Conf-930116--20

Paper to be submitted to American Nuclear Society Probabilistic Safety Assessment '93, Clearwater Beach, Florida, January 26-29, 1993.

ANL/CP--75951

DE93 004846

The EBR-II Probabilistic Risk Assessment: Results and Insights*

by

D. J. Hill, W. A. Ragland, and J. Roglans

Reactor Analysis Division
Argonne National Laboratory

*Work supported by the U.S. Department of Energy, Nuclear Energy Programs under contract W-31-109-ENG-38.

THE EBR-II PROBABILISITIC RISK ASSESSMENT: RESULTS AND INSIGHTS

D. J. HILL, W. A. RAGLAND, and J. ROGLANS
Argonne National Laboratory
9700 South Cass Avenue
Argonne, Illinois 60439-4842, USA

ABSTRACT

This paper summarizes the results from the recently completed EBR-II Probabilistic Risk Assessment (PRA) and provides an analysis of the source of risk of the operation of EBR-II from both internal and external initiating events. The EBR-II PRA explicitly accounts for the role of reactivity feedbacks in reducing fuel damage. The results show that the expected core damage frequency from internal initiating events at EBR-II is very low, 1.6 10^{-6} yr^{-1}, even with a wide definition of core damage (essentially that of exceeding Technical Specification limits). The probability of damage, primarily due to liquid metal fires, from externally initiated events (excluding earthquake) is 3.6 10^{-6} yr^{-1}. Overall these results are considerably better than results for other research reactors and the nuclear industry in general and stem from three main sources: low likelihood of loss of coolant due to low system pressure and top entry double vessels; low likelihood of loss of decay heat removal due to reliance on passive means; and low likelihood of power/flow mismatch due to both passive feedbacks and reliability of rod scram capability.

INTRODUCTION

A Level 1 Probabilistic Risk Assessment (PRA) including external events has been completed for the Experimental Breeder Reactor-II (EBR-II). There were several objectives for this project; to provide a quantitative estimate of the risk associated with the operation of EBR-II, to provide a framework for managerial decision-making for the management of risk at the facility, to provide insights into the nature of the risk of EBR-II that can be applied in the design of future LMRs, and to train a group of scientific staff in the methodology of PRA so that the lessons learned can be appropriately translated into plant management and design. The release of Revision 2.0 of the EBR-II PRA signals the attainment of these goals.[1]

EBR-II is a 60 MW(e) liquid sodium cooled, pool type fast reactor which has operated successfully as a power reactor and irradiation facility for over 25 years. A detailed description of the EBR-II plant can be found elsewhere.[2] The EBR-II PRA investigated the possibility of the EBR-II core incurring damage as a result of accidents which could conceivably occur. The accidents considered include those which are classified as arising from internal events, e.g., pump failures, and external events, e.g., fires, floods. At this time the seismic PRA is still in process. Although the primary product of the study is a quantitative statement of the risk of operation of EBR-II, the inherent uncertainties in the numerical results mean that the more valuable insights arise from the qualitative, risk management, insights concerning relative importance of events, systems, and maintenance practices. The significance of the EBR-II PRA lies in the comprehensive nature of the evaluation of the whole facility using a systematic integrated process.

The study focused upon the identification of possible core damage scenarios from plant operations and external phenomena. The analysis was halted at core damage and furthermore the meaning that was given to 'core damage' included scenarios that led to temperatures that exceeded Technical Specification limits even though no clad breach and radionuclide release from driver fuel was predicted. The analysis did not extend to the analysis of subsequent scenarios for release of radioactive materials to and beyond the containment. The scope of the EBR-II included shutdown accidents, refuelling accidents and external events, (the seismic portion is not yet complete). Passive safety response -- both passive reactivity shutdown and passive decay heat removal -- was explicitly accounted for in the event trees.

This approach was taken for a number of reasons. Firstly, if a transient were to occur that led to fuel or structural temperatures in excess of those in the Technical Specification limits, the consequences for future operability of the EBR-II reactor would be severe and it is, therefore, desirable to estimate this type of "availability risk" as well as that of the more conventional core damage. Secondly, there are a whole class of transients which will not lead to any radionuclide release because of the passive safety response of EBR-II and it is desirable to derive a quantitative measure of this feature of EBR-II. These events correspond to the Accommodated ATWS category in the ALMR safety documents.[3]

In order to facilitate the understanding of the nature of risk at EBR-II the sequences identified were divided into six categories, labelled P1 to P6. These six categories represent fundamentally different types of accident sequence. In addition, various different measures of core damage were used. Table 1 describes the plant transient categories and the core damage definitions.

