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APPLICATIONS OF THE EBR-II PROBABILISTIC RISK ASSESSMENT*

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ABSTRACT

A Probabilistic Risk Assessment (PRA) of the Experimental Breeder Reactor II (EBR-II), a Department of Energy (DOE) Category A research reactor, has recently been completed at Argonne National Laboratory (ANL), and has been performed with close collaboration between PRA analysts and engineering and operations staff. A product of this involvement of plant personnel has been a excellent acceptance of the PRA as a tool, which has already resulted in a variety of applications of the EBR-II PRA. The EBR-II has been used in support of plant hardware and procedure modifications and in new system design work. A new application in support of the refueling safety analysis will be completed in the near future.

I. THE EBR-II PRA

EBR-II is a DOE Category A research reactor located at ANL West in Idaho. It is a 62.5 Mw-thermal Liquid Metal Reactor (LMR), which supplies, at full power, 20 Mw-electric to the Idaho National Engineering Laboratory (INEL) loop. EBR-II started operation in 1964 and it has been used in a variety of research programs, recently as a testbed in the Integral Fast Reactor (IFR) Program. The PRA for EBR-II[1] started in 1989 after the National Academy of Sciences recommended that probabilistic risk assessments be performed for DOE Category A reactors.

Since completion, the EBR-II PRA has been successfully accepted as an additional safety analysis tool for EBR-II. Two characteristics of the PRA development have been particularly instrumental in achieving the PRA acceptance and use, namely:

- the PRA was developed at ANL by ANL staff members, some of whom are normally involved with safety analyses in support of EBR-II operations and experimental programs;

- the systems models and the initiating event characterization were developed with direct input from plant engineering and operations staff, who were involved with the PRA effort from the beginning and also performed several stages of review of the PRA document and models[2].

The recognition of the PRA as a useful tool has resulted in a number of PRA applications in EBR-II. The uses of the EBR-II PRA can be generally

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arranged in two groups, differentiated in the way the PRA is applied. The first group corresponds to situations that the PRA identified as a risk contributor but that could be amended with relative ease. The second type of applications corresponds to uses of the PRA models and tools in support of plant modifications and safety evaluations or in backing new system design. Both of these types of applications form part of a successful risk management program. Examples of the two types of applications will be briefly described in the following sections.

II. RISK REDUCTION ACTIONS IDENTIFIED IN THE EBR-II PRA

The first set of applications of the EBR-II PRA was the identification of some situations that increased the plant damage risk or system unavailability and that were amenable to relatively simple correction actions. Included in these findings were some deficiencies detected during plant seismic walkdowns. These findings were expected from the previous experience of walkdowns at commercial plants. Inadequacies in the anchorage of some electrical cabinets and battery racks were observed. Although it has been later determined that the premature seismic failure of these items due to poor anchorage would result in a negligible effect on the seismic risk at EBR-II, the availability of electrical systems after a seismic event is still desirable and important. Therefore, appropriate actions are still implemented to correct the deficiencies.

Risk reduction actions were identified during the course of the EBR-II PRA. Most actions would only result in a marginal reduction in the risk of fuel damage, but there was a particular item that presented a more significant and unexpected problem. During the system analysis a dependency was identified in the clutch control power supply of the two primary pumps.

Because of the passive safety characteristics of the EBR-II reactor, many unprotected loss of flow (LOF) transients can be accommodated by the reactor without resulting in damage to the fuel elements, as was demonstrated in a series of tests in 1986[3]. There are two primary pumps in EBR-II whose power is supplied from a pair of motor generator (MG) sets coupled with magnetic clutches. One of the most severe LOF transients is the outcome of the simultaneous loss of both MG set clutches, which results in a fast coastdown of the pumps. An unprotected double pump LOF initiated by the double clutch failure leads to severe core damage, the highest damage category distinguished in the EBR-II PRA[4].

