CONF-9010151--6

WSRC-RP--89-1404

DE91 005101

EXTERNAL EVENTS ANALYSIS FOR THE SAVANNAH RIVER SITE K REACTOR (U)

by

M. D. Brandyberry and H. E. Wingo

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

The information contained in this paper was developed during the course of work 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.

M8912032

EXTERNAL EVENTS ANALYSIS FOR THE SAVANNAH RIVER SITE K REACTOR (U)

M.D. Brandyberry and H.E. Wingo
Westinghouse Savannah River Company
Savannah River Site
Aiken, SC 29808
(803)725-1706

ABSTRACT

The probabilistic external events analysis performed for the Savannah River Site K-reactor PRA considered many different events which are generally perceived to be "external" to the reactor and its systems, such as fires, floods, seismic events, and transportation accidents (as well as many others).

Events which have been shown to be significant contributors to risk include seismic events, tornados, a crane failure scenario, fires and dam failures. The total contribution to the core melt frequency from external initiators has been found to be 2.2×10^{-4} per year, from which seismic events are the major contributor (1.2×10^{-4} per year). Fire initiated events contribute 1.4×10^{-7} per year, tornados 5.8×10^{-7} per year, dam failures 1.5×10^{-6} per year and the crane failure scenario less than 10^{-4} per year to the core melt frequency.

I. INTRODUCTION

The analysis of "external events" consists of characterizing the effects on reactor safety of events which are "external" to the reactor's systems. These events can be weather related events such as tornados, hurricanes, or ice storms; natural events such as forest fires, earthquakes or tsunamis; or human related events such as fires or vehicle crashes. These events may affect the reactor in various ways depending upon whether they occur inside or outside the reactor buildings (or both as in the case of an earthquake). Also important is whether an event has the ability to breach the reactor building and physically damage the reactor. This could occur either physically as with an aircraft impact or by way of such things as a ventilation system in the event of a chemical spill. Some events may also directly impact the reactor systems as in the case of a fire.

The list of external events analyzed here was developed largely from collective knowledge of such events considered in other PRAs and SAR-type studies. From that basis, the list is considered exhaustive. All events have been examined from the standpoint of hazard to the reactor and those which have been found to present a significant hazard have been analyzed in detail using probabilistic approaches.

Identified events may be grouped in the manner shown in Table 1 (fire and seismic are separated out due to their generally perceived significance). There is nothing unique to this ordering of events. It is merely convenient due to similarities between events in the various categories.

Results from the internal event analysis were utilized where appropriate throughout the external events analysis. However, this external event analysis was performed concurrently with the internal event analysis for Level 1 of this PRA, and, therefore, many internal event results were available very late in the analysis. Every attempt has been made to remain consistent in logic models, assumptions, and data (where appropriate) with the internal event analysis.

II. EXTERNAL EVENTS RESULTS SUMMARY

The core damage frequency associated with all significant external contributors to risk is shown in Table 2.

Due to the methodology employed by the internal events analysis, the wind and flood risks shown in Table 2 are duplicated in the internal events analysis. Tornado and Dam failures are included in the internal events loss of river water sequences. These events could also affect offsite power, but their contribution to the Loss of Offsite Power (LOSP) frequency (0.7 per year from the internal events study) is a very small part. Thus, the total external event core damage contribution which should be combined with the internal events core damage frequency is given by the values shown in the third column of Table 2.

As can be seen here, crane failures and seismic are the dominant external contributors to core damage frequency. The crane failure analysis scenario involves the dropping of a heat exchanger back into the heat exchanger bay during a heat exchanger changeout while there is still fuel in the tank.

The results of the seismic risk quantification for the reactor in the base case configuration indicated that five out of 19 total sequences contribute 95% of the mean frequency of core damage of 1.2×10^{-4} per reactor year. This base case assessment was evaluated for the reactor in an unmodified state, with few upgrades for seismic strengthening, and was done to help serve as a measure of improvement for subsequent modifications.

An uncertainty analysis indicated the distribution on the frequency of core damage ranged from 4.2×10^{-6} per reactor year at the 5th percentile to 5.5×10^{-4} per reactor year at the 95th percentile. Sequences contributing most to core damage were:

- Failure of circulation of D_2O (primary coolant), Sequence 4, which comprised 32.3 percent of the total risk. This sequence was dominated by relay chatter that caused closure of rotovalves in the D_2O system

- Failure of cooling water piping, Sequence 17, which comprised 25.9 percent of the total risk. This sequence was dominated by failure of underground piping from soil consolidation.

