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SRTC
Savannah River Technology Center

WFO 94-002
APPENDIX A

WSRC-RP-93-1521

EMERGENCY PREPAREDNESS TECHNOLOGY SUPPORT TO THE HEALTH & SAFETY EXECUTIVE (HSE), NUCLEAR INSTALLATIONS INSPECTORATE (NII) OF THE UNITED KINGDOM (U)

A Work For Others Proposal to

Science Applications International Corporation

To Support Methodology Request From The United Kingdom Health & Safety Executive's Nuclear Installations Inspectorate

UNCLASSIFIED
DOES NOT CONTAIN
UNCLASSIFIED CONTROLLED
NUCLEAR INFORMATION

ADC & Reviewing Official: K. R. O'Kula
Kevin R. O'Kula, Manager
Data & Consequence Analysis Group
Process Safety Technology Section

Date: 3-3-94

Westinghouse Savannah River Company
Savannah River Site
Aiken, South Carolina 29808

SAFETY • RESPONSIBILITY • SECURITY
SRS
SAVANNAH RIVER SITE

PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT NO. DE-AC09-89SR18035

DISCLAIMER

APPROVAL PAGE

PROJECT: UK NII Emergency Preparedness Technology Proposal

DOCUMENT: WSRC-RP-93-1521

TITLE: Emergency Preparedness Technology Support to the Health & Safety Executive (HSE), Nuclear Installations Inspectorate (NII) of the United Kingdom (U)

PREPARED BY:

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Waste & Environmental Safety Technology Section

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Safety Technology Department

HSE

TABLE OF CONTENTS

LIST OF ACRONYMS AND KEY DEFINITIONS

ANS	American Nuclear Society
APET	Accident Progression Event Tree
Contract data	technical data first produced in the performance of subcontract, technical data which are specified to be delivered under the subcontract, ..., or technical data actually delivered in connection with the subcontract
DBA	Design Basis Accident
DOE	Department of Energy, United States
EAL	Emergency Action Level
FTE	Full-Time Equivalent
HSE	Health and Safety Executive, United Kingdom
KDAM	Knowledge-Based Decision Analysis Methodology
MELCOR	Severe Accident modeling code produced by Sandia National Laboratories
NII	Nuclear Installations Inspectorate
NRPB	National Radiological Protection Board, United Kingdom
SAIC	Science Applications International Corporation
SRS	Savannah River Site
SRTC	Savannah River Technology Center
STD	Safety Technology Department
Technical data	recorded information regardless of form or characteristic, of a scientific or technical nature. It may, for example, document research, experimental, developmental, or demonstration, or engineering work, or be usable or used to define a design or process, or to procure, produce, support, maintain, or operate material. The data may be ... computer software (including computer programs, computer software data bases, and computer software documentation).
UK	United Kingdom
WFO	Work For Others, Defined in DOE Order 4300.2B, Non-Department of Energy Funded Work (Work For Others), Change 2: 2-7-92
WSRC	Westinghouse Savannah River Company

EXECUTIVE SUMMARY

Presentations made at the April, 1993 ANS Emergency Preparedness Topical meeting described a methodology for assisting emergency response personnel during a nuclear facility accident. The talks were heard by representatives of the Nuclear Installations Inspectorate (NII) of the United Kingdom (UK). The NII raised the possibility of preparing a similar methodology for British commercial plants. The approach uses an accident progression logic model method developed by Westinghouse Savannah River Company (WSRC) and Science Applications International Corporation (SAIC) for K Reactor to predict the magnitude and timing of radioactivity releases (the source term) based on an advanced logic model methodology. Predicted releases are output from the personal computer-based model in a level-of-confidence format. Additional technical discussions eventually led to a request from the NII to develop a proposal for assembling a similar technology to predict source terms for the UK's advanced gas-cooled reactor (AGR) type.

To respond to this request, WSRC is submitting a proposal to provide contractual assistance as specified in the Scope of Work. The work will produce, document, and transfer technology associated with a Decision-Oriented Source Term Estimator for Emergency Preparedness (DOSE-EP) for the NII to apply to AGRs in the United Kingdom. The prototype will incorporate inputs from plant instrumentation and the emergency response analyst(s) to forecast the source term. The project will provide the technical basis for the NII to advise the National Radiological Protection Board of the UK on impending releases from British commercial plants under accident conditions. In return, WSRC will preserve reactor core competence in several key areas, transfer technology to help ensure greater overall nuclear safety in a global context, and develop in-house capabilities to extend the approach to onsite facilities.

Safety analysts from both WSRC and SAIC will be coordinated during the project, with SAIC-Albuquerque providing overall management. The work is to be performed in two parts, or modules:

Module	**1.**	**Feasibility and Technical Baseline Development (3 months)**
	1.a	Feasibility of Technology Application to AGRs
	1.b	Inputs & Technical Baseline for Model Development
Module	**2.**	**Source Term Predictor Model Development & Technology Transfer (~ 1 calendar year)**

Module 1.a will be performed partially at the offices of the NII in the United Kingdom. The balance of Module 1.a and all of Module 1.b will be conducted at WSRC Aiken, SC offices and at SAIC in Albuquerque, NM. The scope of work

and level of detail for Module 2 will be developed in detail during Module 1.b. At minimum, it is expected to last approximately one calendar year.

This response to the request for technical support offers the following for consideration regarding WSRC in performing this work:

- Qualification of Personnel and Resource Commitment
- Proposed Work
- Schedule of Milestones
- Job Cost Estimate for Work For Others
- Payment Schedule.

WSRC will provide approximately half of the engineering effort over the course of the program, integrated over Modules 1 and 2.

It must be noted that although either WSRC or SAIC could alone fulfill this project, the revised, considerably longer schedule for all requisite activities would preclude timely completion relative to the funding cycle of the customer, the NII.

QUALIFICATION OF PERSONNEL AND RESOURCE COMMITMENT

Personnel

Qualified personnel from both the Savannah River Technology Center's Safety Technology Department (STD) and Science Applications International Corporation's (SAIC's) Albuquerque office will perform the work described under "Proposed Work". The lead contractor for this effort shall be WSRC in the management of the project. Personnel from both organizations shall be integrated in the project to ensure timely completion of schedule and specified milestones. The work is divided into a short-term (three months) Module 1, and a longer-term Module 2 and is described in the next section.

The WSRC professionals supporting this project include, but are not limited to:

Module 1

- D. S. Cramer (Fellow Engineer) Data, Systems Analysis
- D. A. Kalinich (Engineer) Accident Phenomenology; APET Methods

Module 2

- D. S. Cramer (Fellow Engineer) Data, Systems Analysis
- R. P. Taylor, Jr. (Senior Adm. Engineer) Systems Analysis, APET
- D. A. Kalinich (Engineer) Accident Phenomenology; APET Methods
- D. Allison (Senior Engineer) Severe Accident Phenomenology.

K. R. O'Kula of the SRTC Safety Technology Department will be the overall project manager for this activity, and will coordinate Module 1 and 2 activities for WSRC. M. Leonard of SAIC will serve as the overall program coordinator.

Period of Performance

Upon initiation of the WFO contract, activities discussed in the Proposed Work section of this document are expected to continue throughout a combined total of four months for Module 1 (Module 1.a - ~1 month, Modules 1.b - 1.e ~3 months). Module 2 work scope must still be finalized. Its duration is expected to be about one calendar year.

