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EXTERNAL EVENT PROBABILISTIC RISK ASSESSMENT FOR THE HIGH FLUX ISOTOPE REACTOR (HFIR)

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ABSTRACT

The High Flux Isotope Reactor (HFIR) is a high performance isotope production and research reactor which has been in operation at Oak Ridge National Laboratory (ORNL) since 1965. In late 1986 the reactor was shut down as a result of discovery of unexpected neutron embrittlement of the reactor vessel.

In January of 1988 a level 1 Probabilistic Risk Assessment (PRA) (excluding external events) was published as part of the response to the many reviews that followed the shutdown and for use by ORNL to prioritize action items intended to upgrade the safety of the reactor. A conservative estimate of the core damage frequency initiated by internal events for HFIR was 3.11×10^{-4}. In June 1989 a draft external events initiated PRA was published. The dominant contributions from external events came from seismic, wind, and fires. The overall external event contribution to core damage frequency is about 50% of the internal event initiated contribution and is dominated by seismic events.

INTRODUCTION

The High Flux Isotope Reactor (HFIR) is a high performance isotope production and research reactor which has been in operation at the Oak Ridge National Laboratory (ORNL) since 1965. Its main missions are the production of transuranic and cobalt isotopes, materials irradiation research, and neutron scattering research.

In late 1986 a special internal post-Chernobyl review of HFIR discovered unexpected neutron embrittlement of the reactor vessel. As a result of the discovery the reactor was shutdown in November 1986. The Department of Energy (DOE) and ORNL began an extensive review of the reactor design, safety, operation, maintenance, and management. Over twenty reviews of various depths have been conducted to date by DOE, ORNL and independent oversite groups such as the National Academy of Science/National Research Council and the Advisory Committee on Nuclear Facility Safety.

Partly as a result of this review process, a Probabilistic Risk Assessment (PRA),[1] of HFIR was completed for internal initiated events in January 1988. This was the first PRA on a large research reactor in the United States. The PRA initiated by external events was completed in draft form in June 1989. The approach (used for the external events initiators) and results of the external events assessments will be presented in this paper.

HIGH FLUX ISOTOPE REACTOR DESIGN

The HFIR is an 85 MW flux trap reactor. A schematic of the reactor is contained in Figure 1. It is water cooled and beryllium moderated. It operates at 468 psi pressure with an inlet temperature of 120°F and outlet temperature of 158°F. The peak thermal flux in the flux trap is 5 x 10^{15} n/cm^2-sec which makes the HFIR the highest thermal flux reactor in the world. The core of the reactor is small (17 1/2 inches diameter, 24 inches in height) with a 5 inch diameter target hole through its center. The core contains about 9.6 kg of highly enriched (93%) U^{235}, arranged in two concentric cylindrical elements. The inner element contains 171 involuted plates and the outer 369 involuted plates. The core is made up of a U_3O_8/Al mixture clad in aluminum. The core is replaced every 24 days. The Be moderator surrounds the core and is about 1 ft thick. Control is achieved by 4 safety plates arranged in a cylinder around a solid control cylinder. The outer cylinder is raised and the inner lowered to increase reactivity and keep a symmetric flux profile. These control cylinders are sandwiched between the core and the Be reflector and are composed of Eu_2O_3, and Ta.

The reactor core is contained in an 8 foot diameter pressure vessel that is about 19 ft high. The pressure vessel is located near the bottom of a large pool (36 ft deep and about 18 ft across) containing 85,000 gallons of water.

The pressure of 468 psi is maintained by compressing the primary system water using a pressurizer pump in combination with a system of letdown valves. The flow (16,000 gal/min) is achieved by 3 out of 4 AC motor driven primary pumps and it is downward through the core and target regions. Decay heat is removed using a small DC motor to drive the primary pumps. The power to the DC motor is supplied using a dedicated battery power supply or by using off-site power, on-site diesel generators, or portable diesel generators (AEPG's) connected to inverters.

A schematic of the HFIR process flow system is included in Figure 2 and a schematic of the electrical power distribution system is included in Figure 3.

The reactor is contained in a large reactor building 128 x 160 x 110 ft which is maintained at a slight vacuum. Exhaust fans continuously pull air from the building through a series of filters and exhaust up a 250 ft stack. The building, filters, fans and stack act as a dynamic confinement in the event of an accident.

The reactor was built in 1965 to Uniform Building Code Seismic Standards resulting in a seismic design acceleration of about 0.08g's. The primary coolant system was upgraded in 1987 to enable it to withstand 0.15g's, which is the safe shutdown earthquake for the HFIR.

Since the external events Probabilistic Risk Assessment (PRA) made extensive use of the earlier published internal event initiated assessment, a brief summary of the results of the internal PRA are presented.

HFIR PROBABILISTIC RISK ASSESSMENT (INTERNAL EVENTS)

The HFIR PRA[1] was developed with several uses in mind. Foremost, it was required by the DOE design review team; within ORNL it is used for safety improvement and to help prioritize the many design and administrative changes required by the numerous review committees. In addition it is also used for operator and engineer training, emergency planning, technical specification modification, maintenance improvements, and to help define and document the safety design basis of the plant.

