CONF-930402--2

WSRC-MS-92-347

PROBABILISTIC RISK ASSESSMENT SUPPORT OF EMERGENCY PREPAREDNESS AT THE SAVANNAH RIVER SITE (U)

by K. R. O'Kula

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

WSRC-MS--92-347

DE93 007634

Other Authors:

W. H. Baker
(WSRC)
R. P. Taylor
(WSRC)
K. C. Wagner
Science Applications International Corporation
Albuquerque, NM
C. N. Amos
(SAIC)

A paper proposed for Presentation/Publication
at/in the Fourth Topical Meeting on Emergency Preparedness & Response
Huntington, NY
04/26-29/93

This paper was prepared in connection with work done under Contract No. DE-AC09-89SR18035 with the U. S. Department of Energy. By acceptance of this paper, the publisher and/or recipient acknowledges the U. S. Government's right to retain a nonexclusive, royalty-free license in and to any copyright covering this paper, along with the right to reproduce and to authorize others to reproduce all or part of the copyrighted paper.

PROBABILISTIC RISK ASSESSMENT SUPPORT OF EMERGENCY PREPAREDNESS AT THE SAVANNAH RIVER SITE (U)

K. R. O'Kula, A. A. Simpkins, and R. P. Taylor
Westinghouse Savannah River Company
992W-1 Savannah River Technology Center
Aiken, South Carolina 29803-7657
(803) 644-5440

K. C. Wagner and C. N. Amos
Science Applications International Corporation
2109 Air Park Road SE
Albuquerque, New Mexico 87106
(505) 247-8787

ABSTRACT

Integration of the Probabilistic Risk Assessment (PRA) for K Reactor operation into related technical areas at the Savannah River Site (SRS) includes coordination with several onsite organizations responsible for maintaining and upgrading emergency preparedness capabilities. Major functional categories of the PRA application are scenario development and source term algorithm enhancement. Insights and technologies from the SRS PRA have facilitated development of: (1) credible timelines for scenarios; (2) algorithms tied to plant instrumentation to provide best-estimate source terms for dose projection; and (3) expert-system logic models to implement informed counter-measures to assure onsite and offsite safety following accidental releases. The latter methodology, in particular, is readily transferable to other reactor and non-reactor facilities at SRS and represents a distinct advance relative to emergency preparedness capabilities elsewhere in the DOE complex.

I. INTRODUCTION

The Savannah River Site (SRS), operated by the Westinghouse Savannah River Company (WSRC) for the Department of Energy (DOE), is the primary center for the production and processing of nuclear materials for national defense, deep-space exploration, and medical treatment applications in the United States. In recent years, SRS has also become a leading technology transfer point in the DOE complex from both reactor and non-reactor programs, particularly in areas of equipment qualification, bioremediation, fire safety, and Probabilistic Risk Assessment (PRA) technology applications. The last area is based on the ongoing K Reactor PRA conducted to evaluate current operational and shutdown risks. The reactor PRA has led to identification of improvements to eliminate vulnerabilities and to other enhancements that promote safe facility operation at Savannah River.

PRA support to outside technical groups has evolved over the last four years as the baseline probabilistic program has matured and as major portions have been completed. Assistance for emergency planning is coordinated by Environmental Transport staff of the Savannah River Technology Center (previously, the Savannah River Laboratory) for the emergency preparedness organization at SRS. The reactor risk evaluation process, in particular, has focused on the following programs undergirding emergency planning: (1) scenario development for emergency response drills; (2) extension of the current emergency preparedness source term algorithm to forecast projected releases for beyond-design-basis events; and (3) development of an expert system framework to project likely release modes and corresponding source terms, given plant instrument readings, observations, and operator input. This paper discusses the coordinated PRA applications support program to emergency planning and response at SRS.

II. REACTOR OPERATION & CURRENT PROBABILISTIC SAFETY PROGRAMS

The Savannah River K Reactor is a heavy-water moderated and cooled reactor (HWR) operated with uranium-aluminum (U/Al) fuel contained in a low-temperature and low-pressure primary system. Fuel and target material is configured in concentric tubular elements in hexagonal arrays of assemblies rather than in commercial fuel "pins". Anticipated remaining operation of K Reactor is below 800 MW_{th}. The reactor has a confinement, rather than a containment, system to mitigate the effects of operational and accidental releases. These features are the basis for conditions expected during highly unlikely accident occurrences, including release of fission products and tritium from the reactor core, the subsequent deposition/transport through the confinement system, and the potential source term to the environment. PRA and accident management programs at SRS indicate the course of hypothetical accidents in a HWR would be significantly different from core melt progression and containment response in commercial light water reactors.

