progress; installation is expected February 14.

Work on load cells to remotely monitor the liquid nitrogen levels in the dewars was initiated.

b. Analysis Development

A work release will be issued to better define the complete scope of work under the Analysis Development Work Package, and the expected completion date will be advanced if possible. A target date of May has been set for completion of the SFD-ST data analysis, with completion of a final results report about August 1983.

2. Summary of Work Performed in November 1982 (Continued)

c. Severe Fuel Damage Scoping Test (SFD-ST)

The SFD-ST quick look analysis was reported and presentations of the results were given to the NRC and PBF personnel. Efforts began on grab sample processing and sample line plateout determination.

d. Severe Fuel Damage Test 1-1 (SFD 1-1)

Conceptual design work continued on required modifications for the sample system in SFD 1-1. Additional shielding will be added to the detector enclosure and purge air will be provided; the cost and schedule for these and other required changes are being developed.

e. Series I Chemistry

Sectioning of the Fission Product Detection System steam sample line was initiated. Analysis of the samples will begin in December.

f. Series II Measurements Development

Measurement system objectives were established and confirmed with NRC, Oak Ridge National Laboratory, and Battelle Columbus Laboratories in a meeting on November 19. Investigation of the application of light scattering for mass concentration and particle size measurements was initiated. A study of deposition coupon surface coatings and instrument compatibility was started.

g. Fission Product Signature Analysis

A letter report on the feasibility of on-line fuel condition monitoring was transmitted to meet the milestone. The milestone date for the report on the RFKM model to calculate fission product release rate coefficients was revised to 12-31-82E due to direct impact on available personnel of the Severe Fuel Damage Scoping Test Quick Look Report preparation.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

a. Fission Product Detection System (FPDS) Upgrade Completion

Efforts to reutilize a PDP-11/34 will continue, and acquisition of peripheral equipment will be initiated. Installation of shielding plugs for the detector calibration will be completed. Planning and design of the rerouted steam line, and of the load cells for the nitrogen dewars will continue. Addition of a liquid nitrogen line to remotely fill the dewars in Cubicle 13, and design of additional neutron and gamma shielding of the detectors will be initiated. Spectrometer calibration efforts will begin.

b. Analysis Development

Detailed scope and schedule will be defined and a work release issued. The availability of the final computer routine will be determined.

c. Severe Fuel Damage Scoping Test (SFD-ST)

Grab sample processing will continue; preliminary results will be reported by an internal letter. Sample line sections will be removed from PBF and spectral gamma scans will be conducted.

d. Severe Fuel Damage Test 1-1 (SFD 1-1)

System modifications to improve performance during the next SFD test will continue. Designs will be reviewed and the in-plant work will begin. Automatic data acquisition routines and collimator control software changes will be implemented.

e. Series I Chemistry

Preliminary radiochemical analyses of steam samples will be performed.

f. Series II Measurements Development

Investigation of the application of light scattering and a continuous effluent sample system will continue.

g. Fission Product Signature Analysis

The RFKM model report will be completed to meet the milestone.

5. Problems and Potential Problems

The scope of the Fission Product Detection System upgrade completion exceeds funding; management will review scope and budget to determine the best resolution of the problem.

A6351: Core Melt Mitigation

EG&G Idaho Program/Technical Monitor: W. A. Spencer/H. J. Reilly
DOE-ID Technical Monitor: J. R. Sanders
NRC Technical Monitor: R. T. Curtis

The objective of this study was to make an evaluation of systems proposed to mitigate the consequences of severe accidents with special attention to detailed engineering problems associated with backfitting the systems to the specific plants under analysis.

1. Scheduled Milestones for November 1982

   None.

2. Summary of Work Performed in November 1982

   EG&G-PR-5633 was prepared for publication (final editing, composition, word processing).

3. Scheduled Milestones for December 1982

   None.

4. Summary of Work to be Performed in December 1982

   EGG-PR-5633 will be published as a NUREG report.

5. Problems and Potential Problems

   None.

A6352: NRC Representative to KfK

EG&G Program/Technical Monitor: W. A. Spencer/P. E. MacDonald
DOE Technical Monitor: D. L. Rose
NRC Technical Monitor: R. R. Sherry

The objective of this program is to provide information on severe fuel damage, fission product behavior, and core melt research in Germany to the NRC. The information will be used to compliment the NRC's Severe Fuel Damage Research Program.

1. Scheduled Milestones for November 1982

   None.

2. Summary of Work Performed in November 1982

   This task is reported separately in bimonthly reports prepared by the NRC representative to KfK and transmitted under separate cover.

3. Scheduled Milestones for December 1982

   None.

4. Summary of Work to be Performed in December 1982

   None.

5. Problems and Potential Problems

   None.

A6372: Fission Product Behavior During Past Accidents

EG&G Program/Technical Monitor: W. A. Spencer/P. E. MacDonald
DOE Technical Monitor: D. L. Rose
NRC Technical Monitor: R. R. Sherry

The objective of this program is to investigate fission product behavior during past accidents and destructive tests. Well-characterized accidents were selected for detailed analysis. The remaining task is to analyze the Plutonium Recycle Test Reactor accident using TRAP-MELT to evaluate models regarding fission product release from fuel, transport of fission products through various containment barriers, potential physiochemical forms of fission products, and effects of water on fission product transport.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

None.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

No further work will be done until the new version of TRAP-MELT is received from Battelle Columbus Laboratories.

5. Problems and Potential Problems

None.

THERMAL FUELS BEHAVIOR PROGRAM

CHANGE CONTROL BOARD ACTIONS

CHANGE CONTROL BOARD STATUS

Cost Account	CCB #	Description	Status	Date
42XXXXX	83-01	TFBP FY-1983 Baseline	Deferred	11/12/82

CHANGE CONTROL BOARD ACTION

($000)

CCB #	Description	FY-1983	FY-1984	FY-1985/Beyond	Total Approved Action
		None			

MONTHLY REPORT FOR

NOVEMBER 1982

2D/3D PROGRAM

Paul North

P. North, Manager

John P. Crouch

J. P. Crouch
Plans and Budget Representative

2D/3D PROGRAM

COST SUMMARY AND COMMENTS

2D/3D PROGRAM

189	Title	(1) Obligational Authority Carried Over From FY-1982	(2) New FY-1983 Obligational Authority	(3) Total Obligational Aughority	(4) November FY-1983 YTD Costs	(5) (3)-(4) Balance	(6) Outstanding Commitments November FY-1983	(7) (5)-(6) Balance
A6100	3D Technical Support and Instrumentation	$ 834.2K	$0.0K	$ 834.2K	$219.6K	$ 614.6K	$ 66.8K	$ 547.8K
A6282	Fluid Distribution Grids	421.0K	0.0K	421.0K	50.0K	371.0K	110.0K	261.0K
A6289	UPTF DAS	747.4K	0.0K	747.4K	82.4K	665.0K	0.0K	665.0K
	2D/3D TOTALS	$2,002.6K	$0.0K	$2,002.6K	$352.0K	$1,650.6K	$176.8K	$1,473.8K

2D/3D PROGRAM

MILESTONE SCHEDULES

2D/3D PROGRAM

November 1982

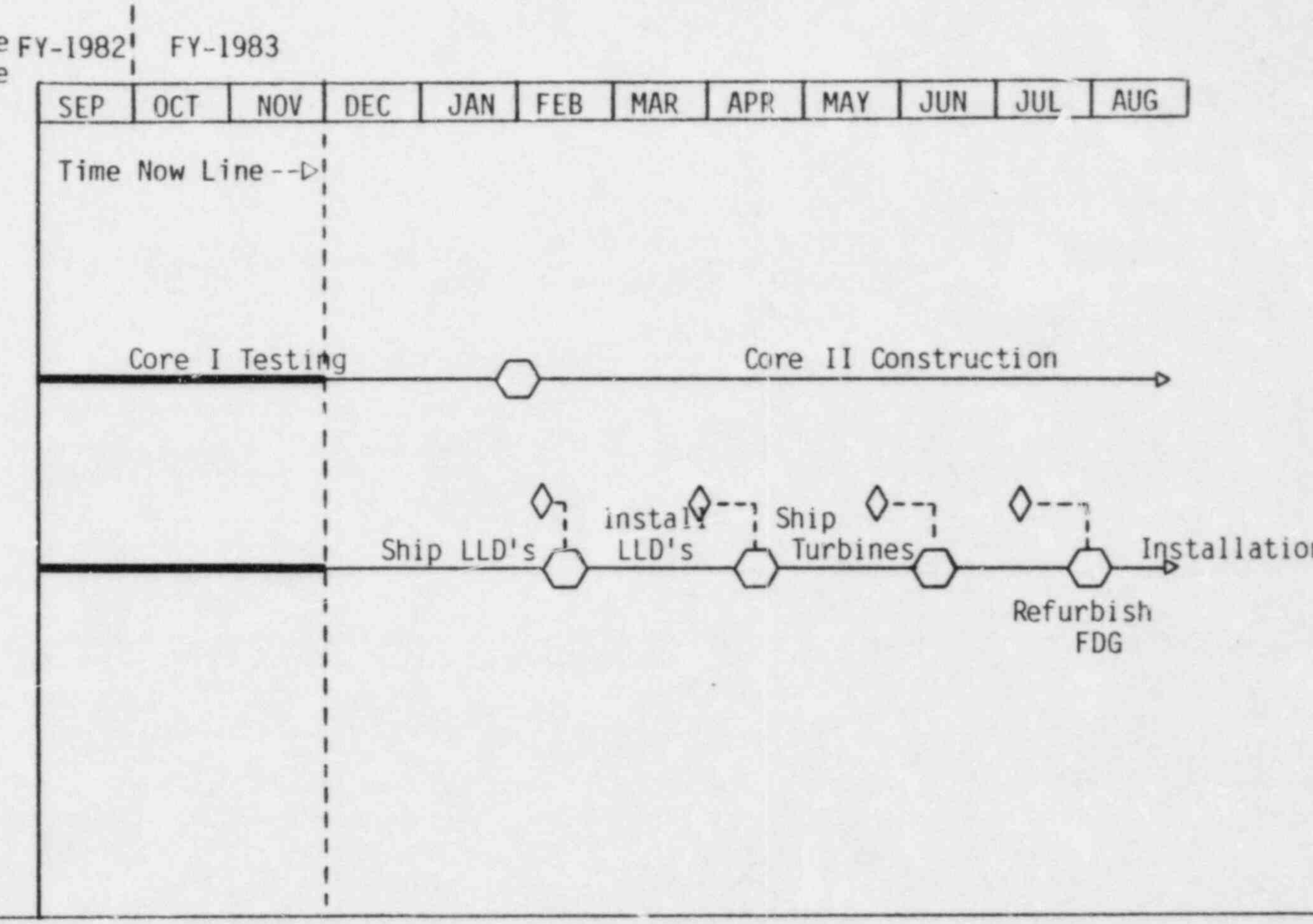
SCTF Projects (A6100)

LEGEND

- Completed Major Milestone
- Scheduled Major Milestone
- Changed Major Milestone
- Completed Secondary Milestone
- Scheduled Secondary Milestone
- Changed Secondary Milestone
- Actual Completion Date
- Scheduled Completion Date

Facility

Refurbishment (SCTF-II)



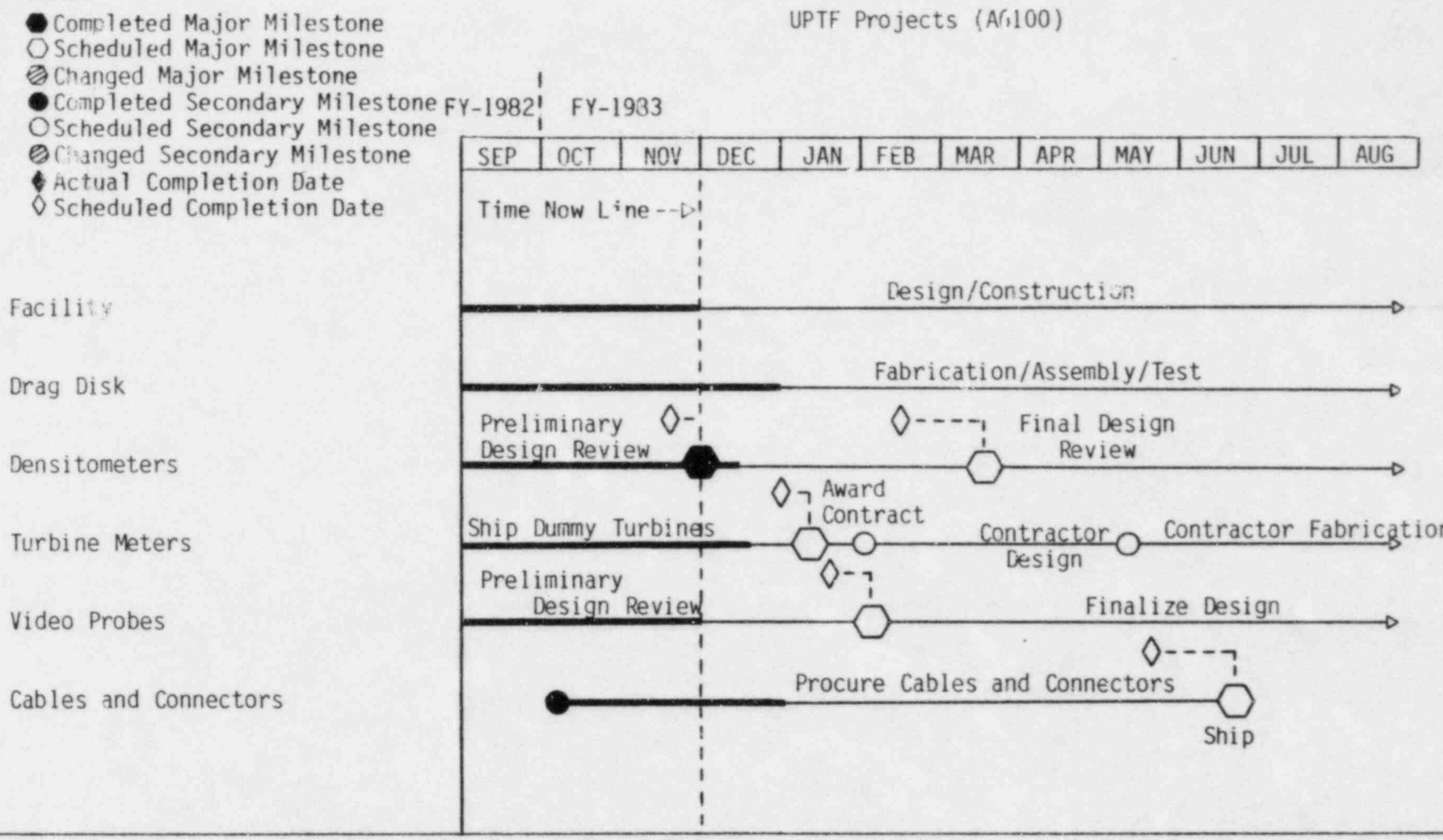
NOTES:

2D/3D PROGRAM

UPTF Projects (A6100)

November 1982

LEGEND



NOTES:

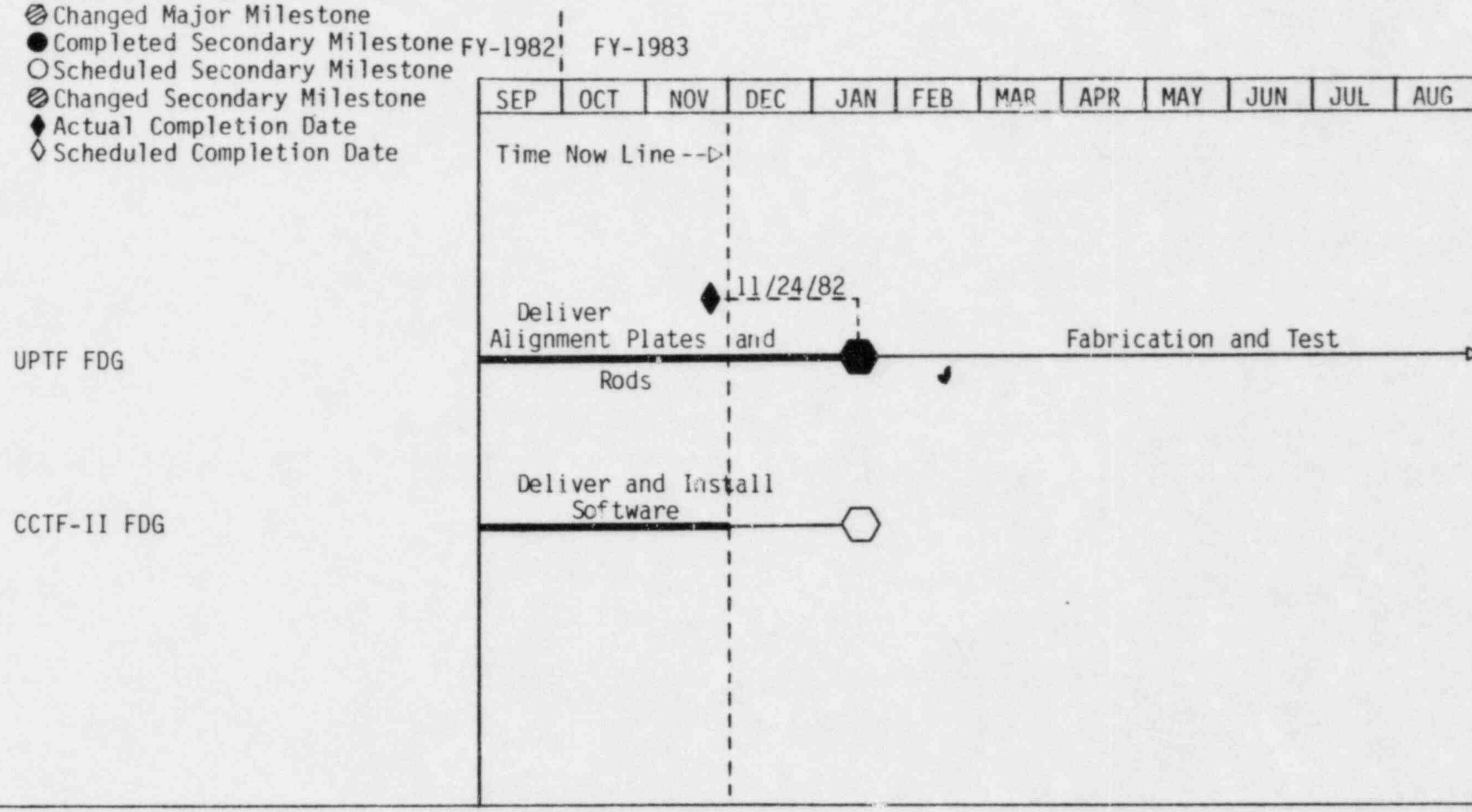
2D/3D PROGRAM

November 1982

Fluid Distribution Grids (A6282)

LEGEND

- ⬢ Completed Major Milestone
- ⬡ Scheduled Major Milestone
- Changed Major Milestone
- ● Completed Secondary Milestone
- ○ Scheduled Secondary Milestone
- Changed Secondary Milestone
- ♦ Actual Completion Date
- ◊ Scheduled Completion Date