PROPOSED WORK

Background For Proposed Program

Technical papers presented at the April, 1993 Emergency Preparedness Topical meeting described a methodology for assisting emergency response personnel during a nuclear facility accident.[1-3] The presentations were heard by representatives of the United Kingdom Nuclear Installations Inspectorate (NII). The NII subsequently discussed with SRTC the possibility of preparing a similar predictive methodology for estimating radioactivity releases from British commercial power reactors. The approach uses an accident progression logic model methodology developed by WSRC and SAIC during the K Reactor safety upgrades to forecast the radiological source term in a level-of-confidence format. The overall methodology has been labeled, *Decision-Oriented Source-term Estimator for Emergency Preparedness (DOSE-EP)*.[4] Additional technical discussions ensued through the summer and early fall, among NII, WSRC, and SAIC parties leading to a verbal agreement that a joint work plan be developed supporting the NII in this area.

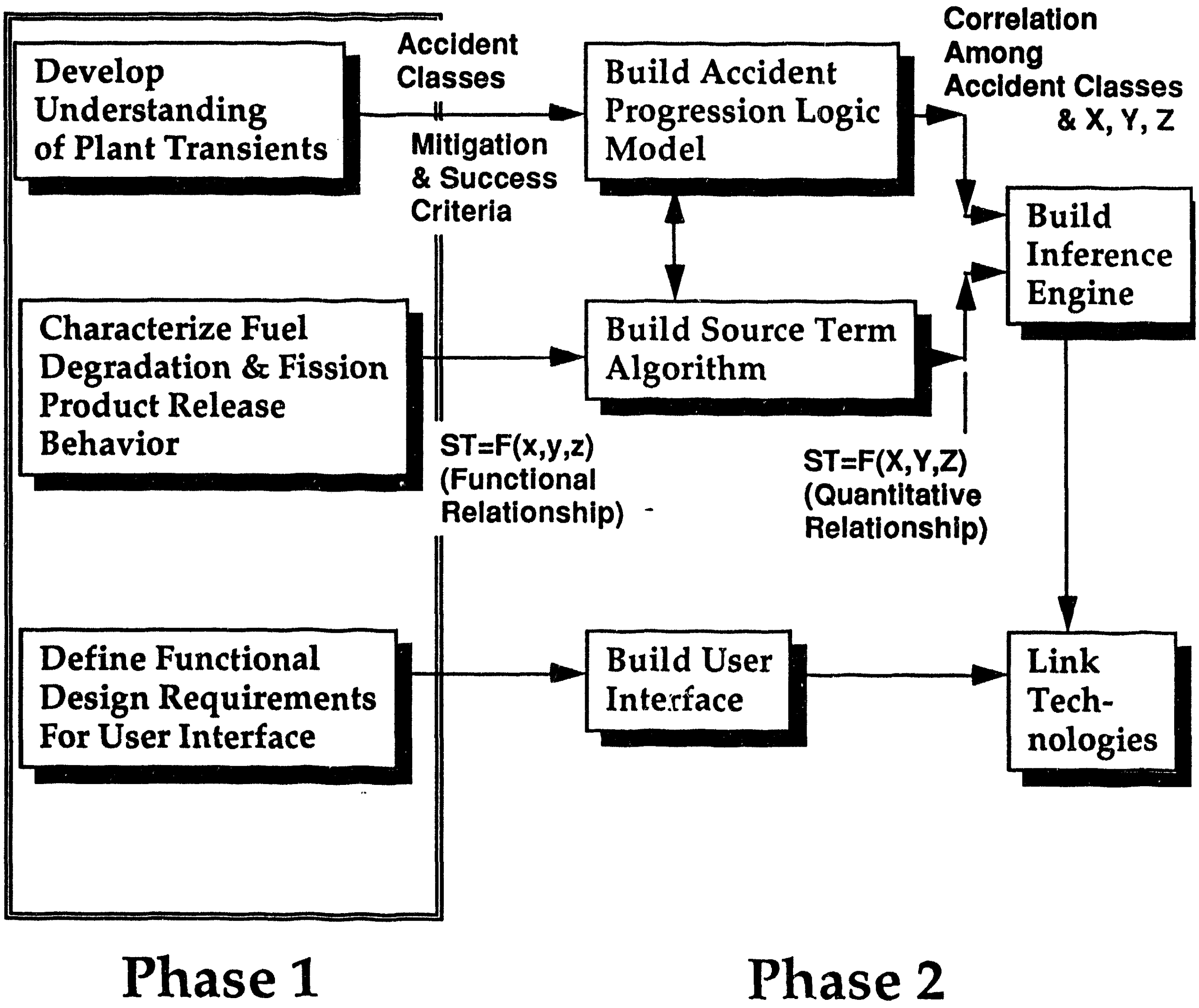
To help plan the upcoming work, SAIC-Albuquerque met with the NII in the UK late this summer. The substance of these discussions was relayed to Westinghouse Savannah River Company during planning meetings in September, 1993 and it was proposed that a joint work scope for the British be arranged to produce a DOSE-EP prototype. The NII approved the coordinated arrangement of WSRC and SAIC and asked that a scope of work be developed. The requested scope has been finalized and submitted to the NII. The current document details the support planned using WSRC resources, and applies an improved WSRC/SAIC emergency preparedness technology to help the NII meet its regulatory & monitoring requirements for the UK Advanced Gas-Cooled Reactor (AGR) class.

Application of the Technology

A Scope of Work has been prepared based on inquiries made by the NII (ref. file NUC 40/1/05, Attachment A), telecons among WSRC (O'Kula), NII (Whitehead), and SAIC-Albuquerque (Leonard), and working-level discussions held in Aiken, SC among all parties during the week of November 8 - 11, 1993 (Attachment B). The program described below integrates available expertise from the Safety Technology Department of the Savannah River Technology Center with the accident progression and safety analysts from the Science Applications International Corporation's (SAIC's) Albuquerque, NM office. However, only the WSRC activities are costed in this proposal.

The work is scheduled to be conducted in two modules and is based on the process flow path represented in Figure 1. To ensure completion of the overall projection without undue contractual interruption, SAIC shall serve as the

Figure 1: Work For Others: Source Term Predictor Technology Process Flow Path



prime contractor to the NII in the two modules: (1) Feasibility and Technical Baseline Module; and (2) Prototype Development Module.

Module 1 of the model development program consists of a literature review and feasibility study, along with preliminary conceptual model development. Module 2 follows with detailed quantitative model development and integration of the model with the WINDOWS™ - based interface.

Module 1. Feasibility and Technical Baseline Development

This work will gather information about UK advanced gas-cooled reactors (AGRs) from current safety documentation (e.g. case studies). A feasibility memorandum will be generated discussing the likelihood that the overall program and prototype development can be completed as anticipated. It will also identify problem areas or information "gaps" that could turn into critical path items as the program progresses. Secondly, the team would establish requisite blocks of information for understanding plant transients, characterizing fuel degradation and fission product release behavior, and defining functional design requirements for the user interface.

Module 1.a

Purpose: Establish Feasibility of Technology Application to AGRs

Module 1.a consists of a literature review and feasibility study that provides a preliminary technical basis for the DOSE-EP model development. The basis for the study will be a baseline research and information gathering session at NII offices. This onsite technical visit will assemble existing information on AGR operations, potential transients, core behavior during upset conditions, and fission product release phenomena for AGR fuel. The quantity, quality, and applicability of this information will be used to determine the feasibility of AGR model development. If the conclusion is reached that existing information and technical bases are insufficient to produce a DOSE-EP model yielding estimates with sufficient level of confidence, then it will be recommended that subsequent technical tasks not be performed. In this case, specific activities will be recommended to provide the basis necessary to proceed forward with the DOSE-EP model development. If the opposite conclusion is reached, i.e. existing information and technical bases are sufficient, then a written recommendation will be provided to proceed with the remainder of Module 1. Additionally, an upper bound on the number of accident classes sufficient to plan model development will be made.

