Identification and Quantification of Risk Scenarios for a Unique Nuclear Reactor – a Historical Example

D.H. Johnson[a], M.A. Linn[b], and C.T. Ramsey[b]
[a] B John Garrick Institute for the Risk Sciences, Los Angeles, USA
[b] Oak Ridge National Laboratory, Oak Ridge, USA

Abstract: Today a number of organizations are pursuing innovative reactor designs for eventual deployment. PRA will play an important role in the design and eventual licensing of these reactors. Indeed, PRA is well suited to support design and licensing decisions, but the unique nature of some of these designs can prove a challenge for the PRA analyst.

To illustrate the process of identifying, and equally important, quantifying risk scenarios associated with a unique design, the High Flux Isotope Reactor (HFIR) PRA is revisited. This reactor located at the Oak Ridge National Laboratory, has been in operation since 1965 and was the subject of the first PRA of a "class A" DOE research reactor in 1988. The high power density (approximately 1 MW/liter), aluminum fuel cladding and narrow coolant channels were key factors in the exploration of scenarios that potentially result in fuel damage and the release of radioactive material. Examples of the articulation and quantification of scenarios resulting from fuel channel blockage and fuel plate manufacturing errors are given, with explicit attention paid to the uncertainty of the scenario frequencies. Examples are also given to demonstrate how the PRA was used to improve design changes considered at the time of the PRA.

While the features that influence the risk scenarios at HFIR are unique to that plant, the process followed in the conduct of the HFIR PRA is a valuable example to today's reactor designers.

Keywords: PRA, scenario structuring, quantification of scenarios, uncertainty

1. INTRODUCTION

In 1986, the results of an irradiation specimen evaluation at the HFIR uncovered significantly more neutron embrittlement of the reactor vessel than had been anticipated. The reactor was shut down in November 1986 and became the focus of reviews by the Oak Ridge National Laboratory (ORNL), DOE, NAS/NRC and the ACNFS. [1]

These reviews resulted in changes to staffing, operations and design of the reactor. The reactor power level, operating pressure and inlet temperature were lowered. The design changes included the addition of an emergency depressurization system whose design purpose was to lessen the stress on the vessel during overpressure events.

The HFIR PRA was conducted as part of additional safety reviews that were conducted. The level 1+ internal events portion of the PRA was completed in January 1988 [2]. External events were added to the PRA in 1989. Additional design changes made during the conduct of the external events PRA included the addition of what later became known as FLEX equipment (specifically the addition of mobile power sources that can be transported to the site in a timely manner).

2. DESCRIPTION OF HFIR

HFIR is an 85 MW flux trap research reactor located at ORNL. It is water cooled and beryllium moderated and operated at 468 psi with inlet and outlet temperatures of 120F and 158F, respectively.

Peak thermal flux is 5 x 10^{15} n/cm^2-sec. The primary pressure is maintained by a pressurization and let-down system resulting in the primary system being "water solid."

Operation began in 1965 with primary missions to provide transuranic and cobalt isotopes, materials radiation research and neutron scattering research. Upgrades made to the facility include a cold neutron source and the facility today is a world class neutron science resource.

The core is comprised of two cylindrical fuel elements. The inner fuel element is comprised of 171 fuel plates while the outer element is comprised of 369 fuel plates. The core is small: the diameter is 17.5 inches and the height 24 inches, making the power density on the order of 1 MW/liter. Each fuel plate is made up of highly enriched U_3O_8-Al mixture in an Al cladding matrix. Each plate is approximately 50 mils (1.27 mm) thick; coolant channels are also approximately 50 mils.

The two fuel elements are shown in Figure 1.