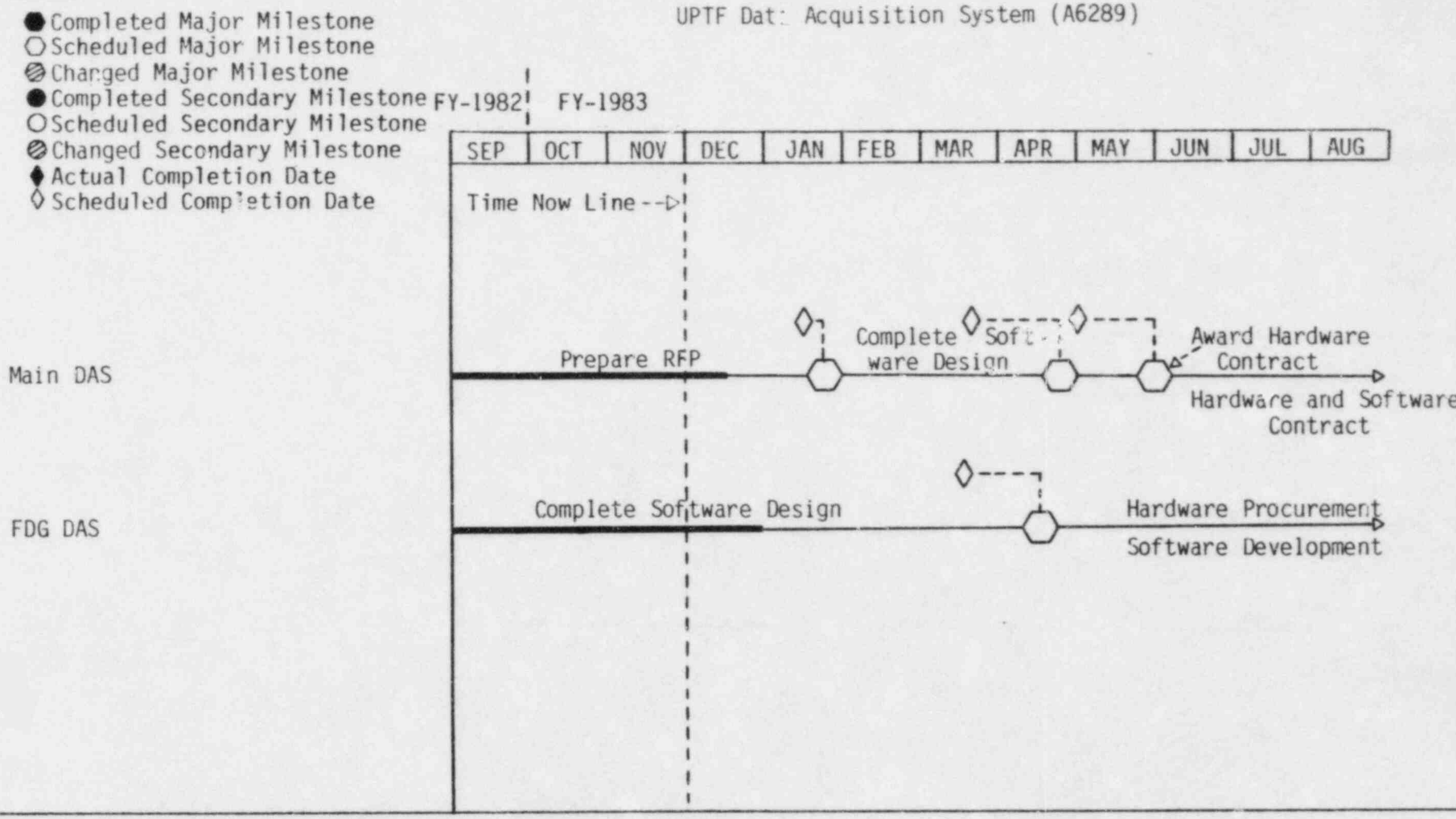


NOTES:

2D/3D PROGRAM

November 1982

UPTF Data Acquisition System (A6289)



NOTES:

2D/3D PROGRAM

TECHNICAL REVIEW AND SUMMARY

PROGRAM MANAGERS

SUMMARY AND HIGHLIGHTS

A preliminary design review for the UPTF Gamma Densitometer System was held on November 19, 1982. The alignment plates and dummy rods for the UPTF Fluid Distribution Grid/Liquid Level Detectors were shipped to Germany on November 24, 1982 approximately two weeks ahead of the scheduled date of December 7, 1982. The mechanical drawings for the optical sensor and dummy rod were released and sent to Germany.

A6100: 3D Technical Support and Instrumentation
EG&G Idaho Technical Monitor: J. B. Colson
DOE-ID Technical Monitor: W. R. Young
NRC Technical Monitor: Y. S. Chen

The 3D Technical Support and Instrumentation Project provides instrumentation and technical support for the 2D/3D Refill and Reflood Program. This is a multinational program under an international agreement among the United States Nuclear Regulatory Commission (USNRC), the Federal Minister for Research and Technology (BMFT) of the Federal Republic of Germany (FRG) and the Japan Atomic Energy Research Institute (JAERI). This program is designed as an analytical and experimental study of the thermal-hydraulic behavior of emergency core coolant during the refill and reflood phases of a postulated Loss-of-Coolant Accident (LOCA) in a pressurized water reactor (PWR). Instrumentation is being provided for the Cylindrical Core Test Facility (CCTF) and Slab Core Test Facility (SCTF) in Japan and the Primary Coolant Loop (PKL) and the Upper Plenum Test Facility (UPTF) in FRG. These instruments, which are based on advanced instrumentation developed at the Idaho National Engineering Laboratory (INEL), are being designed, fabricated, tested, and installed in the test facilities. The NRC is being supported in a staff capacity for all aspects of the 2D/3D Program including experimental design, operational support and analysis of test results.

1. Scheduled Milestones for November 1982

Node	Description	Due Date	Actual Date
	Preliminary Design Review for UPTF Densitometer	11-19-82	11-19-82

2. Summary of Work Performed in November 1982

A. Federal Republic of Germany (FRG) Upper Plenum Test Facility

1. 453071000 - Drag Disks

Draft procedures for acceptance testing the drag-disk transducers have been completed. Fabrication is proceeding on a low priority basis.

2. 453072000 - Gamma Densitometers

A preliminary design review of the densitometer system was conducted on 11-19-82.

2A. Summary of Work Performed in November 1982 (continued)

3. 453073000 - Turbine Meters

A report on the graphite turbine bearing tests was completed and transmitted to DOE/NRC. The procurement package for the UPTF turbine meter systems was completed and submitted to procurement for bid solicitation. Drawings were completed and work initiated on two turbine meter dummy probes for UPTF. All action items from the envelope design review were resolved and a letter of resolution transmitted.

B. Japan Atomic Energy Research Institution (JAERI) Slab Core Test Facility

1. 453091000 - Core II Refurbishment

The design of eight UCSP turbine meters for SCTF-II is 60% complete. The conductivity tips were modified and are ready to be used in probe assemblies.

2. 453092000 - Core III Refurbishment

No activity.

C. Operational Support

1. 453013000 - FRG Operational Support

No activity.

2. 453023000 - JAERI Operational Support

Repair of the LSI-11 and peripherals and a successful checkout completed.

The CCTF OLLD STALK repair is 10% complete. The shipment invoices for the graphics terminal and LSI system were transmitted to Dr. Hirano at JAERI and preparations began on fabrication of shipping crates.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

A. FRG Upper Plenum Test Facility

1. 453071000 - Drag Disks

Fabrication will continue on a low priority basis, i.e., to be worked on only when the shop needs work.

2. 453072000 - Gamma Densitometers

A design review report will be issued. The final design of the systems will be initiated.

3. 45307300 - Turbine Meters

The UPTF turbine meter dummy probes will be fabricated and prepared for shipment to Germany. Bids on the turbine meter systems are expected by December 31, 1982.

B. JAERI Slab Core Test Facility

1. 453091000 - Core II Refurbishment

The design of eight UCSP turbine meters for SCTF-II will be completed. Paperwork will be issued to obtain the eight UCSP turbine meters. The conductivity probes will be assembled for SCTF-II incore stalks.

2. 453092000 - Core III Refurbishment

No planned activity.

C. Operational Support

1. 453013000 - FRG Operational Support

No activity planned.

2. 453023000 - JAERI Operational Support

Verification of the LSI-11 operation will be demonstrated with JAERI personnel present at INEL. The LSI-11, peripherals, and graphics terminal will be shipped to Japan.

5. Problems and Potential Problems

None.

A6282: Fluid Distribution Grid System for 3D Program Facilities
EG&G Idaho Technical Monitor: J. B. Colson
DOE-ID Technical Monitor: W. R. Young
NRC Technical Monitor: Y. S. Chen

The fluid distribution measurement systems measure liquid level, and detect gross local voids and water distribution in various regions of each facility simulated core vessel for the 2D/3D Refill and Reflood Program. This is a multinational program under an international agreement among the United States Nuclear Regulatory Commission (USNRC), the Federal Minister for Research and Technology (BMFT) of the Federal Republi o Germany (FRG) and the Japan Atomic Energy Research Institute (JAERI). This program is designed as an analytical and experimental study of the thermal-hydraulic behavior of emergency core coolant during the refill and reflood phases of a postulated Loss-of-Coolant Accident (LOCA) in a pressurized water reactor (PWR). This instrumentation is being provided for the Cylindrical Core Test Facility (CCTF) and Slab Core Test Facility (SCTF) in Japan and the Upper Plenum Test Facility (UPTF) in FRG.

1. Scheduled Milestones for November 1982

Node	Description	Due Date	Actual Date
	Snip alignment plates and dummy rods for the UPTF FDG/LLD	12-7-82	11-24-82

2. Summary of Work Performed in November 1982

A. 451012000 - JAERI Cylindrical Core Test Facility Core-II Fluid Distribution Grid

All programming was completed. Software documentation was completed and delivered to word processing. A test procedure to be used in the acceptance demonstration was completed. A dry run of the acceptance demonstration was held successfully.

B. 451013000 - FRG Upper Plenum Test Facility Fluid Distribution Grid

The mechanical drawings for the optical sensor and the dummy rod were released and sent to Germany. The mechanical final design resolution letter was issued. The alignment plates and dummy rods for the upper plenum and incore FDG/LLD stalks were fabricated and shipped to Germany before the scheduled delivery date. A site work release was issued for the fabrication and assembly of the electronic signal conditioners.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

A. 451012000 - JAERI Cylindrical Core Test Facility Core II Fluid Distribution Grid System

The acceptance demonstration for JAERI personnel will be completed.

The documentation will be completed and the display equipment, software, and documentation will be shipped to Japan.

B. 451013000 - FRG Upper Plenum Test Facility Distribution Grid System

The optical fiber and tips will be received from the vendors. The conax seals for the upper plenum FDG/LLD stalks will be ordered. Ordering of support material for fabrication of the UPTF FDG/LLD optical probes will be completed. Purchase requisitions will commence for the electronic parts for the signal conditioners.

5. Problems and Potential Problems

None.

A6289: FRG Upper Plenum Test Facility Data Acquisition System
EG&G Idaho Technical Monitor: J. B. Colson
DOE-ID Technical Monitor: W. R. Young
NRC Technical Monitor: Y. S. Chen

The Data Acquisition System (DAS) for the Upper Plenum Test Facility (UPTF) Project provides an electronic data acquisition system for the experimental measurements in UPTF. This test facility is part of a multinational program under international agreement among the United States Nuclear Regulatory Commission (USNRC), the Federal Minister for Research and Technology (BMFT) of the Federal Republic of Germany (FRG) and the Japan Atomic Energy Research Institute (JAERI). This Program is designed as an analytical and experimental study of the thermal-hydraulic behavior of emergency core coolant during the refill and reflood phases of a postulated Loss-of-Coolant (LOCA) in a Pressurized Water Reactor (PWR). The UPTF is to be constructed in Germany.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

Work continued on revisions to the Implementation Plan and the Hardware Specification. Both of these documents are 95% complete. The Main DAS Software Design Review was held and revisions are being made to the Software Specification. Work on the Statement of Work and the Bidders Instructions continued but was delayed because of higher-priority work on the Implementation Plan and the two Specifications. A draft system study for the FDG System was completed and is being reviewed.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

The Hardware and Software Specifications, Implementation Plan, Statement of Work, and Bidders Instructions will be completed. The System Study for the FDG System will be completed and work will start on the FDG Hardware Specification. The Implementation Plan will be sent to DOE-ID for approval. The RFP package will be assembled.

5. Problems and Potential Problems

The availability of sufficient software personnel now and during the next six months continues to be a problem. The schedule may begin to slip if this condition is not improved. A solution to this problem is being pursued.

MONTHLY REPORT FOR

NOVEMBER 1982

NUCLEAR SAFETY METHODS DIVISION

F. Aguilar

F. Aguilar, Manager

L. Morgan

L. Morgan
Plans and Budget Representative

NUCLEAR SAFETY METHODS DIVISION

COST SUMMARY AND COMMENTS

NUCLEAR SAFETY METHODS DIVISION

189	Title	(1) Obligational Authority Carried Over From FY-1982	(2) New FY-1983 Obligational Authority	(3) Total Obligational Authority	(4) November FY-1983 YTD Costs	(5) (3)-(4) Balance	(6) Outstanding Commitments November FY-1983	(7) (5)-(6) Balance
A6050	Fuel Behavior Model Development	$ 0.0K	$ 90.0K	$ 90.0K	$ 43.5K	$ 46.5K	$ 0.0K	$ 46.5K
A6052	Code Development and Improvement - Transient Analysis	76.7K	160.0K	236.7K	115.1K	121.6K	0.0K	121.6K
A6278	TRAC-BWR Heat Transfer	112.0K	0.0K	112.0K	39.6K	72.4K	1.2K	71.2K
A6329	Code Development and Improvement - LOCA Analysis	0.0K	0.0K	0.0K	0.0K	0.0K	0.0K	0.0K
A6330	RELAP5	0.0K	0.0K	0.0K	119.6K	119.6K	10.0K	129.6K
A6360	Modeling Severe Fuel Damage	1.6K	150.0K	151.6K	134.8K	16.8K	0.0K	16.8K
		$190.3K	$400.0K	$590.3K	$452.6K	$137.7K	$11.2K	$126.5K

NUCLEAR SAFETY METHODS DIVISION

MILESTONE SCHEDULES

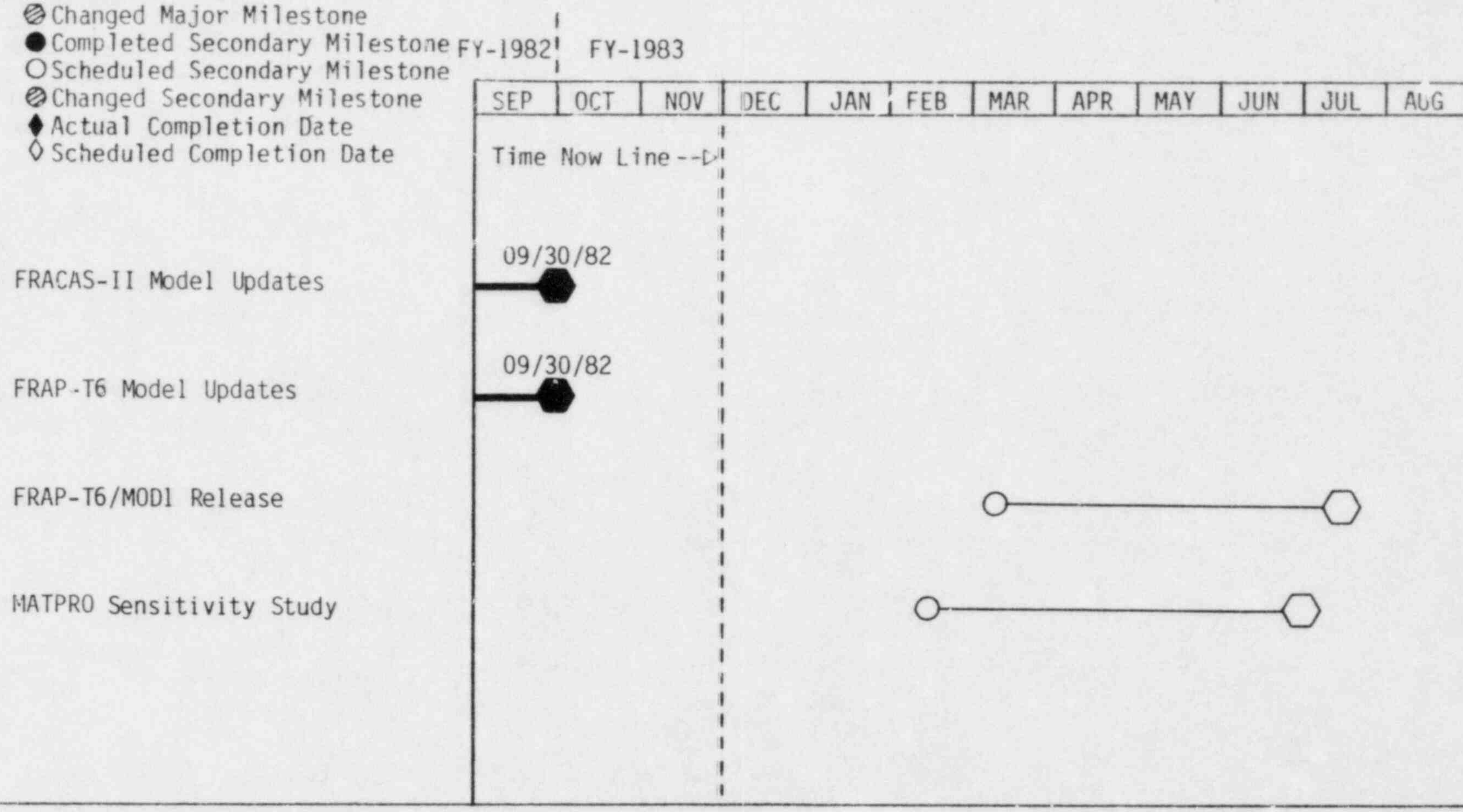
NUCLEAR SAFETY METHODS DIVISION

November 1982

Fuel Behavior Model Development (A6050)

LEGEND

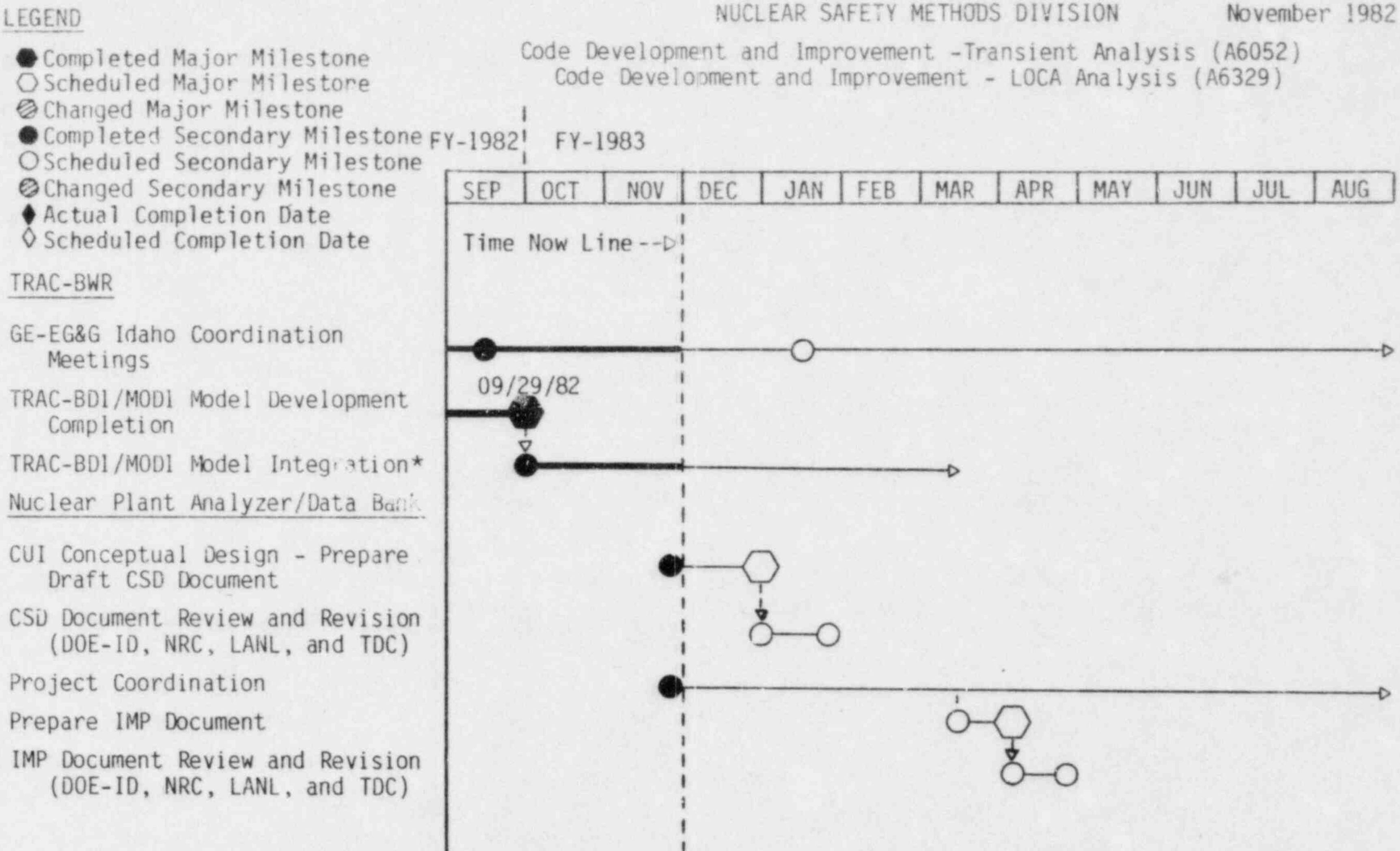
- ⬢ Completed Major Milestone
- ⬡ Scheduled Major Milestone
- Changed Major Milestone
- ● Completed Secondary Milestone
- ○ Scheduled Secondary Milestone
- Changed Secondary Milestone
- ♦ Actual Completion Date
- ◊ Scheduled Completion Date



NOTES: These are proposed milestones that are being negotiated with NRC.