Remaining tasks in Module 1 will be initiated only with written approval of the NII project manager.

Module 1.a

Establish Feasibility of Technology Application to AGRs

Level of Effort: This work will be completed by SAIC.

Tasks:

1. Collect information related to AGR reactor design, operation, and accident response characteristics from current safety documentation including, but not limited to, *case studies*, safety and accident analyses.
2. Assess available information for understanding plant transients, modeling fission product release (abnormal occurrence, design basis, and severe accident) processes, and defining functional design requirement for user interface.
3. Identify "gaps" in information and/or supplemental analyses needed to support prototype source term model.
4. Develop AGR "baseline" understanding of systems and key components.

Deliverable:

1. Feasibility and technical justification memorandum for application of technology.

Module 1.b

Purpose: Develop Accident Classes

This phase of work will develop accident classes for the AGR type reactors based on the inputs from Module 1.a. Accident class is defined as a collection of accident scenarios that present similar initial and boundary conditions to the evolution of fission product release. Accident classes will be delineated by grouping postulated accident scenarios that exhibit similar characteristics with respect to initiating events and the availability or failure of reactor and plant systems, and active/passive engineering safety features. The development of accident classes is a required input to the success criteria development task.

Module 1.c

Purpose: Establish Success Criteria

The DOSE-EP work will postulate three potential phases, given the plant is in an upset condition

- Fuel damage vulnerable - a first phase begins with the initiating event and covers the period in which fuel damage is averted. Under the success criteria module, conditions will be developed that must be satisfied to prevent fuel damage once upset conditions are detected. In other words, if these criteria are met, then the plant has successfully met the challenge posed, and the transient will not progress past the first accident phase.

- Limited fission product release - if one or more of the success criteria from the first phase is(are) not met, than the accident will progress into a limited fission product phase. Some, but limited, fission product release occurs from the fuel, consistent with limiting design basis events (DBEs). A second set of criteria will be established that must be achieved to prevent core damage from progressing beyond the DBE phase. Tabulate source terms via *RASCAL*[4] format for incorporation in Source Term Predictor Prototype's off-normal database.

- Extended core damage - A third and final phase assumes that second phase criteria have not been met. Consequently, severe accident damage occurs involving significant portions of the core. The criteria for accident termination once extended core damage has occurred are addressed in the development of the functional source term relationships and the accident progression logic.

Module 1.d

Purpose: **Identify Functional Source Term Relationships**

Relationships will be developed between the accident phenomena and the source term parameters to correctly estimate the source term, given an accident sequence is identified. The objective of this sub-module is to establish qualitative relationships describing the influence of severe accident progression on fission product release and subsequent transport through the primary system and into the containment/confinement system. Qualitative cause-effect relationships between distinct accident phenomena and source term characteristics are defined in this segment of work.

Tasks for Modules 1.b through 1.d:

A technical basis report will be prepared, consisting of three technical sections, and a fourth section on level of effort and scope of work.

1. Section 1: Contains qualitative description of known accident classes and the basis for delineation of those classes.
2. Section 2: Contains qualitative description of the mitigation success criteria for each accident class. Technical basis for these success criteria will also be documented.
3. Section 3: Contains description of the qualitative source term relationships. This description will identify the dominant cause-effect relationships between distinct accident phenomena and source term characteristics.

Deliverables:

1. Technical report providing overall technical basis and containing discussion outlined above.
2. Level of effort and work scope for completion of Module 2. Any addenda or corrections to the statement of work presented in the next section would be provided at this time, if required.

Module 1.e

Purpose: **Provide "Strawman" User Interface**

The user interface is an important ingredient toward the overall success of this project. the objective of this task is to provide the NII with an initial look at the major features of the user interface that will be integrated into the final DOSE-EP model. The interface will be a strawman, in that it will illustrate a proposed format to be used in an emergency situation. The interface will be linked to an existing source term estimator and will be fully functional. However, this baseline model will not represent proposed AGR accident progression characteristics. Upon review of the preliminary interface, the customer can transmit a list of improvements/preferences for incorporation to the interface. Implementation of these items would be part of the Module 2 scope of work.

Task for Modules 1.e:

A demonstration model interface shall be provided to elicit customer comment for the final interface.

Deliverable:

1. A demonstration interface with simple instructions and menu to follow for emergency assessment situations A full users manual will be prepared as part of Module 2.

Modules 1.b - 1.e

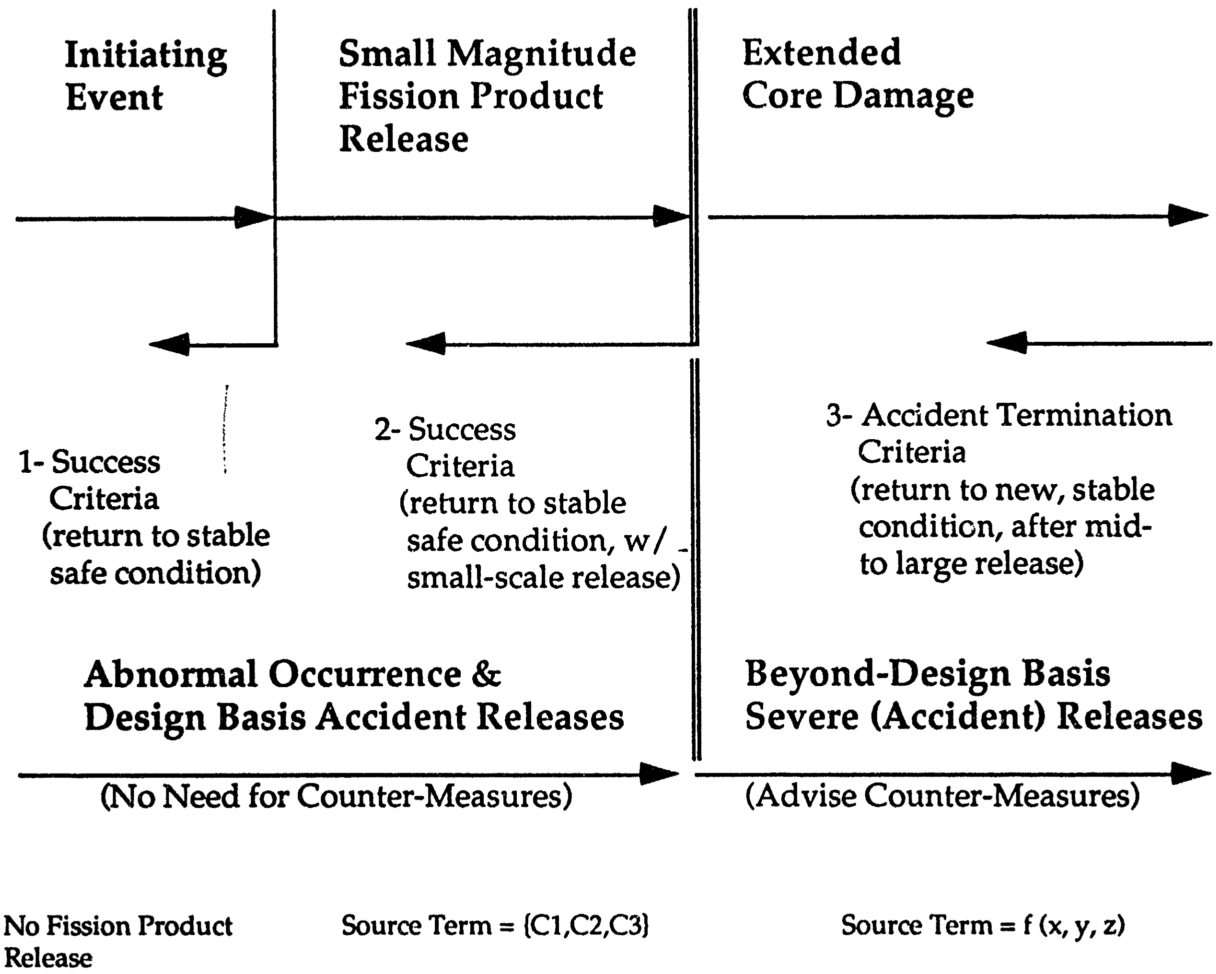
Level of Effort:

