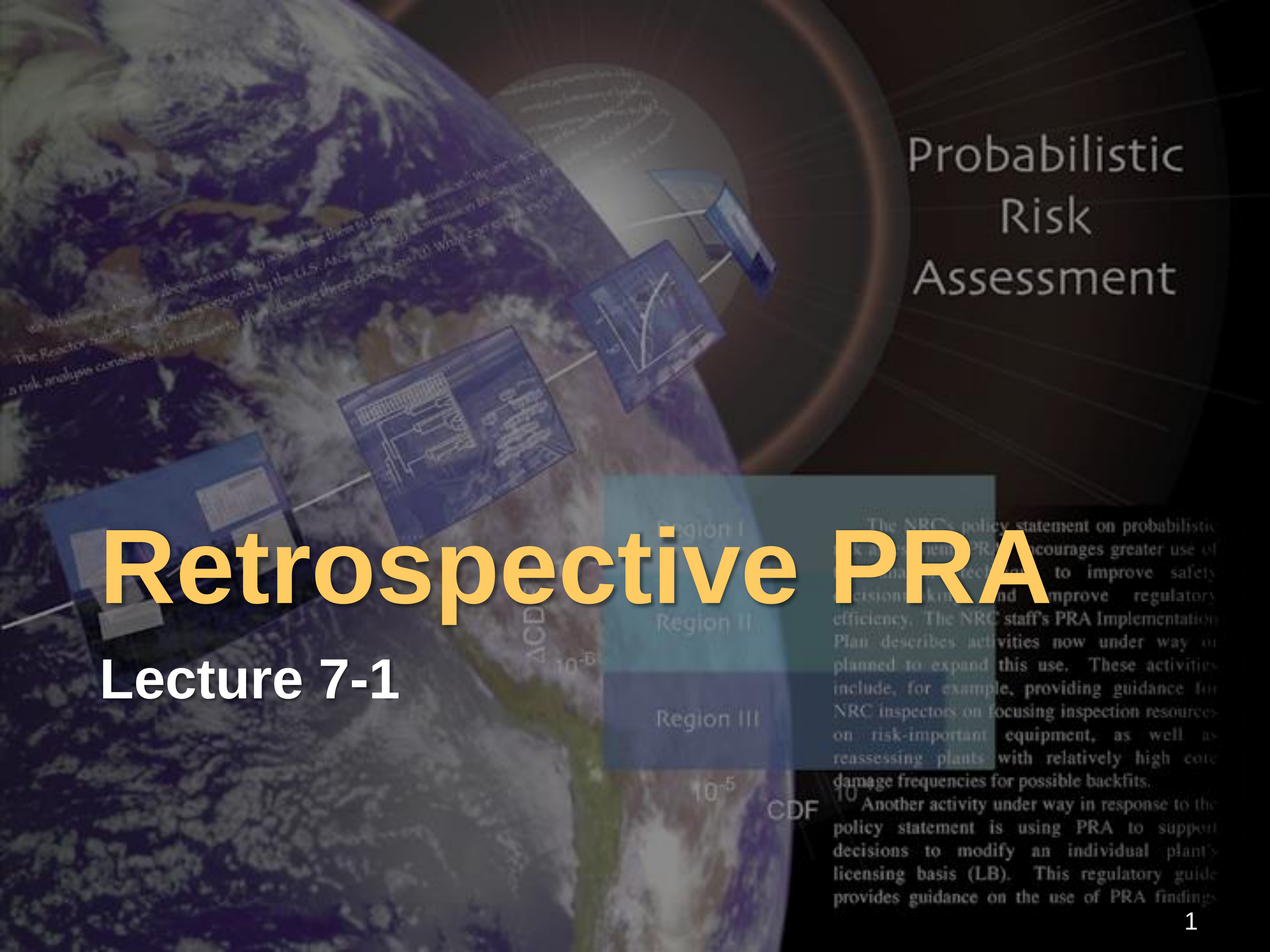

Retrospective PRA

Lecture 7-1

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Schedule

	Wednesday 1/16	Thursday 1/17	Friday 1/18	Tuesday 1/22	Wednesday 1/23
Module	***1: Introduction***	***3: Characterizing Uncertainty***	***5: Basic Events***	***7: Learning from Operational Events***	***9: The PRA Frontier***
9:00-9:45	L1-1: What is RIDM?	L3-1: Probabilistic modeling for NPP PRA	L5-1: Evidence and estimation	L7-1: Retrospective PRA	L9-1: Challenges for NPP PRA
9:45-10:00	**Break**	**Break**	**Break**	**Break**	**Break**
10:00-11:00	L1-2: RIDM in the nuclear industry	L3-2: Uncertainty and uncertainties	L5-2: Human Reliability Analysis (HRA)	L7-2: Notable events and lessons for PRA	L9-2: Improved PRA using existing technology
11:00-12:00	W1: Risk-informed thinking	W2: Characterizing uncertainties	W4: Bayesian estimation	W6: Retrospective Analysis	L9-3: The frontier: grand challenges and advanced methods
12:00-1:30	**Lunch**	**Lunch**	**Lunch**	**Lunch**	**Lunch**
Module	***2: PRA Overview***	***4: Accident Sequence Modeling***	***6: Special Technical Topics***	***8: Applications and Challenges***	***10: Recap***
1:30-2:15	L2-1: NPP PRA and RIDM: early history	L4-1: Initiating events	L6-1: Dependent failures	L8-1: Risk-informed regulatory applications L8-2: PRA and RIDM infrastructure	L10-1: Summary and closing remarks
2:15-2:30	**Break**	**Break**	**Break**	**Break**	