- Failure of the cooling water basin, Sequence 16, which comprised 18.9 percent of the total risk. This sequence was dominated by failure of the Building 186 cooling water reservoir from soil consolidation.

- Loss of river water supply to the cooling water reservoir followed by failure to respond to the loss, Sequence 2, which comprised 14.8 percent of the total risk. This sequence was dominated by failure of operating personnel to respond to the alarm for loss of cooling water supply to the cooling water reservoir.

- Failure of the integrity of the D_2O system, Sequence 5, which comprised 2.7 percent of the total risk. This sequence was dominated by a small leak in the D_2O system and failure of the operating personnel to isolate the leak.

The fire event core melt frequency is seen to be very low with respect to frequencies generated in other plant fire risk analyses. The methodology used to assess the fire risk at K-reactor has been applied in other situations (e.g., Hanford N-reactor) with significantly higher fire induced core melt frequency results. Thus, the methodology does not drive this low result. The low frequency at K-reactor is due to the independence and redundancy of important reactor systems.

III. SEISMIC ANALYSIS

There are four main parts to the seismic risk assessment: These are:

- seismic hazard assessment
- seismic fragility analysis
- system model
- risk quantifications

The seismic hazard quantifies the frequency of ground motion that is estimated to occur at the site. As part of the hazard analysis the effects of surficial soil deposits on ground motion are included. The uncertainty in the seismic hazard is characterized in terms of a family of discrete hazard curves with a probability weight assigned to each curve. The weights are a measure of the degree to which it is believed that a curve represents the true site hazard. The sum of the weights add to unity. The Electric Power Research Institute and Seismicity Owners Group seismic hazard evaluation project was used to estimate the seismic hazard. The mean hazard for the Savannah River Site is shown in Figure 1.

The fragility analysis evaluates the conditional fraction of failure of plant structures and equipment as a function of ground motion level. Component (structure or equipment item) failure is generally defined in terms of a response level that would lead to a loss of function. The seismically-intiated failure of plant components is expressed in terms of the same ground motion parameter as that used in the hazard analysis, which for this analysis was the spectral acceleration at 5.0 Hz. The 5.0 Hz frequency is selected because it is in between the fundamental frequency (i.e., 2.0 Hz) of the reactor building, which houses much of the safety-related equipment, and the fundamental frequencies of the equipment. As part of the fragility analysis, the uncertainty in the capacity of components was also evaluated. A selected sample of component fragility values is shown in Table 3.

To describe the performance of the reactor system, given the occurrence of a seismic event, a logic model was developed that considered sequences of events and the consequences of seismically induced component failure. A seismic event tree was constructed that considered the response of major structures and safety systems (equipment and operator actions) to earthquake ground motion. For each safety system included in the analysis, a fault tree was developed that defined the seismically-initiated and random component failures that could lead to the top failure event.

An attempt was first made to use the entire internal events analysis and simply add seismic failures to ensure completeness. However this proved to be too cumbersome because of the large size of the internal events analysis models.

To quantify the risk of core damage, the seismic hazard, fragility, and system portions of the analysis were assembled. The quantification was performed in two steps, First, the plant logic model was quantified by evaluating the seismic event tree using component fragility information. This produced a fragility curve for each core damage sequence, which defined the conditional fraction of times the sequence occurred as a function of ground motion level. A total plant level fragility was then determined that quantified the conditional fraction of times core damage occurred as a function of ground motion level for all seismic sequences.

In the second step the frequency of core damage was estimated by combining the sequence fragility curves generated in the first step with the seismic hazard (frequency that ground motion levels occur). The frequency of occurrence of each sequence as well as the total frequency of core damage was determined.

A point (mean) estimate of the seismic risk was determined and an uncertainty analysis was performed. To quantify the uncertainty in the seismic risk, the uncertainty in each part of the analysis was propagated through the analysis. In the first step of the quantification process, the uncertainty in component capacities was propagated through the plant logic model to quantify the uncertainty in the sequence fragility curves. The product of this evaluation was a probability distribution on the conditional failure fraction as a function of ground motion level. By combining the uncertain sequence fragility curves with the family of hazard curves, the probability distribution on the frequency of core damage was determined.