The overall probabilistic risk study is patterned after commercial reactor PRA practices using state-of-the-art NUREG-1150 methodologies[1] tailored for SRS reactor application. Plant configuration and power levels have changed since 1987, thereby demanding that PRA "snapshots" be taken to provide updated assessments of reactor risk after sets of changes have been implemented. The current assessment includes the impacts of reduced power level, several major seismic upgrades, more internal plant redundancy to maintain core cooling, enhanced operator training, changeout of cadmium safety rods with boron carbide elements, and the introduction of a natural draft cooling tower.

Formally, the K Reactor probabilistic safety analysis consists of three distinct phases of analysis, or "Levels." The Level 1 analysis is an assessment of the frequency of accidents that lead to damage of the nuclear fuel assemblies in the core. Accidents initiated by both internal causes (equipment and system random failures, human error) and external causes (seismic events and fires) are considered. The Level 1 PRA utilizes sequence event trees (ETs) quantified with fault trees (FTs) and plant system data to track system failures and thereby identify core melt sequences. Sequences resulting from the Level 1 anlysis are grouped on the basis of similarities in the condition of the plant during the accident sequences. These groupings are termed plant damage states (PDSs). The Level 2 assessment evaluates accident progression from the ensemble of plant damage states using an accident progression event tree (APET) logic model to probabilistically follow thousands of possible sequences. Groups of sequences that possess similar timing and fission product release characteristics are identified on the basis of the Level 2 PRA and are referred to as radiological source terms. The final portion of the PRA, the Level 3, assesses the consequences (health, contamination, and economic effects) of the source terms weighted by likely meteorological conditions at the time of the release.

The analysis of the K Reactor response to accident conditions in the PRA accounts for LWR-HWR differences and is the basis for assistance to emergency preparedness programs at SRS. The Levels 1 and 2 of the PRA specifically provide: (1) relative likelihood, or frequency of occurrence, guidance for practice response drills and possible accident scenarios; and (2) timing and source term characteristics incorporating real-time plant instrumentation data.

III. CURRENT PROBABILISTIC SAFETY SUPPORT

Support of emergency planning and response by the PRA program at SRS currently concentrates on the following programs: (1) scenario development for emergency response drills; (2) extension of the current emergency preparedness source term algorithm to forecast projected releases for beyond-design-basis events; and (3) development of an expert logic system framework to predict likely release modes and corresponding source terms.

Scenario Development

Reactor safety analysts are responsible for two activities in this role: (1) scenario review and critique; and (2) source term construction. Rehearsal of all elements of the emergency planning organization at SRS is conducted regularly to demonstrate readiness and DOE Order compliance. PRA review of planned scenarios and preparedness drills is coordinated with environmental transport and emergency preparedness personnel. The practice strives to attain timelines that are realistic, credible sequences, yet "exercise" all parts of the response organization. Additionally, the exercise timeline must trigger predetermined emergency response actions. These particular sets of actions are dictated by the appropriate emergency classification level (ECL) which is based on observed plant conditions and predicted radiological outcomes relative to SRS onsite protective guidelines and offsite protective action guides (PAGs). Anticipated source terms also must be developed that logically account for plant instrumentation readings as a function of time into the scenario.

Table 1 illustrates an abbreviated loss-of-coolant accident (LOCA) timeline established by the emergency preparedness organization with call out of major events and identification of emergency classification levels triggered by the chain of events. Current ECL hierarchy at SRS in ascending degrees is noted: NOUE - Notice of Unusual Event, ALERT, SAE - Site Area Emergency, and GE - General Emergency. An overall frequency of occurrence is developed for the planned scenario in an annotated format, and inconsistencies are removed. Typically, several iterations are required before emergency planning and reactor safety analysts reach a consensus on the validity of the exercise.

Source term construction includes the application of integrated analysis codes, CONTAIN/SR[2] or MELCOR/SR[3] to predict the time rate of release of radioactivity from the confinement system into the environment. Figure 1 shows the tritiated water release associated with a hypothetical loss-of-pumping accident sequence. The release in this case is limited to tritium because the reactor confinement does not retain tritium, and fuel assemblies are cooled sufficiently to preclude melting since ECCS is available throughout the sequence (unlike the Table 1 LOCA illustration). The figure shows that of the 3.5 kg of tritiated water vapor released in this accident, approximately 0.5 kg would be retained in the confinement (walls, floors, and water disposal system) with ~ 3 kg released from the stack. Again, a number of iterations are necessary among environmental transport, emergency preparedness, and PRA personnel before the source term is deemed appropriate and consistent to the scenario.