a. 260 person-hours, 1.5 person-months (2 personnel @ 50% time over two calendar-months elapsed time)
b. Travel to SAIC-Albuquerque prepare, document, and complete deliverables for Modules 1.b through 1.e.

5 days x $94*/day x 2 persons	=	$940
2 round-trips @ $1200/trip	=	$2400
Travel total	=	$3340

* $94/day = $60- lodging + $34 - meals = March, 1994 per diem for government travel.

Figure 2: Accident Release Regimes Considered For Source Term Predictor Prototype - British AGRs



Module 2. Predictive Source Term Model - AGR Prototype

This module would be based on the information developed in Modules 1 regarding the identification of accident classes, success criteria, and desired user interface. An accident progression logic model and associated source term algorithm would be built. The user interface according to the application environment specified by the NII would be finalized. The ASTM standard guide E 622 for developing computerized systems shall be followed at least in part, with functional requirements, functional design, implementation design, system assembly, and system evaluation phases planned.[5] Completion of the accident progression and source term algorithm components then provides inputs required to finish the "inference engine" as the overall assessment tool (Figure 1). Linkage of the technologies, documentation of the prototype, and technology transfer via training session are the final work products to conclude Module 2.

The second module will be performed through four tasks. A description of each task and deliverables follows the level of effort summary (below). The level of effort and timeline required to complete Module 2 shall be completed as a product of Module 1.

<u>Module 2</u>

Purpose:	**Build AGR Prototype Model and Transfer Technology to the Nuclear Installations Inspectorate**		
Level of Effort:	**a. To be defined at the end of Module 1** **b. Travel to UK NII Offices For Transfer of Technology at end of Module 2**		
	7 days x $150/day x 3 persons	**=**	**$3150**
	3 round-trips x $1500/trip	**=**	**$4500**
	Travel total	**=**	**$7650**
Tentative Module Dates:	**Start - May - June, 1994** **Best-Estimate of Early Finish - FY 1995**		

Module 2.a

Purpose: **Develop/Finalize Accident Progression Logic**

The relationships developed in Module 1.d mapping accident phenomena to source term characteristics will be converted into a traceable logic model. The relationships will be defined in the form of logic rules, such that the DOSE-EP model will reflect the most likely accident progression for a given set of initial and boundary conditions. A quantitative relationship will be documented for each rule. An example of the quantitative relationship and the rule logic used is the following: given Event A, there is ninety percent probability that Event B will occur. In addition, the relationships between plant symptoms, accident classes, and mitigation success criteria developed under Modules 1.a and 1.b will be quantified and translated into the overall logic model.

Module 2.b

Purpose: **Formulate the Source Term Algorithm & Quantify**

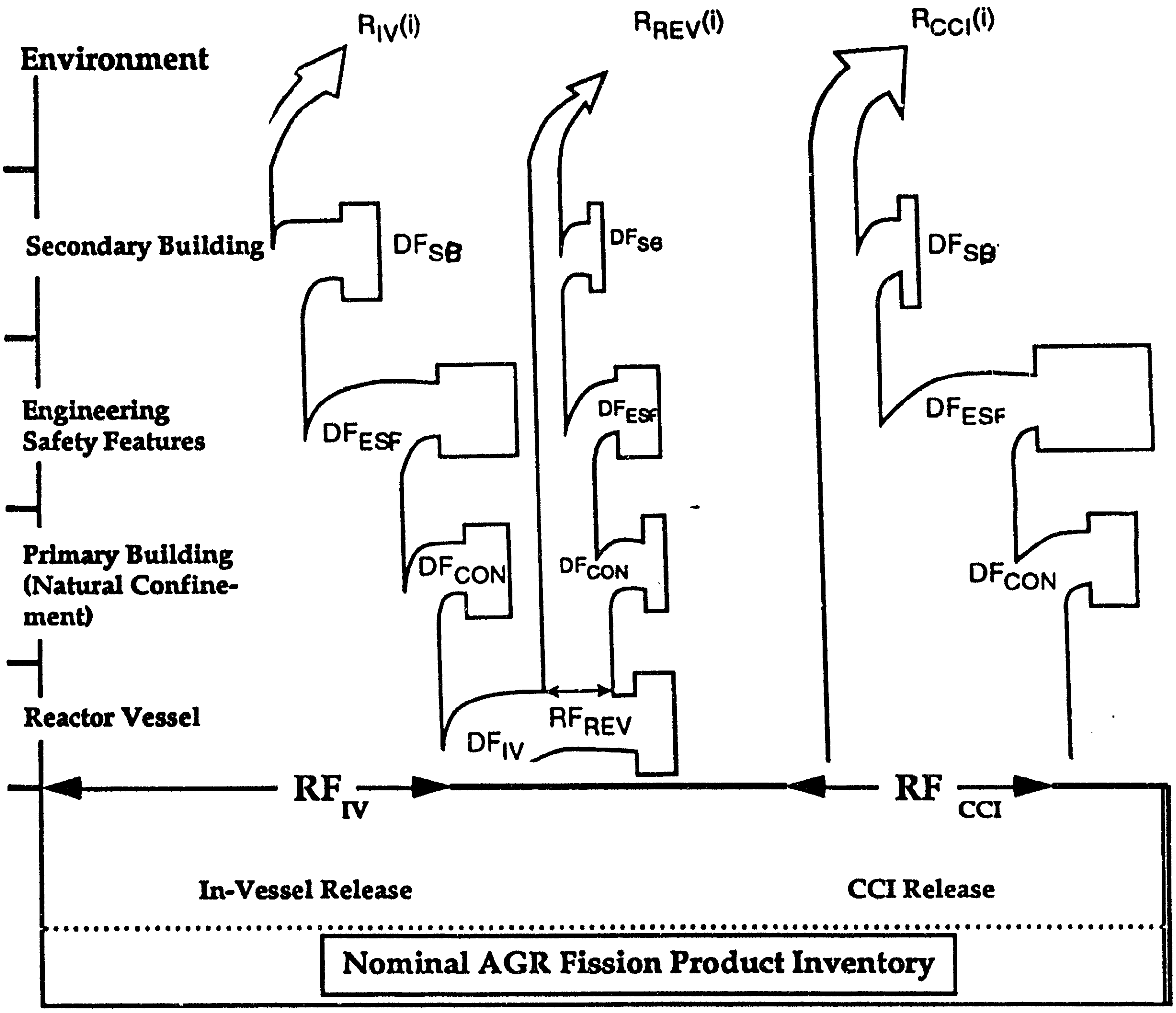
This module of work has the objective to translate the qualitative functional source term relationships developed under Module 1.d into quantitative expressions using a two-step procedure. First, simple mathematical expressions are developed manipulating the source term parameters defined earlier to calculate the magnitude of the fission product release to the environment. The source term parameters considered are the release fractions (RFs) and decontamination factors (DFs) for three fission product groups. These parameters are defined based on the outcome of physical event in the accident progression logic model.

