WSRC-RP--89-1405

DE91 004257

RESULTS OF THE LEVEL 1 PROBABILISTIC RISK ASSESSMENT (PRA) OF INTERNAL EVENTS FOR HEAVY WATER PRODUCTION REACTORS (U)

by

S. P. Tinnes, D. S. Cramer, V. E. Logan, S. V. Topp, J. A. Smith, and M. D. Brandyberry

Westinghouse Savannah River Company
Savannah River Site
Aiken, SC, 29808

A paper for presentation at the
Topical Meeting on the Safety, Status, and
Future of Non-Commercial Reactors
Boise, ID
September 30 - October 4, 1990

and for publication in the Proceedings

RESULTS OF THE LEVEL 1 PROBABILISTIC RISK ASSESSMENT (PRA) OF INTERNAL EVENTS FOR HEAVY WATER PRODUCTION REACTORS (U)

S. P. Tinnes, D. S. Cramer, V. E. Logan, S. V. Topp,
J. A. Smith, and M. D. Brandyberry
Westinghouse Savannah River Company
Savannah River Site
Aiken, SC 29808
(803)725-5309

ABSTRACT

A full-scope probabilistic risk assessment (PRA) is being performed for the Savannah River Site (SRS) production reactors. The Level 1 PRA for the K Reactor has been completed and includes the assessment of reactor systems response to accidents and estimates of the severe core melt frequency (SCMF). The internal events spectrum includes those events related directly to plant systems and safety functions for which transients or failures may initiate an accident.

I. OBJECTIVES OF THE PRA

The SRS PRA has three principal objectives:

- Improved understanding of SRS reactor safety issues through discovery and understanding of the mechanisms involved.
- Improved risk management capability through tools for assessing the safety impact of both current standard operations and proposed revisions.
- A quantitative measure of the risks posed by SRS reactor operation to employees and the general public, to allow comparison with declared goals and other societal risks.

II. REACTOR DESCRIPTION

The SRS is located in South Carolina approximately 25 miles southeast of Augusta, Georgia. It is situated in portions of Aiken, Allendale, and Barnwell counties on an area of about 300 square miles. The site, which is owned by the U. S. Government, is set aside as a controlled area for the production of special materials for national defense. The site produces nuclear materials, primarily plutonium and tritium, for the Department of Energy (DOE).

The SRS facilities include three operating reactors (P, L, and K) and two non-operating reactors (C and R) that are almost identical to each other in design and administrative controls. The reactors are sited a minimum of 2.5 miles apart and a minimum of 4.5 miles from the site boundary. The SRS reactors began operation in 1953, with the primary purpose of producing plutonium and tritium for nuclear weapons. The reactors are presently in the late stages of a program to upgrade seismic resistance, reactor systems control, and administrative controls to become more nearly equivalent (in risk terms) to those used in the U. S. nuclear utility industry.

This PRA applies to the Mark 22 core design in the K Reactor with the physical layout that existed in June 1987. Physical changes that have been made since then have not been incorporated, with one exception: a fourth emergency coolant injection path has been incorporated. The Mark 22 charge is composed of relatively light-weight, highly-enriched uranium-aluminum alloy tubular fuel containing tubular lithium targets, for the purpose of tritium production. The core is moderated by heavy water (D_2O), and the primary coolant is also D_2O. The reactor is cooled by six primary loops with redundancies in pump motors and electric power supplies. The six primary loops are cooled by twelve heat exchangers that are supplied by cooling water from a large basin that, in turn, is supplied by water from the Savannah River.

It is useful in understanding this PRA to note some of the major differences between the SRS reactors and the light water cooled reactors used for commercial production of electricity (LWRs). Since the SRS reactors are not required to produce steam, they operate at low temperature (maximum about 90 degrees C) and low pressure (maximum about 200 psi). There are no concerns with steam blowdown, turbine trip, or other scenarios related to the high stored energy and electricity generation aspects of an LWR. On the other hand, uranium-aluminum alloy fuel clad with aluminum for the SRS reactors melts at a much lower temperature than the uranium oxide, zirconium-clad fuel for LWRs, and also has lower heat capacity. The SRS reactors therefore require much faster action to maintain adequate core cooling under accident scenarios.