The project was subcontracted to Pickard, Lowe and Garrick Inc. (PL&G), Newport

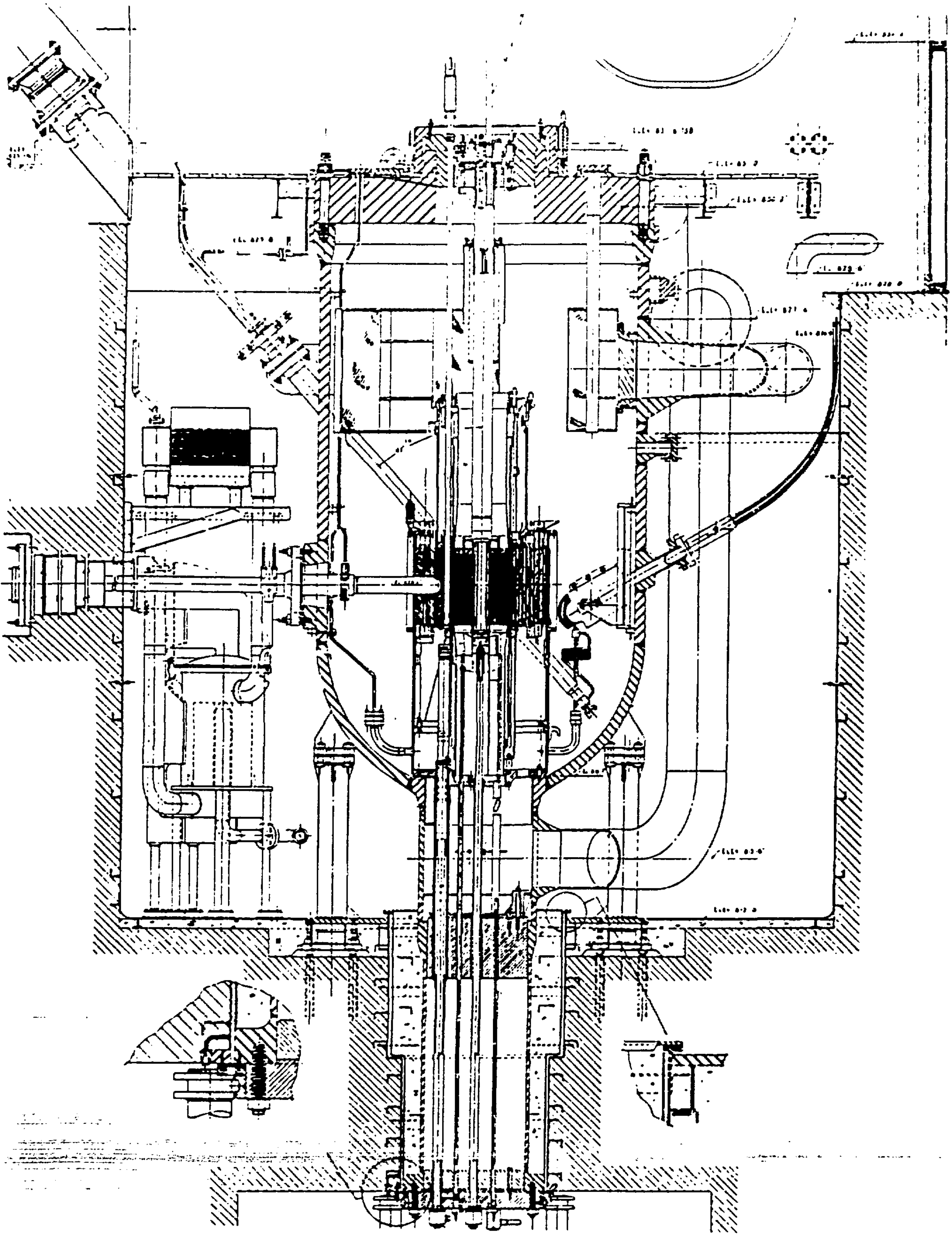

Figure 1. Vertical Section of HFIR Reactor Vessel and Core

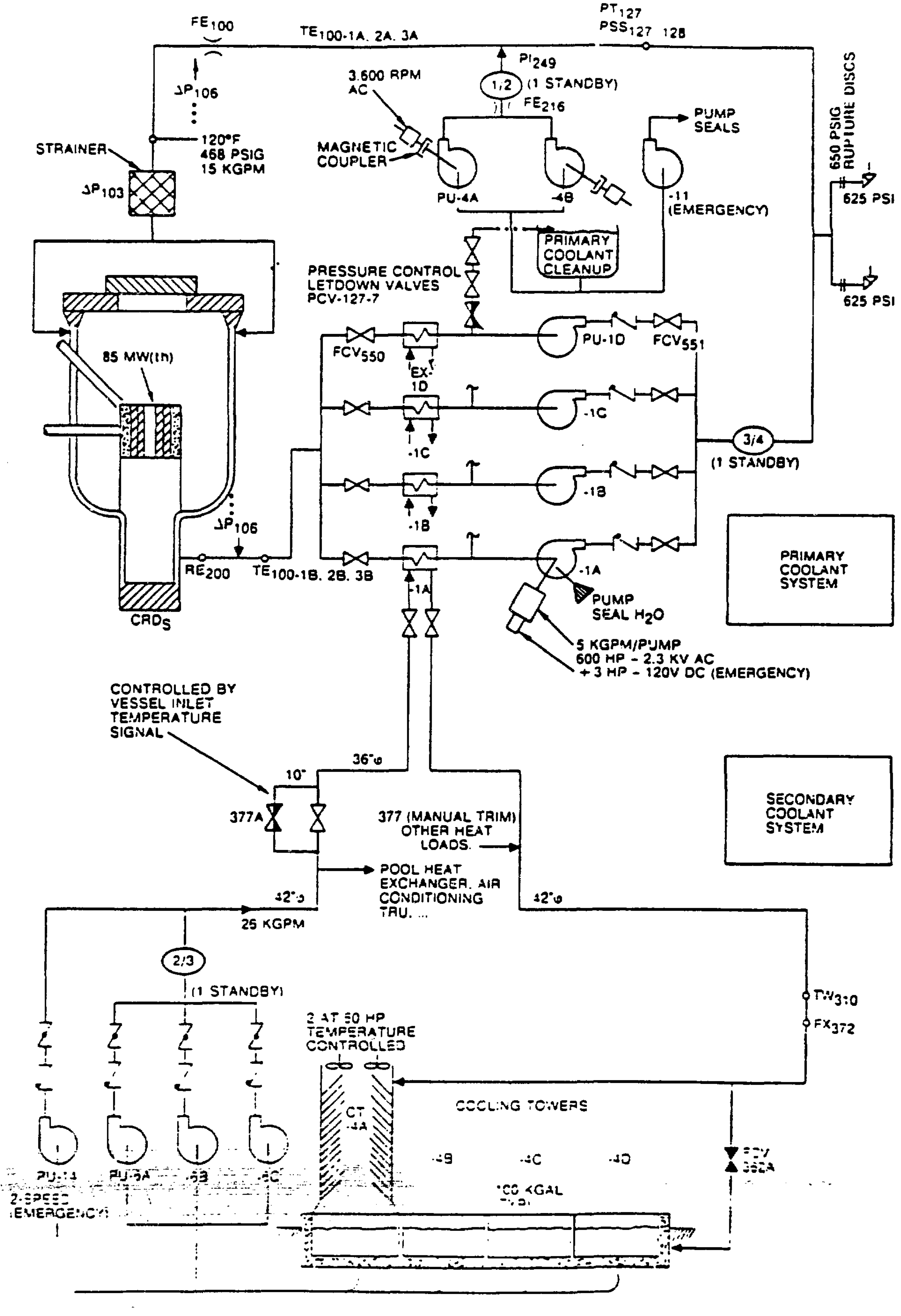

Figure 2. HFIR Process Flow Schematic (primary and secondary systems)