Sequences contributing most to core damage were as follows (also see Table 4):

Failure of circulation of D_2O in sequence 4 was dominated by relay chatter that caused closure of rotovalves in the D_2O system. Relay chatter was a dominant contributor to sequence four because relays control motion of fast acting rotovalves that could operate and completely close off D_2O flow to the core. The operating circuit for all rotovalves is housed in a single unanchored electrical cabinet at ground level in the reactor building. Analysis indicated that for relay chatter to cause closure of the rotovalves, the close contacts must be made up by vibration that continued for 1/2 second, the open contacts must not be made up, the stop relays must not vibrate closed, and the unanchored relay cabinet housing the contacts must not be overturned. A fault tree with this logic was constructed and analyzed to evaluate the probability of seismically-initiated valve closure from relay chatter.

Failure of cooling water piping in sequence 17 was dominated by failure of underground piping from soil consolidation. A complicating factor in this sequence was the fact that the piping material was subject to non-ductile cracking during the coldest weather. A probabilistic analysis was used to determine the probability of concurrent occurrences of coldest weather with damaging earthquakes to determine the fragility of this underground piping.

Failure of the 25 million gallon cooling water basin in sequence 16 was dominated by failure of the Building 186 cooling water reservoir from soil consolidation. The cooling water reservoir was constructed with no requirements for seismic design, but had been shown to remain intact for design basis earthquakes with no soil consolidation. During normal operation water from the Savannah River is continuously pumped to the cooling water basin, and then through the reactor heat exchangers to dissipate heat. The river water system consists of about 50 miles of sections of reinforced concrete underground piping that was not constructed with seismic requirements, and was assumed in the analysis to fail at low seismic challenges. This failure coupled with failure of the cooling water reservoir depletes the supply of cooling water for the reactor and leads to failure because cooling water cannot be conserved by recirculation to and from the cooling water reservoir.

Loss of river water supply to the cooling water reservoir followed by failure of the operating crew to respond to the loss in sequence 2 was dominated by failure of operating personnel to respond to the alarm for loss of cooling water supply to the cooling water reservoir. The reservoir contains a significant amount of water and several hours are available to permit response by the operating crew. The values estimated for operator error used an analysis methodology[1] that permitted evaluation of the human error probability for the presence of several crew members, for times available for coping with the accident ranging from one minute to several hours, and for various stress levels.

Failure of the integrity of the D_2O system in sequence 5 resulted from a small leak in the D_2O system and failure of the operating personnel to isolate the leak. Failure of the operating crew dominated the failure probability for the sequence. The small leak was calculated to be caused by failure at low seismic challenges of the tank used to pressurize seals for the main circulating pumps. The seal system was calculated to have a 50% probability of failure at a peak ground acceleration of only 0.21 g. The same methodology[1] used to evaluate operator action in sequence 2 was used to evaluate the probability of operator error in not isolating the leak.

Values estimated for operator error had a significant impact on the calculated core damage frequency. The best estimate of operator error was determined using the methodology of [1], which provided for evaluation of errors with varying number of personnel, amount of time available, and stress level. When the values were changed from these values to more pessimistic values, the calculated core damage frequency increased by a factor of at least three.

The error probability for performing a specific task by one operator under low stress conditions was determined from the internal events analysis. Low-stress conditions were considered to occur following seismic events if normal electrical power was available. The probability of operator error under stressful conditions depended upon the time available to perform the required actions, which in turn depended upon the particular sequence of events. The following equation (derived from information in Ref. 1) was used to calculate the total operator error probability for the limiting sequence requiring the most operator actions in the minimum time.

$$HEP_6 = 0.9\,(t_{4\,op})^{-0.67} + 0.75\,(t_{2op})^{-0.53} + 4.8 \times 10^{-5}(t_{sd})^{-0.46} \qquad (1)$$

where HEP_6 is the operator error for sequence 6.

The total time available during sequence 6 in minutes is (When river water is not available, 7.9 hrs is the time it takes to drain the cooling water basin without recirculation of cooling water effluent):

$$T6 = 7.9\ hr(60) = t_{4op} + t_{2op} + t_{sd} \qquad (2)$$

Let t_{sd} = time for reactor shutdown = 1 minute for each of two actions to maximize this error.

Let t_4 be the time for each of the six actions with four operators in the control room, which are considered equal in duration.

Let t_2 be the time for each of the three actions with two operators outside the control room, which are considered equal in duration.

The time available for actions with four operators (t_{4op}) is calculated from Eqs. 1 and 2 with t_{sd} set to one minute. The distribution of the probability of operator error was calculated by varying t_{4op} in 5 minute steps from 5 minutes to 137 minutes and was then assumed to be represented by a lognormal distribution from which an error factor was determined. The calculation of operator error for other sequences with different operator actions was performed similarly in order to derive the operator error probabilities and error factors shown in Table 5 under the column Seismic Error Factor. Operator errors determined for the internal events analysis are also shown, and the final adjustment made to encompass both factors.