2:30-3:30	L2-2: NPP PRA models and results	L4-2: Modeling plant and system response	L6-2: Spatial hazards and dependencies	L8-3: Risk-informed fire protection	Discussion: course feedback
3:30-4:30	L2-3: PRA and RIDM: point-counterpoint	W3: Plant systems modeling	L6-3: Other operational modes L6-4: Level 2/3 PRA: beyond core damage	L8-4: Risk communication	Open Discussion
4:30-4:45	**Break**	**Break**	**Break**	**Break**	
4:45-5:30	Open Discussion	W3: Plant systems modeling (cont.)	W5: External Hazards modeling	Open Discussion	
5:30-6:00		Open Discussion	Open Discussion		

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Learning Objectives

- Retrospective PRA – concept and use
- NRC Accident Sequence Precursor (ASP) program and key results
- Related activities
- Other uses

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Resources

- I. Gifford, C. Hunter, and A. Gilbertson, "U.S. Nuclear Regulatory Commission Accident Sequence Precursor Program: 2017 Annual Report," May 2018. (ADAMS ML18130A856)
- U.S. Nuclear Regulatory Commission, "Accident Sequence Precursor (ASP) Program https://www.nrc.gov/about-nrc/regulatory/research/asp.html
- N. Siu, et al., "Accidents, near misses, and probabilistic analysis: on the use of CCDPs in enterprise risk monitoring and management," *Proceedings of ANS International Topical Meeting on Probabilistic Safety Assessment (PSA 2017)*, Pittsburgh, PA, September 24-28, 2017.