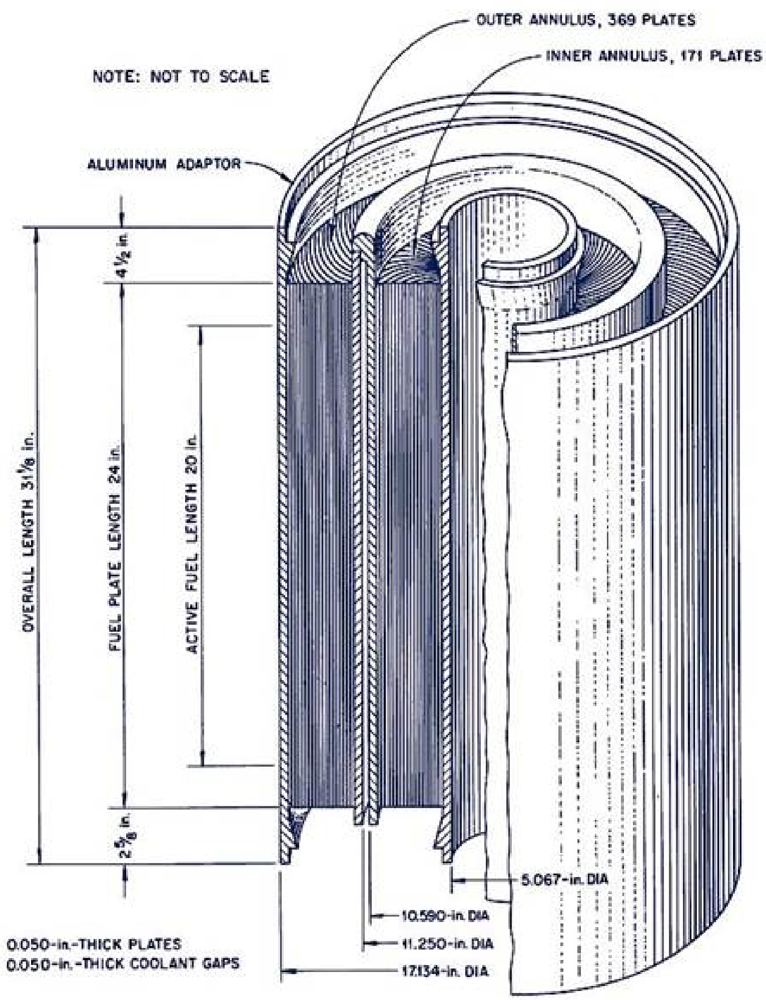
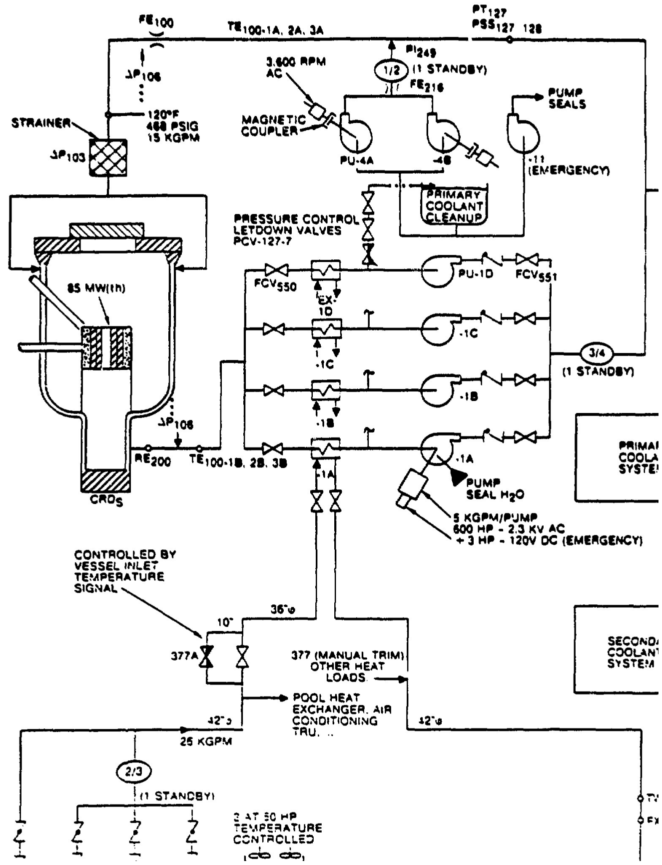


Figure 1 The HFIR Fuel Elements

The reactor is operated at full power for approximately 25 days at which time it is shut down and both fuel elements are replaced. Typically, 6 to 7 fuel cycles are accomplished in a calendar year.