Benefits of the emergency cooling water system in reducing risks from seismic events were estimated to be small, and upgrades to strengthen the seismic resistance without concurrent upgrades to emergency power systems did not significantly decrease the risk of operation.

Values used to estimate the seismic hazard for the Savannah River Site had a significant impact on the calculated risk of operation. As part of the SRS seismic PRA effort, Lawrence Livermore National Laboratory (LLNL) was contracted to use the results of the Seismic Hazard Characterization Project (SHCP[2]) to calculate the seismic hazard at the SRS. While the seismic hazard methodologies used by the Seismic Owners Group/Electric Power Research Institute (SOG/EPRI) and LLNL are similar, the two seismic hazard estimates for the SRS are different. Figure 2 shows the SHCP seismic hazard results for the Savannah River Site[3] in terms of Peak Ground Acceleration (PGA). A comparison of the SOG/EPRI and LLNL results indicates there is a major difference in the estimate of the 0.85 fractile hazard curves and the mean. In comparison, in most cases the median or 0.50 fractile level hazard curves are typically within a factor of 5 or less of each other. Reasons for the differences in these estimates of the SRS seismic hazard are not completely understood at this time, but are being investigated.

Referring to the median (0.50 level) plant fragility of Figure 3, the median capacity of the reactor is approximately 0.64g S_a (Spectral Acceleration) and the plant HCLPF (High Confidence Low Probability of Failure) value is 0.18g S_a. In terms of PGA these values are 0.30 g and 0.085 g, respectively, as compared to the plant seismic design basis of 0.20g (PGA). The median capacity of the plant suggests that the reactor, as modeled in the base case without major upgrades, can withstand the impact of a design basis seismic event. However, the relatively low HCLPF level suggests a limited margin exists above the design basis earthquake magnitude.

IV. FIRE ANALYSIS

The fire analysis was performed with the assistance of Sandia National Laboratory utilizing methods described in [4]. It characterizes the fire induced core melt frequency for K-reactor as of June 1987, with no postulated or installed upgrades or modifications since that time included.

A unique aspect of the K-reactor fire analysis was the availability of an extensive operating history and site specific fire experience. Between 1958 and 1987, 94 reactor operating years and 20 significant fire events were recorded. This led to the availability of plant specific fire frequencies for the reactor building and the diesel generator buildings (2) of 0.12 and 0.03 per year respectively. A control room fire has never occurred at a SRS reactor.

The extensive operating data allowed the quantification of the core damage frequency with both plant specific data and with commercial fire data[5] updated using bayesian techniques[6] and the plant specific data. The results of the core melt frequency quantification with either set of data were virtually identical (within a factor of 2). The core melt frequency reported (mean of 1.4×10^{-7}) utilizes the bayesian updated fire occurrence frequencies since this was the more conservative of the two results.

There are a number of important plant-specific design features which cause the fire-induced core damage frequency at K-Reactor to be lower than that for other reactor plants even though identical analytical techniques were applied.

1. The cooling water system can sustain a complete loss of pumping power and still supply sufficient cooling water for decay heat removal and to all safety system heat loads after shutdown by operating in a gravity feed mode.

Critical cooling water system isolation valves cannot shut with the cooling water pumps running, and their cabling is not routed in similar or adjacent areas to power cabling for coolant system pumps. Therefore, even if a fire-induced spurious valve actuation signal would occur the system would still be functional, since a fire cannot both stop the pumps and close the valves.

2. The process water pump DC motor power is independent of the control room. There are no plant areas where a fire can fail any more than three of the six process water (Bingham) pumps. After reactor shutdown, the process water AC motors are tripped and no more than three DC driven pumps are required to remove decay heat. Therefore, additional random failures of the process water pump systems are required to fail the pumps.

3. There are several plant areas where fire induced damage can totally fail the ECCS system, but these areas are widely separated from DC power cable routing for the Bingham pumps. Therefore, random failures of the process water system must again occur to cause core damage. One cable run was discovered with control cabling for valves in both the ECCS and process water systems. However, a minimum of five spurious actuations would be required to fail the process water system. This scenario was judged to be probabilistically insignificant.

4. No fire induced method for defeating all shutdown systems (safety rods and SSS - poison injection) was identified.

V. WEATHER RELATED EVENTS

The risk of reactor building flooding from external sources is negligible due to the siting of the reactor. The K reactor building is sited at a minimum of 270 ft above mean sea level (MSL). The nearest significant body of water is L lake which is estimated to have a probable maximum flood (PMF) stage which is still 80 ft below K-reactor grade. The Savannah River (5 miles from K-reactor) PMF stage is over 130 ft below K-reactor grade.