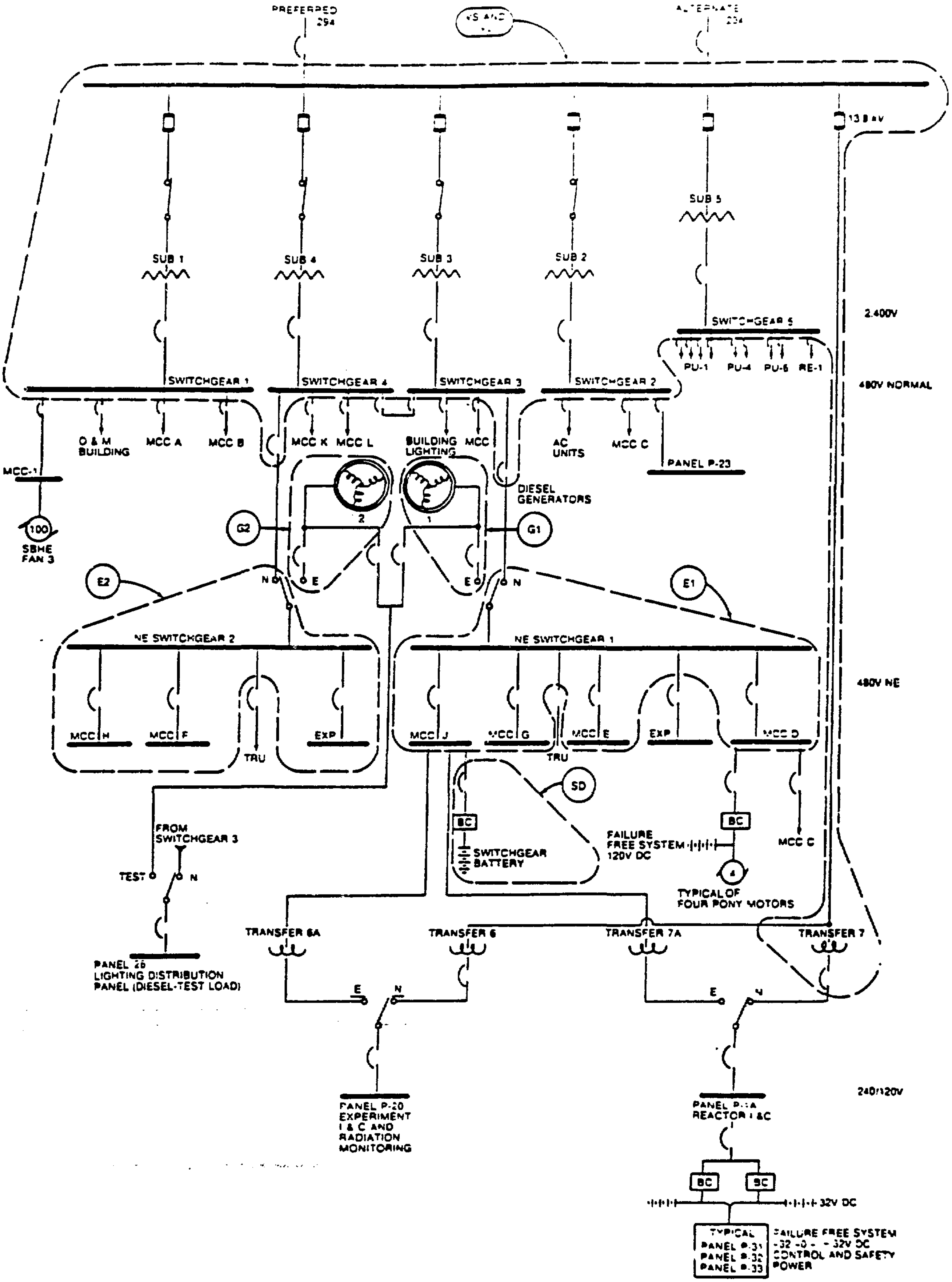

Figure 3. HFIR Electrical System Schematic

Beach, CA and work began in July 1987 with the final report (excluding external events) issued in January 1988. Several basic assumptions were set forth to guide the work. These are shown in Table 1.

The system was modeled using the so-called PL&G methodology commonly used on several commercial nuclear power plants. It consists of large event trees with only sparse use of fault trees. This is different than the approach used in the classic Reactor Safety Study (WASH-1400)[2] and used by the Nuclear Regulatory Commission in several of their internal assessments.

The internal initiating events were selected by applying the following six steps: (1) examine the 20 years of operating history and the quarterly technical reports, (2) review the HFIR Accident Analysis Report,[3] (3) review the HFIR design, drawings, and operational procedures, (4) hold discussions with the original HFIR design team, (5) extensively review the incidents at other research reactors and applicable commercial nuclear power reactor experience, and (6) create a master logic diagram (MLD) which generally examines how the HFIR core could be damaged.

Because of the simplicity of the HFIR design, the plant models were rather simple to develop and were expressed using event sequence diagrams (ESD) in addition to using event trees. The ESD was found to be helpful in explaining accident scenarios to review teams, management, and operators. For the most part, the HFIR PRA used established assessment methods.

There were two initiating events, however, that were unique to HFIR and required a new innovative approach. These were flow blockage events and fuel damage resulting from manufacturing defects and assembly errors.

Flow Blockage

Because of the very high power density and narrow fuel channels in the HFIR and experience at other research reactors it was acknowledged by the original designers that core damage could occur as a result of small flow blockages. In order to reduce this vulnerability, a strainer containing small orifices was inserted in the inlet primary coolant pipe, and the plant confinement, shielding, and water clean-up systems were designed to accommodate a core melt which might be expected to occur during the lifetime of the plant.

There are no system models or explicit data to use for modeling and quantifying flow blockage scenarios, since such blockages are not expected in the commercial nuclear power plants.

A very structured expert opinion approach[4] was developed by S. Kaplan of PL&G which used the HFIR designers, operators, and engineers as the knowledge base.

Fuel Element Defects

Because of the high power density and narrow flow channels, fuel defects associated with the manufacturing and assembly of the fuel plates and elements could cause narrowing of the flow channels, in turn causing surface hot spots or flow starvation. These may be of sufficient severity so as to cause fuel damage.