The reactor core is housed in an 8 foot diameter pressure vessel that is about 19 feet high. The vessel sits at the bottom of an 85,000 gal pool of water measuring approximately 36 feet deep and 18 feet across. Spent fuel elements are stored at one end of the pool until they are shipped off site.

A simplified flow diagram for HFIR is shown in Figure 2.



Material samples to be irradiated can be placed in fixed positions located inside the inner fuel element cylinder (the “flux trap”) or moved to this region via rabbit tubes. This region interior to the inner element is 5 inches in diameter.

The reactor is housed in a confinement building kept at a slight negative pressure with fans pulling air through filters exhausting to a 250 foot stack.

3. THE HFIR PRA

There is no doubt that if the ASME Non LWR PRA Standard had been available in the late 1980s, it would have proven to be a useful resource for the HFIR PRA team. However a significant body of experience based on post WASH 1400 PRAs on commercial nuclear power plants had been completed and was readily available. In addition, methodology guidance such as found in NUREG/CR 2300, as well as the foundational Kaplan/Garrick definition of risk [3] was available. This PRA knowledge base provided a strong foundation for developing what turned out to be the first completed PRA for a DOE Class A research reactor.

Today, the developers of advanced reactor designs often start their safety evaluations with a process hazard assessment (PHA). While no formal PHA was conducted, the process of conducting the HFIR PRA started with identifying the available hazard sources and developing a comprehensive understanding of the operational and safety 'sensitivities' of the design. Candidate initiating events were selected following:

- An examination of the available 20 years of operating history, including the quarterly technical reports
- Review of the HFIR Accident Analysis Report as well as supplemental safety analyses
- Review of the HFIR design, drawings, and operational procedures
- Extensive discussions with the original HFIR design and engineering teams
- Review of incidents that had occurred at other research reactors as well as applicable events at commercial power reactors
- Creation of a Master Logic Diagram.

The internal initiating events identified are listed in Table 1.

Table 1: Internal Event Initiator Categories

Initiator Identifier and Category
1A Manual Scram
1B Inadvertent Control Blade Drop
1C Inadvertent Scram
2A Complete Loss of Offsite Power
2B Loss of Preferred Power
2C Loss of Switchgear DC
3A Runaway Pressurizer Pump
3B Loss of Running Pressurizer Pump
4 Flow Blockage and Fuel Manufacturing Defects
5 Small LOCA
6 Large LOCA
7 Beam Tube Failure
8A Reactivity Insertion
8B Degraded Secondary Cooling
8C Loss of Instrument Air
8D Degraded Primary Flow

As can be seen, except for Category 4, and perhaps Categories 3A and 3B, the initiator categories are somewhat typical compared to other reactor PRAs: reactivity events, pipe breaks and loss of support system events. (Of course, the plant response to all initiators are unique to the design of the HFIR). The design-specific sensitives identified centered on the high power density of the aluminum based fuel. Too rapid a decrease in pressure could result in exceeding the critical heat flux of the fuel. In fact, the LOCAs would have been better characterized as "Loss of Pressure Accidents." Likewise blockage of coolant channels could result in overheating. Flow blockage could result from the introduction of foreign material or from loss of integrity of a fuel plate.

Flow blockage became a significant focus for the analysis team. The process of identifying flow blockage scenarios can be thought of as paralleling that of a PHA. By explicitly identifying the characteristics, source, size and initial location of the blocking material, opportunities for risk reduction actions were identified. Fuel manufacturing defects also were modelled as specific initiators.

Important unknowns were treated probabilistically in the PRA. For example, it was not known with certainty how long after shutdown forced primary flow was required, nor how long the batteries providing this post scram cooling would reliably last.