The only significant flooding scenario involves dam failures upstream of SRS on the Savannah River destroying the River Water Pump houses and depriving the reactor areas of the normal source of makeup water to the 25 million gallon basins. Dam failures are modeled as an initiator in the internal event loss of river water sequences and are not explicitly treated as an external event.

The effects of the majority of weather related events (ice and hail, snow, etc.) on reactor operation are only from a loss of electric power standpoint. The reactor building which contains virtually all of the equipment necessary to shutdown and cool the reactor is constructed of reinforced concrete which varies from 7 ft to 1.5 ft in thickness and was designed to withstand blast pressure of 1000 lbs/ft^2. It has been shown that even tornado missiles can not damage the reactor directly[7].

One tornado scenario which can directly affect the reactor system involves the river water pump houses in much the same manner as dam failures do. The frequency of tornado damage to the pump houses is therefore again included as an initiator in the internal events loss of river water sequences and is not explicitly treated as an external event.

VI. HUMAN RELATED EVENTS

Transportation accidents were analyzed in detail. The only conceivable direct impact on the reactor itself would be from an aircraft impact. Since K-reactor is 20+ miles from an airport, not on an airway (5 miles from a low altitude airway) and built with blast resistant construction, the risk from aircraft impact was shown to be negligible. Also, the nearest public highway is 2 miles away, the nearest pipeline 17 miles away, the nearest public railway 2.8 miles away and there is no ship traffic on the Savannah River.

Considering the risk from chemical releases from transportation accidents, onsite transportation risks were assumed to dominate the risk from public sources due to the distances from public thoroughfares and the fact that many different types of flammable/ toxic/ radioactive materials are transported around the SRS in much closer proximity to the reactors.

A study of the risks from onsite transportation of hazardous substances has been performed for the SRS[8]. Utilizing the frequency and type of releases described in this study and postulating that any release would cause control room evacuation, the accident frequencies are coupled with the conditional frequency of core melt (calculated in the Internal Events analysis) due to loss of heat sink to arrive at a core melt frequency from onsite transportation accidents of approximately 10^{-7} per year.

A final human related event which turned out to be important was a scenario involving the dropping of a heat exchanger during changeout. If a heat exchanger were to be dropped back into the heat exchanger bay, the 100 ton heat exchanger could cause a primary or secondary system LOCA event. If irradiated fuel were still in the reactor, damage could occur. Since heat exchangers have been removed with hot fuel in the reactor (at least twice since 1971), a very conservative analysis was performed to assess the core damage frequency from this scenario. It is estimated that one heat exchanger is replaced approximately every two years. However, during an unknown (majority) of these changeouts, there is no fuel in the tank. It was not possible to substantiate this, however, so a changeout frequency of 0.5 per year was used.

The scenario is effectively driven by crane operator error since there are no mechanical safety features on the crane to stop a load drop, and a drop probability per changeout of 1.0×10^{-3} per changeout was assumed. Also, no credit was taken for the fact that the scenario must occur a minimum of 10 to 15 hours after reactor shutdown in order for the heat exchanger to be removed from the system itself, and a conditional probability of core melt due to a LOCA event of 0.7 from the Internal Events analysis was assumed. A more realistic analysis is planned, but the preliminary results indicate a core damage frequency from this scenario to be on the order of 10^{-4} per year.

Several changes in procedure or hardware would substantially change these numbers. Obviously, if heat exchangers were never removed with fuel in the tank, the core melt frequency would reduce to 0.0. Over a factor of 10 reduction to 8.4×10^{-6} per year is achieved by merely not allowing changeout while fuel is in the core during normal shutdown periods (assuming a constant unscheduled replacement rate of 0.04 = two unscheduled replacements since 1971 in approximately 50 reactor years). Hardware changes to the crane to make it difficult for the load to drop upon operator error (drop speed limiting devices, etc.) would allow the mechanical reliability factors to become dominant and thus lower the accident frequencies greatly.

ACKNOWLEDGMENT

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

REFERENCES

1. R. J. BUDNITZ, H. E. LAMBERT, E. E. HILL, "Relay Chatter and Operator Response After a Large Earthquake," NUREG/CR-4910, August, 1987.