All systems, engineering safety features (ESFs), and phenomena that have the potential to affect fission product transport must be considered in formulating the source term algorithm. Figure 3 illustrates the charting of fission product transport and removal under this paradigm given fuel degradation has occurred. There are three stages of release for fission products from fuel that are typically postulated in a severe accident

- In-vessel fuel degradation release;
- Fission product release from core material initially deposited in the vessel, but then revolatilized at a later time in the core melt progression;
- Ex-vessel release from core-concrete interaction (CCI).

Release from all three stages are examined to determine the fraction of initial core inventory that has the potential to reach the environment. Several

Figure 3: Conceptual Diagram of Fission Product Release and Source Term Algorithm Parameters



deposition processes operate that gradually decrease the transported inventory before an environmental release is predicted. Decontamination of this transported material can occur via natural deposition in the primary reactor building, by the ESFs in the building, and by natural deposition in the secondary reactor building. The sum total of the release from the three conceptual stages that is not held with the reactor building boundary is the estimated source term.

The second step in the model involves quantifying the individual source term RF and DF algorithm parameters. Quantification will be performed by reviewing the information developed under Module 1.a. The information describing the behavior of the fuel matrix under the postulated upset conditions is used to determine the RFs at the various stages of an accident. The DFs will be determined by evaluating any plant-specific calculations that have been performed or through a comparative assessment by evaluating calculations performed for similar plant types.

Module 2.c

Purpose: Integrate Overall Model Components

Integration of the various components of the DOSE-EP model must be linked to achieve a working tool. The products of the logic model, the source term algorithm, and user interface development activities are linked. The resulting methodology is reviewed and tested as a predictive tool using evaluation basis source terms discussed earlier.

Module 2.d

Purpose: Refine User Interface and Transfer Technology

Final adjustments to the user interface are made based on comments received from the Inspectorate during Module 1.e. In addition, a complete and concise user's manual shall be developed explaining the DOSE-EP model, the interpretation of accident class and output by user(s) and emergency response analyst(s).

Training will be conducted in NII offices to fully acclimate the customer to the DOSE-EP technology applied to the UK's AGRs.

Module 2 Deliverables

Module	Deliverable
2.a	A technical note shall be provided describing the basis for the accident progression logic and quantification of its component parts.
2.b	A technical report describing the basis for the source term algorithm and quantification of RF and DF source term parameters will be authored.
2.c	A functioning prototype DOSE-EP model will be assembled and tested. A test version may be transmitted to the NII prior to customer training.
2.d	A user's manual is completed and transmitted to the NII at the time of the training meeting. The training orientation is conducted to fully instruct NII inspectors on the DOSE-EP model and to provide start-up guidance in applying the manual to extract the required information from the computer model and in understanding the interface.

SCHEDULE OF MILESTONES

The following is a tentative schedule for milestones for this project. Target completion dates assume a February 1, 1994 project startup.

Module	Duration (months after project authorization)	Target Completion*
1.a	1 month (Completed by SAIC)	28 Feb , 1994
1.b - 1.e	2 months	May 30, 1994
2	to be determined at the end of Module 1	FY 1995

* Work is assumed to start 2/1/94 overall.

JOB COST ESTIMATE FOR WORK FOR OTHERS

The following are "Job Cost Estimate Pricing of Work Performed for Others" worksheets prepared by SRTC Financial Resources. The worksheets indicate the estimated cost of work performed by WSRC, and travel/lodging spendout while fulfilling tasks under this WFO in SAIC in Albuquerque.

The overall total estimated cost for this project is the sum of all cost components, and including travel. The total for Module 1 is $ 35,917, and includes personnel travel/lodging costs to the SAIC offices in Albuquerque, NM. The total for Module 2 shall be established at the end of Module 1.

Module	WSRC FTE-hours $	SAIC-Albuquerque FTE-hours $	WSRC Travel/Lodging $
1.a SRTC+SAIC	0 300	300	
1.b through 1.e SRTC+SAIC	260 760	500	3340
1 Total SRTC+SAIC	260 1060	800	3340
2 SRTC+SAIC			
TOTALS			

TOTAL LABOR: 260 FTE-hours

TOTAL COST: $35,917 (labor + travel/lodging)

Job Cost Estimate Pricing of Work Performed for Others - Sheet 1

SAVANNAH RIVER TECHNOLOGY CENTER
JOB COST ESTIMATE FORM
FOR WORK FOR OTHERS - NON-FEDERAL ENTITIES

JOB DESCRIPTION: MODULE 1B - 1E
ESTABLISH FEASIBILITY OF TECHNOLOGY APPLICATION
TO AGRs AND ESTABLISH INPUTS & TECHNICAL BASELINE
FOR MODEL DEVELOPMENT
CUSTOMER: HEALTH & SAFETY EXECUTIVE NUCLEAR
INSTALLATIONS INSPECTORATE, UNITED KINGDOM THRU SAIC

PERFORMING ORGS: L1500 AND L1700
SRTC CONTACT: KEVIN O'KULA
DATE: 3/9/94

1. CLS DIRECT LABOR

LABOR TYPE	PERF.ORG	LABOR RATE		EST. HOURS		DIRECT LABOR
EXEMPT	L1500	28.70	X	110	=	$3,157
	L1700	28.10	X	150	=	$4,215
		0.00	X	0	=	$0
		0.00	X	0	=	$0
				TOTAL EXEMPT		$7,372
WEEKLY		0.00	X	0	=	$0
		0.00	X	0	=	$0
		0.00	X	0	=	$0
		0.00	X	0	=	$0
				TOTAL WEEKLY		$0
HOURLY		0.00	X	0	=	$0
		0.00	X	0	=	$0
		0.00	X	0	=	$0
		0.00	X	0	=	$0
				TOTAL HOURLY		$0

TOTAL DIRECT LABOR $7,372

2. DIRECT MATERIALS

	DESRIPTION	
A.		$0
B.		$0
C.		$0

TOTAL DIRECT MATLS $0

3. OTHER DIRECT COST

A.	SUBCONTRACT	$0
B.	TRAVEL	$3,340
C.	SUPPLIES	$0
D.	OTHER	$0

TOTAL OTHER DIRECT COST $3,340

TOTAL DIRECT COST $10,712

4. CLS OVERHEAD

PERF ORG.	OH RATE		TOTAL DIR. LABOR		OVERHEAD AMT
L1500	252.0%	X	$3,157	=	$7,957
L1700	256.8%	X	$4,215	=	$10,825
	000.0%	X	$0	=	$0
	000.0%	X	$0	=	$0

TOTAL OVERHEAD $18,782

DEPRECIATION 4.50% X $7,372 $332

TOTAL DEPRECIATION $332

TOTAL CLS OHD'S & DEPR COST $19,114

	DOE RATE		DIRECT COST PLUS OH AND DEP LESS TRAVEL	TOTAL
5. DOE EXPENSE FACTOR	15.40%	X	$26,486	$4,079
DOE-SR OVERHEAD	7.60%	X	$26,486	$2,013

TOTAL DOE ADDED FACTORS $6,092

TOTAL ESTIMATED COST $35,917

MANAGER SRTC-FR | MANAGER - PERF. ORG. | DOE FINANCE REPRESENTATIVE | MANAGER - SITE COST ACCTNG.