TECHNICAL INSIGHTS

Some of the insights from this analysis are summarized below. The results are discussed in greater detail in the PRA main report.[1]

Probability of core damage

- The mean probability of core damage from internal event generated sequences is 1.6 10^{-6} per reactor year.
- The mean probability of core damage from external event generated sequences is approximately 3.6 10^{-6} per reactor year.
- These results are considerably better than results for other research reactors and the nuclear industry in general as the following table indicates. Table 2 compares published results from PRAs on USDOE Category A reactors. Where comparisons can be made it can be seen that EBR-II compares favorably with the other facilities. Comparisons of this sort should be viewed with caution as the widely varying reactor types and definitions of damage could lead to a bias in the results. Nevertheless, the favorable comparison is not surprising as the EBR-II mission has been centered upon developing and demonstrating the safety concepts of LMRs for commercial application and EBR-II is viewed as a prototype of a particular design concept for LMRs, the Integral Fast Reactor (IFR).

IMPORTANCE OF INDIVIDUAL INITIATING EVENTS

The risk of operation of EBR-II is not dominated by any one event or class of events. The most important initiating event is an unprotected loss of normal power. The resulting sequences would not damage driver fuel and are included by virtue of the wide definition of damage in the EBR-II PRA. (In fact, the transient is identical to a transient test that was purposely performed as SHRT45 in the Inherent Safety Demonstration Tests.[2]) Generally, reactivity insertion events have been found to be unimportant. Direct damage events and local faults are not significant contributors to damage.

IMPORTANCE OF ACCIDENT END STATES

One unique feature of the EBR-II PRA was the explicit accounting of accident sequences that could be shown not to lead to core damage i.e, those sequences where engineered safety features failed to function correctly but inherent characteristics of the plant ensured that no damage to driver fuel occurred.

Table 1. Definition of Transient and Damage Categories.

DAMAGE CATEGORIES	
CD	CORE DAMAGE - Average driver subassembly reaches sodium boiling, fuel melting, or pin failure
MCD	MINOR CORE DAMAGE - Sodium boiling, fuel melting or pin failure occurs in the hottest driver subassembly, but no damage occurs in average driver assemblies
PED	POTENTIAL EXPERIMENT DAMAGE - Clad temperature in hottest driver pin exceeds 816 °C (1500 °F) or lies between the eutectic temperature and 816 °C for over 60 s
CSD	CORE/STRUCTURAL DAMAGE - Primary sodium bulk temperature in the primary tank exceeds 538 °C (1000 °F) and some core and structural damage is assumed to take place
ND	NO DAMAGE - Clad temperature never exceeds the eutectic temperature or lies between the eutectic temperature and 816 °C for less than 60 s
TRANSIENT CATEGORIES	
P1	Direct damage events, resulting from primary piping or tank leaks or rupture, structural failures
P2	Protected loss of heat sink events, with neutronic shutdown, leading to uniform overheating of the system and a long term core vulnerability
P3	ATWS events, leading to a short term core vulnerability
P4	Protected events that lead to minor core damage or potential experiment damage
P5	Faults leading to degraded containment function and where core damage can not be ruled out, e.g., steam generator tube ruptures.
P6	Local faults, i.e. events occurring within one subassembly and lead to damage to that subassembly and possibly its neighbors

Table 2. Mean Core Damage Frequencies for USDOE Facilities

		Mean Core Damage Frequencies (Yr^{-1})					
Ref		Total Internal	Seismic	Fire	Wind/ Tornado	Total External	Total
	EBR-II	$1.6\ 10^{-6}$		$3.6\ 10^{-6}$	-		
[4]	ATR	$1.8\ 10^{-4}$	-	-	-	-	-
[5]	HFBR	$3.5\ 10^{-4}$	-	-	-	-	-
[6]	HFIR	$3.1\ 10^{-4}$	$1.2\ 10^{-4}$	$1.9\ 10^{-5}$	$2.9\ 10^{-4}$	$4.3\ 10^{-4}$	$7.4\ 10^{-4}$
[7]	N-Reactor	$6.7\ 10^{-5}$	$1.7\ 10^{-4}$	$1.7\ 10^{-5}$	-	$1.9\ 10^{-4}$	$2.5\ 10^{-4}$
[8] [9]	SPR K-Reactor	$2.1\ 10^{-4}$	$1.2\ 10^{-4}$	$1.4\ 10^{-7}$	-	$2.2\ 10^{-4}$	$4.3\ 10^{-4}$

For internally initiated events leading to both protected and unprotected plant transients the probability of occurrence is estimated to be 3.1 10^{-3} yr^{-1}. Of these transients,

- 95% lead to no damage; i.e. temperatures will not exceed Technical Specification limits,

and of those which do lead to some form of damage,

- 37% lead to temperatures exceeding Technical Specification limits but not necessarily fuel failure and radionuclide release,

- 33% lead to radionuclide release from driver fuel,

- 30% lead to a uniform overheating of core, vessel, and sodium or damage to a structural component.