The dependency that was identified between the two clutch power supply circuits implied that a single failure could result in the simultaneous loss of both clutches, and therefore, both primary pumps. Furthermore, it was also found that the two clutch power supplies were not physically separated, but rather they where located not only in the same cabinet but also in the same chassis. This lack of physical separation added some possibilities for common cause or external event induced failures, such as fire or humidity. Indeed, this physical dependency had a role in the risk assessment of inadvertent actuations of the fire protection system, as is described in a following section.

Engineering work to reverse the power supply dependency situation started soon after it had been identified. The logic models that were developed for the PRA are being used along with the PRA tools to design the separated control system and to compare alternative options.

III. APPLICATION OF THE EBR-II PRA IN PLANT MODIFICATIONS AND SYSTEM DESIGN

Of equal interest are the applications that have involved the PRA models and methods in support of safety analysis for plant or procedure modifications and in support of system design. The request for this type of applications have been originated not among the PRA staff but rather in the plant engineering and operations groups. The most interesting of these applications are described in the following subsections.

A. Control Rod Motion Testing Procedure.

The PRA has been used in support of the revision of an operating procedure in EBR-II. The EBR-II geometry consists of a primary loop immersed in a sodium pool. The sodium pool is contained in a primary tank and the reactor core is located in a vessel inside the tank. To test that the control rods in EBR-II can be moved from their operating position when required, rod motions tests have been carried out periodically. These tests are designed to detect control rod jamming, which could hypothetically be caused by rod bowing or binding, by foreign material buildup in the rod drives, or by tilting of the reactor vessel cover which might compress all rod drives simultaneously. The rod motion tests are not required for reactor operation and are only performed as preventive maintenance, for early problem detection in the rod drives. If problems were detected during the motion tests the reactor would be preventively shutdown until the problem were resolved.

The original control rod motion test procedure consisted in moving all control rods (there are nine control rods in EBR-II in the current core configuration) daily, two at a time in opposite direction to maintain steady power. The rods were moved 3 inches from their original position, which is a considerable length in the EBR-II core, with an fuel region height of 13 inches. There existed a potential for mishaps, if the positive reactivity worth of the control rod that was being moved up (inserted in the core, in EBR-II) during the test was not fully compensated by the negative worth of the rod dropped (withdrawn from the core). To prevent undesirable situations, alternative rod motion test procedures were preferred, in which, without significantly affecting the overall operating risk at EBR-II, the test frequency, the length of the rod movement, or both could be reduced.

Alternative rod test procedures were analyzed from the PRA perspective. In particular, changes of the two parameters of interest, the testing period and the rod travel distance, were studied for their impact on the scram function availability and frequency of reactivity insertion initiating events. The control rod motion test procedure affects the results of the EBR-II PRA in the following way.

The probability of the mechanical failure to scram is dominated by the common mode failure of the rods to move. Unless the rod motion is periodically tested, the different hypothetical contributors to rod jamming could be undetected until a scram is required. The periodic testing of the rod motion is therefore important for problem detection. The probability of an undetected rod drive blockage increases with the testing interval. Therefore, increasing the rod testing interval results in an increase in the risk of failure to scram. On the other hand, the probability of a reactivity insertion initiating event depends upon, among other things, the number of "up motion" demands for the control rod (i.e., possibility of the up switch sticking in the on position). From the standpoint of reactivity insertion transients, a long rod motion test interval is more desirable. The optimum test interval will be a trade off between the two effects.

The effect of the length of the rod movement is a more uncertain issue. The mechanisms that could lead to rod jamming are not fully known except for the reactor vessel tilting. Because of the small clearance of the control rod drives through the vessel cover, even a small tipping to the cover would result in the complete blockage of at least one control rod (the farthest one from the tilting point). The other blocking mechanisms, however, are likely to affect the rod motion at certain parts of the rod travel only, or by increasing the motion resistance at certain sections.