2. D. L. BERNREUTER, J. B. SAVY, R. W. MENSING, AND J. C. CHEN, "Seismic Hazard Characterizatron of 69 Nuclear Plant Sites East of the Rocky Mountains, Methodology, Input Data, and Comparisons to Previous Results," NUREG/CR-5250, UCID 21517, Vol 1-8, January, 1989.

3. J. B. SAVY, "Seismic Hazard Characterization of the Savannah River Site," (SRP), UCID-21596, November, 1988.

4. M. P. BOHN, AND J. A. LAMBRIGHT. "Recommended Procedures for Simplified External Event Risk Analyses," (draft) Sandia National Laboratories, Albuquerque, NM, NUREG/CR-4840 February, 1988.

5. "Users Guide for a Personal-Computer-Based Nuclear Power Plant Fire Data Base," Sandia National Laboratories, Albuquerque, NM, SAND86-0300, NUREG/CR-4586, August 1986.

6. R. L. IMAN, S. C. NORA, "Modeling Time to Recovery and Initiating Event Frequency for Loss of Off-Site Power Incidents at Nuclear Power Plants," Sandia National Laboratories, Albuquerque, NM, SAND87-2428, NUREG/CR-5032, January 1988.

7. D. SHARP, "Tornado Hazards to Production Reactors at Savannah River Plant," DPST-86-579, July, 1986.

8. "Evaluation of Accident Risks in the Transportation of Hazardous Materials by Truck and Rail at the Savannah River Plant," WRSC-RP-89-715, September 1989.

Seismic and Fire	**Weather related events**	**Flooding**	**Other nature related events**	**Human related events**
	Extreme cold	Flash Flood	Forest fires	Transportation accidents i. Aircraft crash ii. Train/Vehicle crash iii. Ship crash iv. Pipeline explosion
	Ice storm	Area Flood	Volcanoes	
	Snow storm	Dam failure	Glaciation	
	Hail	Tsunami	Animal/biota	
	Extreme heat/drought	Seiche	Meteorite	
	Dust Storm			River contamination
	Severe electrical storm			Turbine missile
	High wind i. Hurricane ii. Tornado			Crane failure/ construction forces
				Site chemicals/other SRS facilities
				Firearm discharge

Table 1. Categorization of External Events Eonsidered

EVENT	CORE MELT FREQUENCY (PER YEAR)	CORE MELT FREQUENCY (PER YEAR) TO BE ADDED TO INTERNAL EVENTS RESULTS
Seismic	1.2×10^{-4}	1.2×10^{-4}
Crane failures	$< 10^{-4}$	$< 10^{-4}$
Flood	1.5×10^{-6}	included in internal events analysis
Wind	5.8×10^{-7}	included in internal events analysis
Fire	1.4×10^{-7}	1.4×10^{-7}
Transportation	$< 10^{-7}$	$< 10^{-7}$
Total external event core damage frequency	2.2×10^{-4}	2.2×10^{-4}

Table 2. External Event Core Melt Frequencies

COMPONENT	Median (G) PGA	S_a	β_r	β_u	FAILURE MODE
REACTOR BUILDING	1.15	2.44	0.27	0.38	SHEAR WALL
COOLING WATER RESERVOIR	0.47	1.00	0.40	0.5	SOIL CONSOLIDATION
DIESEL FUEL TRANSFER PUMP	1.03	2.18	0.24	0.28	INCIPIENT SLIDING
LUBE OIL HEAT EXCHANGER	1.45	3.07	0.25	0.46	ANCHOR BOLTS
EMERGENCY BUS ELECTRICAL CABINET	0.36	0.76	0.24	0.24	ROCKING, not anchored
HEAT EXCHANGER	0.92	1.95	0.23	0.34	CONNECTION BOLT
SEAL HEAD TANK	0.28	0.59	0.26	0.33	ROCKING
CAT DIESEL	>3	>6	0.30	0.30	ANCHOR BOLTS
CONDUIT/CABLE TRAY	0.92	1.95	0.26	0.51	GROSS DISTORTION OF CONDUIT/TRAYS
GM DIESEL BATTERIES	0.31	0.66	0.31	0.52	BLOCK WALL FAILURE
RELAY CHATTER, MAIN REACTOR VALVES	0.19	0.40	0.26	0.24	INCIPIENT ROCKING
UNDERGROUND PIPING	2.0	4.24	0.42	0.60	VIBRATORY MOTION
COOLING WATER PIPING, NIL DUCTILITY	1.45	3.07	0.49	0.50	PIPE RUPTURE
EFFLUENT SUMP BUILDING	1.65	3.50	0.40	0.45	SOIL CONSOLIDATION