Other References

- K.A. Coyne, “Risk-Informed Regulation at the U.S. Nuclear Regulatory Commission,” April 14, 2016. (ADAMS ML16105A427)
- U.S. Nuclear Regulatory Commission, “Workshop on the Use of PRA Methodology for the Analysis of Reactor Events and Operational Data,” *NUREG/CP-0124*, 1992.
- V.M. Bier (ed.), *Accident Sequence Precursors and Probabilistic Risk Analysis*, University of Wisconsin Press, Madison, WI, 1998.
- Nuclear Energy Agency, “Proceedings of the Workshop on Precursor Analysis,” *NEA/CSNI/R(2003)11*, 2003.
- J.R. Phimister, V.M. Bier, and H.C. Kunreuther, *Accident Precursor Analysis and Management: Reducing Technological Risk Through Diligence*, Committee on Precursors, National Academy of Engineering, National Academies Press, New York, 2004.
- J.W. Minarick and C.A. Kukielka, “Precursors to Potential Severe Core Damage Accidents: 1969-1979, a Status Report,” NUREG/CR-2497, June 1982.
- G. Apostolakis and A. Mosleh, “Expert opinion and statistical evidence: an application to reactor core melt frequency,” *Nuclear Science and Engineering*, **70**, 135-149, 1979.
- U.S. Nuclear Regulatory Commission, “ROP References” https://www.nrc.gov/reactors/operating/oversight/program-documents.html
- U.S. Nuclear Regulatory Commission, “NRC Incident Investigation Program,” *Management Directive 8.3*, June 25, 2014. (ADAMS ML13175A294)

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What is a Retrospective PRA?

- Preceding lectures address “prospective PRA analysis” – identifying and prioritizing possibilities to assist forward-looking decision making
- “Retrospective PRA analysis” applies a PRA modeling framework and “what-if” thinking to past events: how close did an incident come to becoming an accident?

What can go wrong?
What are the consequences?
How likely is it?

→

What could have gone wrong?
What would have been the consequences?
How likely was it?

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Why Use Retrospective PRA?

- Support risk-informed prioritization of events for attention and further investigation, possible “early warning” signals
- Support risk-informed, graded responses to inspection findings
- Provide a different (but still risk-oriented) perspective on plant safety

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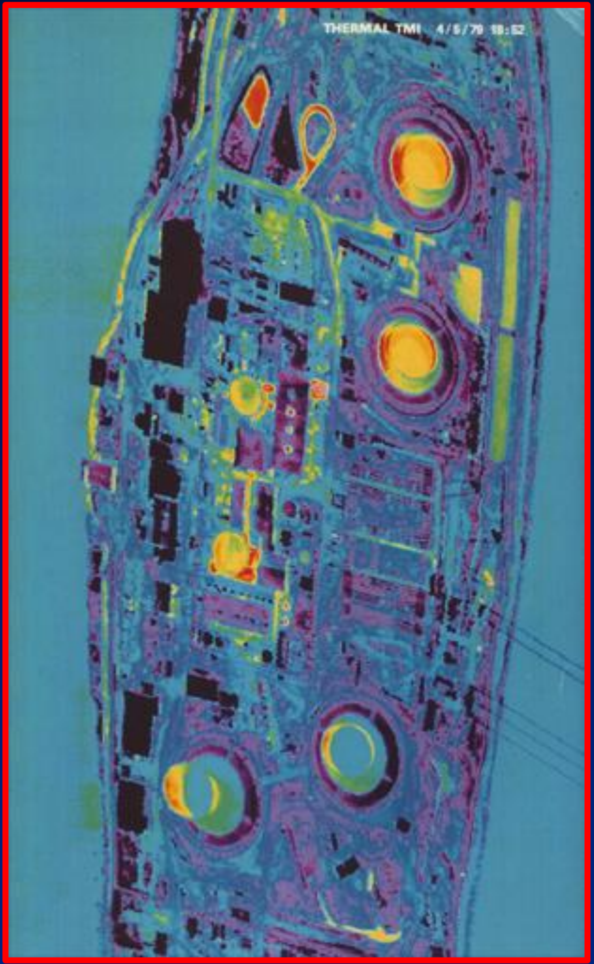
Early Warning Potential