PAYMENT SCHEDULE

WFO 94-002 Total Authorized Funding = $ 35,917.

Date*	Payment Due ($)	Cumulative Funding Collected ($)
04/1/94**		
05/1/94		
06/1/94		
07/1/94		
08/1/94		
09/1/94		
10/1/94		
11/1/94		
12/1/94		
....		

* This schedule assumes a contract term of 02/1/94 to 05/31/94 for Module 1, and 06/1/94 to ______ for Module 2.

** The U.S. Department of Energy requires that a continuous ninety-day advance of funds must be maintained during the life of the WFO project. WSRC will review costs as they are incurred and may make adjustments to the payment schedule, if required, in order to maintain the ninety-day advance of funds.

REFERENCES

1. K. R. O'Kula, A. A. Simpkins, R. P. Taylor, K. C. Wagner, C. N. Amos, *Probabilistic Risk Assessment Support of Emergency Preparedness at the Savannah River Site,* Proc. of the Fourth Topical Meeting on Emergency Preparedness and Response, Long Island, NY, (April, 1993) pp. 99-103.

2. A. A. Simpkins, K. R. O'Kula, and C. H. Hunter, *Reactor Accident Source Term Algorithm for Emergency Response at the Savannah River Site,* Proc. of the Fourth Topical Meeting on Emergency Preparedness and Response, Long Island, NY, (April, 1993) pp. 91-94.

3. A. A. Simpkins, K. R. O'Kula, R. P. Taylor, Jr., and D. P. Kearnaghan, *Enhancement of the Source Term Algorithm for Emergency Response at the Savannah River Site,* Proc. of the Fourth Topical Meeting on Emergency Preparedness and Response, Long Island, NY, (April, 1993) pp. 95-98.

4. SAIC Technical/Cost Proposal, "A Proposal to Develop an Advanced Gas-Cooled Reactor Emergency Preparedness Source-Term Estimator," SAIC Proposal No. 01-1120-71-0940-009, (December 31, 1993).

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6. American Society for Testing and Materials E 622 Standard Guide for Developing Computerized Systems (1993).

7. J. M. Griesmeyer and L. N. Smith, *A Reference Manual for the Event Progression Analysis Code (EVNTRE),* NUREG/CR-5174, SAND88-1607, Sandia National Laboratories, Albuquerque, NM, (September, 1989).

Attachment A/1
Request For Technical Support From UK NII

HSE
Health & Safety
Executive

NUCLEAR INSTALLATIONS INSPECTORATE

Mr Kevin R O'Kula, Manager
Risk and Source Term Technology Group
Safety Technology Section
Westinghouse Savannah River Company
CCC Building 992W-1
1991 South Centennial Avenue
Aiken, South Caroliona 29803-7657
USA

Your Ref: SRT-RST-930256

Our Ref: NUC 40/1/05

24 June 1993

Dear Kevin

Proposed Visit To Westinghouse Savannah River Company
Emergency Preparedness Methodology Discussions

Thankyou for your letter of June 17. As discussed in our telephone conversation I am proposing that Mr Paul Harvey and myself visit Aiken during the week commencing 2 August 1993, with a duration of 3/4 days. I anticipate formal authorisation for the visit from my management shortly and I will ask you to make accomodation arrangements on my behalf when this is received.

The purpose of the visit will be to explore the assistance that Westinghouse Savannah River company can lend to the Inspectorate in respect of source term evaluation. Assuming our discussions confirm your assistance would be of benefit, I would hope that a preliminary outline of work could be agreed during our visit.

I attach an outline specifcation of the Inspectorate's needs, and the areas of your methodology we would wish to explore during our visit. I hope this is sufficient for the purpose of an agenda.

Please do not hesitate to contact me if you have any queries.

Yours sincerely

T Whitehead
HM Inspector (Nuclear Installations)

Attachment A/2
Request For Technical Support From UK NII

1. Source Term Evaluation - Product Requirements

General Concept:

A simple, reliable and user friendly tool, that can be operated without highly specialised knowledge on a stand alone PC, to provide a rapid estimation of source term, based primarily on plant conditions/damage, in the event of a radiological release from major nuclear facilities.

Types of facilities to be catered for include Magnox, Advanced Gas Cooled and Pressurised Water reactors and chemical (reprocessing) plant.

2. Input Data:

Plant operating conditions
Plant damage states
Perimeter Gamma monitoring results (if available)
Field monitoring data (if available)

The tool must be functional without monitoring data.

3. Output Data:

Estimated core damage
Fraction of fission product inventory released (iodine and noble gases will be sufficient)
Predicted duration of release
Likelihhood of further releases
Confidence associated with predictions

The Inspectorate has no need for information on predicted doses.

4. In Aitken, we would particularly wish to:

a) Become familiar with the basic logic steps in moving from plant information through to plant damage states and finally through to prediction of source term, preferably illustrated with simple examples.

b) Witness a demonstration of the software currently in operation, for a range of plant damage states.

c) Explore the logic behind PRAST and particularly the extent of its dependence, if any, on quantified PSAs.

d) Examine at what stage of the logic field data could be input to refine the source term.

e) Examine evidence of verification/validation studies and also any sensitivity studies that have been carried out (ie how sensitive the output is to spurious/incorrect input data).

f) Examine the raw data requirements - ie the information WSRC would need to be supplied with to begin work.

Attachment A/3
Request For Technical Support From UK NII

HSE
Health & Safety
Executive

NUCLEAR INSTALLATIONS INSPECTORATE

Mr Kevin R O'Kula, Manager
Risk and Source Term Technology Group
Safety Technology Section
Westinghouse Savannah River Company
CCC Building 992W-1
1991 South Centennial Avenue
Aiken, South Caroliona 29803-7657
USA

Your Ref: SRT-RST-930291

Our Ref: NUC 40/1/05

3 August 1993

Dear Kevin

Westinghouse Savannah River Company - Emergency Preparedness Methodology

Thankyou for your letter of 30 July. I can confirm it is the intent of my unit within NII to pursue emergency management methodolgy support from the Risk and Source Term Technology Group of Westinghouse Savannah River Company. The UK's fiscal years begin and end in April and I can advise that funds of up to approximately $100,000 have been allocated for this work for the period ending 31 March 1995.

As you are aware, no decision has yet been reached by the NII's senior management on whether collaboration between our organisations can go ahead. It is unlikely that a decision will be reached until the end of this month given the extent of staff leave.

You can be assured that I am actively pursuing the matter internally and, as a matter of courtesy, I will let you know as soon as there are any developments.

Yours sincerely

T Whitehead
HM Inspector (Nuclear Installations)

Attachment A/4
Request For Technical Support From UK NII

TO: Mr L G Williams
Head of Branch E

cc: NUC 40/1/05

FROM: Mr J A Driscoll
Head of Unit E1
Room 612
SPH
Extn 4160

DATE: 13 July 1993

NII ROLE IN THE EVENT OF A NUCLEAR EMERGENCY
(Source Term Library)

Staff Notice No 23 is incorporated into the NII Emergency Procedures Handbook and includes the following instruction. "The scope of NII advice (to Government Departments and the Government Technical Advisor) will be centred on the situation on the site. Advice in relation to the situation off the site is primarily one of independent assessment of the source term".