Interaction with fuel experts at ORNL and Argonne National Laboratory identified a comprehensive list of 10 potential defects. Based on further interaction with fuel experts and detailed examination of the manufacturing process including the inspection and quality control procedures, the 10 defects were reduced to five categories which consist of 3 fuel inhomogeneities and 2 assembly errors. Specific details on the manufacture and inspection processes relating to the defects on the short list were obtained from the fuel manufacturer and the ORNL fuel experts. Based on the approach taken for the analysis of flow blockages, small event trees were developed and branch points were quantified using expert opinion coupled with data from fuel manufacturing, inspection and quality control procedures.

Table 1. Basic Assumptions Used in the HFIR PRA

1. Core damage will be defined as occurring at the onset of incipient boiling.

2. The reactor configuration assessed would be that at restart (includes power reduction and all pre-restart design modifications) with the addition of the portable diesel generators.

3. The probability of vessel failure would come from the "Evaluation of HFIR Pressure-vessel Integrity Considering Radiation Embrittlement" - ORNL/TM-10444, edited by R. D. Cheverton.

4. The plant specific HFIR data is to be used wherever possible and to the extent possible.

5. Consideration should be given to accidents which have occurred at other research reactors when exploring initiating events.

6. Results should be expressed in such a way as to facilitate ease in calculation of off-site consequences.

7. Models should be fluid (easily modified as the design changes) in order to make the assessment a "living PRA".

Results of the HFIR PRA (Internal Events)

In order to facilitate source term determination and subsequent off-site consequence analysis, the results of the HFIR PRA were expressed in terms of plant damage states. The plant damage state matrix is shown in Figure 4. The matrix categorizes the end state of an event tree as to (1) the extent of the damage, (2) whether the primary system is intact following the accident, (3) in case of a loss of coolant accident (LOCA), whether the break is inside or outside the reactor pool (which provides fission product scrubbing), and (4) whether power is available to one, two or all three exhaust fans.

The results of the PRA are also expressed in terms of frequency of core damage. Table 2 indicates the overall frequency of core damage as a result of internal initiated events.

EXTERNAL EVENTS PRA APPROACH

Following the internal events assessment the same subcontractor began to examine the risks associated with external event initiators. The initiators considered consisted of the 9 major categories below:

Seismic
Wind/Tornado
Fire/Smoke
Floods (External and Internal)
Spray (Steam and Water)
Explosions
Missiles
Caustic Attack
Falling Objects

Except for the first two initiators, the other contributors were all assessed using the same general approach. These initiators will be referred to as Internal Hazard Initiators (IHI).

Internal Hazard Initiators (Approach and Results)

The assessment of the internal hazard initiators begins with an identification of initiators and an assessment of potential interactions between the hazard and the plant equipment, referred to as spatial interactions. This is accomplished by an extensive examination of plant drawings, plant layout, and a detailed plant walk-down. In the case of the HFIR, 207 possible accident scenarios involving IHI's were identified for further analysis.

The 207 scenarios were grouped into one of 5 categories: (1) the scenario does not affect any safety system and does not cause any initiating events; (2) the scenario directly causes an initiating event; (3) the scenario affects one or more systems relative to plant safety; (4) the scenario has the potential to directly cause plant or core damage; and (5) the scenario has the potential to propagate to vital areas of the plant, especially the main and/or auxiliary control room.

A screening process to reduce the number of scenarios to a more manageable number for detailed analysis was implemented based on the following rules.

Scenarios in category 1 were eliminated from further analysis. Because of the importance of the main and auxiliary control rooms to the scram function of the reactor, all scenarios in category 5 were retained for detailed analysis. A conservative estimate of the frequency of occurrence and an assessment of the potential plant damage state was assigned to scenarios in the remaining categories 2-4. For each plant damage state, the scenarios were ranked by frequency. The scenario was eliminated from further analysis if the scenario contribution to a damage state was less than 2% of the frequency contribution of the internal events to that damage state.

After screening, 62 scenarios remained in categories 2-4 and 16 scenarios were retained in category 5. Fires appeared to dominate the internal hazard scenarios and are discussed in detail in the following paragraph.

After the screening, the fire scenarios

	EXTENT OF CORE DAMAGE											
	PARTIAL CORE DAMAGE						TOTAL CORE DAMAGE					
	PRIMARY SYSTEM INTACT?						PRIMARY SYSTEM INTACT?					
	YES		NO				YES		NO			
	—		BREAK IN POOL?				—		BREAK IN POOL?			
	—		YES		NO		—		YES		NO	
	SBHE AVAILABLE?											
	YES	NO	YES	NO	YES	NO	YES	NO	YES	NO	YES	NO
PLANT DAMAGE STATE:	1	2	3	4	5	6	7	8	9	10	11	12

NOTE: AN "E" SUFFIX IS ATTACHED FOR THOSE SEQUENCES IN WHICH CORE DAMAGE OCCURS EARLY IN CORE LIFE. A "B" SUFFIX INDICATES SCENARIOS WITH CONFINEMENT BUILDING FAILURE. A "D" SUFFIX INDICATES A SCENARIO WITH A DRY POOL.

Figure 4. HFIR Plant Damage State Matrix

Table 2. Initiating Event Categories, Mean Frequencies, and Contribution to Core Damage

Category	Mean Frequency (yr^{-1})	Contribution to Core Damage Frequency (percent)	Mean Core Damage Frequency
1A Manual Scram	21.4	7.1	2.2-5
1B Inadvertent Control Plate Drop	3.16	1.2	3.7-6
1C Inadvertent Scram	1.01-1	0.1	3.6-7
2A Complete Loss of Offsite Power	4.43-1	18.0	5.6-5
2B Loss of Preferred Feeder	4.10-1	2.4	7.4-6
2C Loss of Switchgear DC	3.79-3	0.1	3.9-7
3A Runaway Pressurizer Pump	1.34-1	1.5	4.8-6
3B Loss of Running Pressurizer Pump	9.65-1	0.5	1.6-6
4A L1 Scenarios	1.5-5	4.8	1.5-5
4B L2 Scenarios			
L2E	2.02-5	6.8	2.1-5
L2MCP	3.42-5	10.9	3.4-5
4C L3 Scenarios	2.00-5	6.4	2.0-5
4D L4 Scenarios (Total)	2.12-5	6.8	2.1-5
5 Small Break LOCA	4.56-3	5.1	1.6-5
6 Large Break LOCA	3.30-5	10.6	3.3-5
7 Beam Tube Failure	5.94-4	0.7	2.1-6
8A Reactivity Insertion	1.60-1	4.2	1.3-5
8B Degraded Secondary Cooling	2.32-1	0.6	2.0-6
8C Loss of Instrument Air	8.13-2	5.5	1.7-5
8D Degraded Primary Flow	2.68-1	7.1	2.2-5

Note: Exponential notation is indicated in abbreviated form; i.e., 2.9-5 = 2.9×10^{-5}.