Source Term Algorithm Upgrades

Enhancements of the computer code used to develop anticipated reactor releases has been a second major role for PRA analysts. Validation and verification of the current Reactor Accident Program (RAP) to predict the source term, and

---

Table 1. Timeline For SRS Emergency Preparedness Scenario

Initial Conditions

1. K Reactor Operating at 700 MW, Approximate Exposure is Mid-Cycle
2. Offsite Supply Transformer Is Down To Repair Incoming Line
3. One Emergency Diesel is Tagged Out For Repair of Intake Manifold; Another Operating Emergency Diesel Is Undergoing Functional Testing

Time	Event	Emergency Classification Level
0600	Emergency Diesel Fails On Overspeed Trip	NOUE
0606	Begin Controlled Reactor Shutdown	
0625	Fire In Transformer Yard	ALERT
0630	Reactor Scrammed	
0720	Transformer Trips Due To Grounding Caused By Transformer Yard Fire; Complete Loss of AC Power	
0735	Still No Offsite Electrical Power; All Transformer Rooms Are Lost & All Emergency Diesels Are Inoperative	SAE
0746	Decreasing Air Pressure Due To Loss of All AC	
0840	Heavy Water Leak Of ~1000 GPM On Process Water System Is Detected Via Reactor	

Building Tritium Alarms

0845 ECCS Booster Pump Diesel Fails GE

0930 Reactor Vessel Water Level Is ≤ 1 ft.

0935 Indications of Fuel Damage; Gamma Radiation Levels Rise

0936 Noble Gas Release From Building Stack Begins

0950 Incoming transformer line is restored; Commercial power is brought in; ECCS operation is available & utilized to cover core preventing additional damage

1015 Emergency Diesel Repaired and Returned To Service

1128 Monitoring teams verifying dry deposition levels have one contamination case

1230 Enter Recovery Mode

---

extension to a beyond-design basis condition mode is nearing completion.

The beyond-design basis operation of RAP is a contingency option to be made available upon determination that activity monitors on the plant are inoperable, or after severe accidents have adversely impacted the confinement system operation. The result of these improvements meets objectives set forth in a DOE Operational Readiness Review and yields software product versatility approximately equivalent to the commercial reactor code, RASCAL.[4] Major aspects of this work are discussed elsewhere.

Expert System & Source Term Ranking

The most significant technology transfer to emergency preparedness involves collaboration with Science Applications International Corporation to implement PRA-based expert system logic models in a user-friendly format to predict ranked source terms and estimated likelihood of occurrence. This aspect of the PRA support is an ongoing activity promising broad, sitewide applicability to enhance other operating facilities' emergency preparedness.

The ranked source term and frequency prediction is based on available plant indicators and on understanding of the reactor and confinement systems response developed from Levels 1 and 2 of the reactor operation PRA. The expert system logic model (ESLM) coordinates plant monitoring inputs, revised APET models, and ECL decision-making requirements to provide near real-time evaluation of the current conditions in the plant, prediction of conditions that could unfold, and safety margins available before exercise of preventative counter-measures are prudent. Figure 2 illustrates key module coordination, feedback, and information strategy for the expert logical system.

The methodology requires a final form of the Level 1 and 2 PRA, the probabilistic APET under-girding the risk assessment, and a source term algorithm developed for quantification of the reactor risk. Other mid- to high-hazard DOE facilities, including non-reactors, could take advantage of this approach. It must be emphasized that the computer models will lead to an "expert" system that is usable and sensible only if the plant cognizant safety engineers participating in the PRA process are also leading the expert model activity. Involvement of PRA analysts ensures overall model credibility and technical scrutability.

As a result of this PRA application, the emergency operations staff at SRS will in time have online, near real-time capability to generate the ranked predictions, as well as guidance to implement the best accident management strategies to mitigate dose consequences.

IV. LIMITATIONS

The approaches outlined in this summary are probabilistically based and have been developed after considerable lead time to formulate credible PRA evaluations for the K Reactor at Savannah River. In many cases, the information provided from the PRA tools are supplementary decision-making aids that should be regarded as neither sufficient nor complete. However, as conditions deteriorate in a facility during real or practice drill accident sequences, a PRA offers a best-estimate perspective that may be lacking otherwise. It is postulated that recovery is more probable if emergency preparedness personnel have the best possible PRA "roadmaps" for mitigation and/or prevention of plant damage and radiological release.