4. IDENTIFICATION AND QUANTIFICATION OF FLOW BLOCKAGE SCENARIOS

The initial steps of identifying flow blockage scenarios began the categorizing the source location of the material. This facilitated the identification of specific potential material sources that could be transported to the fuel. The categorizing scheme is shown in Figure 3. L1 and L2 scenarios involve material originating within the primary system, upstream and downstream of the primary system strainer, respectively. A third category, L3, reflects the potential that the material may originate outside the primary system and is introduced during refuelling or maintenance when the primary system is open to the reactor pool.

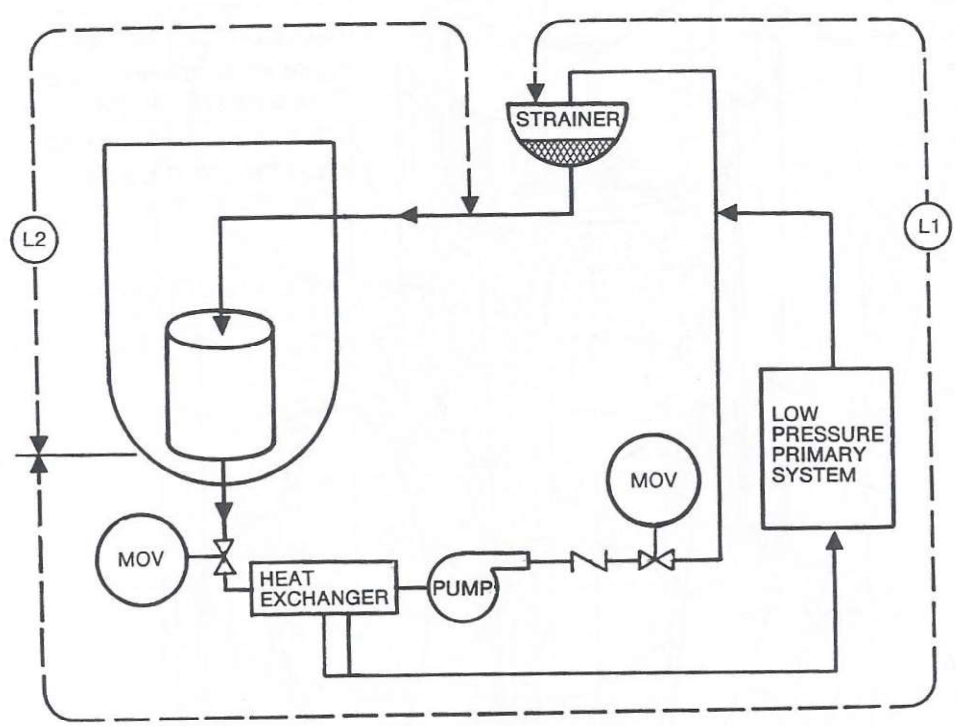


Figure 3 Sources of Fuel Blockage Material

A review of historical events as well as system design and operations indicated that blockage scenarios could further be characterized according to the type and size of material: metal, elastomeric, crud and corrosion products, biological material, plastics, etc.

Event trees were developed for each subtype of flow blockage scenario: originating location and material type. These scenarios included the potential for discovery and removal, trapping by the primary strainer or flow distributor, passing to a 'dead space' of vessel, etc.

The logic used to evaluate flow blockage scenarios is shown in Figure 4. Figure 5 shows an event tree developed to evaluate L1 category events (where the blocking material originates upstream of the primary system strainer).