Table 3. Representative Fragility Values

SEQUENCE	DESCRIPTION	MEAN FREQUENCY
4	LOSS OF PROCESS WATER CIRCULATION	3.94×10^{-5}
17	FAILURE OF COOLING WATER PIPING	3.16×10^{-5}
16	FAILURE OF COOLING WATER RESERVOIR	2.31×10^{-5}
2	LOSS OF RIVER FLOW TO RESERVOIR & FAILURE TO RESPOND IN TIME	1.81×10^{-5}
5	FAILURE OF PROCESS WATER SYSTEM	3.33×10^{-6}
ALL OTHERS	VARIOUS	6.47×10^{-6}
	TOTAL	1.22×10^{-4}

Table 4. Seismic As-Is Results

EVENT DEFINITION	ERROR FACTOR FROM SEISMIC ANALYSIS	ERROR FACTOR FROM INTERNAL EVENT ANALYSIS	OVERALL ERROR FACTOR	ERROR PROBABILITY
Failure of operating crew to respond to loss of cooling water flow to reservoir, normal electric power available	-(no stress)	3 to 4	3 to 4	1.8×10^{-3}
Failure of operator to throttle effluent valves, no power available	1.4	3 to 4	6	3.9×10^{-2}
Failure of operator to start emergency generator	1.4	3 to 4	6	1.9×10^{-2}
Failure of operator to isolate pump seal leakage & provide alternate seal pressure source, no power available	1.9	3 to 4	6	1.9×10^{-2}
Failure of operator to actuate explosive valves to shutdown reactor	4.0	10 to 40	60	5.6×10^{-5}

Table 5. Selected Error Probability for Operator Actions Following Seismic Events