Although the cover tilting was estimated to be the largest contributor to common mode control rod blocking problems, it is desirable that the rod motion testing be designed to protect against the other contributions. Thus the criterion to establish the desirable length of the control rod motion (short of moving the rod its entire length or installing torque measurements in the drive motors) is to have, at every test, the combined rod travel that is required to completely bring the reactor down from full power. Examining the power reactivity decrement (the excess reactivity lost from hot standby to full power) and rod worth records for EBR-II, the required combined rod length to shut the reactor down with the usual core configurations is about 6 in. and with alternative configurations does not exceed 8 in. The combined rod motion of 6 to 8 inches can be accomplished for example, by moving only 2 or 3 rods the original distance of 3 in, or moving all the rods 1 in.

A variety of test intervals and distances have been studied. In general, increasing the test interval from the baseline daily test always decreases the scram function availability, with moderate penalties in overall risk of about 3% for a two-day interval and up to 18% for a weekly test pattern. Test patterns of moving only a subset of rods daily on a rotational basis were assumed to have no impact on the probability of undetected rod jamming, since the important blockage sources are common cause contributions.

The best options from the point of view of not increasing the risk of undetected rod jamming are:

- daily rod tests of 3-inch movement for 2 or 3 controls rods, on a rotation pattern.
- daily rod tests of 1-inch movement for all 9 control rods. A two day-test interval would not have a significant impact on risk.

Other patterns analyzed either increased the probability of undetected problems or resulted in a heavier operator burden, which was not only undesirable for operations personnel, but also would have a detrimental impact in the reactivity insertion transient probability by increasing the human error contributions. EBR-II operations is in the process of deciding on the proposed rod test procedures. The rotational daily testing of 2 or 3 rods is the desirable approach since it would require fewer number of rod motions.

B. Upgrade of the Fire Protection System in the Cable Routing Room.

The cable routing room (CRR) in EBR-II contains a variety of safety-related electric equipment, and is equipped with a Halon fire protection system. A recent audit of fire protection practices at the plant recommended an upgrade of the cable routing room fire protection by installing a water sprinkler system. The argument in favor of water is that Halon lacks the cooling power of water, and a fire extinguished with Halon can easily ignite again after the Halon reserves are exhausted. Water-based systems are also believed to be more effective in preventing the spread of fires. In addition, there is a general move against Halon and other cloro-fluoro-carbons known to damage the atmospheric ozone layer.

A Halon system had originally been installed in the CRR because it contained electrical equipment. Water extinguishing systems are traditionally not used in electrical fires to avoid water damage (short circuits) or wetting of the electrical equipment. It is generally agreed that in case of an actual fire where electrical equipment is present water sprinkling cannot worsen the situation. Water sprinkler systems have proved to be very reliable and effective in combating fires. There is little argument, therefore, that a water system in the CRR is desirable if a fire occurs. However, the possibility of an inadvertent actuation of the water sprinkler system, in the absence of a fire, and its effect on the safety-related equipment in the CRR was a concern. A study from the PRA perspective of the possible adverse effects of the spurious actuation of the fire protection system was initiated.

The fire protection water system designed for the CRR is a wet pipe sprinkler system with 11 sprinkler heads of the ON-OFF type (after actuation, they automatically shut off when the temperature has decreased below a threshold). A flow alarm announces the actuation of the system to the reactor operators and the fire department simultaneously. After ruling out a false alarm, the operators are instructed by procedure to initiate an anticipated shutdown. The cable penetrations on the top of the electrical cabinets in the CRR are sealed with a silicone-based material. The system and procedure are designed such that the reactor is in a safe shutdown mode by the time water can penetrate the cabinets and start wetting the electrical equipment. However, if the flow alarm malfunctions or the operators cannot respond quickly enough, wetting of some safety-related equipment may occur when the reactor is still at power.

An event tree was constructed to analyze the possible sequences of events after an inadvertent actuation of the water sprinkler system in the CRR. The event tree is shown in Figure 1. The frequency of the initiating event was estimated mostly based on a DOE database on sprinkler system[5], and the probability of a steam leak from the space heating system in the CRR.