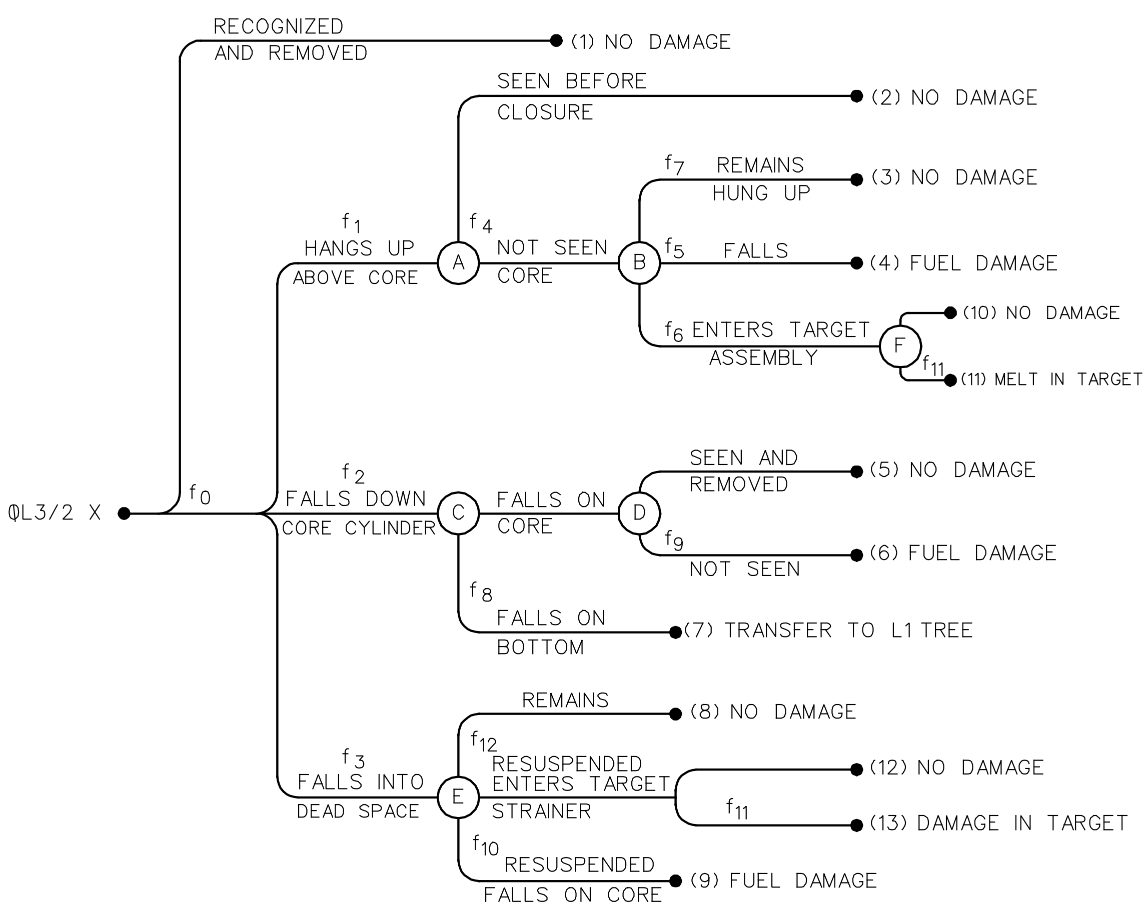


Figure 4 Flow Blockage Evaluation Logic

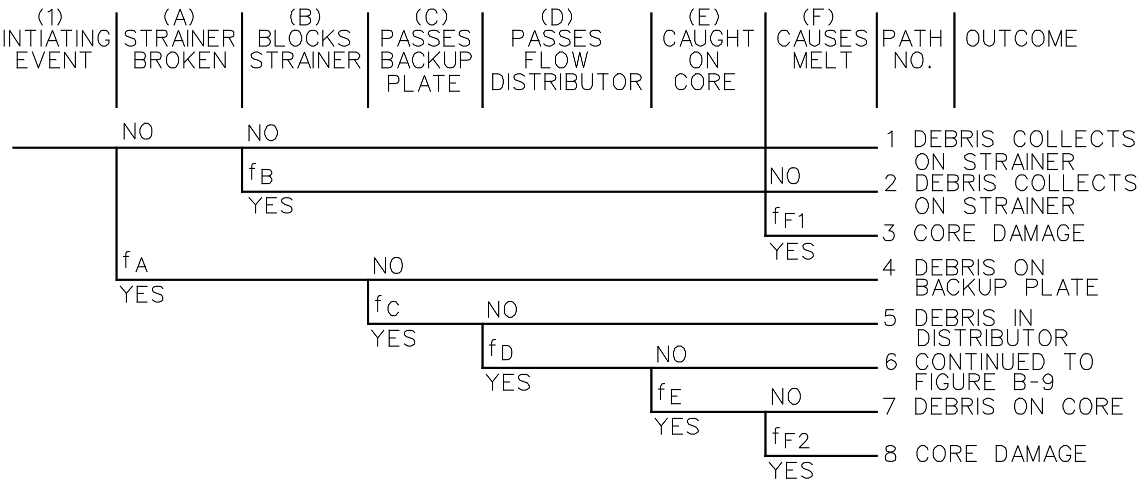


Figure 5 Event Tree for Evaluating Material Originating Upstream of Primary System Strainer

