EGG-M-92288
CONF-930116--29

EGG-M-92288

ADVANCED TEST REACTOR PROBABILISTIC RISK ASSESSMENT[a]

EGG-M--92288

DE93 005256

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ABSTRACT

A Level 1 probabilistic risk assessment (PRA) incorporating a full-scope external events analysis has been completed for the Advanced Test Reactor (ATR) located at the Idaho National Engineering Laboratory. The major goals of the Level 1 PRA program are the following:

1. to be comparable to the best studies for Department of Energy (DOE) and commercial nuclear power reactors

2. to include an aggressive and comprehensive risk reduction effort for identified significant vulnerabilities or weaknesses

3. to provide comprehensive risk management applications for improved facility operations and safety

4. to transfer ATR PRA insights and improvements in PRA technology to others.

All four goals were met by Revision 1 of the ATR PRA and its supporting analyses and subsequent applications.

THE ADVANCED TEST REACTOR

The ATR, in operation since 1968, is a 250-MW_{th} nuclear test reactor designed to study the effects of intense irradiation on samples of reactor materials. ATR is significantly different from a pressurized water power reactor, having a small, compact core (with aluminum clad plate-type fuel elements), high power density, low primary coolant system pressure and temperature (subcooled at atmospheric conditions), a greater ratio of coolant mass to power, full power heat rejection through a cooling tower instead of a steam turbine, and a confinement structure rather than a containment building.

A plan view of the ATR core is in Figure 1. which shows the unique clover-leaf arrangement of the ATR fuel elements so as to contain nine flux traps within the core. Double contained in-pile tubes for high pressure experiment loops pass through these flux traps. Although the loops are independent of the reactor system, they are reactivity-coupled to the core and can initiate or respond to reactivity transients.

The ATR is regularly scrammed (by a fast insertion of the safety rods) to shutdown for experiment changes and refueling, as often as 15 times a year. Also, the experiments may scram the reactor upon an abnormality to protect the experiment even if the reactor is not threatened. Experiment scrams are the leading cause of spurious scrams for the ATR. A scram is not a significant transient for the ATR because of the full power heat rejection to a cooling tower (no secondary system adjustments are required) and the large reactivity worth of the safety rods (no boron injection is needed to maintain subcriticality). Therefore, a scram or reactor shutdown simply requires continued heat removal for decay power but no other functions. The frequent scrams

[a] Work performed under the auspices of the U.S. Department of Energy, DOE Contract No. DE-AC07-76ID01570.

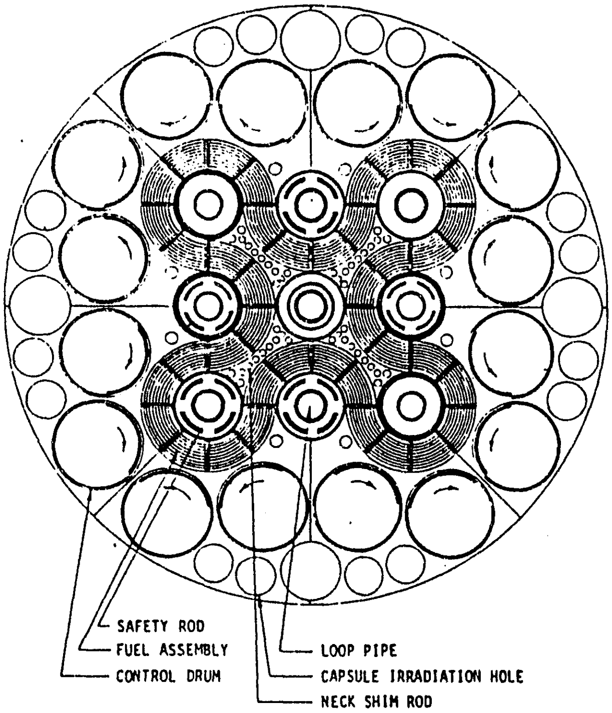

Figure 1 ATR Core Cross-Section

Another characteristic of the ATR which was important to the PRA is the capability for adequate decay heat removal by several different options:

1. by normal primary and secondary coolant circulation powered from off-site power

2. by primary circulation by either of two emergency flow pumps powered by diverse electrical systems (one of which is a continuously running diesel generator) with backup power capabilities

3. by secondary circulation from utility cooling water pumps provided power by a diesel generator or without secondary circulation by direct heat rejection through the uninsulated primary piping to the confinement atmosphere and structures

4. by natural circulation within the reactor vessel as long as coolant makeup is provided (or for more than 14 hrs even without makeup)

5. by primary system depressurization and injection of emergency firewater (which can be supplied by gravity flow or diesel-engine-driven fire pumps).