Total (Internal Events) = 3.11-4.

were analyzed in detail as to (1) the occurrence rate of the fire, (2) the physical effect of the fire and (3) the response of the plant to the fire.

The frequency of the plant damage state due to a fire is a multiple of 5 factors: (1) the annual frequency of a fire in room (Z), (2) the fraction of fires in a specific area (J) of room (Z) (Geometry Factor), (3) fraction of fires in room (Z), area (J) with severity sufficient to cause damage (severity factor), (4) fraction of fires that are not suppressed before damage occurs to equipment (non-suppression factor) and (5) the conditional frequency of reaching damage state (X) due to failure of equipment exposured to a fire.

The annual frequency of a fire at HFIR outside of the main and auxiliary control room was estimated at 0.0117 fires/yr. This was based on the fire occurrence data from nuclear power plants modified by zero reported fires at HFIR since it went critical in 1965.

The main and auxiliary control room fire occurrence frequency was estimated to be 6.55×10^{-4}/yr and 5.49×10^{-4}/yr, respectively. These are generic frequencies modified by the HFIR information.

The dominant sequence initiated by a fire resulted in a total loss of all AC power to the HFIR (fire in the electrical building) and a failure to maintain decay heat removal capability using the Auxiliary Electric Power Generators (AEPG's). The mean frequency of this scenario was 1.08×10^{-5}/yr with the total fire initiated mean core damage frequency being 1.83×10^{-5}/yr. Thus 60% of the risk was a result of this single scenario.

All other internal hazard initiators were analyzed in the same manner as the fire hazard. Table 3 summarizes the internal hazard initiators contribution to the mean frequency of core damage for the HFIR.

Seismic Risk

The seismic risk analysis consisted of 5 steps: (1) determine the seismic hazard for the HFIR site (frequency of ground motion acceleration of various sizes), (2) perform a fragility analysis (response of structures and/or components to various magnitudes of ground acceleration), (3) analyze the plant response to the seismic failures resulting from steps 1 and 2, (4) obtain a mean (point estimate) of the core damage frequency and assign core damage states resulting from a combination of steps 1-3, (5) finally, perform an uncertainty analysis for those scenarios found to be dominant contributors to the seismic risk.

The seismic hazard curves were obtained by combining the site specific curves generated by the Electric Power Research Institute (EPRI) and those generated by Lawrence Livermore National Laboratory. The combination process did not include the results of Livermore ground motion expert #5, because this data set was an outlier by those performing the analysis. The sensitivity of the HFIR core damage frequency to this assumption will be addressed later in this paper. The statistical combination of the two current hazard curves was performed by Risk Engineering Inc., the results are portrayed in Figure 5.

The seismic fragilities for the HFIR were generated by EQE Engineering Inc. The components and structures for which fragilities were generated were based on the results from the internal events PRA. Earthquake characteristics, system damping, load combinations, model combination, combination of responses to earthquake directional components, and structural modeling considerations were all included in the development of the fragilities. The failure criteria for the majority of components was defined as a failure of the component to perform its design function and for structures, the failure criteria was defined as inelastic deformation sufficient to interfere with the operability of equipment. Table 4 show the fragilities for the components and structures which were used in the PRA.

The plant logic and assembly of seismic hazard and fragility information used current

Table 3. Contribution of Fire, Flood, and Other Environmental Hazards to Core Damage Frequency

Description	Contribution Core Damage Frequency	Percent of Internal Initiating Event Core Damage Frequency
Fire Scenarios	1.83×10^{-5} per year	5.88
Flood and Other Environmental Hazard Scenarios	1.81×10^{-6} per year	0.58
Fire, Flood, and Other Environmental Hazard Scenarios	2.01×10^{-5} per year	6.46
Internal Initiating Event	3.11×10^{-4} per year	