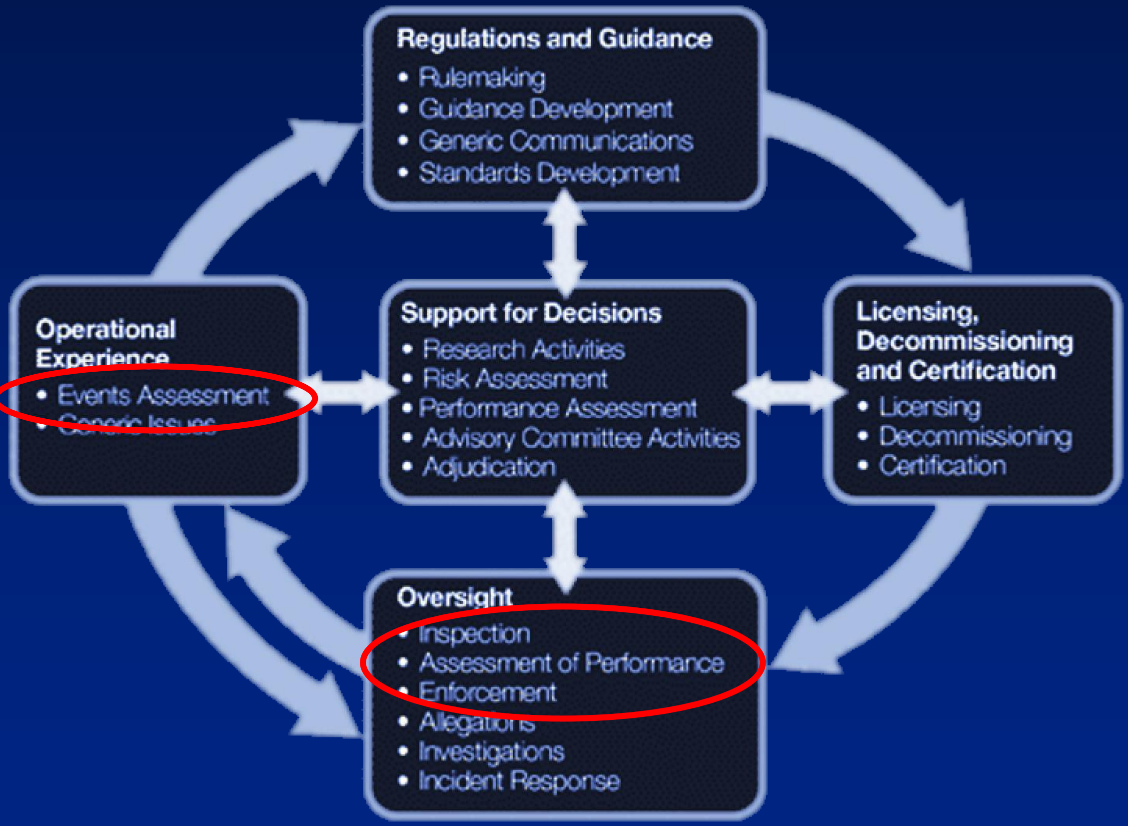
- Davis-Besse (1977) - LER 346/77-016
  – Partial loss of feedwater; stuck-open pressurizer PORV; operators failed to recognize stuck-open PORV
  – CCDP = 7×10^{-2} (analysis ~1982)*
- Three Mile Island 2 (1979) - LER 320/79-012
  – Total loss of feedwater; stuck-open pressurizer PORV; operators failed to recognize stuck-open PORV; subsequent operator errors led to core damage
  – CCDP = 1



Adapted from cover page, M. Rogovin and G.T, Frampton, Jr., “Three Mile Island: A Report to the Commissioners and to the Public,” Nuclear Regulatory Commission Special Review Group, January 1980.