ATR PRA METHODOLOGY

The best available methodologies were used in order to ensure that the ATR PRA was state-of-the-art to provide the best available risk information for ATR safety bases and decision making. Some new methods were also developed for the ATR PRA. The ATR PRA methodology and results are summarized in Reference 2. Significant features of the ATR PRA include the following:

* A comprehensive search for initiating events was performed involving use of the extensive accident event list generated for the ATR Plant Protection System design basis analysis and Technical Specifications Upgrade in 1976, comprehensive engineering evaluation, and master logic diagram approaches.

* The 20 year ATR experience base was used for defining the frequencies of all but rare initiating events. A Bayesian update of a noninformative prior was used to account for the ATR data.

* System fault tree models were used to quantify yearly frequencies for electrical power support system failure initiating events because of the complex nature of the electrical support system (for loss of the diesel-commercial swing bus, loss of diesel generator power, and loss of all ac power).

* State-of-the-art generic data bases were developed for component failure data,[3] piping system component leakages and ruptures,[4] and inadvertent actuation of fire suppression systems.[5]

* ATR data was used to supplement the generic failure data and test and maintenance outage times for most of the important components. The failure rates were obtained using a Bayesian update with the generic value as the prior and the ATR data as the evidence.

* A five-year running average was used to evaluate the frequencies of expected transients or component failure data to reflect the effect of plant changes and recent plant performance when these effects were significant.

* A comprehensive estimation was performed for potential fuel damaging large primary coolant system rupture frequencies based on the results of break spectrum analyses and on several diverse methods (statistical data analysis for historical data sources,

probabilistic fracture mechanics, and expert judgement elicitation). Three different approaches for the statistical data estimations were used (U.S. nuclear reactor experience, basic piping systems elemental data, and the Thomas piping systems elemental approach[6]). The results of the five estimation approaches were combined using a weighted geometric averaging technique with a large uncertainty (an error factor of 30) estimated for this rare event.

* An approach similar to the above was also used to estimate the event frequency for a large experiment loop rupture that could potentially result in a large reactivity increase sufficient to threaten core fuel damage. This event frequency was very low due to the very narrow range of conditions for a core threatening event and the double containment for the loops within the core.

* All the different types of dependent failures were accounted for including those resulting from spatial dependencies in the external events analysis. In addition to explicit modeling of any known dependent failure mechanisms, parametric modeling of dependent failures of similar components was performed. Similar components within systems were identified and parametric dependent failure events were added to the system fault trees.

* Detailed human reliability analyses (HRA) were performed for the human errors important to the event sequences. The ATR PRA HRA consisted of three subtasks: screening analysis of pre-accident human errors, screening analysis of operator errors during an accident, and refined, detailed HRA of the resulting dominant human errors. The refined HRA included a combination of the technique for human error rate prediction (THERP)[7] and human cognitive reliability (HCR).[8] The HRA was important to the ATR PRA because of the long times available for accident recovery for many of the event sequences.

* A comprehensive screening of all types of external events was performed (ten events were evaluated in detail when they did not screen out)

* A comprehensive seismic risk analysis was performed including seismic-fire-flood interactions, relay chatter for risk-significant relays, and detailed human reliability analysis specific to seismic event conditions as a function of instrumentation availability.

* A comprehensive internal fire risk analysis was performed including use of the location transformation technique (vital area analysis) for screening, COMPBRN-III analyses, and evaluation of fire-flood interactions. Fire-flood interactions were an important contributor to the external events analysis.

* A comprehensive internal flooding risk analysis was performed including consideration of leakage (spray) and rupture events, inadvertent actuation of fire suppression systems, and use of the location transformation technique for screening.

* A comprehensive high wind risk analysis was performed similarly to the seismic risk analysis with consideration of wind-fire-flood interactions.

The PRA models and results, including external events, are loaded on the CAFTA/RMQS and SAPHIRE 4.0 PRA PC work station code packages for ongoing risk assessment and risk management applications.