Furthermore, of the 33% which would lead to radionuclide release from driver fuel, 5.2 10^{-7} yr^{-1} (60%) resides in one event which has a simple recovery action which was not accounted for in the PRA because it was not described in any procedure. Application of this recovery action, simply deenergising the secondary pump, would significantly reduce this risk. The results demonstrate the effects of the reactivity feedbacks in limiting the severity of the transients, precluding any form of damage in most of the more likely ATWS events. The reactivity feedbacks are partially responsible for the difference between the total frequency of transients and the total damage frequency.

A moderate fraction of the damage frequency resides in the transient classes related to local faults and steam generator tube ruptures. Table 3 provides the contribution of each initiator group to the different transient and damage categories. No particular accident sequences are dominant in EBR-II. The more likely damaging sequences are summarized in Table 4.

RISK REDUCTION AND APPLICATIONS

The results show only nine internally initiated accident sequences which have a contribution to any form of damage of greater than 2%, (3.2 10^{-8} yr^{-1}). The risk is distributed among a variety of different types of accident with no one accident or class of accidents dominating the results. This result is evidence of the long operating history of EBR-II where attention has always been paid to improvements which reduce the risk of operation, such as separation of the power supplies to the primary pump motor - generator sets and simplification of the scram system. As this study progressed and the important sequences were characterized, areas of potential improvement suggested themselves. One, the separation of the power supplies to the clutches on the motor-generator sets, is already being actively pursued.

Any program put in place to apply the result of the PRA to EBR-II operations would include:

- Use of PRA in the design and modification process to evaluate how plant modifications might affect risk

- Use of PRA results to provide input into EBR-II safety analyses

- Use of PRA results to enhance operator training

All of these can be thought of as elements of a risk management program.

INTERPRETATION OF RESULTS

Consider first the initiating events which require reactor scram. The PRA for EBR-II identified about four per year as the estimated frequency based upon a conservative analysis of the last 15 years of EBR-II operation. The scram reliability for EBR-II was typically estimated at 5 10^{-6} per demand (the actual value is initiating-event dependent). This result depends greatly upon hypothetical common cause events in the fault tree analysis. (Note that a modern scram system can achieve much higher reliability than this, 10^{-7}, as long as the design has sufficient redundancy and diversity.) The probability of an unprotected internally initiated accident sequence at EBR-II is therefore 2 10^{-5} per year; the vast majority of these sequences lead, however, to an absolutely benign outcome because of the passive reactivity feedbacks. Even adopting the wide definition of damage which was used in the EBR-II PRA, (the definition classifies all temperatures which exceed

Table 3. EBR-II Transient Annual Frequency Distribution in Categories

CONTRIBUTION TO DAMAGE CATEGORIES					
DAMAGE CATEGORY	CD	MCD	PED	CSD	ND
INTERNAL EVENTS	$9.0\ 10^{-8}$	$4.3\ 10^{-7}$	$5.9\ 10^{-7}$	$4.6\ 10^{-7}$	$2.9\ 10^{-5}$
EXTERNAL EVENTS*	$< 10^{-10}$			$3.6\ 10^{-6}$	
TOTAL	$9.0\ 10^{-8}$	$4.3\ 10^{-7}$	$5.9\ 10^{-7}$	$4.1\ 10^{-6}$	$2.9\ 10^{-5}$

CONTRIBUTION TO TRANSIENT CATEGORIES						
TRANSIENT CLASS	P1	P2	P3	P4	P5	P6
INTERNAL EVENTS	$2.9\ 10^{-8}$	$2.7\ 10^{-7}$	$1.4\ 10^{-5}$	$1.4\ 10^{-8}$	$1.7\ 10^{-5}$	$4.2\ 10^{-8}$
EXTERNAL EVENTS*		$3.6\ 10^{-6}$	$2.1\ 10^{-8}$			
TOTAL	$2.9\ 10^{-8}$	$3.9\ 10^{-6}$	$1.4\ 10^{-5}$	$1.4\ 10^{-8}$	$1.7\ 10^{-5}$	$4.2\ 10^{-8}$

* External events exclude seismic initiators.