V. CONCLUSION

The PRA for evaluation of K Reactor safety has been applied to support emergency preparedness at the Savannah River Site. The application stresses availability of a near final form Level 1 PRA (initiator identification and plant damage state evaluation) and Level 2 PRA (accident progression and source term development) to enhance:

- Preparedness drill timeline and source term development;
- Beyond-design basis capabilities, especially as conditions deviate from limited core damage assumptions; and
- Emergency response through the development of a expert system logic model.

The approach is vulnerable to several limitations and is not a complete answer when responding to emergency conditions in a DOE facility. However, the best-estimate nature of the PRA process allows response to take advantage of the most realistic picture of the plant especially if plant instrumentation is questionable and if accident progression insights are otherwise lacking.

ACKNOWLEDGMENT

This abstract was prepared in connection with work done under Contract No. DE-AC09-89SR18035 with the U. S. Department of Energy.

REFERENCES

1. U.S. NUCLEAR REGULATORY COMMISSION, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, (December 1990).

2. K. R. O'KULA, "The CONTAIN/SR Code For Analysis of SRS Reactor Confinement Response During Severe Accidents," Westinghouse Savannah River Company, WSRC-TR-90-326, (1990).

3. Science Applications International Corporation, "MELCOR/SR Computer Code Manual," SAIC 92/6506 (1992).

4. G. F. ATHEY, A. L. SJOREEN, and T.J. MCKENNA, "RASCAL Version 1.3 User's Guide", Oak Ridge National Laboratory, Prepared for the Division of Operational Assessment, Office for analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, (September, 1989).