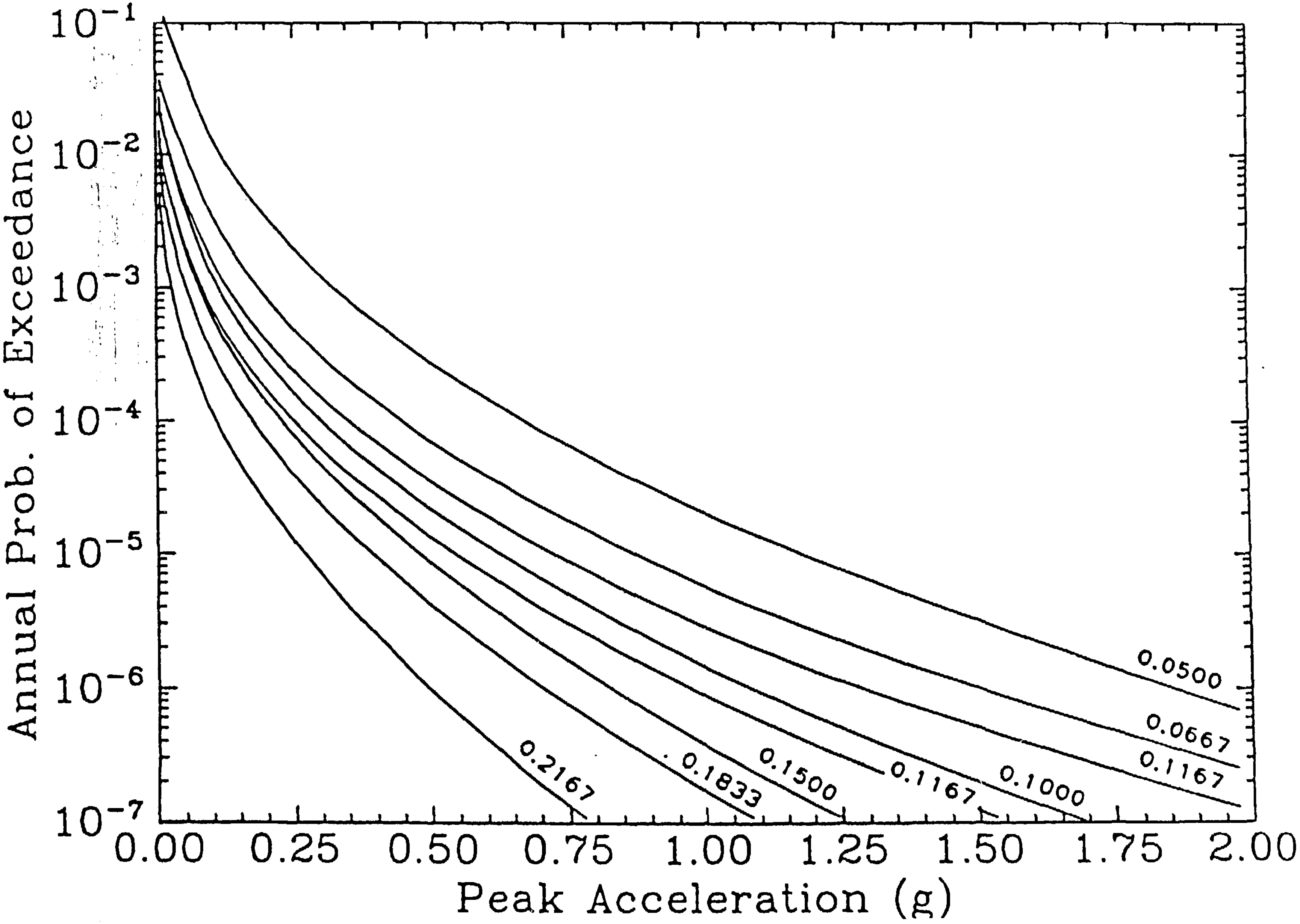

Figure 5. Aggregate Seismic Hazard Curves [EPRI and LLNL (w/o Expert #5)] for the HFIR Site

Table 4. **Seismic Fragility Summary**

Item Number	Component	$\tilde{a}$(g)	β_r	β_u	β_c	HCLPF (g)
1	Pool Caution Exchanger	0.11	0.29	0.20	0.35	0.05
2	Pool Cleanup Filter	.15	.29	.20	.35	.07
3	Electrical Building	.16	.38	.45	.59	.04
4	Vent Stack Inner Liner	.20	0.25	0.30	.39	.08
5	Pool Demineralizer	.23	.29	.20	.35	.10
6	Control Room Ceiling	.25	.25	.30	.39	.10
7	CHOG/SBHE MCC Block Wall	.25	.25	.30	.39	.10
8	Offsite Power	.31	.25	.43	.50	.10
9	Pool Anion Exchanger	.35	.29	.20	.35	.16
10	Reactor Building High Bay Leakage	.38	.26	.22	.34	.17
11	Reactor Building Control Bay Masonry Walls	.40	.26	.30	.40	.16
12	Water Wing Masonry Walls	.40	.26	.30	.40	.16
13	Instrument Air	.40	.26	.30	.4	.16
14	Reactor Building Control Bay Roof	.48	.27	.40	.48	.16
15	Reactor Building Observation Window	.48	.27	.40	.48	.16
16	Pony Motor Batteries	.54	.29	.19	.35	.24
17	Pool Deaerator	.77	.26	.20	.33	.36
18	Vent Stack Outer Stall	.83	.45	.44	.63	.19
19	Reactor Vessel Supports	.88	.32	.35	.47	.29
20	Pony Motor Battery Room Masonry Walls	1.0	0.26	0.30	0.40	0.40
21	Primary Pump Support	1.1	0.33	0.33	0.45	0.36
22	Precondenser C6-13	1.1	0.31	0.57	0.65	0.25
23	After Condenser C6-C	1.1	0.31	0.47	0.56	0.30

Table 4. **Seismic Fragility Summary** (continued)

Item Number	Component	≈ a(g)	β_r	β_u	β_c	HCLPF (g)
24	Retrofit Block Walls (primary shielding)	1.1	0.27	0.46	0.53	0.33
25	Reactor Building High Bay Frame	1.2	0.30	0.35	0.46	0.41
26	Control Rod Drives	1.2	0.20	0.35	0.40	0.48
27	Pool Demineralizer after Filter	1.2	0.29	0.20	0.35	0.54
28	Pressurizer Pumps	1.3	0.20	0.40	0.45	0.48
29	Reactor Building Substructures	1.5	0.30	0.35	0.46	0.51
30	Emergency Depressurization System Valves	1.5	0.20	0.50	0.54	0.47
31	CHOG/SBHE Steel Shed	1.7	0.35	0.40	0.53	0.50
32	Piping	1.8	0.30	0.50	0.58	0.48

accepted practices and is shown schematically in Figure 6.

The dominant seismic initiated-sequence involves a total loss of AC power to the HFIR and a failure to obtain and connect at least one of the two AEPG's within the 6 hours of battery lifetime. With the loss of the pumps to circulate the coolant, the decay heat cannot be removed resulting in core damage. The mean frequency of this scenario is 4.92 x 10^{-5}/yr. The total seismic initiated core damage frequency is 1.55 x 10^{-4}/yr. For the majority of the sequences, the reactor primary system is intact and/or the pool is available for scrubbing fission products. However, one dominant sequence does involve a seismically induced loss of coolant accident with loss of pool integrity with a mean frequency of 9.65 x 10^{-6}/yr.

High Winds and Tornados

The analysis of the effects of high winds and tornados on the HFIR follows the same approach as for seismic analysis. The steps are: (1) create tornado/wind hazard curves (frequency of wind events at various velocities), (2) perform a fragility analysis, (3) perform a tornado missile analysis, (4) combine steps 1-3 with the plant logic and obtain an estimate of core damage frequency and plant damage states, and (5) perform an uncertainty analysis.

The wind hazard curves were generated by EQE Engineering, Inc. Because of the large distances to the seacoast, hurricanes were excluded from the data sources. The Reinhold-Ellingwood model was used for tornado hazards with variability. The variability included (1) the uncertainty in plant site (2) uncertainty in the tornado data (frequency of occurrence, area, and path length), and (3) uncertainty in tornado damage area and length. Extratropical winds were derived from data from the Knoxville Weather Services and combined with the tornado frequency curves. The results for tornados and extratropical wind hazard are shown in Figure 7.

The wind fragilities were also generated by EQE Engineering Inc. The structures that were considered in the analysis included: (1) major elements of the reactor building and the control water wing; (2) the electrical building, and (3) the exhaust stack. The fragilities are shown in Table 5.