ATR RISK REDUCTION

The ATR risk reduction effort was initiated at the beginning of the Level 1 PRA program to respond to risk-significant vulnerabilities or findings as they were defined by the ongoing PRA. Sensitivity studies were performed to identify the dominant contributors to the fuel damage frequency (fdf) and to identify the most risk-effective upgrades that could reduce the fdf. Risk-significant findings were communicated to management with recommendations for the most effective upgrades from both a risk reduction and a cost effectiveness point of view. One of the effective risk reduction efforts was an extensive upgrade of the emergency procedures incorporating insights and guidance from the Level 1 PRA for internal events.

The preliminary total Level 1 PRA fdf at the conclusion of the external events analysis was noted to be an outlier for nuclear reactors at a mean value of approximately 1.0E-3/yr. However, the analyses had also determined that relatively few sequences dominated the fdf and relatively inexpensive upgrades or appropriate guidance for planned and ongoing plant safety and operational upgrades might significantly reduce the total fdf. At that point, goals were set for the internal and external events and numerous sensitivity cases were run to identify the most risk-significant and cost effective upgrades. A recommended list of plant upgrades was transmitted to and accepted by management including seismic restraint improvements, various simple upgrades, and a relocation of the Utility Battery-Backed Power System (which eliminates the dominant fuel damage threat, flooding propagation from the diesel generator area to the switchgear area and Utility Battery-Backed Power System, reducing the mean fdf by 70 percent), and providing input to the design of the new process control system. Action was started on the upgrades, including methods to prevent and mitigate water propagation to sensitive electrical equipment should a large fire and subsequent flooding occur in the diesel generator area, and protection of the emergency depressurization system power supplies from fire suppression system spray. The final version of Revision 1 of the ATR Level 1 PRA[1] reflects the plant configuration with the completion of these upgrades which are to be completed in early 1993. The upgrades reduced the total mean fdf by about a factor of 20 from the preliminary result without upgrades.

ATR PRA RESULTS

The total ATR Level 1 PRA fdf after completion of the external events upgrades is approximately 5E-5/yr (mean value) with a lower 5% bound of 9.5E-6/yr and an upper 95% bound of 1.4E-4/yr. Internal events and external events contribute about equally to the total fdf. Relative contributions to the total fdf by the type of initiating event are shown in Figure 2. Seismic events are the dominant contributor as is often the case for nuclear reactors after the easy risk reduction upgrades are accomplished. The volcanism contributor, unique to the ATR, is for a simple, conservative assessment for a potential opening of a volcanic rift with significant lava flow. The contribution for this event could be reduced, if it were important to do so, by a detailed analysis or better geologic data. The "reactivity loop" contributor is due to loop experiment upsets which could lead to a reactivity increase in the ATR core.

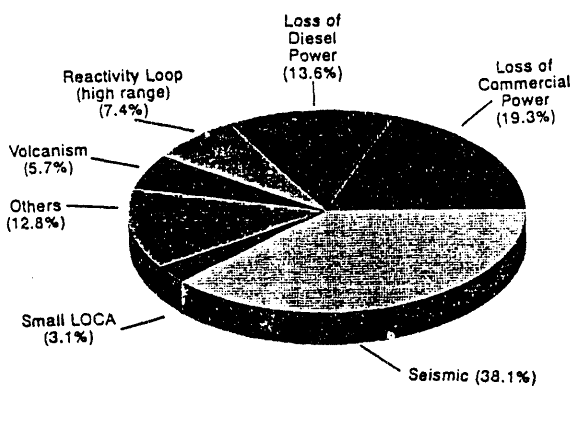

Figure 2 Level 1 Fuel Damage Contributors

ATR RISK MANAGEMENT

A major objective of the ATR PRA program is the application of the results, insights, and models of the PRA for risk-based decision making or risk management. To enable use of the ATR PRA for this purpose, the results and insights and uses of the PRA information and models are being disseminated throughout the ATR operating and support organizations, training lectures are provided to reactor operators, and the SAPHIRE 4.0 ATR PRA software is being loaded onto various PCs for ATR PRA inquiries and sensitivity studies. The development of summary reports to communicate results, insights, and uses of the ATR PRA is described in another paper at this conference. The following are active risk management applications of the ATR PRA:

* for operational improvements such as operator training for transient management and emergency procedures improvements

* review of proposed plant or operational changes for risk issues or for prioritization

* evaluation of operational occurrences and safety or risk concerns or questions to aid management in identifying safety issues, to place occurrences or concerns in perspective to risk, to identify safety issue response options and alternatives, and to prioritize the application of resources

* providing guidance for maintenance improvements and aging and life extension programs (such as developing lists of components prioritized as to their importance to plant risk or their potential importance should they be failed or unavailable and providing input for the risk significance of equipment outage options and outage times)

* provide guidance and input to other ATR safety and operational upgrade programs such as the upgraded safety analysis report, accident monitoring instrumentation, emergency planning, new process and experiment control systems, simulator upgrades, safety parameter display system, fire and life safety upgrades, and safety equipment qualification.