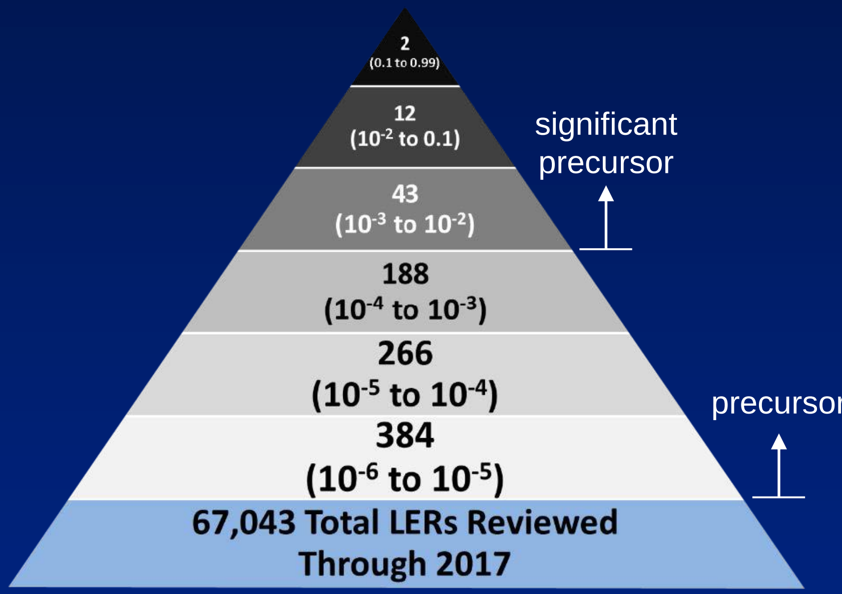
*Based on then-current models. Current estimate ~1E-3 (still a “significant precursor”).

Reminder – NRC Regulatory Functions



Accident Sequence Precursor Program

- Program recommended by WASH-1400 review group (1978)
- Provides risk-informed view of nuclear plant operating experience
  - CCDP (events)
  - ΔCDP (conditions)
- Supports reports to Congress*
- Supported by plant-specific Standardized Plant Analysis Risk models



Licensee Event Reports 1969-2017
(No significant precursors since 2002)

I. Gifford, C. Hunter, and A. Gilbertson, "U.S. Nuclear Regulatory Commission Accident Sequence Precursor Program: 2017 Annual Report," May 2018. (ADAMS ML18130A856)

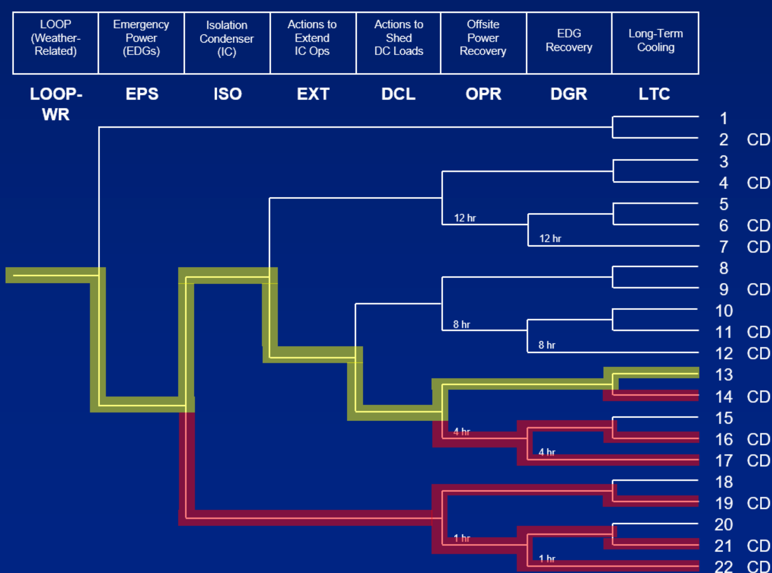
*Reports: Abnormal Occurrence, Congressional Budget Justification, Performance and Accountability

Key Metrics

Events

Conditional Core Damage Probability (CCDP):