III. LEVEL 1 INTERNAL EVENTS METHODOLOGY

Due to the major differences between SRS reactors and commercial power reactors, which affect accident-initiating events, mitigating system designs and the spectrum of possible damage, it is inappropriate to treat this analysis as a perturbation on the PRAs for power reactors. All phases of the analysis are treated as unique to SRS. The duration of all accidents is assumed, by convention, to be 24 hours.

The SRS Level 1 PRA uses the fault-tree linking methodology, a well-established technical approach used in many other PRAs. The detailed technical efforts are divided into eight principal task areas:

1. Initiating Event Identification and Analysis
2. Event Tree Development
3. System Fault Tree Development
4. Electric Power Model Development and Analysis
5. Component Fault Tree (Generic Subtree) Development
6. Component and Human Reliability Data Base Development
7. Sequence Analysis and Compilation of Results
8. Uncertainty Analysis

The task efforts define the expected progression of a potential accident scenario and develop the associated SRS-specific models to represent plant response or to incorporate plant operational experience.

The SRS reactors are unique in that initiating events discovered in analyses associated with other reactor types generally do not apply. The SRS initiating event task conducts a very detailed and systematic search for potential accident initiators. The approach is based on a systematic examination of the failure mechanisms for any and all barriers containing radioactive material. The format for the approach is the dendogram (hierarchical "trees"). The dendograms are used to define the barriers and the failure modes of the barriers in terms of basic physical phenomena (e.g., melt, chemical damage, mechanical damage, etc.). The output is a listing of primal initiators that are assigned to more conventional classes of reactor accidents. This task identified five different classes of accidents that might yield reactor severe core melt scenarios:

- Transient: An upset of the reactor power-to-flow ratio.
- Primary Loss of Coolant Accident: A leak in the primary (D_2O) cooling system.
- Secondary Loss of Coolant Accident: A leak in the light water secondary cooling system which leads to a Loss of Pumping Accident.
- Loss of Heat Sink: Failure of heat transfer between the primary and secondary cooling systems requiring emergency cooling.
- Loss of River Water: Failure of the river water supply to the once-through secondary cooling system requiring conservation of the water on hand.

For each of these accident classes, the potential accident progression scenarios are developed and modeled as event trees. For the branchpoints, the related plant system response are identified and system fault trees are developed. The system fault trees are developed down to the component level (e.g., pump, valve, etc.). Generic subtrees are developed for the components themselves in a separate task and linked to the system fault trees.

The generic subtrees trace the supply of electric power back to key buses. The availability and reliability of electric power at buses is represented as a series of basic events in the generic subtrees. These probabilities are analyzed separately in a Markov model that considers the frequencies of losing normal power, the probabilities of failure of emergency power, and the mean times to repair parts of the electric power supply. The theoretical model calculates the probability of finding the system in any particular configuration (i.e., combination of buses On or OFF), given some initial condition. Unavailability is calculated by admitting all possible transitions among configurations, both those caused by failures and those caused by recovery steps. Unreliability is calculated by admitting only failure transitions. The initial condition is prescribed by assigning probabilities to each of 32 possible configurations.

Throughout the development of the PRA models, SRS specific design and operational information is used. Plant design features and response strategies, including operator actions as specified by SRS procedures, are incorporated directly into the models. Actual plant experience, as obtained from incident reports and operator logs, is used to compute SRS specific component reliability data. The SRS experience base of 110 reactor-years of operation is available for these computations and is supplemented by generic industry data only when necessary.

By linking the system fault tree models together in the manner indicated by the event trees to represent the accident sequence of interest, and using the plant-specific data to estimate initiator frequencies and component failure rates, plus estimation of the human error rates, the accident sequence frequencies are quantified. The SETS code is used to solve fault trees for their minimal cut sets. The TEMAC code is used for quantitative evaluation of probabilities, using as input the cut sets derived by SETS and best estimates of component/event probabilities and frequencies.