NUCLEAR SAFETY METHODS DIVISION November 1982

Code Development and Improvement -Transient Analysis (A6052)
Code Development and Improvement - LOCA Analysis (A6329)



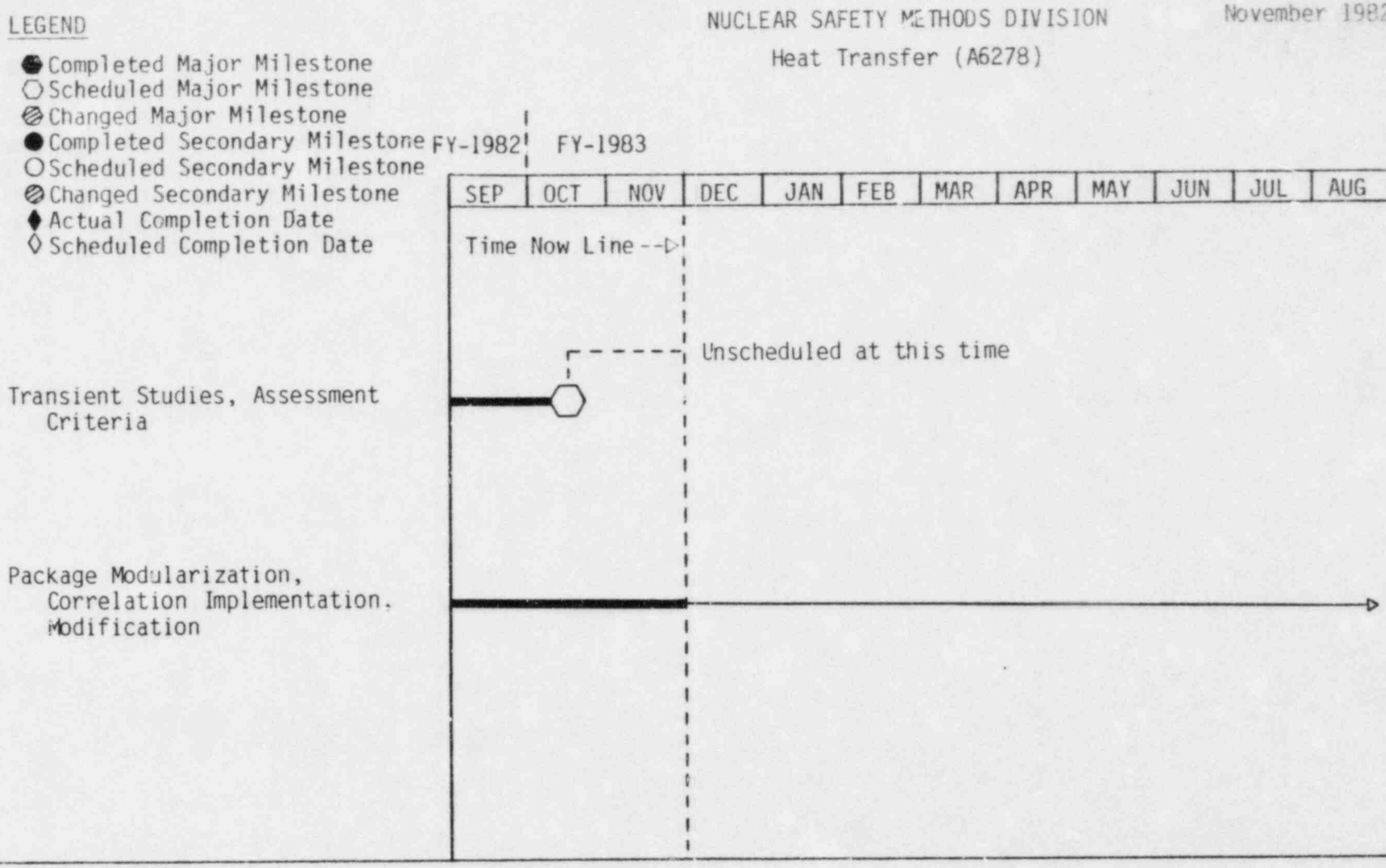
NOTES: * Integration of INEL models is underway. A completion date for model integration including GE's and the content of the MOD1 code is now under review.

The Nuclear Plant Analyzer/Data Bank task has been adapted from FA-121-82, dated December 2, 1982.

NUCLEAR SAFETY METHODS DIVISION

November 1982

Heat Transfer (A6278)

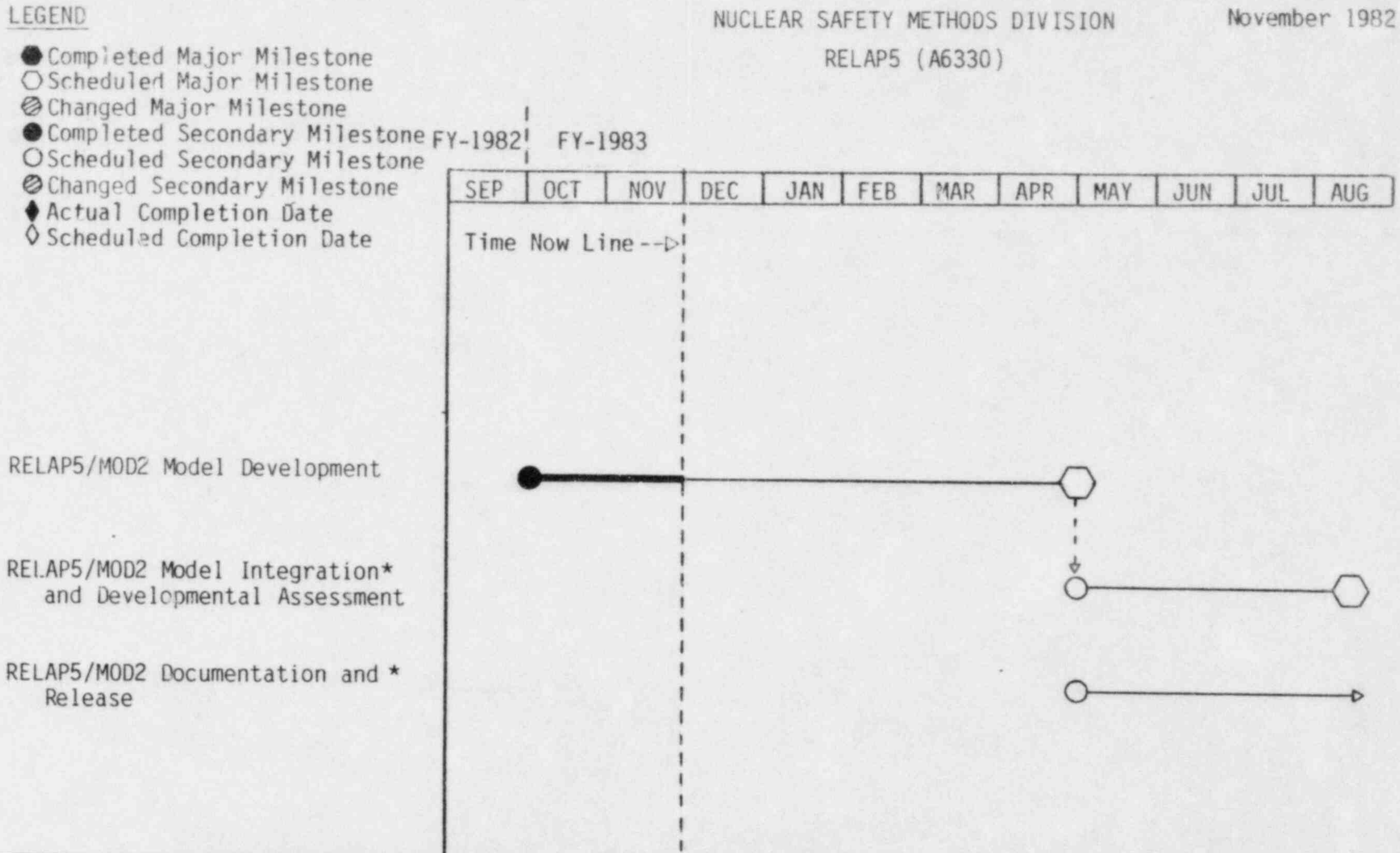


NOTES: The TRAC BWR Heat Transfer milestone chart is adapted from FA-68-81 and has been revised as per FA-154-81. Conclusion of the transient studies now suspended is predicted on resolution of the interfacial shear problem. Interim heat transfer work is defined and progressing under the package modularization task. Final Assessment of the heat transfer package will be scheduled after determination of the TRAC-BD1/MOD1 completion date.

NUCLEAR SAFETY METHODS DIVISION

November 1982

RELAP5 (A6330)



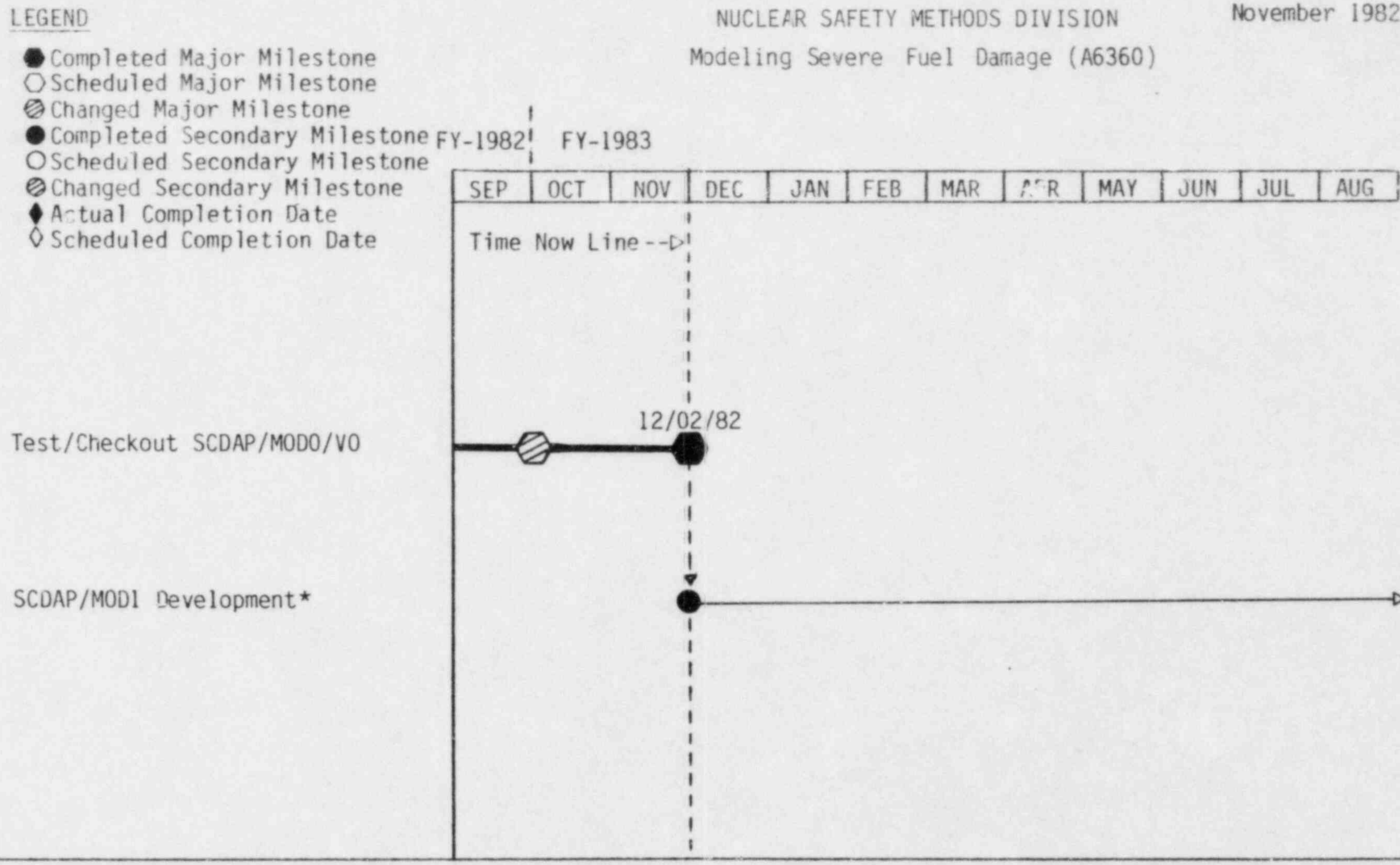
NOTES: Detailed milestones are currently being developed for the RELAP5 Program.

* These are proposed milestones that are being negotiated with NRC.

NUCLEAR SAFETY METHODS DIVISION

November 1982

Modeling Severe Fuel Damage (A6360)



NOTES: Development of FY-1983 milestones has been delayed until the SCDAP/MOD0 checkout and testing activity has been completed.

* SCDAP/MOD1 development schedule is being negotiated with NRC.

NUCLEAR SAFETY METHODS DIVISION

TECHNICAL REVIEW AND SUMMARY

PROGRAM MANAGER'S

SUMMARY AND HIGHLIGHTS

Successful checkout of SCDAP/MOD0 was completed on December 2, 1982. A SCDAP subcode called SCDCOMP was used for both pre- and post-test analysis of the SFD-ST test of the Thermal Fuels Behavior Program. These results, as well as the results of a length effects study done with SCDCOMP, were presented to NRC personnel.

A plan for doing the conceptual design of the common user interface for the Nuclear Plant Analyzer and Data Bank (NPA & DB) project was developed. This plan will result in two major deliverables. The first will be a conceptual system design document to be used as a reference to guide NPA & DB development. The second deliverable will be an integrated management plan for the joint development of the nuclear plant analyzer at the Los Alamos National Laboratory (LANL), Technology Development Corporation (TDC), and INEL.

A6050: Fuel Behavior Model Development

EG&G Technical Monitor: T. M. Howe
DOE-ID Technical Monitor: D. Majumdar
NRC Technical Monitor: G. P. Marino

The Fuel Behavior Model Development Project provides for development and maintenance of (a) a "best estimate" computer code (FRAP-T) which predicts the thermal-mechanical-chemical behavior of light water reactor fuel rods during anticipated transients and postulated accidents including fuel failure probabilities and the associated release of fission products from the fuel rod after such events, (b) basic transient fuel rod behavior models which are required for the FRAP-T code and the SCDAP code, and LWR fuel rod materials properties models which serve as an environmental package (MATPRO) for the fuel behavior codes. Additionally, experimental data and analytical models from the Idaho National Engineering Laboratory (INEL) and other national laboratories, industry, etc., are reviewed and incorporated in the computer codes as appropriate. The analytical tools developed by this project are used by the Nuclear Regulatory Commission (NRC) to audit licensee submittals and by NRC's contractors to plan and interpret fuel behavior experiments.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

a FRACAS-II

A study was performed using FRAP-T6 with the new FRACAS-II subcode to assess pellet-cladding mechanical interaction modeling. Needed model development for FRACAS-II has been identified and will be incorporated in a letter report describing the study and its results to be issued during December.

b. FRAP-T6

Because of problems encountered in the restart of FRAP-T6 from FRAPCON-2, transmittal of the new version of FRAP-T6 to the NESC was delayed. These problems were resolved during November, and the new code version will be transmitted to NESC in early December. The task to reduce the FRAP-T6 running time has begun and will continue through June.

2. Summary of Work Performed in November 1982 (continued)

c. Transient Fuel Behavior Models

The report describing the SCDAP/MODO fission gas release models was completed. The information necessary for Argonne National Laboratory (ANL) to restructure PARAGRASS was provided to ANL personnel during the November ANS meeting. A letter will be sent to ANL in December requesting changes to PARAGRASS.

3. Scheduled Milestones for November 1982

None.

4. Summary of Work to be Performed in December 1982

a. FRACAS-II

A letter report describing the results of the PCMI study comparing Halden experimental data with FRAP-T6 calculations using FRACAS-II will be issued in December.

b. FRAP-T6

The preliminary design of the task to reduce FRAP-T6 running time will be completed. This will include examining alternative ways to reduce running time such as the method used at PNL on FRAPCON-2, assessing those alternatives, and choosing the most effective course of action.

The FRAP-T6 version with the new FRACAS-II model will be transmitted in early December to the NESC.

5. Problems and Potential Problems

None.

A6052: Code Development and Improvement

EG&G Technical Monitor: A. C. Peterson, Jr.
DOE-ID Technical Monitor: D. Majumdar
NRC Technical Monitor: F. Odar

The primary objective of this program is to develop and improve computer codes to predict the system response of light water reactors to postulated design basis loss-of-coolant accidents, operational transients, and anticipated transients without scram. The current emphasis of this program is the continued improvement of the TRAC-BWR computer code, which is an advanced best estimate code to analyze boiling water reactors. The development of the design of a "Nuclear Plant Analyzer" using advanced computer codes is also being performed.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

a. Boiling Water Reactor (BWR) TRAC Development

Assembly of TRAC-BD1/MOD1 continued. Checkout of Candidate Version 15 continued. A new version of TRAC-BD1/Version 12 was made for the NRC Technical Assistance Program to be compatible with changes to the computer operating system. Coding of the moving mesh reflood model for the channel wall continued. Evaluation of the GE level wall continued. Evaluation of the GE level tracking model continued. Evaluation of the GE separator/dryer model was begun.

b. RELAP4/MOD5 and MOD7 Maintenance

"Level 1" maintenance was provided.

c. Nuclear Plant Analyzer (NPA) Development and Coordination

A plan was developed for doing the conceptual design of the common user interface for the nuclear plant analyzer and data bank. This plan will result in two major deliverables: a conceptual system design document and an integrated management plan. The plan was reviewed and approved by NRC/RES, LANL, and TDC. Development of a set of functional requirements for the nuclear plant analyzer continued. The development of three design alternatives for the common user interface began.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

a. Boiling Water Reactor (BWR) TRAC Development

Assembly of TRAC-BD1/MOD1 will continue. Testing of the non-condensible gas model will be completed and it will be inserted into an official code version. Evaluation of the GE level tracking model will be completed and a rough draft of a completion report will be prepared. Evaluation of the GE separator/dryer model will continue. A meeting will be held with Dr. F. Odar of the NRC to discuss the TRAC-BWR work plan.

b. RELAP4/MOD5 and MOD7 Maintenance

"Level 1" maintenance will be provided.

c. Nuclear Plant Analyzer (NPA) Development and Coordination

A draft set of functional requirements will be published as an NSMD internal technical report. This report will be distributed to NRC, LANL, TDC, and DOE-ID for review, comment, and approval. The development of three design alternatives for the common user interface will continue.