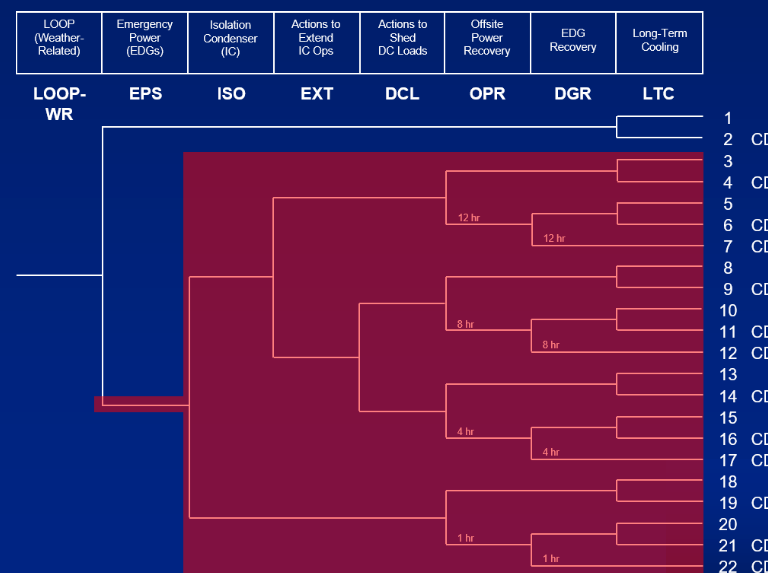
$$CCDP \equiv P\{core\ damage | event\}$$



Conditions

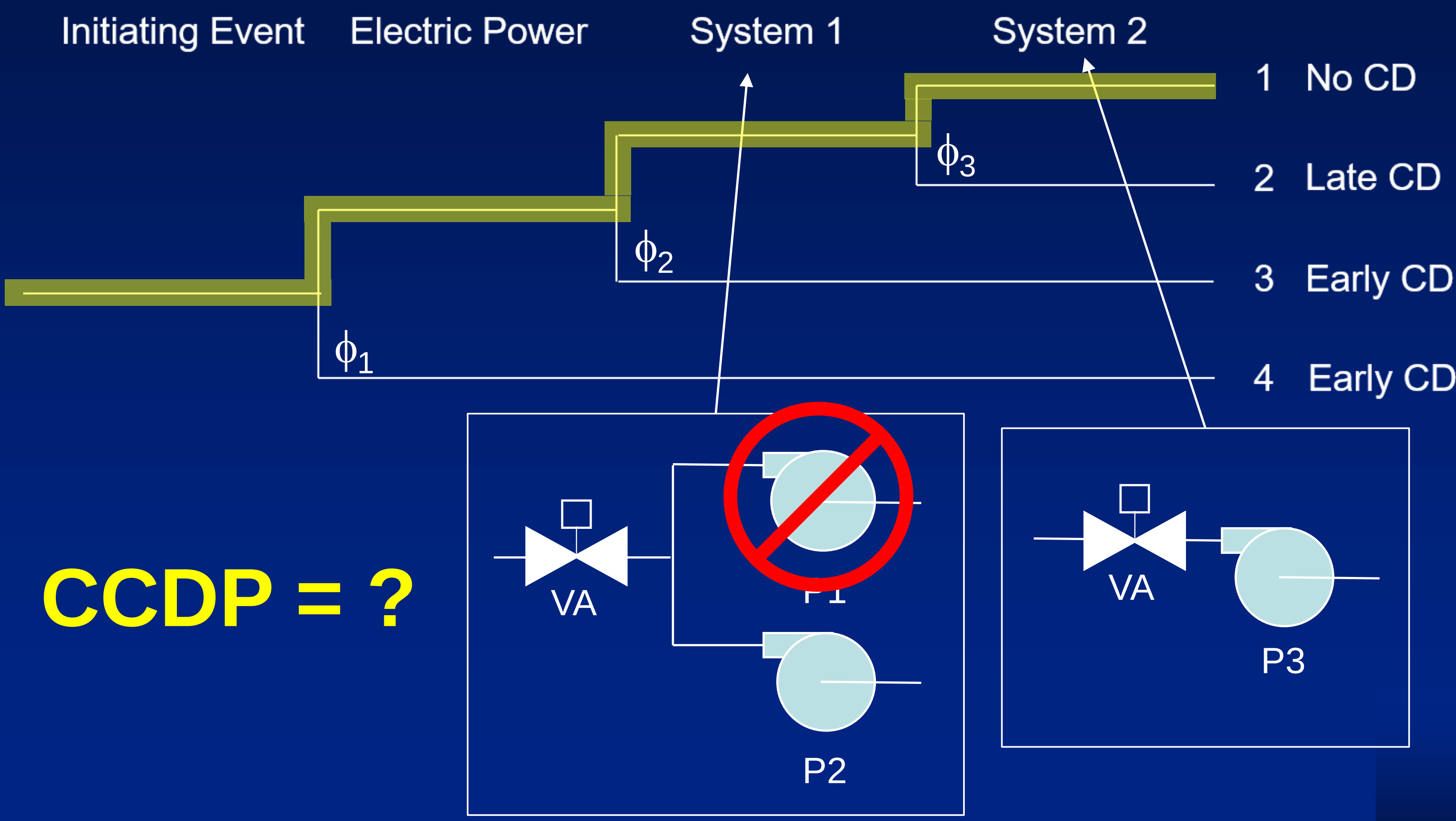
Change in core damage frequency (ΔCDP):*