Uncertainty estimates are made for the total severe core melt frequency by assigning probability distributions to the basic events and propagating them through a simplified model of the SCMF cut set results. Data for the basic events are derived from SRS-empirical or generic data, and uncertainties in the events are taken to be described by either log-normal or "maximum entropy" distributions. Fifth and ninety-fifth percentiles of these distributions are derived from chi-squared confidence interval tests for events where SRS data are available, from error factors quoted for generic data, or from judgement and industry practices for human errors. The simplified SCMF model includes the dominant cut sets from all five contributing classes of accidents, and yields results within 97 percent of the SCMF calculated with the full Level 1 model. The basic event distributions are propagated through this simplified model with the TEMAC computer code using Latin Hypercube Sampling to assess the distribution of the SCMF.

IV. RESULTS OF THE PRA - SEVERE CORE DAMAGE

The severe core melt frequency (point estimate) for the K-Reactor configuration is estimated to be 2.07E-4 per reactor-year from internal initiators. The mean value of the severe core melt distribution (2.34E-4) is slightly higher than the point estimate, with 5% and 95% confidence values of 1.72E-5 and 9.97E-4/reactor-year, respectively. This result lies within the range of values calculated for currently operating nuclear power facilities, using similar techniques. Contributing to the result are 267 different sequences from the different initiating event classes. The distribution of this result among the contributing initiating event classes is tabulated below, along with the associated initiating event frequency and percentage contribution to the SCMF.

Class of Initiating Event	Initiator Frequency per Reactor Year	Severe Core Melt Freq. per Reactor Year	Percent of Total
Primary Loss of Coolant - LOCA	5.6E-3	1.22E-4	59%
Secondary LOCA (Loss of Pumping - LOPA)	3.3E-3	4.90E-5	24%
Transients - TRAN	2.5E+0	1.61E-5	8%
Loss of River Water - LORW	1.2E-3	1.01E-5	5%
Loss of Heat Sink - LOHS	1.2E-4	9.86E-6	4%

Within the Primary LOCA class, expansion joint breaks and subsequent response failures contribute most to the indicated frequency (51 of the 59%). Large break LOCAs contribute the remaining difference. Response to LOCAs in the primary system involves shutdown, emergency injection for makeup of lost coolant inventory, leak control and/or isolation, water removal from the building to avoid flooding of pumps, and direct core cooling in the event of pump flooding.

For the Secondary LOCA class, most of the indicated frequency (23 of the 24%) is related to large breaks, with the small breaks and expansion joint breaks contributing significantly less (< 1% each). Response to LOCAs in the secondary system involves shutdown, leak control and/or isolation, water removal from the building to avoid flooding of pumps, and direct core cooling in the event of pump flooding.

Transients yield 8% of the SCMF and have no subclasses of initiators that dominate. Response to transients involves shutdown only. It is assumed for all classes of transients (i.e., reactivity-addition, flow-reduction, and coolant-heatup transients) that an automatic shutdown (Safety Rod System or SSS ink injection) is required to prevent severe core melt. Once shutdown, process water (D_2O) and cooling water flows (H_2O) are maintained indefinitely to cool the reactor core. Process water flow to the reactor core is maintained by the six online DC Caterpillar diesel generators that drive the six primary cooling pumps. Cooling water flow is maintained to the process water heat exchangers via gravity flow from the 25-million-gallon cooling water reservoir (Building 186).

Within the Loss of River Water initiating event class, loss of plant grid power initiators and river water pump-house initiators contribute about the same indicated frequency (2.5% of the 5%). Response to loss of river water initiators involves event recognition, shutdown, and water conservation. The loss of plant grid initiator analysis is the only set of sequences that includes recovery modeling external to the fault tree models. The recovery deals with the restoration of the plant grid and is based on industry-wide (utility) experience.

The Loss of Heat Sink initiating event class yields 4% of the SCMF and has three subclasses of initiator. The total loss of primary system circulation accounts for 5%, the total loss of secondary cooling due to effluent header failures accounts for less than 1%, and the total loss of secondary cooling due to inlet header failures accounts for less than 1%. Response to loss of heat sink initiators involves shutdown and direct core cooling.