ATR risk management activities and applications of the ATR PRA are compiled and reported to management twice a year in an ATR Risk Management Report. This report includes a review of operational incidents for their risk-significance and reports on an evaluation of operating data for the potential affect on the ATR PRA. Event data are collected and tracked to detect any significant changes or trends that could adversely affect plant risk.

PRA TECHNOLOGY TRANSFER

The ATR PRA, its results and insights, and methodology developments or data have been shared with other DOE reactor PRA programs and analysts as we have worked together to provide quality and applicable risk assessments for the very unique DOE reactors. The PRAs for the very different DOE reactors are not directly comparable or applicable to each other, but the PRA techniques, data development and data bases, and lessons learned are often of use for all of the PRAs. PRA program managers and analysts for the DOE "Class A" reactors have been meeting three times a year to share information and techniques, concerns and problems, and to provide peer review for the different on-going PRAs.

Specific results from several tasks associated with the ATR PRA were also seen as potentially beneficial to a wide number and variety of DOE and commercial power reactor risk assessments. These tasks include the development of data bases needed for the ATR PRA such as the following:

* the development of a generic component failure data base that was obtained primarily from plant-specific data collected for PRAs and reliability studies[3]

* the development of piping system component leakage and rupture frequencies based on Licensee Event Reports[4]

* the development of fire suppression system inadvertent actuation frequencies based on DOE and commercial power reactor experience[5]

The work for all three of the above tasks was performed such that it met the needs of the ATR PRA but was also expanded to meet the needs of other PRAs. The component failure data base, for example, includes data for sodium reactor systems. The resultant data bases were documented in reports made available for unlimited distribution[3,4,5] with over 150 copies of each report sent to PRA practitioners within the United States. The data bases are being used in other DOE and commercial power reactor PRAs, in chemical plant risk analyses, and in hazardous waste processing facility risk analyses.

The fire suppression system inadvertent actuation data analysis covered the following systems: wet-pipe, dry-pipe, preaction, deluge, halon, and carbon dioxide. Surveys of experience with fire suppression systems for DOE facilities from 1952 through 1980 and commercial power reactor experience from 1980 through 1989 were used for the data analysis. Recommended generic frequencies for inadvertent actuation of the various types of fire suppression systems were obtained by combining both sets of data. However, categorization of events by system was in some cases uncertain because of incomplete event descriptions.

The component failure and piping system component leakage and rupture data bases and their development are described in other papers for this conference.

CONCLUSIONS AND OUTLOOK

The major goals of the Level 1 PRA program are the following:

1. to be comparable to the best studies for Department of Energy (DOE) and commercial nuclear power reactors

2. to include an aggressive and comprehensive risk reduction effort for identified significant vulnerabilities or weaknesses

3. to provide comprehensive risk management applications for improved facility operations and safety

4. to transfer ATR PRA insights and improvements in PRA technology to others.

All four goals were met by Revision 1 of the ATR PRA[1] and its supporting analyses and subsequent applications.

The development of the ATR PRA has been an aggressive, integrated program to define a realistic, applicable risk profile of the ATR using the best available methods and to apply the results and insights to appropriate, cost-effective risk reduction actions and support of risk management. There were no example PRAs that could be modified for the ATR. The unique reactor design and operational mission required a full scope PRA including external events and the development of data bases and techniques applicable to the ATR. Additionally, the long successful operating history of the ATR provided a valuable base of data and experience which helped to provide a quality PRA that is directly applicable.

The ATR PRA analyses are to be completed by October 1993. Analyses for Level 2 and Level 3 of the ATR PRA are ongoing with the Level 2 analyses soon to be completed. These analyses are requiring development and use of additional innovative techniques because of the uniqueness of the plant and its mission and because of additional opportunities for cost-effective transient management improvements. PRA analyses for shutdown operations, including fuel and experiment handling operations, have also been started. These analyses can also be expected to result in changes and improvements to the ATR and its operation.

The overall goal of the ATR PRA program is to control the risk and the probability for a severe accident and thereby help to assure safe and cost-effective operation over the continued mission of a unique and valuable facility. The ATR PRA program is helping to meet this goal.

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