5. Problems and Potential Problems

Delivery of the remaining GE LOCA models has been delayed. This delay in their delivery may preclude the inclusion of these models in TRAC-BD1/MOD1.

A6278: TRAC-BWR Heat Transfer

EG&G Technical Monitor: A. C. Peterson, Jr.
DOE-ID Technical Monitor: D. Majumdar
NRC Technical Monitor: M. Young

The primary objective of this program is to develop and assess a best estimate heat transfer package for the analysis of design-basis loss-of-coolant accidents, operational transients, and anticipated transients without scram of boiling water reactors. A best estimate heat transfer package is important for advanced reactor transient analysis computer codes that will be used by the Nuclear Regulatory Commission to audit nuclear power plant safety issues, evaluate operator guidelines, address unresolved safety issues, and design and interpret reactor safety experiments.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

Coding of the moving mesh reflood model for the channel wall continued. Modifications to the interfacial shear package in TRAC-BWR continued. Test calculations with Lehigh data and for CCFL conditions indicated additional modifications are required.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

Testing of the moving mesh reflood model for the channel wall will continue. Modification and testing of the interfacial shear package will continue and a final set of modifications prepared when the test results are considered satisfactory.

5. Problems and Potential Problems

An interfacial shear problem which results in hydraulic oscillations continues to be encountered in the analysis of the Lehigh post-CHF heat transfer data and continue to impact the completion of the transient sensitivity study. The study will be resumed when the shear problem is resolved.

A6330: RELAP5

EG&G Idaho Technical Monitor: T. M. Howe
DOE-ID Technical Monitor: D. Majumdar
NRC Technical Monitor: Y. Chen

The primary objective of the project is to develop and improve the RELAP5 code to predict the system response of light water reactors to postulated design-basis loss-of-coolant accidents, operational transients, and anticipated transients without scram. RELAP5 provides the Nuclear Regulatory Commission (NRC) with a fast-running, economic, best-estimate analytical capability to audit nuclear power plant safety analysis reports, evaluate proposed guidelines and rules, address unresolved safety issues, and design and interpret reactor safety experiments. A secondary objective is to maintain RELAP5 on the Idaho National Engineering Laboratory (INEL) computer facility and provide NRC and its contractor analysts with assistance with the application of RELAP5.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

Testing of the interphase drag model was completed during November. Some additional development of the model was required to resolve difficulties encountered with the stratified choked flow model. A completion report will be issued during December. The interphase mass model improvement task was initiated by stating formulation of a limited two-temperature model and an extended nonequilibrium model. The design proposals for these models will be completed during December and January, respectively. The control system modeling capability was extended by addition of lead/lag, proportional-integrator, and constant control models. A completion report for these models will be issued during December.

Several code problems were resolved during November. These include a kinetics oscillation problem, a pump problem, and a valve error. These corrections were incorporated into RELAP5/MOD1/Cycle 19. Testing of the code version was completed. The updates for Cycle 19 will be issued with the RELAP5 Newsletter.

Two papers were presented at the ANS Winter Annual Meeting in Washington D.C. A draft of the RELAP5/MOD1 developmental assessment report was completed during November and is currently being reviewed. Depending on the extent of comments, the report will be issued during

2. Summary of Work Performed in November 1982 (continued)

December or January. Also, the RELAP5 Newsletter was completed and the final review is in process. The Newsletter will be issued during December.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

Model completion reports will be issued for the interphase drag model and the control system models completed during November. The design proposal for the limited two-temperature model will be completed and model coding will be initiated. Formulation of the extended nonequilibrium model will continue during December. The design proposal for this model will be completed during January. A design proposal will be initiated for the dynamic gap conductance model. This design will be completed during January. The new ANS decay heat model will be incorporated into RELAP5 and the updates tested. The completion report for this model will be issued during January. Planned modification of the internal plot package will be completed during December along with issuance of the completion report for this task.

Review of the RELAP5 Newsletter and the RELAP5/MOD1 developmental assessment report will be completed. The Newsletter will be issued during December. Depending on the number and extent of comments on the developmental assessment report, the report will be issued during December or January. A presentation will be prepared for the Second International Topical Meeting on Nuclear Thermal Hydraulics. A program review and discussion will be held with the NRC project manager.

5. Problems and Potential Problems

None.

A6360: Modeling Severe Fuel Damage

EG&G Technical Monitor: T. M. Howe
DOE-ID Technical Monitor: D. Majumdar
NRC Technical Monitor: G. P. Marino

The Modeling Severe Fuel Damage Project provides for development and maintenance of a mechanistic computer code, SCDAP, to predict the thermal-mechanical-chemical behavior of a light water reactor core during severe reactor accidents. The individual models and integrated code developed in this project are the focal point of knowledge gained from the Nuclear Regulatory Commission's (NRC) Severe Fuel Damage Program as well as from industry and foreign sponsored research. The SCDAP project, coupled with NRC's severe fuel damage experimental programs, provides (a) the analytical methodology needed to identify and understand the phenomena which control LWR core behavior during severe accidents and (b) a capability to plan and interpret severe fuel damage experiments.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

a. SCDAP/MODO Checkout and Testing

Checkout of SCDAP/MODO is complete. This involved running three test cases (SFD-ST and two differing cases of SFD-1) and issuing the SCDAP/MODO User's Manual. The manual includes a model description section, a code usage section, a sample calculation section which presents the results of the three test cases, and a user input section. The TMI-2 case was run on the SCDCOMP/SCDEBRIS version but was not run on the entire code at this time to reduce computer expenditures.

b. Advanced LIQSOL Model Development

The conceptual design of the extension of the LIQSOL model to include calculation of melting and relocation of UO_2 fuel and ZrO_2 which began in October continued through November resulting in a draft of the preliminary design report.

c. SCDAP/MODO Assessment

The assessment effort was maintained at a planned low level of effort through November. Assessment activities will increase during the December through February period.

2. Summary of Work Performed in November 1982 (continued)

d. SCDAP Support

The length-effects study was completed and the results presented to NRC personnel in Washington D.C. This study was performed to determine the effects of fuel rod length (3 versus 12 ft) on rod behavior under severe accident conditions. The pretest and posttest predictions of the PBF SFD-ST test using SCDCOMP were also presented to NRC personnel. A limited evaluation of the modeling approach and capabilities of the MIMAS code was performed and discussed with NRC.

3. Scheduled Milestones for December 1982

Node	Description	Due Date	Actual Date
---	Complete SCDAP/MODO Testing/ Checkout	09/30/82	12/02/82

4. Summary of Work to be Performed in December 1982

a. Advanced LIQSOL Model Development

The extension of the LIQSOL model to include the calculation of melting and relocation of UO_2 and ZrO_2 will continue through December. Coding and testing will be initiated during December.

b. SCDAP/MODO Assessment

A detailed plan to assess SCDAP/MODO using available experimental data and sensitivity studies will be developed in early December. Based on the plan, assessment activities will be conducted through February.

c. SCDAP Support

SCDAP/MODO pretest predictions of the PBF SFD-1 tests will be performed and a letter report written on the results. The lengths effects study will be documented and the results sent to DOE/NRC to provide information for a future ACRS meeting.

5. Problems and Potential Problems

None.

MONTHLY REPORT FOR

NOVEMBER 1982

NRC TECHNICAL ASSISTANCE PROGRAM DIVISION

B F Saffell

B. F. Saffell, Jr., Manager

E. L. Pierson

E. L. Pierson
Plans and Budget Representative

NRC TECHNICAL ASSISTANCE PROGRAM DIVISION

COST SUMMARY AND COMMENTS

NRC TECHNICAL ASSISTANCE PROGRAM DIVISION

189	Title	(1) Obligational Authority Carried Over From FY-1982	(2) New FY-1983 Obligational Authority	(3) Total Obligational Authority	(4) November FY-1983 YTD Costs	(5) (3)-(4) Balance	(6) Outstanding Commitments November FY-1983	(7) (5)-(6) Balance
A6039	Technical Surveillance	$ 200.8K	$ 0.0K	$ 200.8K	$ 101.8K	$ 99.0K	$ 1.1K	$ 97.9K
A6046	Fuel Behavior Analysis	56.5K	0.0K	56.5K	18.6K	37.9K	0.0K	37.9K
A6047	LOCA Analysis Assessment	236.3K	148.0K	384.3K	238.9K	145.4K	0.0K	145.4K
A6102	Data Bank	4.2K	0.0K	4.2K	70.4K	<66.2K>	0.0K	<66.2K>
A6276	LER Evaluation	12.1K	0.0K	12.1K	23.3K	<11.2K>	0.0K	<11.2K>
A6283	Common Cause	1.0K	0.0K	1.0K	17.1K	<16.1K>	0.0K	<16.1K>
A6294	Plant Status Monitoring	63.9K	0.0K	63.9K	26.9K	37.0K	0.4K	36.6K
A6301	Accident Sequence Eval.	176.6K	0.0K	176.6K	78.7K	97.9K	0.0K	97.9K
A6304	Resident Eng. - Germany	3.0K	0.0K	3.0K	0.0K	3.0K	0.0K	3.0K
A6306	HDR Evaluation	6.4K	70.0K	76.4K	47.1K	29.3K	0.0K	29.3K
A6308	Display Design and Eval.	328.8K	0.0K	328.8K	135.0K	193.8K	2.9K	190.9K
A6310	Low Level Waste Risk Meth	108.8K	0.0K	108.8K	71.7K	37.1K	0.0K	37.1K
A6313	Init Evant Data Eval.	48.0K	0.0K	48.0K	15.1K	32.9K	0.0K	32.9K
A6315	Pre-HTGR Siting Eval.	70.1K	0.0K	70.1K	32.2K	37.9K	0.0K	37.9K
A6316	Paramaters Influ Damp.	9.4K	70.0K	79.4K	21.3K	58.1K	0.0K	58.1K
A6317	Data for NREP	8.8K	0.0K	8.8K	44.4K	<35.6K>	0.0K	<35.6K>
A6318	Sys Req & Stnds Dev (RPVs)	135.6K	0.0K	135.6K	26.2K	109.4K	0.0K	109.4K
A6322	EQ Research Program	693.0K	0.0K	693.0K	20.6K	672.4K	0.0K	672.4K
A6326	Integ of Cont Penetration	15.5K	0.0K	15.5K	15.5K	0.0K	0.0K	0.0K
A6331	Emergency Oper Proced Guide	96.9K	0.0K	96.9K	4.4K	92.5K	0.0K	92.5K
A6354	Sev Accident Seq Analysis	322.1K	0.0K	322.1K	133.0K	189.1K	0.0K	189.1K
A6356	Relief Valve Testing	299.2K	0.0K	299.2K	48.7K	250.5K	0.0K	250.5K
A6358	Applied James-Stein Est.	0.7K	0.0K	0.7K	0.6K	0.1K	0.0K	0.1K
A6367	Section XI Support	45.3K	0.0K	45.3K	19.9K	25.4K	0.0K	25.4K
A6369	Nuc Power Plnt Inst Eval.	14.9K	0.0K	14.9K	68.7K	<53.8K>	25.6K	<79.4K>
A6370	Microproc Based Sys Design	94.2K	70.0K	164.2K	58.9K	105.3K	0.0K	105.3K
A6371	Radiological Air Samp.	116.4K	0.0K	116.4K	12.5K	103.9K	0.7K	103.2K
A6376	Two-Phase Inst Eval.	30.0K	0.0K	30.0K	28.6K	1.4K	0.0K	1.4K
A6380	Diagnostic Inst Eval.	10.4K	0.0K	10.4K	26.0K	<15.6K>	0.0K	<15.6K>
A6384	NRC-Halden Delegate	43.5K	145.0K	188.5K	10.1K	178.4K	0.0K	178.4K
	NTAPD TOTALS	$3,252.4K	$503.0K	$3,755.4K	$1,416.2K	$2,339.2K	$30.7K	$2,308.5K

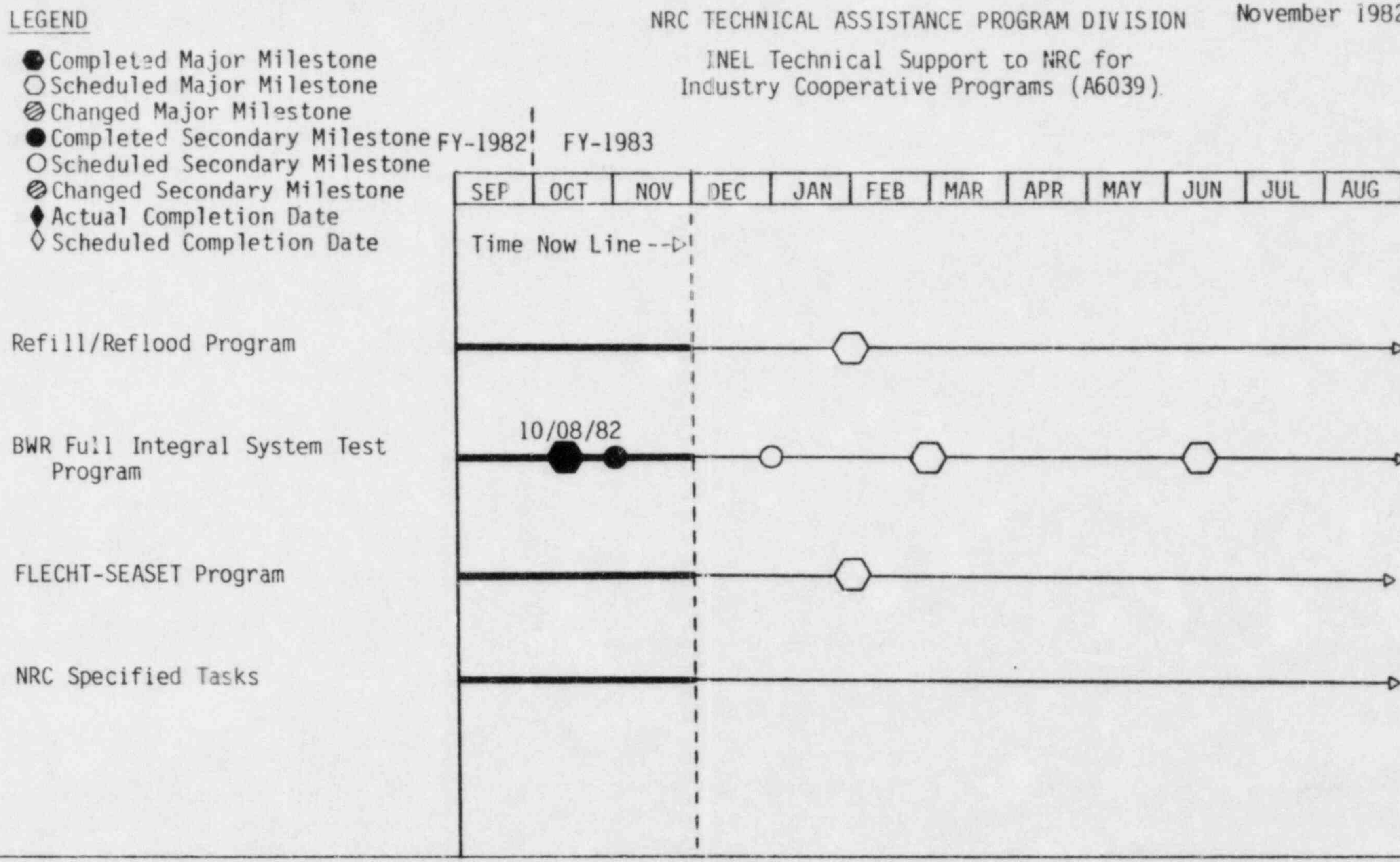
NRC TECHNICAL ASSISTANCE PROGRAM DIVISION

MILESTONE SCHEDULES

November 1982

NRC TECHNICAL ASSISTANCE PROGRAM DIVISION

INEL Technical Support to NRC for Industry Cooperative Programs (A6039)



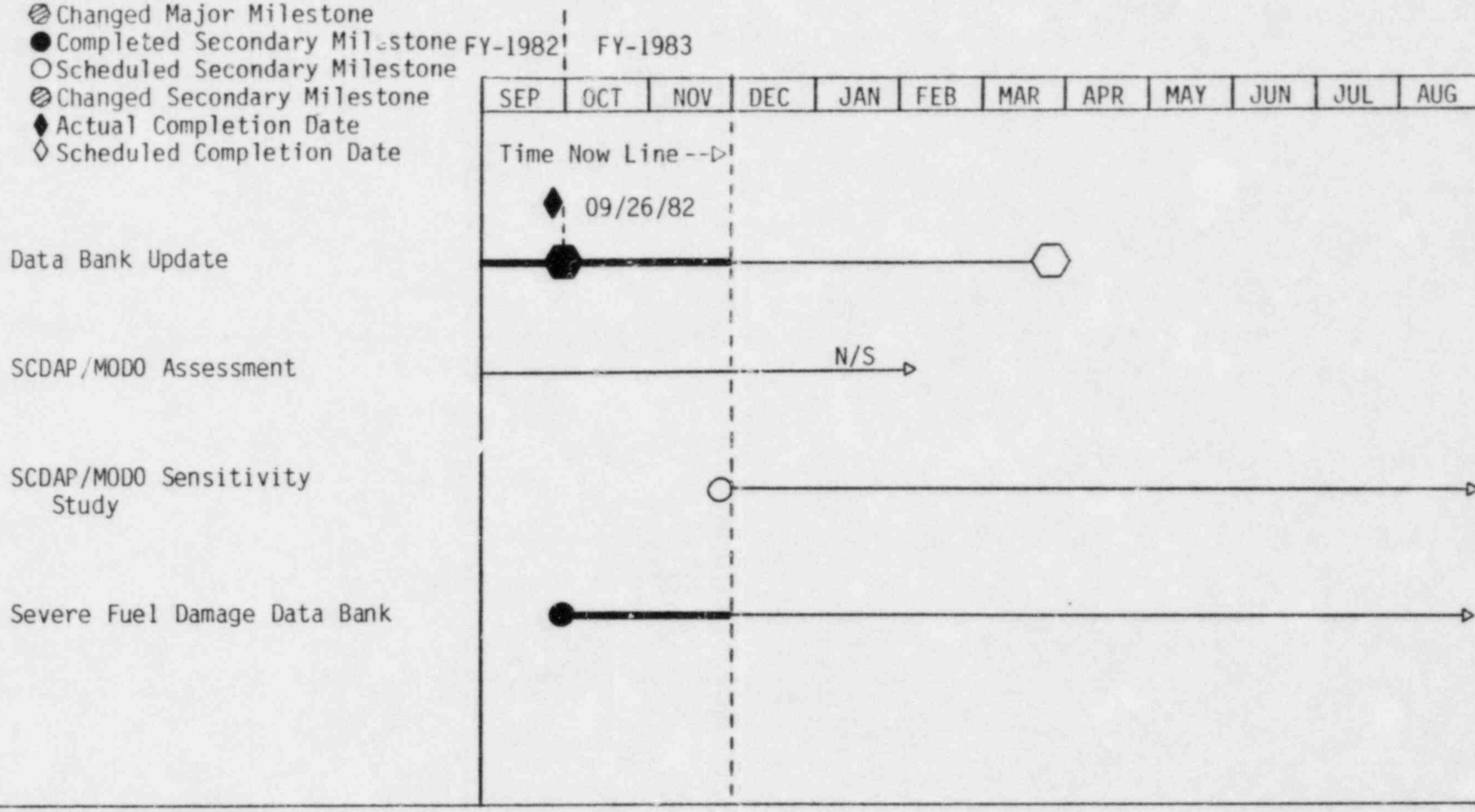
NOTES:

NRC TECHNICAL ASSISTANCE PROGRAM DIVISION November 1982

Fuel Code Assessment (A6046)

LEGEND

● Completed Major Milestone
○ Scheduled Major Milestone
⊘ Changed Major Milestone
● Completed Secondary Milestone
○ Scheduled Secondary Milestone
⊘ Changed Secondary Milestone
♦ Actual Completion Date
◊ Scheduled Completion Date



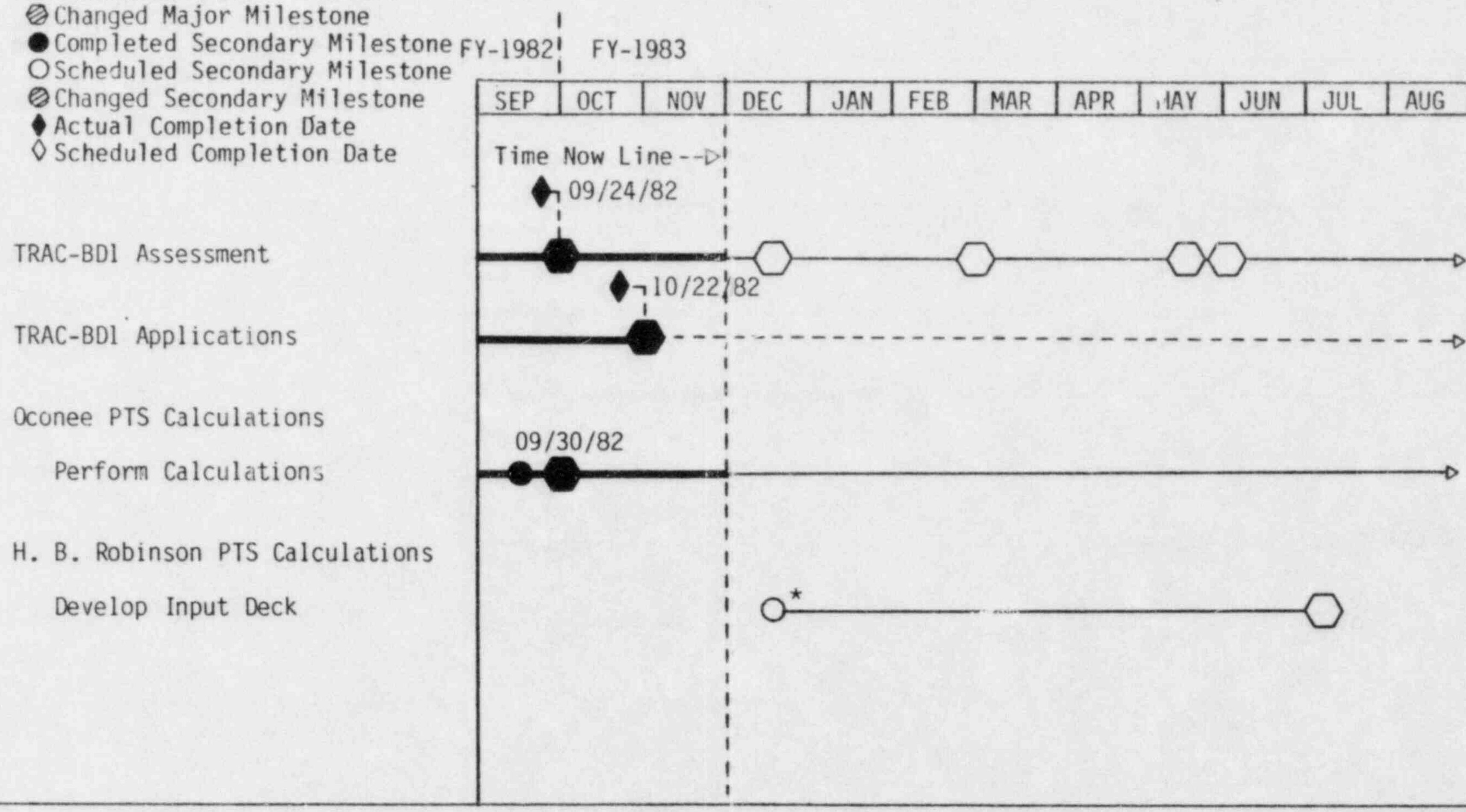
NOTES:

NRC TECHNICAL ASSISTANCE PROGRAM DIVISION November 1982

Code Assessment and Applications (A6047)

LEGEND

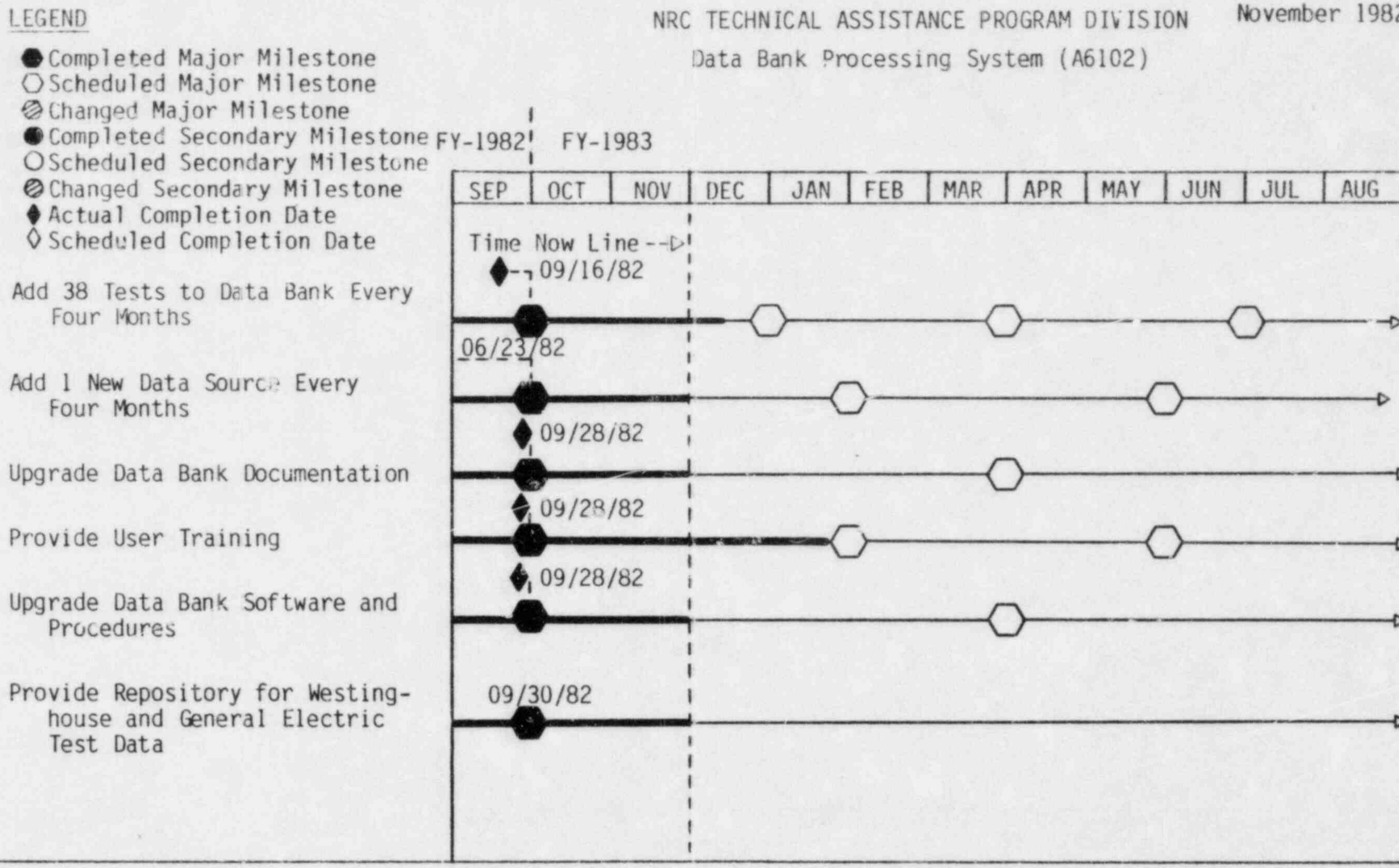
- Completed Major Milestone
- Scheduled Major Milestone
- Changed Major Milestone
- Completed Secondary Milestone
- Scheduled Secondary Milestone
- Changed Secondary Milestone
- Actual Completion Date
- Scheduled Completion Date



NOTES: * Insufficient plant information was received in November to start task.

NRC TECHNICAL ASSISTANCE PROGRAM DIVISION November 1982

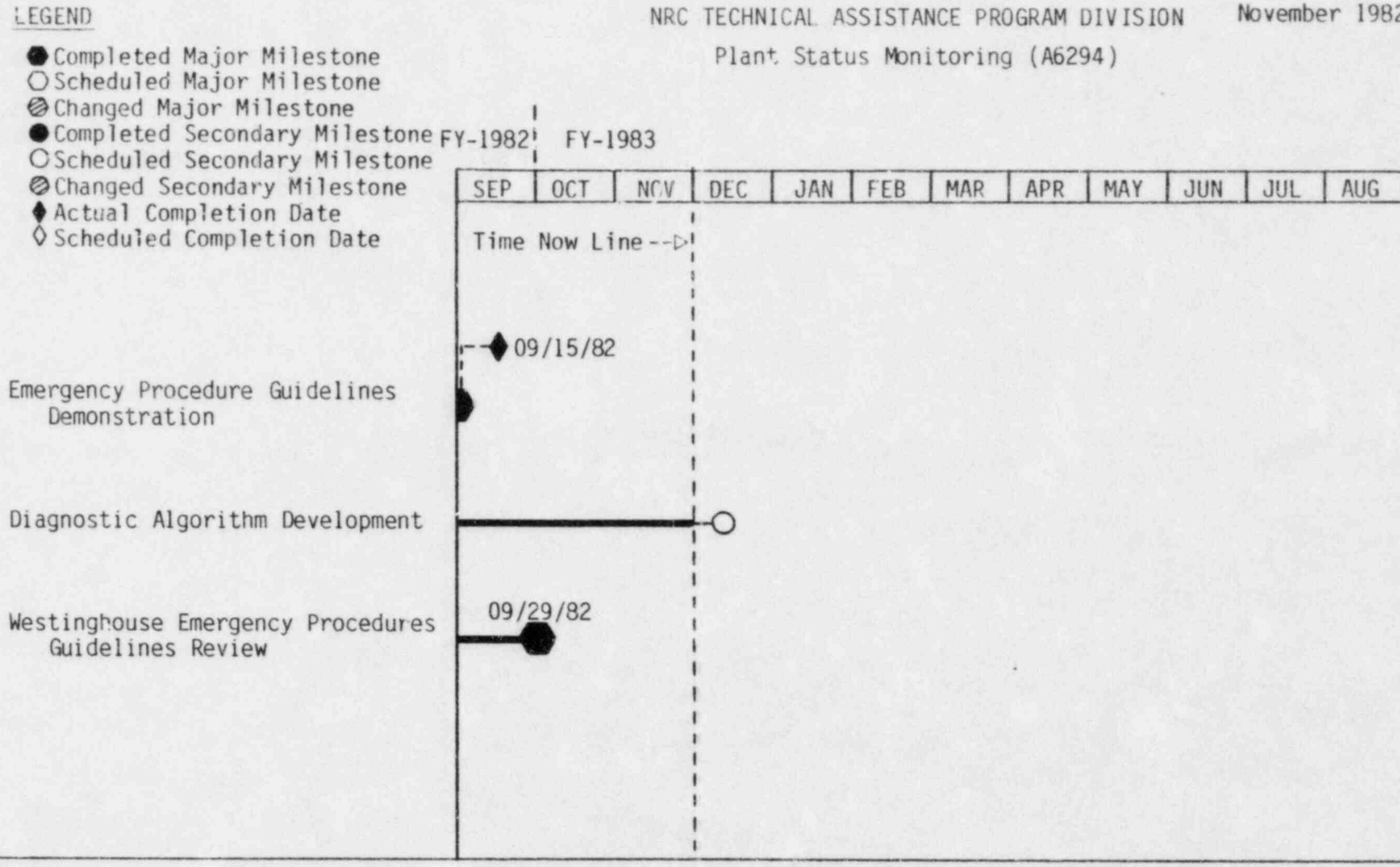
Data Bank Processing System (A6102)



NOTES:

NRC TECHNICAL ASSISTANCE PROGRAM DIVISION November 1982

Plant Status Monitoring (A6294)



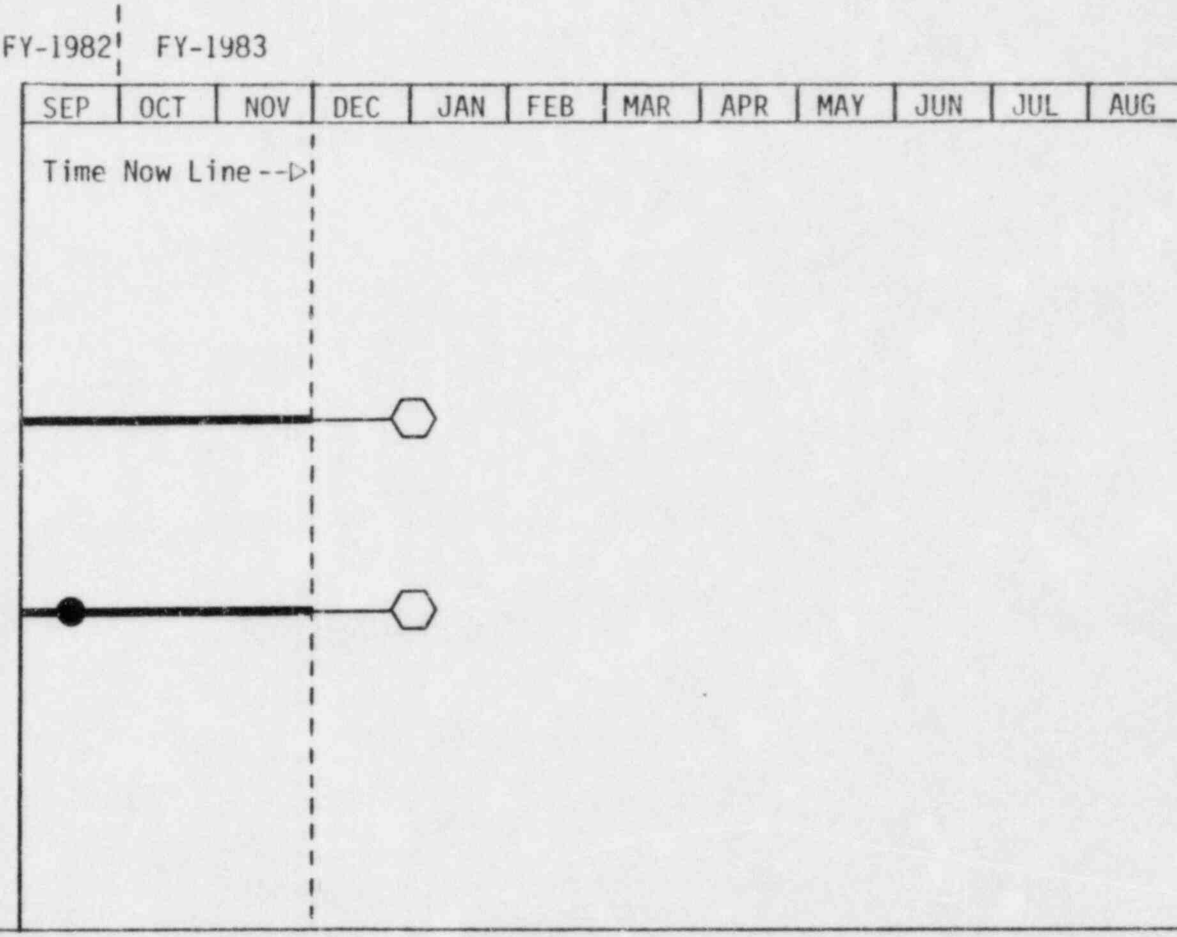
NOTES:

NRC TECHNICAL ASSISTANCE PROGRAM DIVISION November 1982

HDR Mechanical Component Response Analysis (A6306)

LEGEND

- ● Completed Major Milestone
- ○ Scheduled Major Milestone
- ⊘ Changed Major Milestone
- ● Completed Secondary Milestone
- ○ Scheduled Secondary Milestone
- ⊘ Changed Secondary Milestone
- ♦ Actual Completion Date
- ◊ Scheduled Completion Date



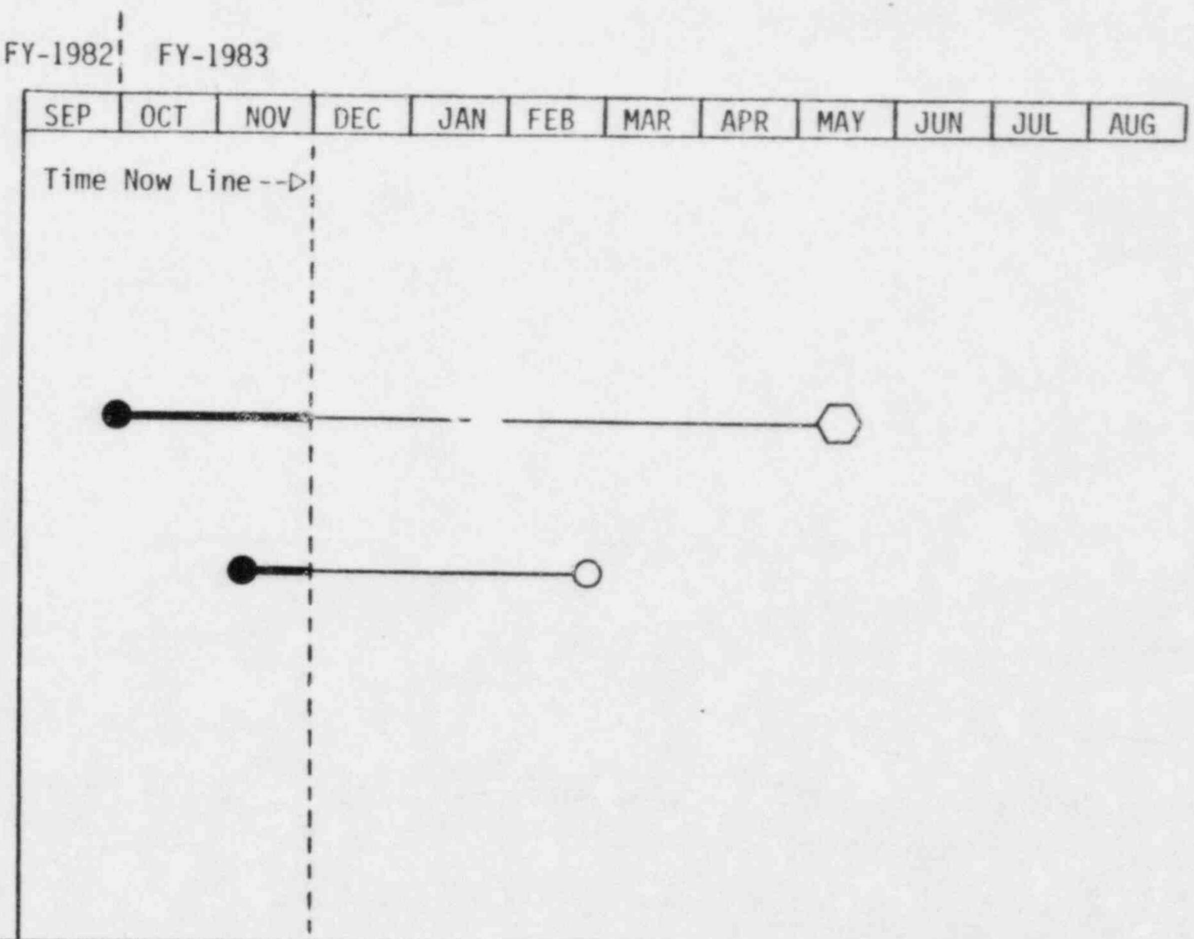
NOTES: All nodes are subject to change based on HDR's schedule.