Based on the above initiating event class results, it is apparent that the SCMF from the Level 1 PRA is dominated by accidents involving pipe breaks. The combination of primary and secondary LOCAs covers 83% of the SCMF. These initiating event classes either immediately or eventually require emergency core cooling system (ECCS) response, as makeup in the primary LOCA and as direct cooling in the secondary LOCA loss of pumping sequences. If the loss of heat sink initiating event contribution is also included as a similar accident requiring direct cooling, the contribution of sequences associated with LOCAs and/or emergency cooling cover about 88% of the SCMF. The remaining percentage is associated with shutdown functions or loss of river accidents.

The five most dominant sequences in the Level 1 PRA, which account for 76% of the total SCMF, are (1) a primary LOCA bellows break and failure to inject emergency coolant (32%), (2) a large secondary LOCA pipe break and failure to inject emergency coolant following flooding (21%), (3) a primary LOCA bellows break and overthrottling of emergency coolant (10%), (4) a transient and failure to shutdown (8%), and (5) a large primary LOCA plenum inlet pipe break in a line containing an ECCS path with throttling of emergency coolant (8%). The dominant sequences are summarized in the following table.

Initiating Event	System Failure	Initiator Frequency per Reactor Year	Severe Core Melt Freq. per Reactor Year	% of Total
Primary LOCA Bellows Break	ECCS Injection	5.6E-3	6.6E-5	32
Large Secondary LOCA	ECCS Injection	5.9E-4	4.2E-5	20
Primary LOCA Bellows Break	ECCS Throttling	5.6E-3	1.9E-5	9
Transient	Automatic Shutdown (Safety Rod System & SSS)	2.5E+0	1.6E-5	8
Large Primary LOCA In ECCS Loop Plenum Inlet Line	ECCS Is Throttled	1.5E-5	1.5E-5	7

The sequence results are also tabulated according to functions or systems that failed in the sequence. The information given in the table below is the percentage of the SCMF associated with the sum of all sequences including a failure of the indicated system or function. These fractions are not additive because sequences may contain more than one system or function failure and thus may be counted more than once in the set of values presented. Thus the sum of the damage percentages shown in the table will add up to more than 100%.

SYSTEM OR FUNCTION	DAMAGE %
Emergency Cooling (Makeup or Direct Cooling)	77%
Leak Isolation For Primary or Secondary LOCAs	22%
Shutdown	8%
Water Disposal System	6%
Loss of River Water Event Recognition	2%
Water Conservation (Following Loss of River Water)	2%
Electric Power	2%

Given that a pipe break occurs, the table shows that the emergency cooling and leak isolation functions dominant the Level 1 PRA severe core melt frequency.

V. APPLICATIONS OF THE PRA

Improvement in understanding the issues involved in SRS reactor safety has been accomplished through analysis of the mechanisms encountered. Insights from this PRA have been applied to reactor safety decision-making since early in the program. To date, 17 formal recommendations have been made ranging from improved reliability of ECCS injection valves to citations of areas for improved operator training and improved operating procedures. PRA methods and results are included among the bases for evaluating reactor restart issues and for assigning priorities to projects in the Reactor Safety Improvement Program.

It is planned to maintain a "living PRA" to keep the analysis abreast of whatever significant changes in operating and safety facilities and procedures may arise. The "living PRA" is expected to be based on a collection of models, some detailed and some simplified. Detailed models will be updated continually to reflect the plant as it exists at any given time. There will be several simplified models that represent sequences by their principal cut sets in order to construct an overall picture. Some models will serve to define the up-to-date allocation of frequencies among damage bins. Others will be modified as needed to estimate the effects on postulated changes.

VI. SUMMARY

The Level 1 PRA for the Savannah River Site K Reactor is completed. The PRA uses a fault-tree linking methodology. The severe core melt frequency (SCMF) from internal initiators is estimated to be 2.07E-4 per reactor-year (point estimate). Contributing to the result are 267 different sequences from five different initiating event classes. Given that a pipe break occurs, the emergency core cooling system (ECCS) and leak isolation functions dominate the SCMF.

VII. ACKNOWLEDGEMENTS

The information contained in this article was developed during the course of work under Contract No. DE-AC09-89SR18035 with the U. S. Department of Energy.

END

DATE FILMED

12/12/90