The PRA team facilitated close working sessions involving reactor designers, safety analysts, operators and maintenance staff to gather and review relevant evidence in describing potential scenarios. Specific material blockage sources (e.g., locking wires, small metal pieces, etc.) were catalogued. Past events at other reactors were compared to HFIR conditions and operations. Changes to HFIR design and operations were reviewed. In many cases, such changes reduced, but did not eliminate, the potential for reoccurrence. In other cases, such as the consideration of additional O-rings discussed below, changes had the potential of increasing the likelihood of fuel damage.

The close interaction of the plant staff and safety analysts with the PRA team established and organized the body of relevant evidence relative to the frequency of occurrence of source material and the other scenario elements. While the process of establishing this body of evidence was the result of all team members participation, it was primarily the responsibility of the plant experts to fully characterize the evidence elements. It was primarily the responsibility of the PRA team to address the question: ‘what probability curve expresses the parameter of interest given the collective evidence?’ Details on handling expert information can be found in Kaplan [4].

5. IDENTIFICATION AND QUANTIFICATION OF FUEL MANUFACTURING SCENAIROS

The HFIR fuel plates are crafted by a process of annealing and rolling aluminium blanks with powered Al-U_3O_8. A number of specific failure scenarios were postulated that could lead to plate failure during operation. These included undetected non-bonds, inhomogenities, cracks and the placement of a plate meant to be in the inner fuel element in the outer element. Each plate is subject to numerous inspections and review before final assembly into the fuel elements. Nevertheless, it is conceivable that a manufacturing flaw escapes the review process and results in structural failure of a plate (resulting in channel blockage and further damage).

Once again, the PRA team worked closely with the fuel experts to identify and quantify the likelihood of a critical undetected fuel manufacturing error being introduced without detection.

6. EXAMPLE RISK MANAGEMENT APPLICATIONS

Early applications of the PRA included the elimination or reduction of specific sources of flow blockage. One example was cited by the NAS/NRC report to DOE "Safety Issus at the DOE Test and Research Reactors:"

> In the course of conducting the PRA, it was quickly discovered that a major pathway to core damage (one referred to as 'flow blockage due to the target tower centering ring coming out of its groove') had been created directly as a result of the modifications to the target tower. Indeed, the mean frequency of core damage attributed to this scenario was calculated to be very high (2.6 x 10^{-2} per reactor year) and immediate action was taken to remedy the situation. An O-ring was removed from the target tower design, eliminating the scenario of principle concern."

The results of implementing various actions to reduce the potential for flow blockage is shown in Figure 6.