Potential tornado missile damage was assessed for the reactor building high bay and the control/water wing. The reactor high bay was protected by the thick concrete walls from penetration. The first floor frequency of penetration was less than 5.7 x 10^{-5}/yr, the control room level less than 7.8 x 10^{-5}/yr and the reactor building roof less than 5.8 x 10^{-5}/yr. Because of the location of internal barriers, presentation area, and restricted angles, the core damage frequency from missiles was less than 10^{-6}/yr. With such a low missile damage frequency, the plant logic models are initiated by failures associated with pressure changes due to the wind. Loss of all on-site AC power is assumed for all wind scenarios due to the low wind velocity damage threshold of the electrical building.

The dominant wind initiated sequence is damage to the reactor building roof, pieces of which fall into the reactor pool and cause primary boundary damage and leading to core damage (large LOCA). The second dominant wind initiated sequence is a loss of all AC power and either a failure of the batteries (due to collapse of one of the interior walls) or failure to transport one of the two AEPG's to the site and connect to the system before the battery lifetime of 6 hours is exceeded. This scenario leads to a loss of decay heat removal capability. The total wind mean core damage frequency is 3.1 x 10^{-5}/yr. For the wind initiators there is always a loss of building confinement and fans, but the pool is intact and therefore there is adequate fission product scrubbing, but with ground level releases.

RESULTS

Table 6 summarizes the core damage frequencies resulting from the major external initiated events and compares this