The different branches of the event tree account for the probability of the reactor being still at power when the water penetrates the electrical cabinets. The possibility of shorts to ground and of losing the constant power source, which results in an automatic scram has been considered likely. In the absence of short to grounds and loss of power the reactor trip channels can be adversely affected and produce anomalous signals in the unsafe direction. Credit was taken for the fact that different signals (i.e., high flux and low flow) must deteriorate in

Figure 1. Event Tree for the Inadvertent Actuation of the Sprinkler System in the CRR.

opposite directions to prevent an automatic scram. In addition to the high flux and low flow channels, the subassembly temperature trip channels were included in the possible sequences. The occurrence of an initiating event not induced by water when the trip channels are degraded is also included.

Two events induced by moisture were also accounted for in the event tree, namely a reactivity insertion and a double pump loss of flow. The double pump loss of flow, caused by a loss of the power supply to the MG set clutches, is included since the power supplies are not physically separated yet (see above). Even if the alarm flow had failed, there remains the possibility of a manual scram when the operators get anomalous readings. This is accounted for in the last branch of the event tree.

The probabilities of the different events involved were assigned based on data used in the EBR-II PRA, and using engineering judgement to account for factors such as physical separation and simultaneous failures of similar components in opposite failure modes. The most probable sequences result in either a scram (SCRM), automatic in most cases and manual in a few, or an anticipated shutdown (ANTC). Of little importance are sequences that represent a small contribution to initiating events for reactivity insertion (RISA) and double pump loss of flow (LF2B) but with the trip channels not deteriorated. The only significant sequence is SPKL-14, which represents the unprotected double pump loss of flow. Its annual probability is very small, $2.3\ 10^{-9}$, but it leads to core damage. Two other sequences that could lead to fuel damage are estimated less probable by at least two orders of magnitude.

The analysis has shown that the added risk due to the inadvertent actuation of the fire protection system in the cable routing room is not very significant. The contribution to core damage risk due to a fire in the CRR with the protection of the original Halon system had been found negligible, although it had been assumed that there was no risk of reignition. If that assumption was correct, installation of a water system represents a net, albeit small, increase in risk of fuel damage. On the other hand, if the reignition were probable, the water system would provide added fire protection with only a marginal increase in risk due to inadvertent actuations. This also shows that the proposed modification is not necessary from a safety standpoint.

Other considerations in deciding the installation of a water sprinkler system, such as the cost of drying and testing wetted electrical equipment after an inadvertent actuation, or cost of down time are more elusive. On the basis of pure equipment replacement costs, probabilistic analysis is still helpful in pointing out that the expected equipment savings in a sprinklered fire outweigh the expected losses in spurious sprinkler actuations. However, the assessment of down time and administrative costs of analyzing the occurrence, its consequences, and testing programs is beyond the PRA methodology.

C. Additional EBR-II PRA Applications.

Other PRA applications at EBR-II are currently under way or planned; one affects a safety system in the Fuel Cycle Facility (FCF) and others are directly related to the operations of the reactor.

The FCF is a fuel reprocessing facility adjacent EBR-II that is part of the Integral Fast Reactor (IFR) Program. The facility is provided with a safety exhaust system designed for high reliability. The system consist of an active train with an operating fan and a closed damper, and a standby train with a fan at rest and an open damper. The sense and command subsystem that controls the operations of the dampers and fans actuates on pressure sensors in some of the cells of the facility. The sense and command subsystem has been analyzed at a fault tree level for inadvertent actuations. Because the system is designed for high reliability, a byproduct is a relatively high rate of inadvertent actuations. The probabilistic assessment of the system is being used in analyzing alternatives that, while maintaining the reliability level, can achieve a lower spurious actuation rate. Thus PRA methodology is actively used in the design of the sense and command subsystem, and parts of the system have already been improved as a result of the probabilistic analysis.

Other PRA applications, some only in the planning stages, have emerged in EBR-II, which normally concern the evaluation of alternative procedures or plant modifications. Of interest among the planned applications is a PRA evaluation of one of the refueling procedures.

The control rods in EBR-II are disconnected from their drives during refueling, in the fully down position. The two safety rods, which operate independently of the control rods, are currently left in the up position during refueling, so they can be scrammed if required. The alternative procedure, of refuelling with the safety rods down, will be evaluated. In other words, the estimated risk of performing the operations with negative reactivity ready to be inserted if needed will be contrasted with the risk of refueling with all the negative reactivity already inserted in the core.

