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David Passehl
Senior Reactor Analyst
USNRC, Region III

Overview

- Operator Actions/Human Errors
- Risk Data
- PRA Standards
- PRA Quality and Peer Review Process

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Operator Actions/Human Errors

Human error contribution to risk can be large.

- Plant studies have indicated that operator error may contribute a large percentage of total nuclear plant risk.
- Human errors may have significantly higher probabilities than hardware failures.
- Humans can circumvent the system design.

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Operator Actions/Human Errors

Human Reliability Analysis

- Starts with the basic premise that humans are, in effect, part of the system.
- Identifies and quantifies the ways in which human actions contribute to the initiation, propagation, or termination of accident sequences.

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Operator Actions/Human Errors

"Human Reliability" is the probability that a person will:

- Correctly perform some system-required activity, and
- Perform no extraneous activity that can degrade the system.

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Operator Actions/Human Errors

Categories of Human Error

- Errors can occur throughout the accident sequence
  - Pre-initiator errors
  - As a contribution or cause to initiating events
  - Post-initiator errors

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Operator Actions/Human Errors

- Generally, two types of human errors are defined:
  - Errors of omission --Failure to perform a required action or step, e.g., failure to monitor makeup tank level
  - Errors of commission-- Action performed incorrectly or wrong action performed, e.g., opening the wrong valve, turning off SI
- Normally only the first type is modeled due to uncertainty in being able to identify errors of commission, and lack of modeling and quantification methods to address such errors
  - ATHEANA research program is directed at errors of commission

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Operator Actions/Human Errors

Some PRA-Based Techniques

- Technique for Human Error Rate Prediction (THERP)
- Accident Sequence Evaluation Program (ASEP)
- Standardized Plant Analysis Risk (SPAR) human reliability analysis method (SPAR-H)*

*NUREG/CR-6883, "The SPAR-H Human Reliability Analysis Method"

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Operator Actions/Human Errors
SPAR-H

SPAR-H segregates human failure events (HFEs) into

- Diagnosis failures (nominal HEP 0.01)
- Action failures (nominal HEP 0.001)
  - Quantifies the two failure types separately.

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Operator Actions/Human Errors
SPAR-H

Nominal HEPs adjusted based on 8 PSFs:

- Available time
- Stress
- Complexity
- Experience/Training
- Procedures
- Ergonomics
- Fitness-for-Duty
- Work Processes

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Operator Actions/Human Errors

Table 3-1. INEEL Results of SPAR Conditional Core Damage Probability Analyses Ranked by Event Importance.

Analysis No.	ASP Reference and Screening Basis Value (CCDP)	Facility	Event Date	LER and AIT Numbers	Risk Importance Measures: SPAR Analysis CCDP	Risk Importance Measures: RAW Factor Increase (CCDP/CDP)	Risk Importance Measures: Event Importance (CCDP-CDP)	Risk Importance Measures: Human Failure Percent Contribution to Event Importance
1	2.1E-04	Wolf Creek Generating Station 1	01/30/96	482-96-001	5.2E-03	24,557	5.2E-03	100
2	2.1E-04	Oconee 2	10/19/92	270-92-004	3.3E-03	95.5	3.2E-03	100
3	1.2E-04	Perry 1	04/19/93	440-93-011	2.1E-03	242.1	2.1E-03	100
4	2.2E-04	Oconee 2	04/21/97	270-97-001	7.1E-04	2.5	4.5E-04	100
5	1.5E-05	Limerick 1	09/11/95	352-95-008	4.8E-04	9.3	4.3E-04	100
6	2.0E-04	Indian Point 2	08/31/99	AIT 50-246/99-03	3.5E-04	25	3.4E-04	100
7	9.5E-05	McGuire 2	12/27/93	370-93-008	4.6E-04	2.4	2.7E-04	82
8	NA	Hatch	01/26/00	321-00-002	2.5E-04	15.2	2.3E-04	100
9	1.1E-04	Robinson 2	07/08/92	261-92-015, 261-92-017, and 261-92-018	2.3E-04	4.2	1.8E-04	100
10	6.5E-05	Haddam Neck	06/24/93	213-93-006 and 213-93-007, AIT 213/93-80	2.0E-04	4.3	1.5E-04	48
11	3.2E-05	Oconee 1, 2, and 3	12/02/92	269-92-018	1.5E-04	125	1.5E-04	100
12	1.3E-05	River Bend 1	09/08/94	458-94-023	1.2E-04	2.5	1.2E-04	100
13	1.8E-04	Sequoyah 1 and 2	12/31/92	327-92-027	1.1E-04	14,103	1.1E-04	100
14	5.5E-05	Beaver Valley 1	10/12/93	334-93-013	6.2E-05	10,690	6.2E-05	100
15	NA	Dresden 3	05/15/96	249-96-004	2.6E-05	15.7	2.4E-05	100
16	1.1E-04	St. Lucie 1	10/27/97	335-97-005	3.8E-05	2.9	2.5E-05	100
17	4.6E-05	Seabrook 1	05/21/96	443-96-005	3.E-05	2.3	2.5E-05	100
18	6.5E-05	Comanche Peak 1	06/11/95	445-95-003 and 445-95-004	1.9E-05	146.2	1.9E-05	10
19	6.0E-05	ANO Unit 2	07/29/95	368-95-001	1.4E-05	73.7	1.4E-05	100
20	5.6E-04	ANO Unit 1	05/16/96	313-96-005	9.6E-05	50.5	9.4E-06	100
21	3.7E-05	D. C. Cook 1	09/12/95	315-95-011	2.3E-05	1.2	4.9E-06	90
22	1.3E-04	LaSalle 1	09/14/93	373-93-015	4.5E-05	1.07	3.0E-06	100
23	7.7E-05	Millstone 2	01/25/95	336-95-002	2.6E-05	1.04	1.0E-06	100