In your Annual Report 1991/92 you reported "The role of the NII during an emergency is now clearly delineated, namely that it will be centred on the situation on the Site. Advice with respect to the situation off site will be limited to an independent assessment of the source term. This information will be relayed to the National Radiological Protection Board who will have the responsibility for considering off site consequences and potential countermeasures. The Branch (E1) has a commitment to develop a "library" of source terms to cover various accident/emergency scenarios. This work has been deferred into the next financial year (92/93) due to lack of resources."

With this commitment in mind a Branch E extra mural support/research bid for the development of the source term library was agreed at the SMG level for the financial years 1991/92 and 1992/93.

Some useful progress has been made in 1992/93 which has resulted following consultations with Branch B in the identification of a possible contractor. The Westinghouse Savannah River Company have developed a method for the determination of accident source terms for nuclear power and chemical plant which warrants, in my view, further examination.

I believe the Branch already have the clearance necessary to pursue this work. The money and a fruitful line of development is available.

J A DRISCOLL

JAD\July93\15.sam

Attachment B
Working Meetings of November 8 - 11,1993
SRTC - UK NII - SAIC

Support to the UK Nuclear Installations Inspectorate: Program Planning Meetings of November 8 - 11, 1993

Kevin R. O'Kula*

Savannah River Technology Center
Westinghouse Savannah River Company

Presented To:

Tom Whitehead and Paul Harvey

United Kingdom Nuclear Installations Inspectorate

November 8, 1993

SRTC
Savannah River Technology Center

SAIC
An Employee-Owned Company

30.1

Context

- **Precedent Established By K Reactor Accident Code/WINDS Revision Completed March, 1993**
  - APET - Source Term Methodology
  - Added Pre- & Post-Processing Logic Structure
  - User-Friendly PC Package To Cope With Operational, DBA, And Severe Conditions
- **Provides Predictive, Level-of-Confidence Information**
  - Supplements Information With Instrumentation Available
  - Technical Basis For Response w/o Plant Instrumentation
  - "Cliff Effects" Capabilities
- **Presented April, 1993 at Emergency Preparedness Topical Meeting**
  - Discussions Among UK NII-WSRC-SAIC For ~ 7 Months
- **Current Visit To Plan Work Scope, Schedule, Format**
  - November 8 Through November 11

Candidate Applications

- **Savannah River Site Facilities**
  - Defense Waste Processing Facility
  - Replacement Tritium Facility
- **U. K. Nuclear Installations Inspectorate**
  - Advanced Gas-cooled Reactors
  - Sellafield Reprocessing Plant
- **U. S. Commercial Reactors**
  - U. S. Nuclear Regulatory Commission

Agenda For November 8th

Time	Location	Purpose	Presenter/Guide
8:30 AM	C, 773-A	Introductions & Agenda WSRC Goals	K. O'Kula, WSRC
8:45 AM		SRTC Perspective In New Technology Development	C. Thiessen, WSRC
9:00 AM		New Mission Directions	
9:10 AM		Initial Remarks & UKNII Perspective	T. Whitehead, NII P. Harvey, NII
9:30 AM		SAIC Remarks & Demonstration of Current KDAM Model	M. Leonard, SAIC S. Ashbaugh, SAIC
10:15 AM		Break	
10:30 AM	773-A	WINDS Visit	R. Addis, WSRC
11:15 AM	E-102, 703-A	SRS Operations Facility/ EOF	C. MacDonald, WSRC
12:00 PM		Lunch	
1:00 PM	12, Centennial	Working Meetings	NII, WSRC, SAIC

SRTC
Savannah River Technology Center

Emergency Preparedness Support to UK NII
Planning Meetings - November 8 - 11, 1993

SAIC
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Agenda For November 9th - 11th

Time	Location	Purpose	Presenter/Guide
Tuesday, November 9			
9:00 AM	11, Centennial	Working Meetings	NII, WSRC, SAIC
12:00 PM		Lunch	
1-5 PM	12, Centennial	Working Meeting	
Wednesday, November 10			
8:00 AM	11, Centennial	UK NII Meetings	NII
10:00 AM	11	Demo Presentation to DWPF	SAIC, WSRC
1 - 5 PM	12, Centennial	NII Meetings	
Thursday November 11			
8:00 AM	32, Centennial	Closeout Meetings	NII, WSRC, SAIC
12:00 PM		Lunch or Adjournment	
1:00 PM		Continuation, if Needed	

SRTC
Savannah River Technology Center

Emergency Preparedness Support to UK NII
Planning Meetings - November 8 - 11, 1993

SAIC
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Goals of the Working Meetings

- Agree On Context For Providing Running KDAM Model To NII
  - Two-Phase Program
  - Based On Addressing AGRs Initially
  - Cost-Effective, Within Funding & Timetable Constraints

- Decide On Format That Will Achieve Multi-Party Objectives

- Identify Personnel To Achieve Overall Success With High Confidence Of Win-Win-Win Result
  - Establish Basis For Meaningful Follow-Up

Expertise Summaries

Kevin O'Kula

Doug Cramer
Don Kalinich
David Allison
Bob Taylor
Enno Nomm

Kevin R. O'Kula

Group Manager, Data and Consequence Analysis

EDUCATION

1975	BS	Applied and Engineering Physics,	Cornell University
1977	MS	Nuclear Engineering	University of Wisconsin
1984	PhD	Nuclear Engineering	University of Wisconsin

Thesis was in the fusion technology area. Considered problems of tritium removal from solid breeding materials; identified improved release enhancements in experimental study.

PROFESSIONAL EXPERIENCE

10/82 - Present

He has been at the Savannah River Site for eleven years, working in reactor shielding, core physics, reactor dosimetry, severe accident analyses, tritium safety evaluation and methodology, and PRA programs in a technical capacity before moving to management.

10/93 - Present
Group Manager, Data and Consequence Analysis (Group size of 14)
Process Safety Technology Section

Managerial Responsibilities: Provide, update, and trend data system and component data for use in risk assessment fault trees (logic models) and risk management evaluations. Develop and apply deterministic and probabilistic consequence methodologies for assessment of postulated accidents in facilities with radiological and chemical source terms. Coordinate technical bases for determination of facility-specific emergency action levels with emergency management and safety analysis groups. Provide DOE Complex-wide methods for sharing risk model data. Develop emergency management tools to predict likely releases. Support DOE facility fire risk modeling and emergency management hazard assessments.
Technical: Testing/validating new tritium transport and pathways accumulation consequence models as participant in international Biospheric Model Validation Study, Phase II Participant - Tritium Working Group. DOE Radiation Worker trained (RWT II). Chemical safety and consequence consultant to DOE through Battelle Pacific contract.

4/92 - 9/93
Group Manager, Risk And Source Term Technology (Group size of 22)
Safety Technology Section

Responsibilities: Complete and document full-scope reactor probabilistic safety assessment (PSA); develop risk and dose management applications to cost effectively prioritize facility modifications; direct resources to cultivate non-traditional business areas for group expertise in accident progression analysis, emergency management, worker safety assurance, and chemical risk reduction.

1/91 - 4/92
Group Manager, Source Term Evaluation (Group size of ~ 5)
Reactor Safety Research Section

Responsibilities: Ensure operational, design basis, and severe accident radiological source term & resultant dose estimates are evaluated in a consistent and technically defensible manner with appropriate methodologies.