Table 4. Summary of Most Important Dominant Sequences

SEQUENCE NAME AND DESCRIPTION		FREQUENCY yr^{-1}	CLASS
INTERNAL EVENTS			
LONP-4	Unprotected loss of normal power with successful LOF scram signal	$4.6\ 10^{-7}$	P3 - PED
OCSL-10	Unprotected overcooling with failure of scram signal	$3.3\ 10^{-7}$	P3 - MCD
TSDC-2	Long térm failure of decay heat removal after losing a shutdown cooler	$1.3\ 10^{-7}$	P2 - CSD
LF2A-6	Unprotected double pump LOF with successful scram signal	$9.6\ 10^{-8}$	P3 - PED
SPTR-3	Small superheater leak escalates; secondary not dumped and no shutdown	$8.4\ 10^{-8}$	P5 - CSD
SPTR-13	Small superheater leak escalates; no pressure relief and no shutdown	$8.0\ 10^{-8}$	P5 - CSD
RISB-6	Unprotected TOP with successful scram signal	$3.5\ 10^{-8}$	P3 - MCD
SHDL-3	Failure of short term decay heat removal at start of long shutdown	$3.3\ 10^{-8}$	P2 - CSD
SPTR-23	Large superheater leak; failure to release pressure; assembly propagation	$3.2\ 10^{-8}$	P5 - CSD
RIFL-1	Argon pressurization ruptures primary tank	$2.9\ 10^{-8}$	P1 - CD
EXTERNAL EVENTS			
SDFR-6	NaK fire in a shutdown cooler degrades remaining shutdown cooler	$2.1\ 10^{-6}$	P2 - CSD
SSFR-6	Secondary sodium fire disables one shutdown cooler and degrades the other	$1.5\ 10^{-6}$	P2 - CSD

the Technical Specification limits as damage), fully 95% of the sequences, given failure to scram, lead to no damage to driver fuel. The conclusion is that about 1.0 10^{-6} per year is the probability that an unprotected accident sequence leading to driver fuel temperatures exceeding EBR-II Technical Specification limits will occur. If desired, this risk at EBR-II could be reduced by at least an order of magnitude for a modern LMR by using a modern scram system design, which can be shown to be immune to common cause events.

In the case of those accidents arising from loss-of-decay heat removal capability, the PRA identified about 11 initiating events per year. (Even a routine shutdown requires the decay heat removal system to be functional and so the seven routine shutdowns a year at EBR-II are included and summed with the four scram events). EBR-II is designed with passive redundant decay heat removal capability through both the Balance-of-Plant and through two shutdown coolers. Through careful design these systems are able to perform their functions without electrical power, which removes an important factor observed in the risk of LWR power plants, namely, electrical dependencies of the decay heat removal function. The mission time for decay heat removal was taken to be 45 days in order to include all the shutdown risk and, even in those circumstances, the probability of failure was estimated to be ~2 10^{-8} per demand, with failure being defined as the sodium pool reaching a temperature of 1000°F (a limit on internal structural components). The resulting overall damage frequency is ~2 10^{-7} per year. This result serves to emphasize the value of passive decay heat removal from the vessel, natural circulation cooling of the core, and a design which provides redundant means of accomplishing this without the requirement for electrical power, thus removing many sources of risk through lack of diesel power, human error, etc. This feature is also important in the external event analysis.

The remaining contributions to the risk of operation of EBR-II, and by extension to any LMR, arise from those rare but conceivable events which are inherently difficult to quantify. These rare events include steam generator tube ruptures, major structural failures, and local faults. Structural failure is a very low probability event, <10^{-7} per year; however, a causative mechanism was identified in EBR-II which allowed for a meaningful evaluation. The hypothetical cause involved over-pressurization of the cover gas by a failure of the argon supply regulating system -- combined with undetected blockages of the relief systems due to sodium vapor condensation. Though not a prominent contributor to risk, this event was judged to be sufficiently important to highlight a generic "lesson learned" requirement for support systems to be designed with the same regard to inherent limitation, (in this case limitation of the ability to over-pressurize) as is used on design of IFR primary systems.