Risk Data

Numerous data collection efforts have been conducted.

- WASH-1400 Study
- Idaho National Laboratory
- Oak Ridge National Laboratory
- Sandia National Laboratory
- Nuclear Regulatory Commission
- Electric Power Research Institute

Risk Data

Examples of Types of Analysis

Estimate the probability of failure on demand

- A commonly used estimate of failure probability is by the fraction (number of failures)/(number of demands). It is a dimensionless quantity.

- The uncertainty in the estimate depends on how many demands and failures were counted; the larger the data set, the smaller the uncertainty in the estimate.

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Risk Data

Examples of Types of Analysis

Estimate the rate at which failures occur in time.

- Data set for this analysis contains a count of failures in some study time. A common estimate of failure rate is the fraction (number of failures)/(total time when such failures could occur). It has dimension 1/time.

- The uncertainty depends on the failure count and the exposure time, with a long exposure time reducing the uncertainty in the estimate.

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Risk Data

Sources:

Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995 (NUREG/CR-5750)

Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants (NUREG/CR-6928)

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PRA Standards

- In the USA, the technical adequacy of licensee PRAs varies widely.
- PRA Standards and industry peer review process have been developed, and can be used to provide an understanding of the strengths and weaknesses of a PRA.
- NRC issued RG 1.200 (and supporting SRP Chapter 19.1 that provides "An Approach for Determining the Technical Adequacy of PRA Results for Risk-Informed Activities"

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PRA Standards

- Standards have been developed for CDF and LERF for:
  - Internal events at full power (ASME)
  - External initiating events (ANS)
  - Low power and shutdown operation (ANS)
  - Internal fires (ANS)

- ASME: Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications (internal initiating events at full power) issued April, 2002, and Addendum A in December, 2003.

- Endorsed in Appendix A to RG 1.200

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PRA Standards

ASME PRA STANDARD

- Provides a Standard for performing and using a PRA
  - Definitions
  - Risk assessment application process
  - Risk assessment technical requirements
  - PRA configuration control
  - Peer review

- The Standard is a "what to do" but not a "how to do" Standard – it does not prescribe specific methods or standard assumptions

- One objective of the peer review is to assess the appropriateness of significant assumptions

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PRA Standards

INDUSTRY PEER REVIEW PROCESS

- NEI-00-02: PRA Peer Review Process Guidance, supported by "sub-tier criteria" and guidance for self assessment against the ASME Standard

- Endorsed in Appendix B to RG 1.200

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PRA Quality and Peer Review Process

- Defined in RG 1.174 and RG 1.200
  - For a given application, PRA Quality is determined by the appropriateness of
    - Scope (internal and external initiating events, full power and low power and shutdown operating modes, CDF, LERF, level 2, level 3
    - Level of detail
    - Technical adequacy

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PRA Quality and Peer Review Process

- Main body of RG 1.200 provides general guidance to licensees on how to use PRA standards (or industry peer review program) to demonstrate and document that the PRA input to a decision is supported by a PRA of sufficient quality.

- Appendices provide Staff regulatory position on the individual Standards or peer review process guidance

- Staff review will focus on those areas where alternatives to the Staff regulatory position are used

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End

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