IV. CONCLUSION

Although it was only recently completed, the EBR-II PRA has already triggered several applications that have resulted in improvements in the plant. Some of the applications consist in correcting or improving situations that became apparent during the performance of the EBR-II PRA. But the EBR-II PRA has done more than prompting corrective actions, by confirming PRA methodology as a tool for safety analysis in the plant. In this trend, PRA techniques have been instrumental in evaluating procedural changes, hardware modifications, and system design. More applications of PRA methods are under way or are planned for the near future. Involvement of plant engineering and operations staff in the EBR-II PRA has been a positive factor in accepting the use of PRA methodology as part of the safety analyses performed in support of procedures or plant modifications.

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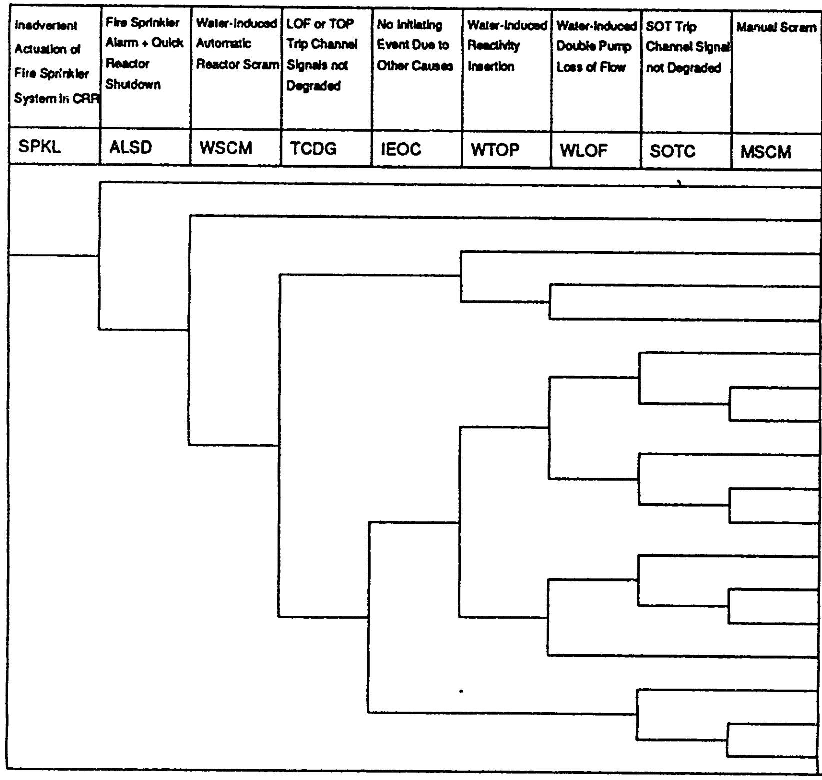
EVENT TREE FOR: Inadvertent Actuation of Fire Sprinkler System in CRR

17-AUG-92



Sequence	Class	Fuel Damage	Frequency
SPKL-1	ANTC		1.3 E-3
SPKL-2	SCRM		9.4 E-6
SPKL-3	RISA		5.2 E-9
SPKL-4	LF2B		4.6 E-7
SPKL-5	ANTC		5.1 E-8
SPKL-6	SCRM		2.6 E-10
SPKL-7	SCRM		2.5 E-10
SPKL-8		CD	1.3 E-11
SPKL-9	SCRM		4.7 E-10
SPKL-10	SCRM		3.8 E-9
SPKL-11		ND	4.2 E-10
SPKL-12	SCRM		4.2 E-7
SPKL-13	SCRM		4.4 E-8
SPKL-14		CD	2.3 E-9
SPKL-15	ANTC		5.1 E-8
SPKL-16	SCRM		8.8 E-12
SPKL-17	SCRM		8.0 E-12
SPKL-18		PCD	8.8 E-13

END

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