Douglas S. Cramer

Group/Section	Data & Consequence Analysis/ Process Safety Technology Section
Title:	Fellow Scientist
Education:	Ph.D., Nuclear Physics, University of Virginia, August 1968 BS, Physics, University of Virginia, June 1963
Expertise:	Applications of probabilistic risk analysis and database development to reactor and non-reactor facilities.
Current Responsibilities:	Provide database development and probabilistic risk analysis to support process safety technology.

Accomplishments:

DuPont/WSRC:

15 years employment at Savannah River
As lead engineer, developed a database for equipment and human reliability with source tractability for over 6,000 failure probabilities including data for uncertainty analyses.
Exchanged database information with off-site users.
For the impact on the Safety Analysis Report excluded specific accidents from the design basis space involving flow reduction and multiple control rod withdrawal.
Evaluated safety implications of removing local powerhouse from the site 115 kV grid.
Evaluated ignition sources in waste tanks in H-Area
Applied concepts of reactor vibration monitoring.
Performed criticality calculations and evaluations.
4 external tech publications and lectures.
30 internal reports.
Promote Boy Scouts, also interest in science in local schools.

Knolls Atomic Power Laboratory, General Electric Company:

Neutron slowing-down spectrometer for high precision assay of irradiated reactor fuel for proof-of-breeding program.
Managed 10 kW nuclear research reactor.
3 external publications.

Community Colleges and Secondary Schools, three US states:

Taught math, physics and chemistry

University of Virginia:

National Aeronautics and Space Administration Fellow
National Science Foundation Fellow
6 external publications.

Donald A. Kalinich

Engineer, Liquid Waste Analysis Group

EDUCATION

1987	BSME	Mechanical Engineering, University of Florida
1990	MS	Mechanical Engineering, University of Florida

Thesis involved the development of an unsteady, quasi 1-D Euler solver to perform scoping calculations for magnetoplasmadynamic compressor concept as Master's Thesis. The solver was later modified to analyze wave rotors.

PROFESSIONAL EXPERIENCE

3/92 - Present
Westinghouse Savannah River Technology Center

Responsible for the development of phenomenological analyses used in the risk assessments of nuclear production reactors and waste treatment facilities. Specific duties have included:

- performing thermal-hydraulic and heat transfer calculations for use in the Savannah River Site K-Reactor Probabilistic Risk Assessment;
- development of accident analyses—involving fires, explosions, chemical releases, and building ventilation response—as part of the update to the Safety Analysis Report (SAR) for the Defense Waste Processing Facility (DWPF);
- quantification of Accident Progression Event Tree analyses for the DWPF SAR;
- oversight of analyses and experimental programs performed by other groups; and
- developing software and models for specific accident phenomena.

8/90 - 2/92
Department of Physiology, University of Florida

Developed modifications for the EX-14 Navy deep-water diving helmet to reduce CO_2 rebreathing and the work of breathing at low fresh-gas flow rates. Specific duties included:

- development of a PC-based data acquisition system with both real-time and post-acquisition analysis capabilities;
- design and construction of experimental equipment and prototypes;
- devising and conducting laboratory experiments that would adequately simulate dive conditions; and
- analysis and evaluation of experimental data, as well as the preparation of quarterly and yearly reports for the contract sponsor.

David Keith Allison

CURRENT POSITION

Senior engineer in the Separations Analysis Group of the Process Safety Technology Section

EDUCATION

BS in nuclear engineering, Texas A&M University - May 1983
MS in nuclear engineering, Texas A&M University - Dec 1984
PhD in nuclear engineering, Texas A&M University - Dec 1990

Dissertation research involved the experimental determination of parameters important to the application of ion channeling to Masked Ion Beam Lithography (MIBL). During the course of the work, silicon membranes were fabricated and the angular distribution of protons transmitted through them recorded as a function of energy and crystal orientation. An analytical model for proton channeling was also developed.

EXPERIENCE

Sept 1990 - Present, Senior Engineer, WSRC

May 1993 - Present

Acted as technical lead for the chemical hazards analysis of the HB-Line facility. Calculated the consequences of H-Canyon Outside Facilities SAR accidents. Authored the H-Canyon Consolidated Facilities Basis for Interim Operation (BIO). Currently analyzing the consequences of chemical releases during HB-Line SAR accidents and extending the previous confinement analysis of the Defense Waste Processing Facility (DWPF).

Jan 1993 - Apr 1993

Modeled the pressure rise resulting from a benzene deflagration in an In-Tank Precipitation (ITP) tank using MELCOR/SR.

Sept 1990 - Dec 1992

Performed analyses related to design basis and beyond design basis accidents for SRS production reactors including assessments of the likelihood and consequences of steam explosions and hydrogen deflagrations. Constructed and presented safety cases before external review bodies.

Robert P. Taylor, Jr.

Title: SGL 37, Senior Administrative Engineer

Group/Section: Liquid Waste Analysis Group
Waste & Environmental Safety Technology Section

Education Background: BS-NE, 1978 -- N. C. State University

PRA/Accident Progression/Emergency Management Expertise:

Extensive Level 1 PRA experience with K Reactor. Skilled in use of CAFTA computer codes for construction/solution of fault trees, event trees, and accident sequences.

Seismic PRA analysis for K Reactor, DWPF, and Tritium Facility. Experience on modification of existing internal event fault trees/event trees to seismic versions. Skilled in use of SHIP (Seismic Hazard Integration Package) computer code used to integrate PRA derived cut sets with seismic hazard to get seismic accident sequence frequencies. Constructed logic for determining what damage state K Reactor would be in based on indication available in the control room for use in the beyond design basis source term estimation computer program constructed for SRS Emergency Management personnel.

Have performed some accident progression work with the DWPF accident analysis, have used the EVNTRE computer code to solve the accident progression event tree (APET) constructed for evaluation of the risks of operating DWPF. Also have some experience in modifying the APET.

Served as primary backup to the technical support manager in the K reactor ERO organization for several years. Participated in the successful test scenario used to verify K Reactor's emergency preparedness status for restart. Provided technical support to the emergency management group for planning exercise scenarios.

Accomplishments:

DuPont/WSRC: In addition to the above, have extensive knowledge of reactor operation and technical support activities. Spent 12 years in various Reactor Engineering jobs both as engineer and manager. Involved in reactor component design, reactor refueling operations, safety system performance analysis, testing, and improvement. Approximately 8 of the 12 years were in jobs involving direct technical support and oversight to C, K, and L Reactor operations. Spent 2 years as Manager of Reactor Operations for K Reactor. In this position I was responsible for managing the shift personnel responsible for running the K production reactor.

Enno Nomm

Title:	Fellow Engineer, (36)
Group/Section:	Risk Assessment Methodology Group, Waste & Environmental Safety Technology Section
Education Background:	BS Physics, Univ. of Illinois; MS Nuc. Engineering, Univ. of Illinois

PRA/Accident Progression/Emergency Management Expertise:
23 years work related to SRS Nuclear Reactor Safety including risk and accident analysis. Involved with initial introduction of PRA methods to SRS nuclear reactor safety.

Current Responsibilities:
Fault tree analysis of plutonium reprocessing facilities at SRS.

Accomplishments:
Served on Technical Review committee representing Reactor Engineering Department for Level One PRA for SRS Nuclear Reactor Facilities.
Initiated the computer data base for unusual occurrences at SRS nuclear reactor facilities.
Contributed to SRS Technical Baseline and Systematic Evaluation Programs for the nuclear reactor facilities.

DISTRIBUTION

J. R. Patterson, DOE Office of Economic Development, 703-46A
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8/4/94

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