NRC TECHNICAL ASSISTANCE PROGRAM DIVISION November 1982

Emergency Operating Procedure Guidelines (A6331)

LEGEND

- ⬢ Completed Major Milestone
- ⬡ Scheduled Major Milestone
- Changed Major Milestone
- ● Completed Secondary Milestone
- ○ Scheduled Secondary Milestone
- Changed Secondary Milestone
- ♦ Actual Completion Date
- ◊ Scheduled Completion Date



GE Emergency Procedure Guideline Demonstration

Training EG&G Idaho Engineer on the Methodology

NOTES:

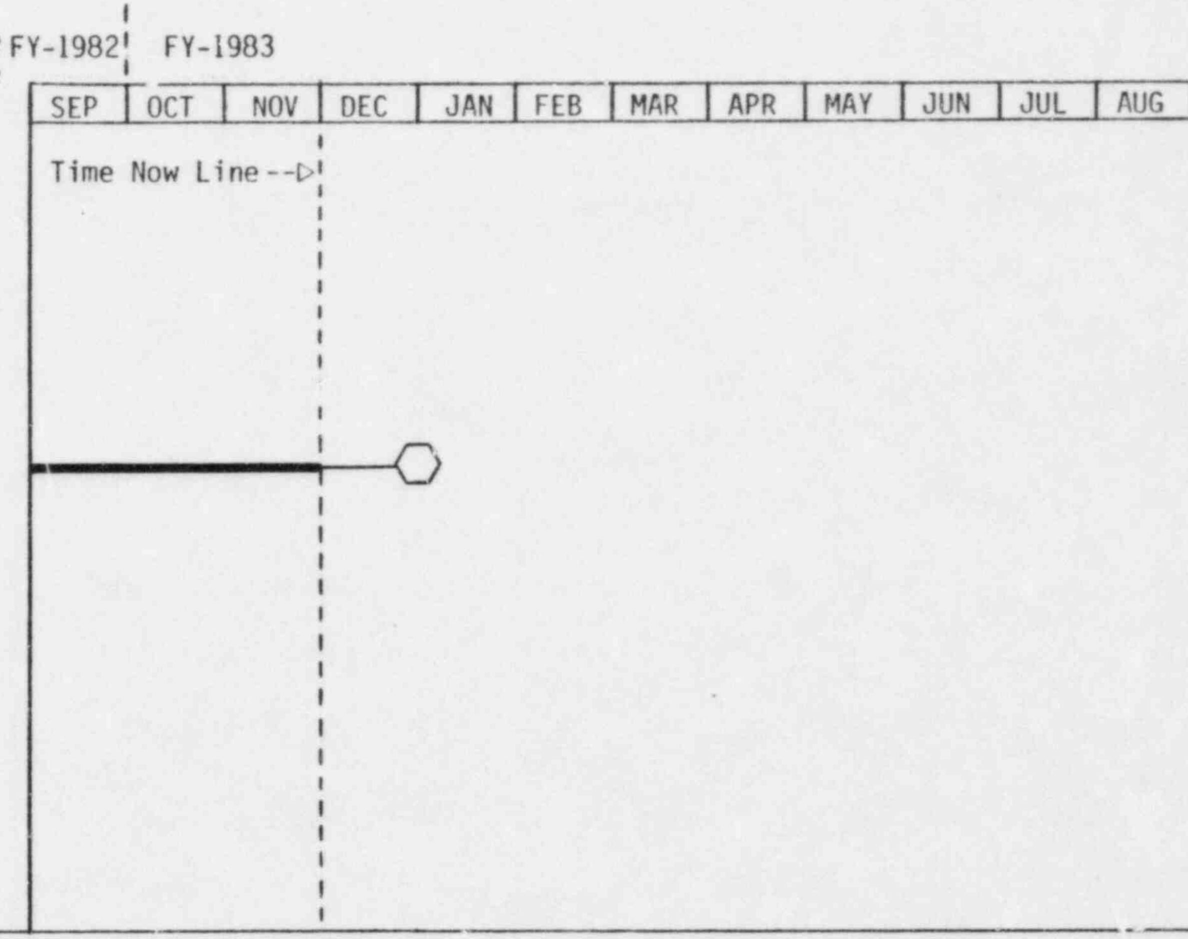
NRC TECHNICAL ASSISTANCE PROGRAM DIVISION November 1982

Kuo-Sheng (A6353)

LEGEND

- ● Completed Major Milestone
- ○ Scheduled Major Milestone
- ⊘ Changed Major Milestone
- ● Completed Secondary Milestone
- ○ Scheduled Secondary Milestone
- ⊘ Changed Secondary Milestone
- ♦ Actual Completion Date
- ◊ Scheduled Completion Date

Safety Relief Valve (SRV) Prediction



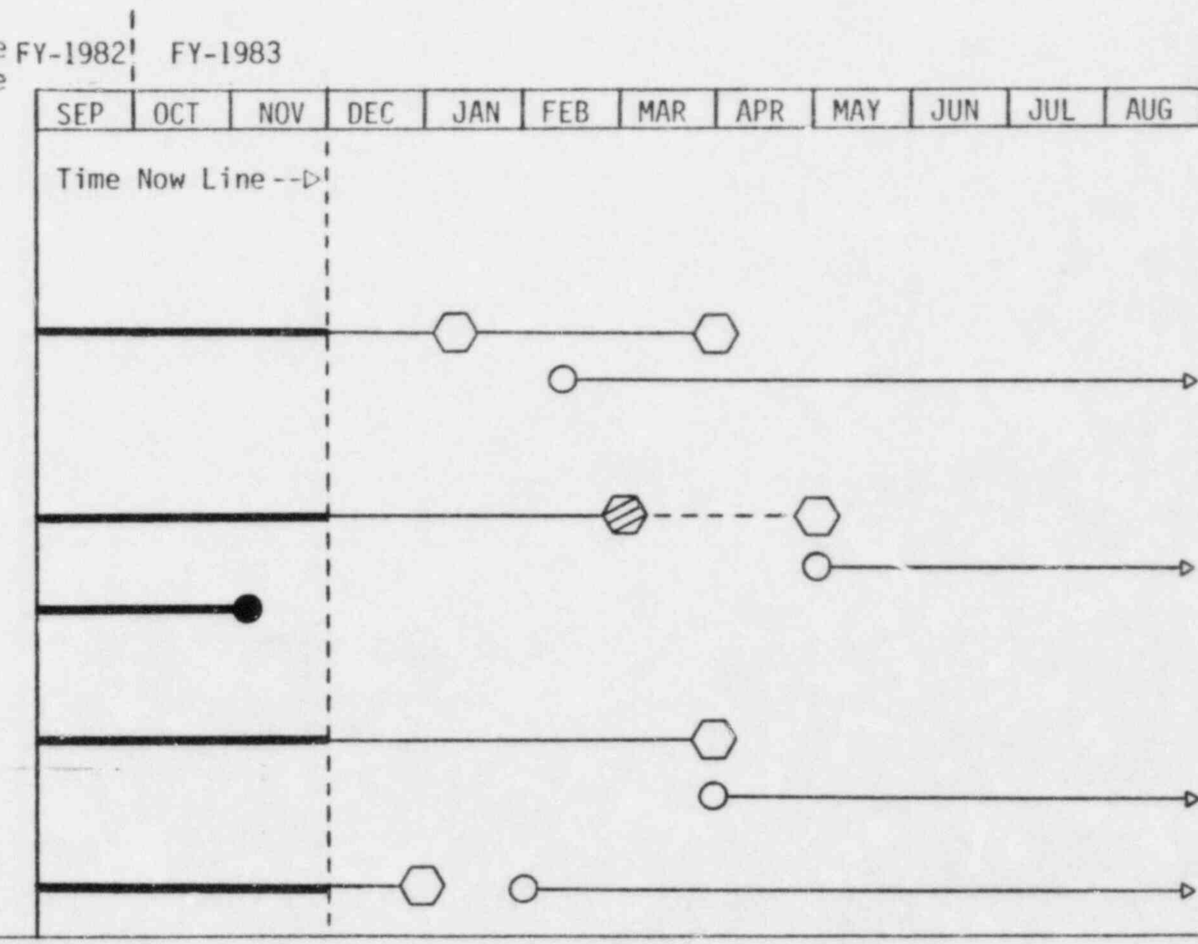
NOTES:

NRC TECHNICAL ASSISTANCE PROGRAM DIVISION November 1982

Severe Accident Sequence Analysis (A6354)

LEGEND

- ● Completed Major Milestone
- ○ Scheduled Major Milestone
- ⊘ Changed Major Milestone
- ● Completed Secondary Milestone
- ○ Scheduled Secondary Milestone
- ⊘ Changed Secondary Milestone
- ♦ Actual Completion Date
- ◊ Scheduled Completion Date



BWR

Browns Ferry IREP Analyses

Preparation

Analysis

PWR

CE Plant Analysis

Preparation

Analysis

CE RELAP5 (No PORV)

Containment Management Support (Bellefonte)

Preparation

Analysis

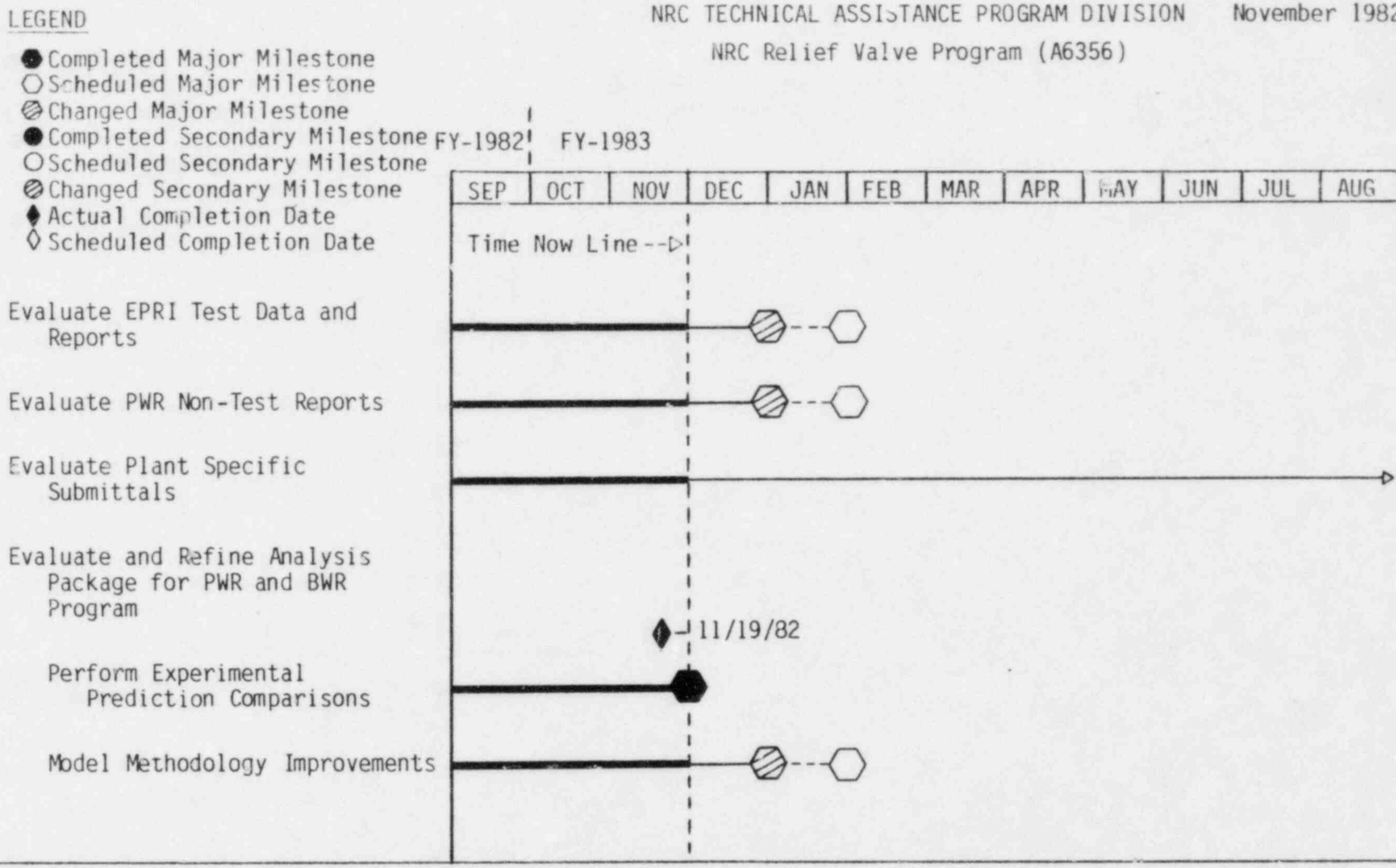
Degraded Core

SCDAP/MODO Hydrogen Studies

NOTES:

NRC TECHNICAL ASSISTANCE PROGRAM DIVISION November 1982

NRC Relief Valve Program (A6356)



NOTES:

NRC TECHNICAL ASSISTANCE PROGRAM DIVISION

TECHNICAL REVIEW AND SUMMARY

PROGRAM MANAGER'S

SUMMARY AND HIGHLIGHTS

A6039: The two FIST blowdown spool pieces were refurbished, calibrated and returned to GE for use in the first blowdown test.

A6316: The report "Parameters that Influence Damping in Nuclear Power Plant Piping Systems", NUREG/CR-3022, was issued.

A6317: The draft NUREG, A Bibliography of Data Bases for Nuclear Plant Risk Assessment, EGG-EA-6100, was transmitted to the NRC for review and comment.

A6356: Reviews of German Standard Problem 4 and an Intermountain Technology, Inc. (ITI) report on the EPRI safety and relief value testing were transmitted to the NRC.

A6039: INEL Technical Support to NRC for Industry, Cooperative Programs
EG&G Program/Technical Monitor: G. E. Wilson
DOE Technical Monitor: P. E. Litteneker
NRC Technical Monitor: W. D. Beckner

The objectives of this work are: To ensure the data from the industry experimental programs are adequate for assessment of thermal-hydraulic analysis models; to ensure the technical expertise available at the Idaho National Engineering Laboratory (INEL) and other national laboratories is transferred and used in the industry experimental programs, and to furnish on-call assistance to the Nuclear Regulatory Commission (NRC).

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

Boiling Water Reactor (BWR) Full Integral Simulation Test (FIST) Program:

The two FIST blowdown spool pieces have been refurbished and cold, single phase calibrated at Idaho National Engineering Laboratory (INEL). The spool pieces were returned to General Electric (GE) by November 14 so that they could be installed and operational for the mock blowdown testing.

The draft FIST Facility Description Report was reviewed and comments transmitted to the NRC (Saff-461-82).

The BWR-6 TRAC Deck has been modified for the power transient calculation and the FIST Deck is currently being modified. Steady state calculations should be complete within the week.

BWR Refill/Reflood (R/R) Program:

Data analysis of Single Heated Bundle (SHB) separate effects tests was continued.

Full Length Emergency Cooling Heat Transfer-System Effects and Separate Effects Tests (FLECHT-SEASET) Program:

Personnel attended the FLECHT-SEASET Program Management Group meeting in Pittsburgh (November 3, 4).

The blockage evaluation task was continued. Additional Flooding Experiment in Blocked Arrays (FEBA) data was put on the INEL data bank. These tests will be included in the data evaluation task.

2. Summary of Work Performed in November 1982 (Continued)

NRC Specified Tasks·

A schedule for the review of the NRC/EPRI/Westinghouse, MB1 Steam Generator Program Plan was developed and sent to DOE-ID and the NRC.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

BWR-FIST Program:

Work will continue on the following items: BWR-FIST TRAC Calculations, Automated Data Qualification software, Gamma Densitometers design, and supply of the second cooled thermocouple velocimeter.

BWR-R/R Program:

Analysis of SHB separate effects data will be completed.

FLECHT-SEASET Program:

The blockage and natural circulation data evaluation tasks will continue.

NRC Specified Tasks:

Review of the NRC/EPRI/Westinghouse, MB1 Steam Generator Program Plan will continue.

5. Problems and Potential Problems

BWR-FIST Program:

The execution of the FIST-BWR calculations is dependant upon receiving the power histories for each calculation from General Electric. Delay of the data beyond December 6 would, in all probability, delay the completion of the calculations.

A6046: Fuel Behavior Analysis Assessment
EG&G Program/Technical Monitor: E. T. Laats
DOE Technical Monitor: D. Majumdar
NRC Technical Monitor: G. P. Marino

The objectives of this program are to independently assess and evaluate the capabilities of the Nuclear Regulatory Commission (NRC) fuel rod behavior codes SCDAP, FRAP-T, and FRAPCON. To support these objectives, this program also maintains a base of experiment data.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

SCDAP/MODO Assessment

A set of seven calculations was performed with the SCDCOMP module of the SCDAP/MODO Code. The intent was to qualitatively assess the differences between responses to the same accident assumed to occur in the Three Mile Island-Unit 2 (TMI-2) Reactor and the Power Burst Facility (PBF). The results indicated that the overall trends calculated to occur in each reactor are identical, even though the rods used in PBF would be 1/4 as long as the TMI-2 rods.

Severe Fuel Damage Data Base

The initial literature search was completed and summarized in an internal letter. The data from about 100 references were noted and categorized for future use.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

The assessment activities will increase. A consolidation of all past calculations using SCDAP or any of its modules will be made. Also, a SCDAP/MODO input deck will be assembled to represent a commercial power plant.

5. Problems and Potential Problems

None.

A6047: LOCA Analysis Assessment and Applications
EG&G Program/Technical Monitor: T. R. Charlton (PWR)
R. R. Schultz (BWR Applications)
G. E. Wilson (BWR Assessment)
DOE Technical Monitor: D. Majumdar
NRC Technical Monitor: F. Odar

The objective of this work is to provide technical suport to the Nuclear Regulatory Commission (NRC) in the assessment and application of advanced thermal-hydraulic safety analysis codes. The assessment results serve to inform the scientific community of the relative capabilities, validity and range of applicability of the NRC developed codes. Application of the codes provide a technical basis for NRC evaluations of calculations performed by reactor vendors, utilities and others during the licensing process.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

The Boiling Water Reactor (BWR) Transient Reactor Analysis Code (TRAC-BD1) assessments with data from the BWR/4 and BWR/6 reference tests in the 30° Steam Sector Test Facility (SSTF) have continued. The BWR-6 transient calculations and draft report are approximately 75% complete. Model development in the BWR/4 task continues.

The Grand Gulf BWR/6 plant model (TRAC-BD1) development continued.

The plans for and data required for pressurized thermal shock (PTS) analyses of H. B. Robinson were presented to Carolina Power and Light.

Five additional sequences for Oconee PTS analysis were received and three of the sequences are being analyzed.

Preparation for a presentation to NRR and RES of the PTS work on Oconee and our plans for H. B. Robinson were completed.

3. Scheduled Milestones for December 1982

Description	Due Date	Actual Date
BWR/6, SSTF V12 Assessement	12/27/82	

4. Summary of Work to be Performed in December 1982

The SSTF, BWR/6 assessment of TRAC-BD1 will be completed. The SSTF, BWR/4 assessment will continue.

The development of the Grand Gulf BWR/6 TRAC-BD1 model will continue.

The following PTS analyses will be completed for the Oconee plant. (Refer to December 1, 1982 List of Oconee Transients).