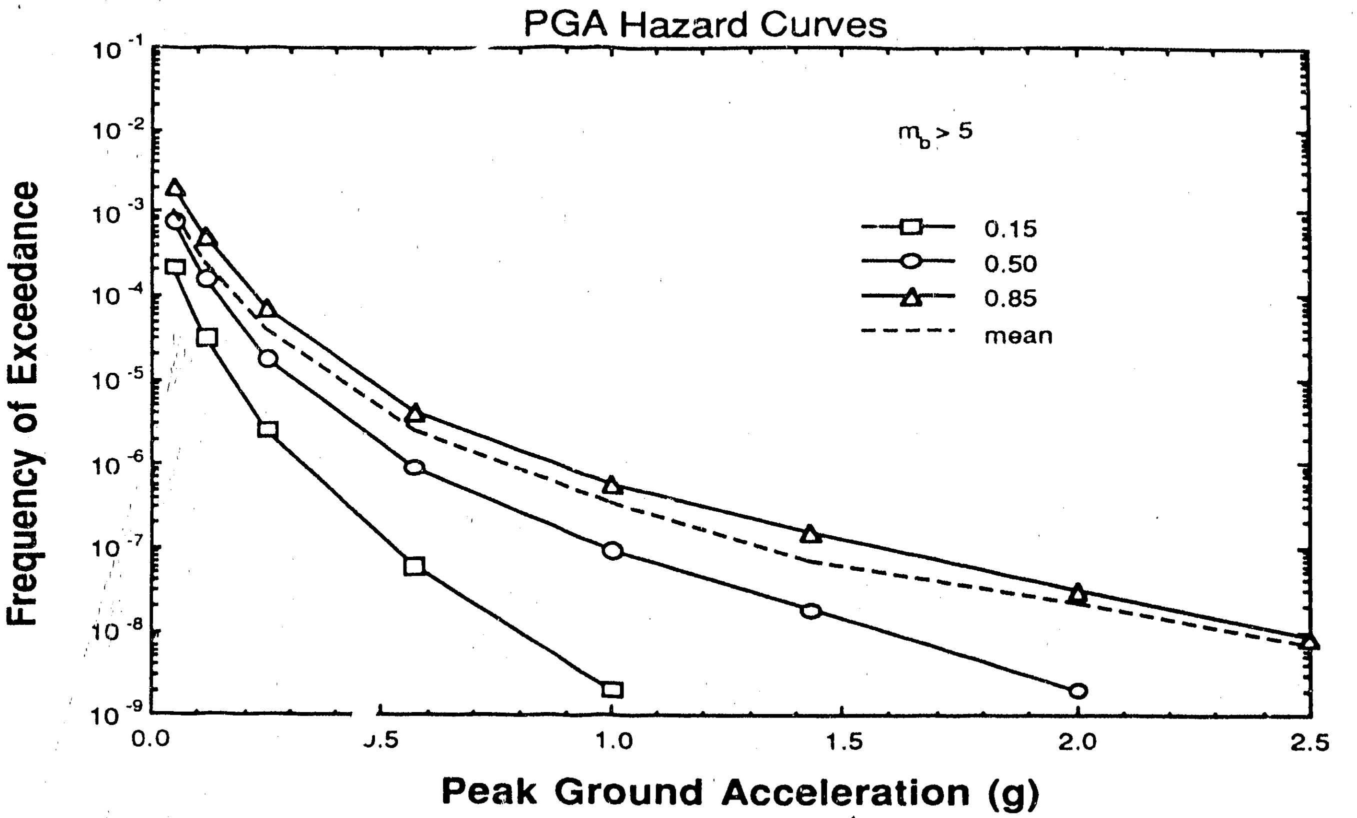

Figure 1. Mean Hazard Curves for the SRS

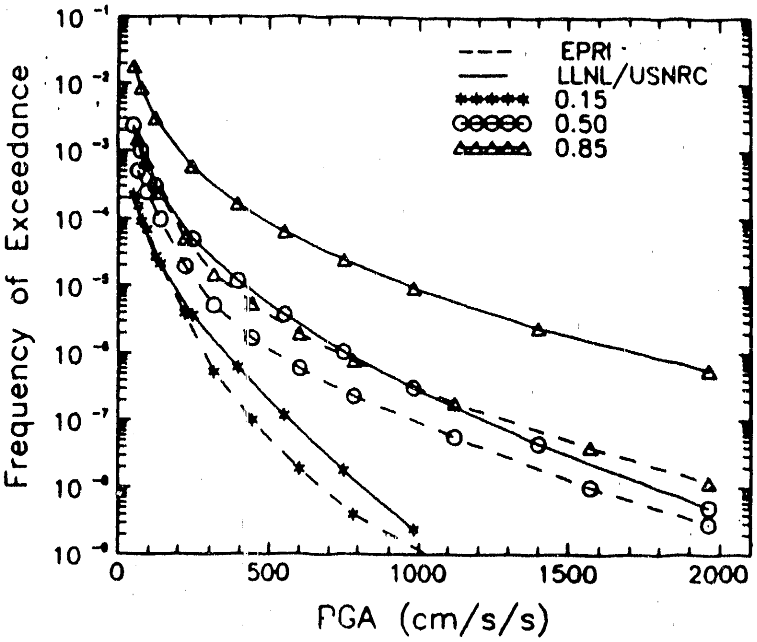

Figure 2. EPRI Fractile Hazard Curves

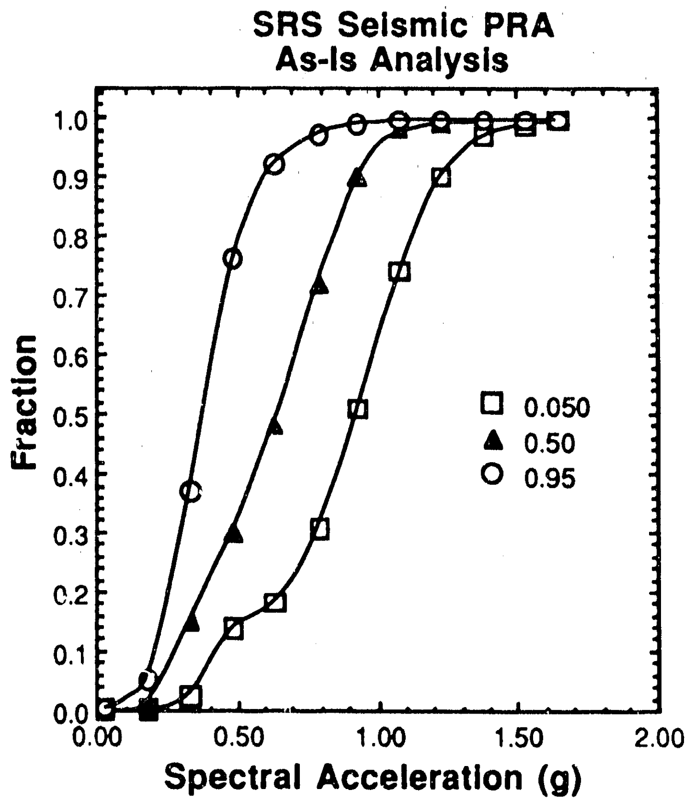

Figure 3. Spectral Acceleration

END

DATE FILMED

01 / 03 / 91