Because of the violent chemical reactions possible in the $Na-H_2O-O_2$ system, steam generators in LMRs have traditionally been a source of much design effort to minimize the chance for $Na-H_2O$ contact. Even so, LMRs in Russia and the United Kingdom have experienced energetic events. The steam generators at EBR-II are of a double-wall design, and have proved to be trouble-free for 25 years of operation. Despite this, steam generator tube ruptures are in general still important contributors to risk -- in the EBR-II case primarily because of the absence of fast-acting water and sodium dumps -- which absence renders the EBR-II IHX vulnerable in the event of an energetic reaction. Thus, the lesson learned here is that effort must be applied on the design of the steam generators and pressure relief systems so as to ensure that the consequences of energetic events do not impair the IHX or primary system integrity.

In all cases of rare events discussed above, a fundamental principle can be extracted. First, in conducting a PRA one can almost never totally eliminate a rare event on the grounds of its probability of occurrence because its probability is poorly known; and the best one can hope for is by intelligent design, which is consciously addressed at each such identified rare event, to reduce the initiating probability to small values (<10^{-7} per year). Therefore, one must in addition take additional measures to reduce the consequences of the rare events with appropriate mitigative systems."

The external event analysis can be divided into two parts those which challenge the decay heat removal capability. Indeed, because of the fact that only one or, at most, two control rods are required to be dropped for shutdown to take place, no external event which could impair the scram function was found. In particular no way was identified in which a fire or flood could lead to a situation in which all the control rods remained energized. The effect of external events on the decay heat removal function for reactors is usually via electrical power, with fires and/or floods leading to disablement of electrical power supplies. In the case of EBR-II, electrical power is not required for decay heat removal. Overall the risk-dominant non-seismic external event for EBR-II is a liquid-metal fire under the deck plates disabling all decay heat removal. The consequences of such an event could be minimized in future IFR designs by argon inert systems and good physical separation of the various decay heat removal paths. Though the seismic PRA for EBR-II is not yet complete, the currently anticipated dominant seismic event is a massive structural failure of the primary tank support system at a median of 0.7g -- leading to failure of the decay heat removal function. (This would be equiva-lent to failure of seismic isolation in a modern seismically-isolated IFR plant.)

CONCLUSION

The risk of operation of EBR-II was found to be very low, (of the order of 10^{-6} per reactor year for even minor core damage), with no one accident or class of accidents dominating the risk profile. The value of this result lies not in the low numerical value of the accident frequency as it compares with other plants but in having extracted the underlying causes of this favorable result in terms of quantifying the payoff of the specific sound engineering design principles which led to the low damage frequency. These lessons learned can now be applied to the future IFR design refinements.

[a]Even though a rare event may lead to no core damage whatsoever, regulatory or political consequences may be quite severe and thus in the design of a modern LMR the focus should be on the first goal -- must seek to minimize the probability of rare initiating events by intelligent design.

REFERENCES

1. D. J. HILL et al., Experimental Breeder Reactor II (EBR-II) Level 2 Probabilistic Risk Assessment, Argonne National Laboratory, Internal Document, June 30 1991.

2. H. P. PLANCHON et al., "The Experimental Breeder Reactor II Inherent Shutdown and Heat Removal Tests - Results and Analysis", Nucl. Engrg. Des. 91 (1986) pp. 287-296.

3. Draft Preapplication Safety Evaluation Report (SER) for the Power Reactor Inherently Safe Module Liquid Metal Reactor, NUREG-1368, US NRC, Sep. 19, 1989.

4. S. A. EIDE, D. M. HENRY, and T. A. THATCHER, "Advanced Test Reactor Level 1 Probabilistic Risk Assessment," Proc. of the ANS Conference on The Safety, Status and Future of Noncommercial Reactors and Irradiation Facilities, Boise, Idaho, September 31 - October 4, 1990.

5. T. L. CHU, M. A. AZARM, R. BARI, J. LEHNER, W. LUCKAS, "Quantification of the Probabilistic Risk Assessment (PRA) of the Higher Flux Beam Reactor (HFBR) at Brookhaven National Laboratory, ibid.

6. G. F. FLANAGAN, "The Use of PRA in Management of Safety Issues at the High Flux Isotope Reactor," ibid.

7. G. D. WYSS, D. M. KUNSMAN, A. L. CAMP, "Results and Insights from the N Reactor Level II/III PRA for Application to Advanced Reactor Design," ibid.

8. S. P. TINNES, D. S. CRAMER, V. E. LOGAN, S. V. TOPP, J. A. SMITH, M. D. BRANDYBERRY, "Results of the Level 1 Probabilistic Risk Assessment (PRA) of Internal Events for Heavy Water Production Reactors (U)," ibid.

9. M. D. BRANDYBERRY, H. E. WINGO, "External Events Analysis for the Savannah River Site K Reactor (U)," ibid.

END

DATE
FILMED
5/03/93