(7a) Small break (2") with full high pressure injection (HPI)

(9) Maximum sustainable main feedwater flow for steam generator overfeed without pump trip from component protection features

(10) Hot standby, 4 turbine bypass valves fail open, feedwater system does not realign to aux header, decay heat 9 MW (one week downtime).

The presentation will be made to NRR and RES on the PTS status.

Work will begin on H. B. Robinson plant deck model if information is received from NRC.

5. Problems and Potential Problems

None.

A6102: NRC/DAE Data Bank
EG&G Program/Technical Monitor: E. T. Laats
DOE Technical Monitor: P. E. Litteneker
NRC Technical Monitor: M. W. Young

The objective of the Nuclear Regulatory Commission/Division of Accident Evaluation (NRC/DAE) Data Bank program is to provide a well controlled, documented repository for experiment data that supports the nuclear reactor safety industry. Toward this goal, the data base is continually being enlarged, assistance is provided to users in the form of training seminars and documentation, and the software employed by the Bank is continually upgraded.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

Data from three test facil ties were added to Data Bank. Data from Single Heated Bundle Faclity (SHBF) Tests 1006, 1028, 1030, 2333, 2332, 2315, 2334 and 2335 were added, as well as Semiscale Tests NC-10, SF-1, SF-2, SF-3, SF-4, and SF-5, along with FEBA Tests 221, 227, and 219.

An overview of the Data Bank program was presented at the Winter American Nuclear Society Meeting held in Washington, DC, on November 17, 1982.

Informal dicussions were held between the NRC, EG&G Idaho, and the Electric Power Research Institute (EPRI) to discuss plans for a joint NRC/EPRI data collection/dissemination program.

Data requests were processed from French, German, Swiss, and Japanese participants of he LOFT program.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

Work will continue on data entry, with continued emphasis being placed on data from the SHBF program.

Software development in the graphics area, bank documentation, and user request activities will also continue.

5. Problems and Potential Problems

None.

A6276: Licensee Event Report (LER) Failure Rate Analysis
EG&G Program/Technical Monitors: J. H. Linebarger/M. E. Stewart
DOE Technical Monitor: P. E. Litteneker
NRC Technical Monitor: R. C. Robinson

The objectives of this project are to summarize and evaluate nuclear power plant component failure data as reported in the LERs and to estimate component failure rates by using the summarized component failure data.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

Review of the draft report on instrumentation and control (I&C) systems led to the recommendation that the results be presented with a focus on plants, on components, and on type of event. Revisions were initiated.

A study of inverters continued.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

Final review comments on the draft I&C report will be incorporated and the report will be finalized.

The study of inverters will continue.

5. Problems and Potential Problems

None.

A6283: Common Cause Data Analysis
EG&G Program/Technical Monitors: J. H. Linebarger/N. D. Cox
DOE Technical Monitor: P. E. Litteneker
NRC Technical Monitor: L. E. Lancaster

The objective of this project is to develop and apply software that uses the Binomial Failure Rates (BFR) model to estimate common cause failure rates with tolerance bounds in support of risk assessment quantification.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

Licensee Event Reports (LERs) on Instrumentation and Controls were being updated. Techniques for displaying tolerance distributions are under development and testing.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

LER updating will be continued. Work on displaying tolerance distributions will also continue.

5. Problems and Potential Problems

None.

A6294: Plant Status Monitoring
EG&G Program/Technical Monitors: J. H. Linebarger/M. E. Stewart
DOE Technical Monitor: P. E. Litteneker
NRC Technical Monitor: M. L. Au

The objective of this project is to define the necessary and sufficient information needed by an operator to unambiguously know the status of the plant during accident conditions.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

A draft report of the final Diagnostic Algorithm document was completed, reviewed, and returned to the authors for revision.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

The Wood-Leaver report on the Emergency Procedure Guidelines (EPG) methodology and its application to the Zion plant will be submitted to the NRC for publication as a NUREG.

The final Diagnostic Algorithm document will be submitted for publication as a NUREG.

5. Problems and Potential Problems

None.

A6301: INEL Accident Sequence Evaluation Program (ASEP)
EG&G Program/Technical Monitors: J. H. Linebarger/W. H. Sullivan
DOE Technical Monitor: P. E. Litteneker
NRC Technical Monitor: P. Baranowsky

The objective of the project is to determine the generic light water reactor (LWR) accident sequences which will be used to investigate licensing issues.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

Completed draft report of ASEP workshop comments. This draft was sent to the Nuclear Regulatory Commission (NRC) Project Manager for review and comment.

Began gathering detailed probabilistic risk assessment (PRA) data for support of future ASEP activities.

Met with Sandia National Laboratory (SNL) Project Manager to develop new plans and schedules for future ASEP activites.

3. Scheduled Milestones for December 1982

Description	Due Date	Actual Date
INEL Report on ASEP Workshop results	12-10-82	

4. Summary of Work to be Performed in December 1982

Will continue to gather and analyze PRA data. Will incorporate any additional comments from the NRC Project Manager into the final report of ASEP Workshop comments and send these to workshop participants for their review.

5. Problems and Potential Problems

None.

A6306: Heiss Dampf Reaktor (HDR) Mechanical Component Response Analysis Testing
EG&G Program/Technical Monitors: B. L. Barnes/R. G. Rahl
DOE Technical Monitor: G. L. Vivian
NRC Technical Monitor: J. O'Brien

The NRC Office of Nuclear Regulatory Research, Division of Reactor Safety Research, has initiated a cooperative effort with the Federal Republic of Germany (FRG) in the Heissdampfreaktor (HDR) testing program to study the response of nuclear power plant systems subjected to ground excitation. The HDR is a decommissioned reactor being used for structural and hydraulic research. This project involves performing experimental impedance testing on the flood water storage tanks and the containment building and evaluation of the change in structural properties with level and type of excitation.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

A draft report of the results of tests performed on two test vessels at HDR was completed and sent out for review and comment.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

The final report on HDR tests will be published with all pertinent comments incorporated. Based on the results reported, a meeting with the NRC Technical Monitor will be held to review the work scope and make recommendations for follow-on work.

5. Problems and Potential Problems

None.

A6308: Display Design and Evaluation
EG&G Program/Technical Monitor: O. R. Meyer
DOE Technical Monitor: P. E. Litteneker
NRC Technical Monitor: J. P. Jenkins

The objective of this work is to provide data to the Nuclear Regulatory Commission (NRC) on evaluation methods and design criteria related to visual display in nuclear power plant control rooms. The data is to serve as a technical basis for NRC standards, guidelines and other regulatory activities.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

Work to upgrade the display experiment facilities is continuing. Test consoles have been selected and are under construction.

Simulator-based display evaluation is in progress.

Review of the checklist for CRT displays continued.

The reports on Response Trees and Predictor Displays are in editing.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

Work to upgrade the display experiment facilities will continue.

The Response Tree Display, operator aid and simulation specifications will be prepared.

Test subjects will continue to be utilized to gather data for the simulator-based display evaluation experiment.

5. Problems and Potential Problems

None.

A6310: Low Level Waste Risk Methodology Development
EG&G Program/Technical Monitors: J. H. Linebarger/N. D. Cox
DOE Technical Monitor: P. E. Litteneker
NRC Technical Monitor: T. J. McCartin

The objective of this project is to develop a low level waste risk assessment methodology to assess the performance of low level waste repositories and define appropriate criteria for low level waste site and design features.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

Operational debugging of the Shallow Land Burial (SLB) code was completed so that all subroutines are now operational. Verification of the ATMOS and DOSET subroutines was completed. Verification of the UNSAT and AQUIFR subroutines was continued. Statistics on many code parameters were gathered and analyzed, particularly with respect to uncertainty. Programming modifications for varying parameters in selected ways was started.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

Verification of ATMOS and AQUIFER will continue. Programming modifications for varying parameters will continue, and sensitivity studies of the atmospheric transport path will begin when these modifications progress far enough.

5. Problems and Potential Problems

None.

A6313: Initiating Event Data Evaluation
EG&G Program/Technical Monitors: J. H. Linebarger/M. E. Stewart
DOE Technical Monitor: P. E. Litteneker
NRC Technical Monitor: R. C. Robinson

The objective of this project is to develop initiating event frequencies for use in these and other probability risk assessments (PRAs).

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

The EPRI initiating event study (NP-2230) was evaluated from the standpoint of whether the plants included are representative of the industry (utilities, vendors, power ratings, architect engineers, and generic classes). In addition, a method was developed for identifying plants which are outliers, having either higher or lower overall occurrence rates.

The applicability of categorical data analysis to separately evaluate plant and aging effects in the data is being investigated.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

An effort to gain data on additional plants will be coupled with a verification of data in EPRI-NP-2230 from plants identified as outliers. Licensee Event Reports, Graybook data, and other sources will be used, with the first phase focusing on BWR plants.

5. Problems and Potential Problems

None.

A6315: Preliminary HTGR Siting Evaluation
EG&G Program/Technical Monitors: H. L. Magleby/H. J. Reilly
DOE Technical Monitor: P. E. Litteneker
NRC Technical Monitor: J. C. Glynn

The objective of this project is to identify and analyze accident sequences whose consequences envelope the consequences of all High Temperature Gas-Cooled Reactor (HTGR) sequences believed to be credible. This will allow evaluation by the Nuclear Regulatory Commission (NRC) of the possibility that the HTGR has significantly different siting characteristics than Light Water Reactors (LWRs). The resolution of which design (HTGR or LWR) presents a lower risk would be of significant benefit to policy makers in deciding whether the current pace of HTGR development should be changed.

The major task to be performed is to develop source terms by identifying and analyzing accident sequences for the 2240 MWt HTGR design whose associated consequences envelope the consequences of credible HTGR accident sequences. A second task is to evaluate the inherent susceptibility of the 2240 MWt HTGR to core damage accidents caused by "externally" initiated events including floods, seismic events and severe wind (tornados, hurricanes). Also, INEL will identify the major areas of and reasons for conservatism in the analysis, and will complete the preparation of the final report.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

A letter was sent to DOE-ID reporting the Denver meeting of October 20, and the revised project schedule.

A decision was made not to do the containment atmosphere response analysis at EG&G Idaho, inasmuch as NRC indicated that they did not wish to provide additional funding for the proposed additional scope. This decision was reported by letter.

Calculations by Brookhaven National Laboratory (BNL) of concrete degradation in the core heatup accident were reviewed and commented on by letter to BNL.

The partial draft main report was revised per NRC comments and forwarded to NRC.

A draft event tree discussion was prepared and forwarded to Gulf Atomic (GA) to assist them in reviewing the EG&G Idaho event trees.

2. Summary of Work Performed in November 1982 (Continued)

A letter was transmitted to NRC detailing questions regarding repair, recovery and operator action to be discussed at a meeting during the week of November 29.

Project personnel (H. Reilly) travelled to La Jolla, California, November 18 to meet with GA regarding their review and comments on project status and results to date.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

Project personnel (H. Reilly) will travel to Washington, DC, the week of November 29 to meet with NRC and BNL on various project problems.

Receive drafts of report sections from other laboratories (scheduled for December 15, 1982).

Complete a draft of appendix on event trees.

Revise draft appendices on susceptibility to fire, windstorms and floods.

Revise main text based on decisions made in meeting during the week of November 29.

Review drafts from other laboratories and request revisions as appropriate.

Start on summary of results by other laboratories.

5. Problems and Potential Problems

None.

A6316: Parameters Influencing Damping in Piping Systems
EG&G Program/Technical Monitors: B. L. Barnes/R. G. Rahl
DOE Technical Monitor: G. L. Vivian
NRC Technical Monitor: J. O'Brien

The objective of this program is to investigate the factors which influence damping in piping systems and provide guidelines for selecting damping values for ues in piping dynamic analyses. Experience and previous investigations have shown that the effects of piping supports are a dominant factor in apparent damping of piping system dynamics. Additionally, the use of higher damping values holds much promise for reduced numbers of seismic supports. This will both reduce system installation costs and also improve system operational reliability for frequent thermal transients.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

The report prepared at the end of FY-1982 was reissued as a formal report (NUREG/CR-3022), "Parameters That Influence Damping in Nuclear Power Plant Piping Systems." A draft test plan was written for EG&G Idaho pipe vibration tests to be conducted in FY-1983. Drafts of two technical papers for presentation at the 1983 ASME Pressure Vessel and Piping conference were written.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work Performed in December 1982

Planning and procurement for FY-1983 tests will be continued.

5. Problems and Potential Problems

None.

A6317: Data for NREP
EG&G Program/Technical Monitors: J. H. Linebarger/M. E. Stewart
DOE Technical Monitor: P. E. Litteneker
NRC Technical Monitor: J. W. Johnson

The objective of this project is to develop a generic reliability data base to be used in the National Reliability Evaluation Program (NREP).

1. Scheduled Milestones for November 1982

Description	Due Date	Actual Date
Draft NUREG of PRA Data Base	11-19-82	11-10-82C Saff-460-82

2. Summary of Work Performed in November 1982

The draft NUREG, A Bibliography of Data Bases for Nuclear Power Plant Risk Assessment (EGG-EA-6100), was transmitted to the NRC for review and comment.

Work for an engineering analysis of component failure modes was scoped and studies of pumps, valves, diesel generators, and instrumentation and control components were started.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

The project will be reoriented to be consistent with the decrease in FY-1983 funding authorized by the NRC.

5. Problems and Potential Problems

None.

A6318: System Requirements and Standards Development for Annealing of Reactor Pressure Vessels
EG&G Program/Technical Monitors: B. L. Barnes/W. L. Server
DOE Technical Monitor: G. L. Vivian
NRC Technical Monitor: A Taboada

Several commercial reactor pressure vessels (RPV's) now in service were manufactured using materials very sensitive to radiation exposure and are reaching a high degree of radiation embrittlement, i.e., nonconformance with current design lifetime requirements. To allow continued safe operation of these reactors, a thermal anneal cycle is under consideration to restore the fracture toughness properties of the RPVs back to an acceptable level.

The primary objectives of this work are to establish criteria for the development of standards to be applied to proposed in-situ thermal annealing procedures for commercial RPVs and to identify those technical areas which require additional research before such criteria can be established.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

A trip was taken by EG&G Idaho personnel to the Amercian Society of Mechanical Engineers Section XI Meeting in Williamsburg, Virginia. There is currently no activity within the Repairs and Replacements Subgroup with regard to the in-situ annealing issue. Involvement with the task group on reference toughness continued as related to application of the methodology to irradiated and annealed pressure vessel materials.

The proposed trip to Europe was postponed due to the tight time schedule. This trip will be rescheduled for the end of February or early March. Emphasis will be placed upon a recent request to visit Cooperheat in England, as well as the BR-3 reactor group in Belgium.

Work continued on the annual report which should be officially submitted next month. Further review of the Electric Power Research Institute/Westinghouse report on annealing continued with a review of the thermal analysis calculations.

The FY-1983 work scope is 9% complete and 9% of the FY-1983 funds have been expended.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

The NRC Technical Monitor will visit the Idaho National Engineering Laboratory to review the work to date. The annual report will be completed and submitted. Preparations for the American Society for Testing and Materials E10 Task Group Meeting (in January 1983) on in-situ annealing will be made.

5. Problems and Potential Problems

None.

A6322: Equipment Qualification Research Program (EQRP)
EG&G Program/Technical Monitor: J. A. Hunter
DOE Technical Monitor: P. E. Litteneker
NRC Technical Monitor: W. E. Campbell

The objective of the program is to provide an improved technical basis for the development of requirements and acceptance criteria for the dynamic (including seismic) and environmental qualification of mechanical equipment and dynamic qualification of electric equipment.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

Personnel from EG&G Idaho participated in a peer review meeting in Rockville, Maryland, for the purpose of reviewing the Southwest Research Institute draft report entitled "Evaluation of Methodology for Seismic Qualification of Nuclear Plant Electrical and Mechanical Equipment."

Work continued to develop a test specification for testing containment purge/vent valves. The test program will be directed at addressing research needs in the areas of extrapolation methodologies and severe accident qualification methodologies. The direction of the test program was changed to more strongly emphasize research equipment qualification needs and to reduce the emphasis of generating a valve performance data base. Consequently the test specification completion was slipped from November to December.

A meeting was held with NUTECH personnel in Idaho Falls to discuss specific issues and possible tasks to include in EQRP. The issues and tasks discussed addressed the area of criteria and methodology for seismic equipment qualification. The meeting was held since NUTECH offered to provide this informal input to the program.

Initial contact was also made with EPRI for the purpose of developing an interface with the Atomic Industrial Forum (AIF) involving equipment qualification (EQ) issues.

A presentation was prepared to present to NRC-NRR that describes the NRC-EQ programs being conducted at INEL. It is anticipated that NRR and RES personnel will participate in the meeting. The presentation will cover NRR and RES sponsored EQ programs.

4. Summary of Work to be Performed in December 1982

A meeting will be held with Energy Technology Engineering Center (ETEC) personnel to review test facilities and to discuss test requirements for containment purge/vent valve testing. In preparation for this meeting, activity will continued to develop a draft test specification for the subject testing.

It is also anticipated that a meeting will be held with ANCO Engineers to discuss their EQ capabilities that would support EQRP. If the situation warrants it, a meeting will also be held with NUTECH to discuss details of the meeting with them held in Idaho Falls in November.

It is planned to attend an AIF EQ meeting in Washington, DC, for the purpose of establishing contact between the EQRP and industry EQ programs

A presentation will be made to RES and NRR personnel in Washington to describe NRC sponsored EQ programs at INEL.

A steering group will be organized to assist in formulating rigorously defined tasks to be incorporated into the EQRP to address the issues associated with extrapolation methodologies, aging technology, qualification test/analysis methodologies, and qualification input characterization. The steering group will focus on defining specific tasks and a technical basis for including them in the EQRP. To assist in executing the working groups, draft EQRP work scopes in the above areas will be developed by discussion by the groups. The purpose of the group is to define specific work scopes for EQRP issue categories which presently are not rigorously defined.

5. Problems and Potential Problems

None.

A6326: Integrity of Containment Penetrations Under Severe Accident Load Conditions
EG&G Program/Technical Monitor: B. L. Barnes
DOE Technical Monitor: G. L. Vivian
NRC Technical Monitor: H. Ashar

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

Beyond answering questions regarding the EG&G Idaho proposal and if requested, making a presentation to the NRC on the EG&G Idaho proposal, no further work on this portion of this task is anticipated. The task as currently defined is 100% complete. This task will not appear in future monthly reports unless the project receives additional funding from the NRC.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

None.

5. Problems and Potential Problems

None.

A6331: Emergency Operating Procedure Guidelines
EG&G Program/Technical Monitors: J. H. Linebarger/M. E. Stewart
DOE Technical Monitor: P. E. Litteneker
NRC Technical Monitor: M. L. Au

The objective of this project is to determine whether emergency procedure guidelines (EPGs), when translated to plant specific procedures, provide unambiguous guidance to the operator under all risk-significant multiple failure accident conditions.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

The subcontract for Wood-Leaver and Associates to evaluate a GE plant using their Emergency Procedure Guidelines (EPG) methodology was approved.

Plans were made to begin the subcontract work the first week in December.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

Work will begin. The EG&G Idaho principals will meet with Wood-Leaver to arrange plans and a schedule for the tasks to be accomplished. Wood-Leaver training for the EG&G Idaho engineer involved in this project will commence.

5. Problems and Potential Problems

None.

A6353: Kuosheng Safety Relief Valve (SRV) Discharge and Piping Vibrational Tests
EG&G Program/Technical Monitors: B. L. Barnes/R. G. Rahl
DOE Technical Monitor: G. L. Vivian
NRC Technical Monitor: J. O'Brien

This task involves evaluation of structural dynamics impedance testing data obtained from the Kuosheng Nuclear Power Plant in Taiwan. Predictions of similar structural dynamic variables for other plants will be based upon the evaluations of the Kuosheng tests.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

Preparation of the report on the Kuosheng SRV discharge analysis and the report presenting experimental data from the Kuosheng Containment area was completed, and both reports are under final review.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

The two reports described above will be printed and issued.