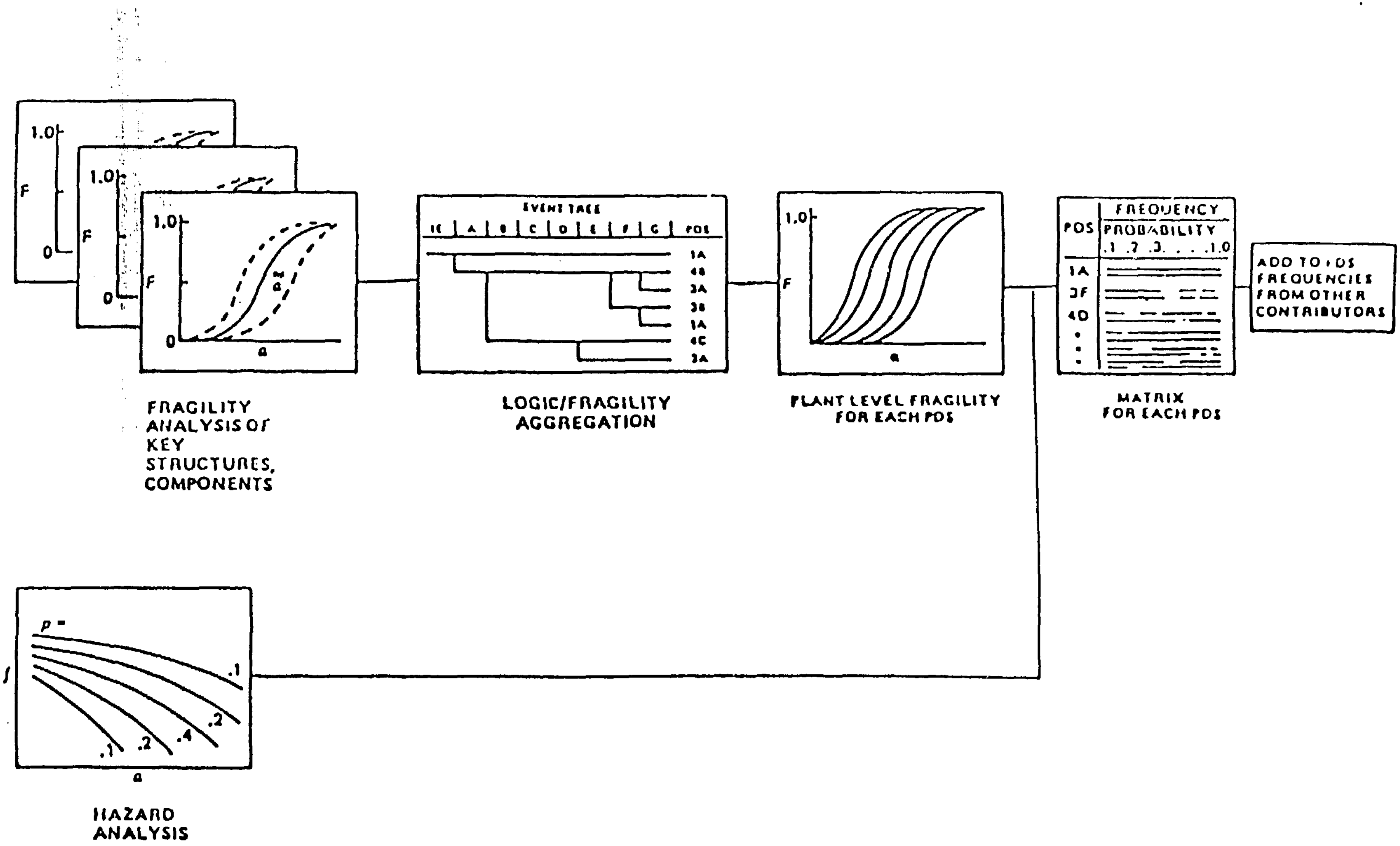

Figure 6. HFIR Seismic and Tornado/Wind PRA Analysis Approach

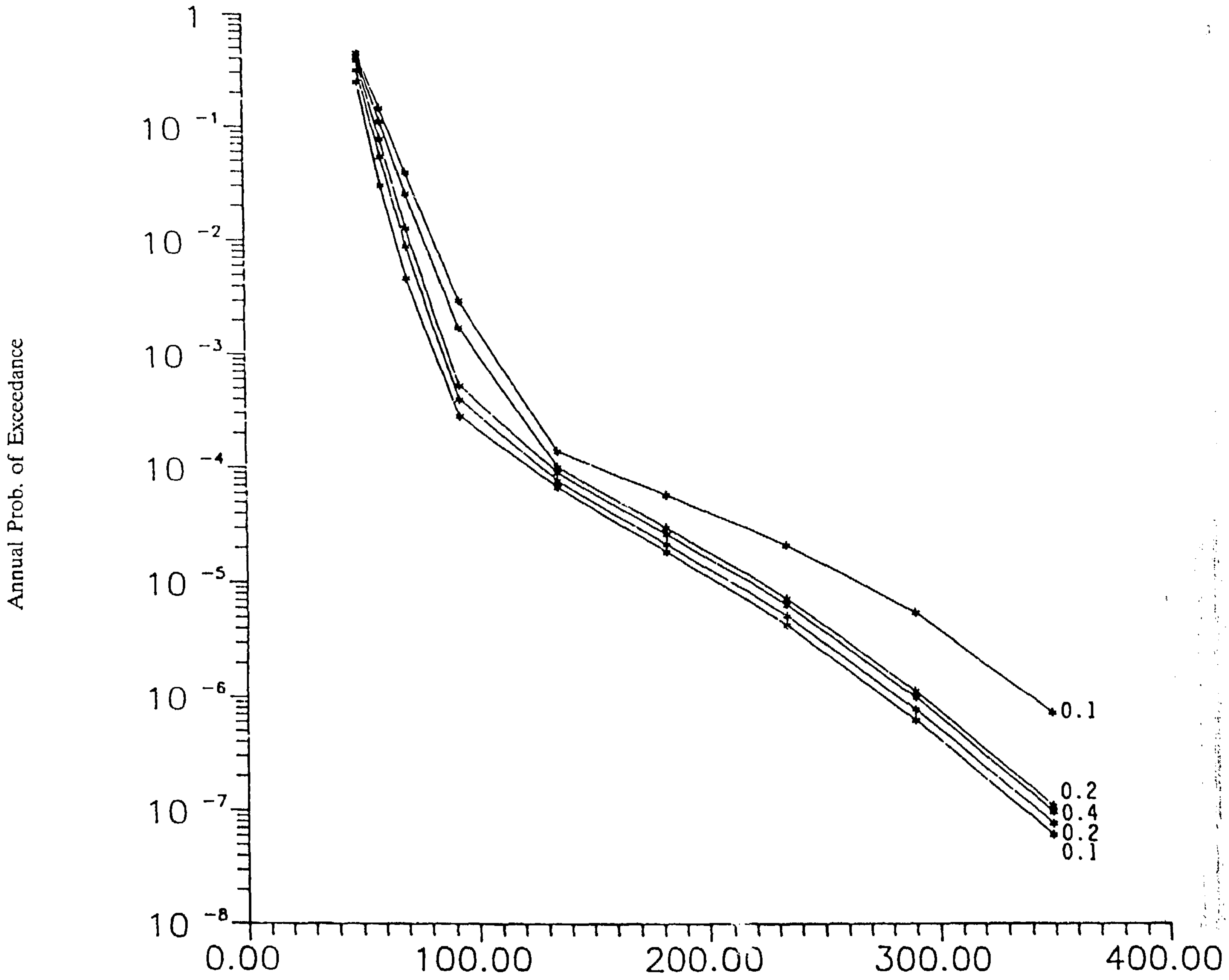

Figure 7. Combined Tornado and Extratopical Wind Hazard for the HFIR

Table 5. **Summary of Structural Wind Pressure Fragilities**

Structure	Failure Mode	V_m (mph)	β_R	β_U
1. Control Bay/Water Wing	Failure of Control Room Block Walls	99	0.06	0.16
2. Control Bay/Water Wing	Failure of Other Block Walls	99	0.06	0.16
3. Reactor Building High Bay	Observation Window Failure	99	0.06	0.16
4. Electrical Building	Failure of East Exterior Block Wall	116	0.08	0.18
5. Vent Stack	Failure of Outer Shell*	175	0.06	0.17
6. Reactor Building High Bay	Roof Beam Failure	214	0.07	0.19
7. Reactor Building	Failure of 4-Inch Pony Motor Battery Room Block Wall**	226	0.08	0.19
8. Reactor Building, Control Bay/Water Wing	Substructure Shear Wall Failure	>300		
9. Interior Shielding Wall	Collapse	>300		

*Only limited probability of impact onto reactor building.

**Fragility is based on proposed conceptual retrofit scheme.

Table 6. Summary of External Events Results

Initiator	Mean Core Damage Frequency
Fire	1.83×10^{-5}/yr
Wind	3.01×10^{-5}/yr
Seismic	1.55×10^{-4}/yr
Other	1.81×10^{-6}/yr
Subtotal - External	2.05×10^{-4}/yr
Subtotal - Internal	3.11×10^{-4}/yr
Total HFIR Mean Core Damage	5.16×10^{-4}/yr

to the internal event initiated core damage frequency. The total core damage (external and internal) frequency for the HFIR is 5.16 x 10^{-4}/yr.

The effect on the HFIR core damage frequency by including the ground motion model of expert #5 in the seismic hazard curves is an increase in the seismic contribution to core damage from 1.55 x 10^{-4}/yr to 4.1 x 10^{-4}/yr, thus increasing the total core damage frequency to 7.71 x 10^{-4}/yr. An effort is under way to technically justify the elimination of expert #5 ground motion model from the data base by using detailed analysis of the data from the recent Saguenay, Canada earthquake.

CONCLUSION

The HFIR PRA results for external events will be addressed as part of the overall HFIR Risk Management Program to ascertain if design changes are needed.

Currently, there appears to be only one area where some significant risk reduction can be achieved at a reasonable cost. The external event initiated core damage can significantly be reduced if one either improves the reliability of the decay heat removal power supplies or proves by experiment and analysis that the HFIR core can be cooled by natural circulation after a period of forced circulation which is shorter than the pony motor battery lifetime, thus eliminating the need for the AEPG's.

The experiment/analysis program is currently underway.

REFERENCES

[1] "The High Flux Isotope Reactor Probabilistic Risk Assessment," ORNL/RRD/INT-36, Jan. 1988.

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[4] S. Kaplan, "Expert Information" vs. "Expert Opinion" Another Approach to the Problem of Eliciting/Combining/Using Expert Opinion in PRA in proceedings of the PSA '89 International Topical Meeting Probability, Reliability, and Safety Assessment conference, pp. 593-602, 1989.

Figure 1. Vertical Section of HFIR Reactor Vessel and Core

Figure 2. HFIR Process Flow Schematic (primary and secondary systems)

Figure 3. HFIR Electrical System Schematic

Figure 4. HFIR Plant Damage State Matrix

Figure 5. Aggregate Seismic Hazard Curves [EPRI and LLNL (w/o Expert #5)] for the HFIR Site

Figure 6. HFIR Seismic and Tornado/Wind PRA Analysis Approach

Figure 7. Combined Tornado and Extratopical Wind Hazard for the HFIR