Figure 1. Loss-of-Pumping Accident
Tritium Release For Source Term Development

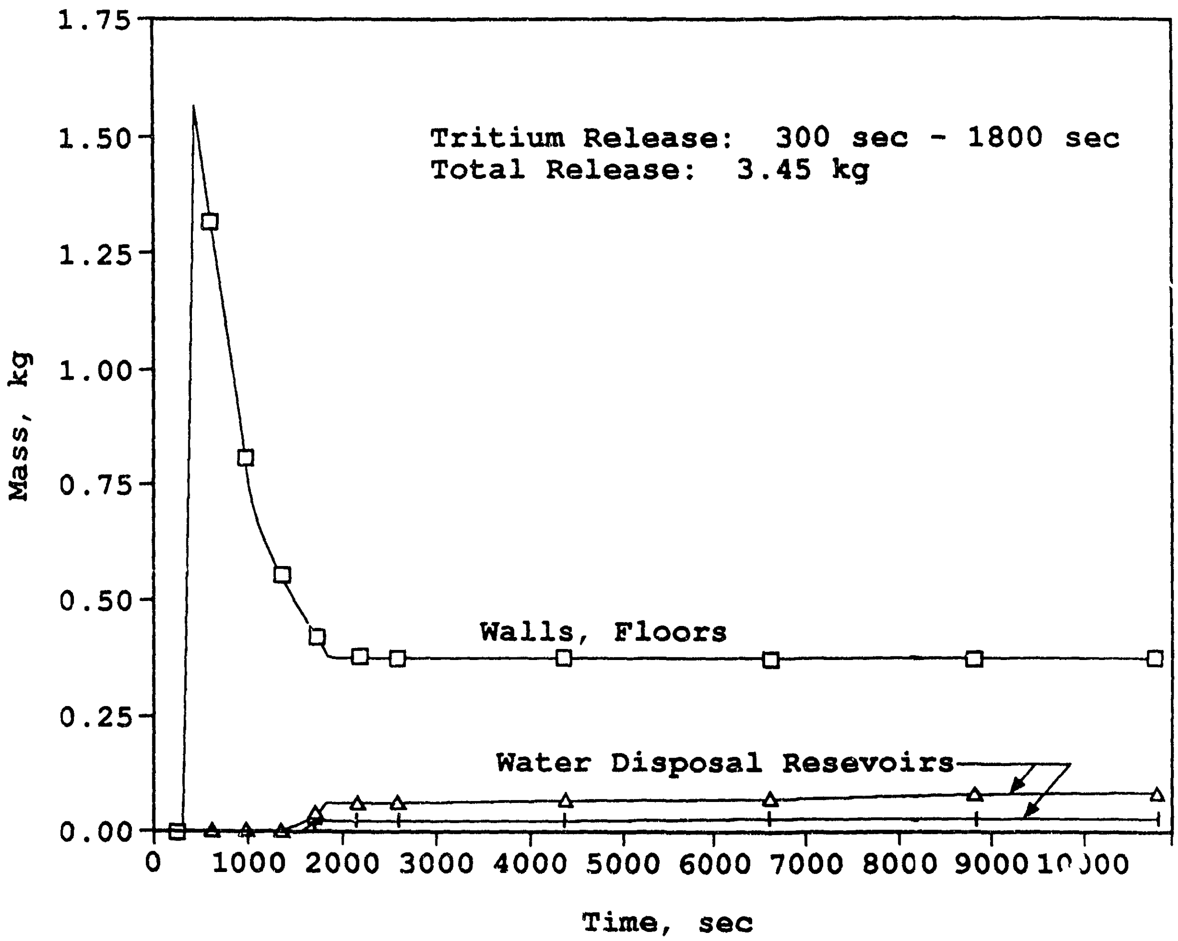
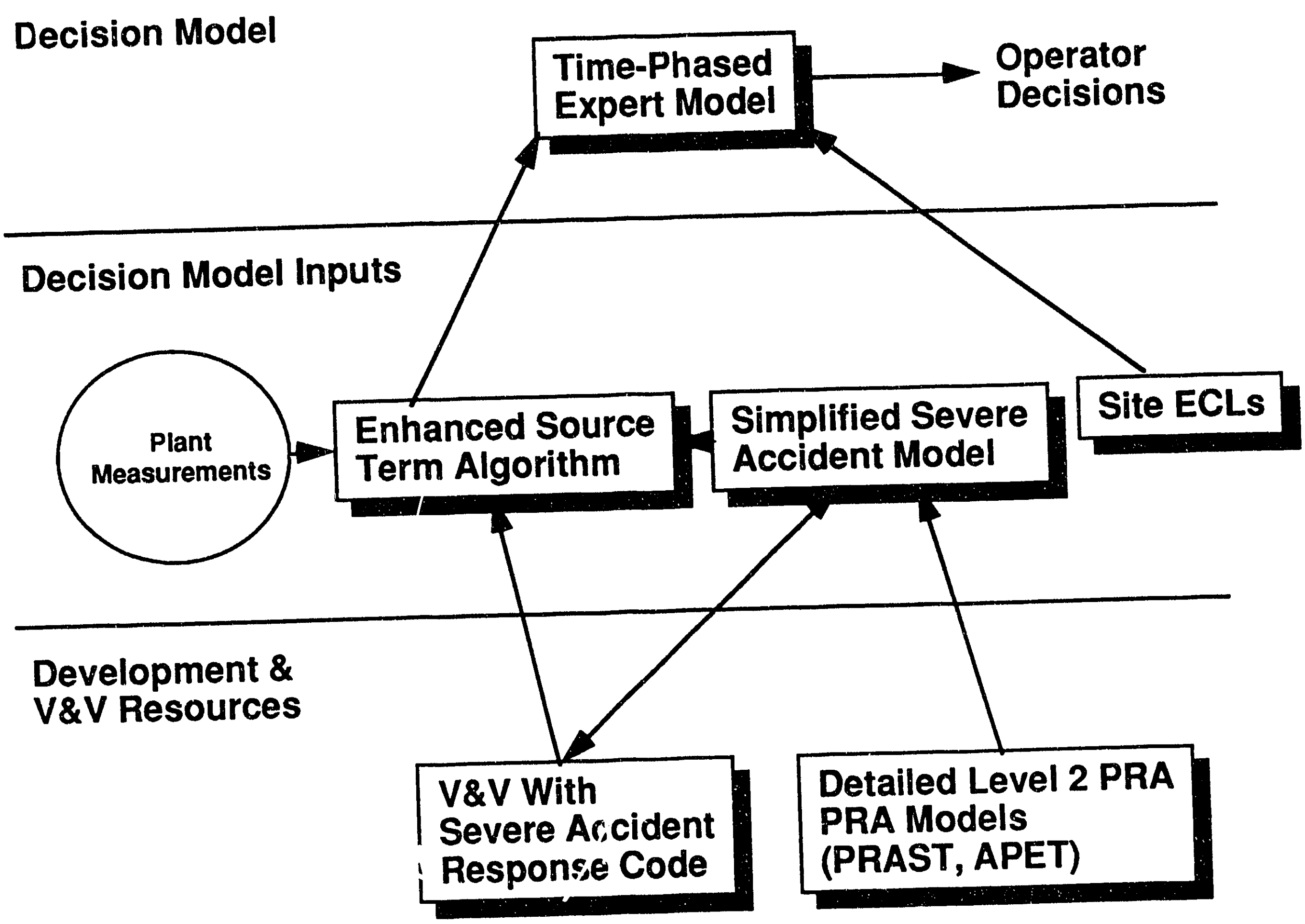


Figure 2. Schematic of Information Flow
For Expert System Logic Model



END

DATE FILMED

5/11/93