5. Problems and Potential Problems

None.

A6354: Severe Accident Sequence Analysis Program (SASA)
EG&G Program/Technical Monitor: J. H. Linebarger
DOE Technical Monitor: P. E. Litteneker
NRC Technical Monitor: R. T. Curtis

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 containment testing, and to point out the need for additional computer code development and experimental data.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

Browns Ferry (BF) Analysis:

Addition of the reactor control systems to the BF model was completed and checkout runs were started. A CONTEMPT containment transient was calculated to check out the BF model and the results are currently being reviewed. Work to select the transients to be run using the BF model continued.

CESSAR 80 Analysis:

Review of the CESSAR 80 model received from Argonne National Laboratory was completed. Virtually the entire model must be constructed or modified. The necessity of obtaining proprietary data has placed the model development temporarily on hold.

Bellefonte Analysis:

Preparation of the RELAP5 input deck is continuing. All of the plant data initially needed has been received or mailed by Tennessee Valley Authority (TVA).

SCDAP Analysis:

A final report detailing the initial SCDAP calculations which were compared with those made by the MARCH code was written, typed, and is now being reviewed.

3. Scheduled Milestones for December 1982

Description	Due Date	Actual Date
Final Report SCDAP/MARCH Hydrogen Calculations	12-31-82	

4. Summary of Work to be Performed in December 1982

Browns Ferry (BF) Analysis:

Work will continue in all the areas mentioned in Item 2 above. The methods of using CONTEMPT with RELAP5 to calculate the influence of the containment on the reactor vessel during a transient will be reviewed and a specific method will be selected.

CESSAR 80 Analysis:

Work will start again if the information mentioned in Item 2 above is received.

Bellefonte Analysis:

Preparation of the RELAP5 input deck will continue. The projected completion goal is 75-80% by the end of December.

SCDAP Analysis:

The report mentioned in Item 2 above will be issued.

5. Problems and Potential Problems

None.

A6356: NRC Safety/Relief Valve Program
EG&G Program/Technical Monitor: J. A. Hunter
DOE Technical Monitor: P. E. Litteneker
NRC Technical Monitors: H. I. Gregg, F. C. Cherny

The Three Mile Island-2 (TMI-2) accident sequence included a failure of a power-operated relief valve to close. This, and other operating experience, raised a significant question about the performance qualification of primary system safety valves, relief valves, associated block valves and piping. As a result, the Nuclear Regulatory Commission (NRC) established requirements that performance verification be provided by full scale prototypical testing. The requirements were first identified in NUREG-0578 and have since been clarified in Sections II.D.1 and II.D.2 of NUREG-0660 and Item II.D.1 of NUREG-0737. The nuclear industry has established programs to provide for the required performance verification. EG&G Idaho is assisting the NRC in monitoring program system integration by monitoring the industry test programs to issues that licensing requirements of the NUREG documents are met. EG&G Idaho is also assisting by providing evaluation of the plant specific submittals to assure the applications of the test results to the specific plants are adequate.

1. Scheduled Milestones for November 1982

Description	Due Date	Actual Date
Perform Experimental Prediction Comparisons	11/30/82	11-19-82*C Saff-94-82 Saff-282-82 Saff-345-82

* Explanation given in Summary of Work Performed in November 1982 Task 4A.

2. Summary of Work Performed in November 1982

In response to the requirements of NUREG-0737, Item II.D.1.A that the utilities conduct performance tests to demonstrate the adequacy of the primary system safety and relief valves, the Pressurized Water Reactor (PWR) Utility Participants transmitted seven Electric Power Research Institute (EPRI) test program reports to the NRC by letter David P. Hoffman to Harold Denton dated September 30, 1982. EG&G Idaho is conducting a systematic review of these reports by having experts in the fields of mechanical design, safety analysis, operations, instrumentation, thermal-hydraulic and structures review the reports for adequacy in each of their specialties. Reports 1 through 5 are reports establishing the valve models, fluid conditions, pressures and flow rates used in the tests. The progress of the review of these five reports is included under Task 2 below. Report 6

2. Summary of Work Performed in November 1982 (Continued)

is a summary report of the test results and the progress of the review is included under Task 1 below. Report 7 presents comparisons of RELAP5 calculations with representative tests and the progress of the review is included under Task 4A below.

EG&G Idaho is also conducting a similar review of the three detailed test reports. The reports are:

EPRI/C-E Safety Valve Test Report July 1982 (10 volumes)
EPRI/Wyle Power Operated Relief Valve Phase III Test Report March 1982 (11 volumes)
EPRI PWR Safety and Relief Valve Test Program PORV Block Valve Information Package May 1982.

The progress of these reviews are included under Task 1 below. A review is also being conducted on the report Review of Pressurized Safety Valve Performance as observed in the EPRI Safety and Relief Valve Test Programs WCAP-10105, which was submitted to the NRC by the Westinghouse Owners Group. The progress of this review is included under Task 2 below.

Task 1: Evaluate EPRI Test Data and Reports

The evaluation of the Safety and Relief Valve Test Report, Report 6 above, continued. Comments from the reviewers are expected in early December for safety related items and by late December for other items.

Review of the EPRI/CE and EPRI/Wyle detailed test reports was initiated.

The review of the EPRI PWR Block Valve Information Package has been completed and comments have been prepared by the reviewers. Work is continuing on combining the comments to provide a final report.

Task 2: Evaluate PWR Non-Test Reports

The evaluation of the valve selection report, the three valve inlet fluid condition reports and the test conditions justification report, reports 1 through 5 above is in progress. Comments are expected from the reviewers by late December.

The thermal-hydraulic review of WCAP-10105 has been completed and comments formulated.

Task 3: Evaluate Plant Specific Submittals

A draft safety evaluation report was prepared for the San Onofre 2 and 3 PWR submittals. A revision will be prepared incorporating the results of the test report reviews, Tasks 1 and 2 above, when these reviews are completed.

2. Summary of Work Performed in November 1982 (Continued)

Task 4: Evaluate and Refine Analysis Package for PWR and BWR Programs

a. Perform Experimental Prediction

This task is considered complete. Comparisons were reported for the German Standard Problem 4 in reports transmitted by Saff-94-82 and Saff-282-82. Comparisons with EPRI safety and relief valve test were reported in the Intermountain Technologies, Inc., (ITI) reports, report 7 above. The EG&G Idaho review of the report was transmitted by Saff-345-82. The evaluation of the review has indicated that, with the instrumentation used in the tests, exact comparison of each parameter preceding and during the transient is not possible. The comparisons reported by ITI are presently considered to be as good as can be done; therefore, no further analyses are planned at this time. If the reviews of the detailed test reports indicate additional comparisons would be beneficial, a new task will be established.

b. Model Methodology Improvements

A study evaluating the EPRI/ITI recommendation as to the number of volume nodes necessary to represent a piping leg in RELAP5 to obtain appropriate values of the hydraulic loads was completed. A task to generate a consistent set of guidelines for application of RELAP5 to plant system analysis and a study applying the guidelines to a plant system was continued. A preliminary report of the guidelines was completed.

A RELAP5 calculation was performed to determine whether a steam or air environment in the piping downstream of a safety/relief valve would result in the larger hydraulic force. This calculation was compared with a hand calculation to help provide a basis for guideline selection.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

Task 1: Evaluate EPRI Test Data and Reports

Evaluations of the Safety and Relief Valve Test Report, the EPRI/CE and EPRI/Wyle detailed test reports, and preparation of a final EPRI PWR Block Valve report will be continued.

4. Summary of Work to be Performed in December 1982 (Continued)

Task 2: Evaluate PWR Non-Test Reports

Evaluations of the valve selection report, the three vendor inlet fluid conditions reports and the test conditions justification report will be continued.

Task 3: Evaluate Plant Specific Submittals

Information necessary to do an audit calculation for San Onofre 2 and 3 will be identified.

Task 4: Evaluate and Refine Analysis Package for PWR and BWR Programs

a. Perform Experimental Prediction Comparison

This task has been complete.

b. Model Methodology Improvements

A letter report for the study determining the number of nodes necessary to represent a piping leg in RELAP5 to obtain appropriate values of the hydraulic loads will be drafted.

The draft guidelines for application of RELAP5 to plant system analysis will be updated based on report evaluations and additional analysis results. A study applying the RELAP5 guidelines to the Summer PWR plant will continue.

Preparation for a meeting with EPRI and NRC personnel to discuss safety/relief valve test results and an evaluation of RELAP5 conducted by ITI/EPRI will be continued.

5. Problems and Potential Problems

None.

A6358: Applied James-Stein Estimators
EG&G Program/Technical Monitors: J. H. Linebarger/N. D. Cox
DOE Technical Monitor: P. E. Litteneker
NRC Technical Monitor: L. E. Lancaster

The objective of this project is to explore James-Stein techniques for pooling data in component failure rate calculations to see if they offer advantages over maximum likelihood techniques.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

Ideas for confidence interval methods were conceived, and theoretical derivations were developed. Simulations of data were continued in order to examine the relative savings with the several James-Stein type estimators.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

The tasks underway in November will continue into December. The programming of one confidence interval method will be started.

5. Problems and Potential Problems

None.

A6367: Support of NRC on ASME Code Section XI Activities
EG&G Program/Technical Monitor: B. L. Barnes
DOE Technical Monitor: G. L. Vivian
NRC Technical Monitor: E. Baker

The objective of this work is to provide technical assistance to the Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research relative to review of the American Society of Mechanical Engineers (ASME) Code Documents, Code Addenda, and Code Cases. Frequently, issues arise relative to Section XI of the ASME Code where the NRC staff involved perceive a need for additional data or evaluation before establishing a staff position. These issues range from the need for data on the number of pipe supports to be exempted by certain code provisions to the reasonable and prudent limits of valve leakage allowable in a nuclear power plant.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

Task 4--Report on Allowable Valve Leakage: EG&G Idaho personnel were notified that NRC review comments on the draft report are complete. The NRC Technical Monitor will add several man-days to the work scope to provide for the re-review of a related document which has been revised since the Task 4 report was originally written.

Task 6--Review of Valve Testing Standards: The preliminary report is being revised, based on comments from the NRC.

Task 7--Review of Supports Examination and Testing Standards: There was no activity on this task during November.

Task 9: Evaluation of the Basis for Section XI Flaw Acceptance Standards: The American Society of Mechanical Engineers (ASME) Section XI Meeting in Williamsburg, Virginia was attended. Special emphasis was directed towards a new, non-mandatory appendix on evaluation of flaws in austenitic steel piping components. EG&G Idaho personnel are involved with this new appendix through task group and working group activities.

Comments were received on the proposed new work scope from S. Tagart of the Electric Power Research Institute (EPRI) and W. Cullen of Materials Engineering Associates. Mr. Tagart's comments were fairly general and related to overall philosophy; Mr. Cullen had no specific technical comments. Additional comments from Mr. W. O'Donnell are expected.

2. Summary of Work Performed in November 1982 (Continued)

The thermal analysis of the thirteen important transients is almost complete. The stress analysis personnel at EG&G Idaho are ready to begin their evaluation as soon as the through-wall temperature gradients are stored on the computer by the thermal analysis personnel.

Task 10--Evaluation of the Impact of Proposed Exemptions of Supports for Section XI: --The abstract, summary, conclusions, and revised PWR data from the final report were presented to the American Society for Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Section XI, Working Group on Component Supports at their November meeting in Williamsburg, Virginia. The final report was officially transmitted and distributed to the membership of the ASME Code Section XI Subcommittee, Subgroup on Water Cooled Systems, Subgroup on General Requirements, and Working Group on Component Supports. This completed all Task 10 work.

Task 12. An EG&G Idaho technical person participated in ASME Code Section XI meetings of the Subcommittee, Subgroup on Water Cooled Systems, Working Group on Nondestructive Examination, Working Group on Component Supports, and the Working Group on Inspection of Class 2 Systems.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

Task 4: Review comments will be incorporated and the new work scope will be completed.

Task 6: Revision of the draft report will continue.

Task 7: No effort is planned until NRC comments are received on the preliminary draft report.

Task 9: An official transmittal of the report describing the new proposed work will be made, once comments are received from Mr. W. O'Donnell. The stress analysis of the operating transients should be completed by the end of December or early January. The computer codes (subroutines) for calculating stress intensity factors should be running on the EG&G system in December if the code is received from EPRI within about a week.

Task 10: This task is complete and will not be included in future reports.

Task 12: No activity is planned until the next meeting in February 1983.

5. <u>Problems and Potential Problems</u>

None.

A6369: Nuclear Power Plant Instrumentation Evaluation
EG&G Program/Technical Monitors: E. W. Roberts/J. A. Rose
DOE Technical Monitor: P. E. Litteneker
NRC Technical Monitor: R. Feit

The general objectives of this program are threefold; (a) to identify problems facing the nuclear industry in meeting the intent of Regulatory Guide (RG) 1.97, Revision 2, with regard to measurement range, accuracy, response time and equipment qualification, (b) to find practical cost effective solutions to those problems and (c) to examine the guide itself to determine adequacy of the current version and to recommend changes as appropriate.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

Computer input of additional plant data, including that from General Electric (GE) and Combustion Engineering (CE), was completed.

Work was started to write an interim report assessing the current status of plant systems being used to meet the intent of RG 1.97 and to identify areas where design or qualification problems exist.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

Work will start to generate a letter report detailing the specific tasks to be accomplished by this program in the near term.

Work on the interim assessment report will continue.

Testing of Three Mile Island-type thermocouples (TCs) will be initiated. The purpose of the testing is to answer questions relating to (a) probable accuracy of output in accident conditions, (b) possible failure modes and, (c) applicability of existing TC codes to commercial TCs during accident conditions.

5. Problems and Potential Problems

None.

A6370: Microprocessor Based Design and Plant Control Automation
EG&G Program/Technical Monitors: E. W. Roberts/D. M. Adams
DOE Technical Monitor: P. E. Litteneker
NRC Technical Monitor: D. W. Boehm

This research project is concerned with the potential safety issues associated with programmable, digital, computer-based nuclear plant control and protection systems and with the adequacy of isolation of isolation methods in nuclear power plants.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

EG&G Idaho has written the requisitions for the Phase I isolators and the test equipment. Isolators are being purchased sole source with a requested delivery date of February 15, 1983. Bids for the test equipment are requested by December 15, 1982.

EG&G Idaho is preparing a test plan for each Phase I isolator. These test plans will be reviewed with the NRC at a future date and prior to testing.

EG&G Idaho has received the Nuclear Regulatory Commission (NRC) comments on the draft report "Preliminary Assessment of Design Issues Related to the Use of Programmable Digital Devices for Safety and Control Systems". Corrections to the draft have been made and the final report will be issued in the near future.

The program is now fully staffed and a budget has been prepared for FY-1983.

The comparative risk assessment task and the interim criteria for digital systems task are both underway.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

The majority of the work for December will deal with the development of the interim criteria for digital systems and the comparative risk assessment task. The Task I test plans will be finalized and discussed with the NRC. Pending receipt of the new statement of work, a new Form 189 will be prepared.

5. Problems and Potential Problems

None, pending the new statement of work.

A6371: Technical Assistance Contract for Evaluation of and Guidance for Radiological Air Sampling
EG&G Program/Technical Monitor: B. L. Rich
DOE Technical Monitor: Pete J. Dirkmaat
NRC Technical Monitor: Alan Roecklein

The objectives of this work are to: Survey current sampling techniques, equipment and plant conditions, test air sampling/monitoring equipment and evaluate current sampling methods and recommend preferred methods.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

Completed characterization of aerosol concentration distribution in the test chamber.

Completed setup and performance tests of the fluidized bed aerosol generator.

Completed first draft of the PAS-NUREG.

Completed testing of wall losses in a typical model of air sampling filter cassette.

3. Scheduled Milestones for December 1982

None.

4. Summary of Work to be Performed in December 1982

The PAS NUREG draft will be reviewed.

Plan and begin surveys of NRC licensee industries which have not yet been covered by site visits.

Plan aerosol release/dispersal studies at typical nuclear facilities on the Idaho National Engineering Laboratory site.

5. Problems and Potential Problems

None.

A6376: Two Phase Instrumentation Evaluation
EG&G Program/Technical Monitors: E. W. Roberts/G. D. Lassahn
DOE Technical Monitor: P. E. Litteneker
NRC Technical Monitor: N. Kondic

The goal of this project is to perform research to evaluate/test instruments/methods for the measurement of parameters which characterize two phase phenomena during normal and accident conditions primarily in the primary system of Pressurized Water Reactors (PWRs). Additionally, this project suggests the testing or investigation of instruments/methods to measure low velocity fluid flow rates, voiding in the steam generator U-tubes, and methods to estimate the location and size of a break in the primary system piping.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

Continued literature search and writing test and evaluation procedures. Started to determine methods to initiate testing proposed instruments/methods in existing test facilities pending NRC approval of instruments selected for further testing. Reviewed currently available instruments for adaptability to distinguish the presence of two-phase conditions in the circulation pump and the steam generator U-tubes. The findings will be discussed in the interim report to be completed in December 1982.

3. Scheduled Milestones for December 1982

Description	Due Date	Actual Date
Interim Report on Required Instruments	12-21-82	

4. Summary of Work to be Performed in December 1982

The identification and evaluation of available or adaptable instrumentation already existing in the Nuclear Steam Supply System (NSSS) and candidate methods for detecting two-phase conditions in the regions of interest has been completed and will be discussed in the interim report to be completed in December. This report will identify and evaluate the capability of instrumentation available or adaptable within the NSSS to measure low velocity fluid flow rates and select and evaluate candidate methods.

Instruments and candidate methods existing or adaptable within the NSSS will be evaluated to determine/estimate break size and location in the Reactor Coolant System (RCS).

4. Summary of Work to be Performed in December 1982 (Continued)

Instruments will be determined for further study using proven physical principles most suitable to be implemented. Design and performance requirements will be recommended for new instruments to be tested to determine their adequacy.

5. Problems and Potential Problems

None.

A6380: Diagnostic Instrumentation Evaluation
EG&G Program/Technical Monitors: E. W. Roberts/G. D. Lassahn
DOE Technical Monitor: P. E. Litteneker
NRC Technical Monitor: N. Kondic

The goals of this project are to identify anticipatory measurements, which are useful in predicting accidents in nuclear power plants; to evaluate the instrumentation available for these measurements; and to recommend fruitful areas of research to develop new measurement techniques for anticipatory measurements.

1. Scheduled Milestones for November 1982

None.

2. Summary of Work Performed in November 1982

Reliability and Statistics Branch personnel have completed independent analyses and have reviewed FSARs to supplement the event trees (see FY-1982 year-end report) as sources of information on possible anticipatory measurements. These analyses also give semi-quantitative information on the importance of the several types of anticipatory measurements. The quarterly report due in December is in progress.

3. Scheduled Milestones for December 1982

Description	Due Date	Actual Date
Quarterly Report	12/21/82	

4. Summary of Work to be Performed in December 1982

The evaluations of the potential anticipatory measurements will be completed and reported.

5. Problems and Potential Problems

None.

MONTHLY REPORT FOR

NOVEMBER 1982

GPP AND LINE ITEMS

R E Rice

R. E. Rice, Manager
Facilities Management Division

R.L.D. Hess

R. L. D. Hess
Planning and Budgets Division

EG&G IDAHO, INC.

GPP ITEM

FY-1983

($000)

PROGRAM WATER REACTOR RESEARCH TEST FACILITIES DIVISION

189 No. A6038

MANAGER P. North

Task Initiated ○
Task Completed △

EA No.	Item Description	Original PA Amount	Current Estimated Cost	Project To Date Costs	O	N	D	J	F	M	A	M	J	J	A	S
93520	WRRTF Water Well Upgrade	$ 125	$ 80	EG&G $ 30.5 M-K $ 44.0												