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Handbook of Parameter Estimation for Probabilistic Risk Assessment

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ABSTRACT

Probabilistic risk assessment (PRA) is a mature technology that can provide a quantitative assessment of the risk from accidents in nuclear power plants. It involves the development of models that delineate the response of systems and operators to accident initiating events. Additional models are generated to identify the component failure modes required to cause the accident mitigating systems to fail. Each component failure mode is represented as an individual "basic event" in the systems models. Estimates of risk are obtained by propagating the uncertainty distributions for each of the parameters through the PRA models.

The data analysis portion of a nuclear power plant PRA provides estimates of the parameters used to determine the frequencies and probabilities of the various events modeled in a PRA. This handbook provides guidance on sources of information and methods for estimating the parameters used in PRA models and for quantifying the uncertainties in the estimates. This includes determination of both plant-specific and generic estimates for initiating event frequencies, component failure rates and unavailabilities, and equipment non-recovery probabilities.

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FOREWORD

During the last several years, both the U.S. Nuclear Regulatory Commission (NRC) and the nuclear industry have recognized that probabilistic risk assessment (PRA) has evolved to the point where it can be used in a variety of applications including as a tool in the regulatory decision-making process. The increased use of PRA has led to the conclusion that the PRA scope and model must be commensurate with the applications. Several procedural guides and standards have been and are being developed that identify requirements for the PRA models. For example, the "Standard For Probabilistic Risk Assessment For Nuclear Power Plant Applications" published by The American Society of Mechanical Engineers (ASME) in 2002 (ASME-RA-S-2002) defines requirements for PRA analysis used to develop risk-informed decisions for commercial nuclear power plants, and describes a process for applying these requirements in specific applications. This handbook was generated to support these documents by providing a compendium of good practices that a PRA analyst can use to generate the parameter distributions required for quantifying PRA models.

The increased use of risk assessment has also helped promote the idea that the collection and analysis of event data is an important activity in and of itself. In particular, the monitoring of equipment performance and evaluation of equipment trends can be used to enhance plant performance and reliability. The reference material provided in this handbook can support those efforts.

This handbook provides references on sources of information and methods for estimating parameter distributions. This includes determination of both plant-specific and generic estimates for initiating event frequencies, component failure rates and unavailability, and equipment non-recovery probabilities, all of which directly supplement the ASME PRA standard.

This handbook provides the basic information needed to generate estimates of the parameters listed above. It begins by describing the probability models and plant data used to evaluate each of the parameters. Possible sources for the plant data are identified and guidance on the collection, screening, and interpretation is provided. The statistical techniques (both Bayesian and classical methods) required to analyze the collected data and test the validity of statistical models are described. Examples are provided to help the PRA analyst utilize the different techniques.

This handbook also provides advanced techniques that address modeling of time trends. Methods for combining data from a number of similar, but not identical, sources are also provided. This includes empirical and hierarchical Bayesian approaches. Again, examples are provided to guide the analyst.

This handbook does not provide guidance on parameter estimation for all types of events included in a PRA. Specifically, common cause failure and human error probabilities are not addressed. In addition, guidance is not provided with regard to the use of expert elicitation. For analysis of these events, the PRA analyst should consult other sources, some of which are cited in Chapter 1.

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ABBREVIATIONS

AC	alternating current
AFW	auxiliary feedwater
ALWR	Advanced Light Water Reactor
ANO	Arkansas Nuclear One
ANSI	American National Standards Institute
ASEP	Accident Sequence Evaluation Program
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
BWR	boiling water reactor
CDF	core damage frequency
c.d.f.	cumulative distribution function
ComEd	Commonwealth Edison
ConEd	Consolidated Edison
CPC	Consumers Power Company
CREDO	Centralized Reliability Data Organization
CVC	chemical and volume control
DC	direct current
DG	diesel generator
DOE	Department of Energy
ECCS	emergency core cooling system
EDF	empirical distribution function
EDG	emergency diesel generator
EEI	Edison Electric Institute
EF	error factor
EPIX	Equipment Performance and Information Exchange System
EPRI	Electric Power Research Institute
FTR	failure to run
FTS	failure to start
GE	General Electric
HCF	Hardware Component Failure
HEP	human error probability
HHRAG	Human and Hardware Reliability Analysis Group
HPCI	high pressure coolant injection
HPP	homogeneous Poisson process
HRA	human reliability analysis
HVAC	heating, ventilation and air conditioning
IEEE	Institute of Electrical and Electronics Engineers
IGSCC	intergranular stress corrosion cracking
i.i.d.	independent identically distributed
INEL	Idaho National Engineering Laboratory
INEEL	Idaho National Engineering and Environmental Laboratory
INPO	Institute of Nuclear Power Operations
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
IPRDS	In-Plant Reliability Data Base for Nuclear Power Plant Components
IREP	Interim Reliability Evaluation Program
LER	Licensee Event Report
LCO	limiting conditions of operation
LOCA	loss of coolant accident
LOSP	loss of offsite power
MCMC	Markov Chain Monte Carlo

MLE	maximum likelihood estimate, or maximum likelihood estimator, depending on the context (see *estimate* in glossary)
MOV	motor-operated valve
MSE	mean square error
MSIV	main steam isolation valve
MTTF	mean time to failure
MTTR	mean time to repair
NA	not applicable
NHPP	nonhomogeneous Poisson process
NPE	Nuclear Power Experience
NPP	Nuclear Power Plant
NPRDS	Nuclear Plant Reliability Data System
NRC	U.S. Nuclear Regulatory Commission
NREP	Nuclear Reliability Evaluation Program
NSAC	Nuclear Safety Analysis Center
NUCLARR	Nuclear Computerized Library for Assessing Reactor Reliability
OREDA	Offshore Reliability Data
PCS	Power Conversion System
PECO	Philadelphia Electric Company
p.d.f.	probability density or distribution function
PLG	Pickard, Lowe and Garrick, Inc.
PORV	power operated relief valve
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PWR	pressurized water reactor
RAC	Reliability Analysis Center
RADS	Reliability and Availability Data System
RCIC	reactor core isolation cooling
RHR	residual heat removal
RMIEP	Risk Methods Integration and Evaluation Program
RPS	reactor protection system
RSSMAP	Reactor Safety Study Methodology Application Program
SAIC	Scientific Applications International Corporation
SBLOCAs	small-break loss-of-coolant accidents
SCSS	Sequence Coding and Search System
SKi	Swedish Nuclear Power Inspectorate
SRV	safety relief valve
SSPI	Safety System Performance Indicator
SwRI	Southwest Research Institute
U.K.	United Kingdom
U.S.	United States