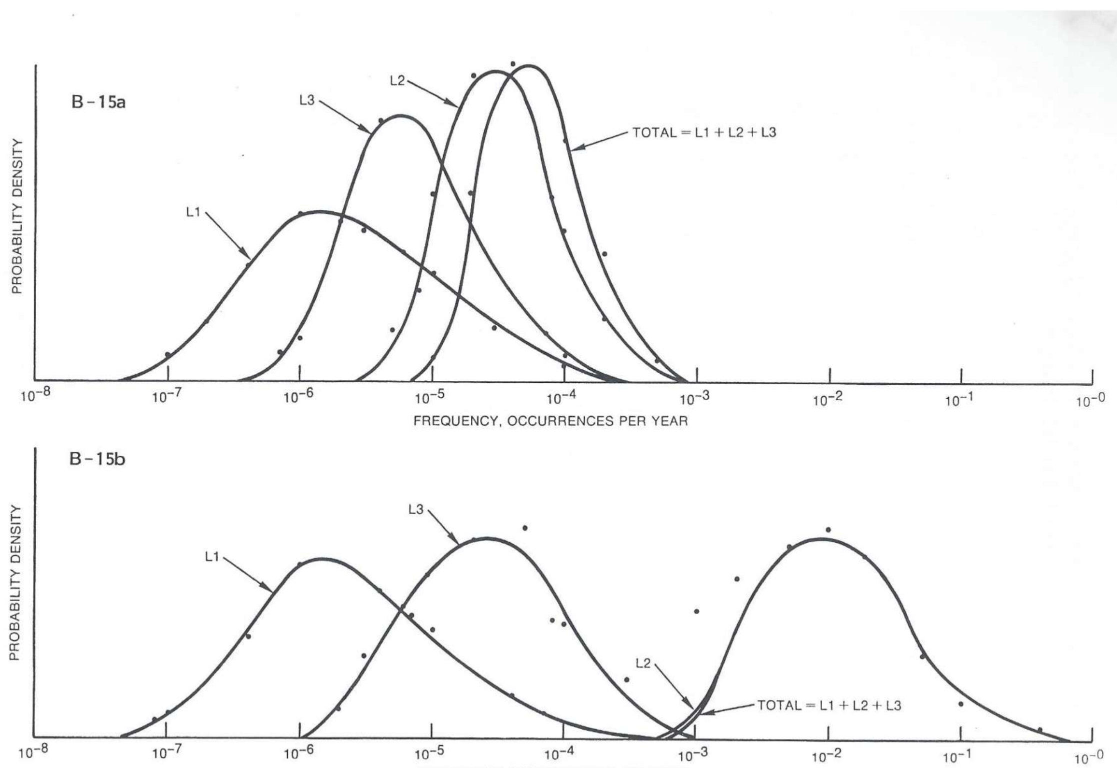


Figure 6 Risk Management of Flow Blockage Events

The important observation was the value of in-depth communication among the operations staff, maintenance staff, safety analyses staff and the PRA team.

A second important risk reduction action resulted in a design modification of a new emergency depressurization system that was being added to protect the system in the unlikely event of a challenge to vessel integrity. The original design of these air operated valves was such that they would 'fail open' on loss of air. The specific concern was to reduce the likelihood of overpressurizing the reactor vessel. The PRA put the modification in a broader context. Given the sensitivity of the fuel to rapid depressurization, this resulted in loss of plant air causing fuel damage. The failure position of these valves were changed to 'fail closed.'

7. CONCLUSIONS

Although conducted in the late 1980s, the PRA of the ORNL High Flux Isotope Reactor offers lessons and examples potentially useful in guiding the design and associated PRA analyses for today's innovative reactor designs. Specifically, the HFIR PRA offers examples of identifying and quantifying unique scenarios. The PRA also offers examples of the effective use of PRA in design.

References

[1] Flanagan, G.F., D.H. Johnson, D. Buttemer, H.F. Perla and S.H. Chien, "External Event Probabilistic Risk Assessment for the High Flux Isotope Reactor (HFIR)," Second DOE Natural Phenomena Hazards Mitigation Conference, 1989, CONF-8910192.

[2] D.H. Johnson, et al, "The High Flux Isotope Reactor Probabilistic Risk Assessment," ORNL/RRD/INT-36, January 1988. Also "The High Flux Isotope Reactor Probabilistic Risk Assessment – Analysis of the Risk from Internal and External Events," PLG-0829, August 1991.

[3] Kaplan, S., and B.J. Garrick, "On the Quantitative Definition of Risk," Risk Analysis, 1, (1981), pp 11-27.

[4] Kaplan, S., 'Expert Information' verses 'Expert Opinions.' Another Approach to the Problem of Eliciting/Combining/Using Expert Knowledge in PRA," Reliability Engineering and System Safety, 35, (1992) pp 61-72.

[5] National Research Council, "Safety Issues at the DOE Test and Research Reactors," Report to the US Department of Energy, National Academy Press, 1988.