$$\Delta CDP \equiv CDP_{condition} - CDP_{base}$$



*Calculated for the duration of the condition.

Knowledge Check



CCDP = ?

Top U.S. Precursors

Plant	Description	CCDP/ ΔCDP	Event Date	Plant Type
Browns Ferry 1	Cable tray fire caused extensive damage and loss of electrical power to safety systems	0.4	03/22/1975	BWR
Rancho Seco	Failure of non-nuclear instrumentation leads to reactor trip and steam generator dry out.	0.3	03/20/1978	PWR
Oyster Creek	Reactor trip results in loss of feedwater with subsequent failure of isolation condenser.	0.03	05/02/1979	BWR
Davis-Besse	Both emergency feedwater pumps found inoperable during testing	0.03	12/11/1977	PWR
Kewaunee	Clogged suction strainers for emergency feedwater pumps	0.03	11/05/1975	PWR
Turkey Point 3	Failure of three emergency feedwater pumps to start during test	0.03	05/08/1974	PWR
Point Beach 1	Clogged suction strainers for emergency feedwater pumps	0.03	04/07/1974	PWR
La Crosse	Loss of offsite power due to switchyard fire	0.02	03/24/1971	BWR
Davis-Besse	Loss of feedwater; scram; operator error fails emergency feedwater; power-operated relief valve fails open.	0.01	06/09/1985	PWR
Hatch 2	Reactor trip with subsequent failure of high-pressure coolant injection pump to start and reactor core isolation cooling unavailable.	0.01	06/03/1979	BWR
Farley 1	Reactor trip with all emergency feedwater pumps ineffective	0.01	03/25/1978	PWR
Cooper	Blown fuse leads to partial loss of feedwater and subsequent reactor trip; reactor core isolation cooling and high-pressure coolant injection pump fail to reach rated speed	0.01	08/03/1977	BWR
Millstone 2	Loss of offsite power with failure of emergency diesel generator load shed signals	0.01	07/20/1976	PWR
Haddam Neck	Loss of offsite power due to ice storm with failure of emergency diesel generator service water pump to start	0.01	01/19/1974	PWR

Most Recent Significant Precursors

Plant	Description	CCDP/ ΔCDP	Event Date	Plant Type
Davis-Besse	Reactor pressure vessel head leakage of control rod drive mechanism nozzles, potential unavailability of sump recirculation due to screen plugging, and potential unavailability of boron precipitation control.	0.006	02/27/2002	PWR
Catawba 2	Plant-centered loss of offsite power (transformer ground faults) with an emergency diesel generator unavailable due to maintenance	0.002	02/06/1996	PWR
Wolf Creek	Reactor coolant system blowdown (9,200 gallons) to the refueling water storage tank	0.003	09/17/1994	PWR
Shearon Harris	High-pressure injection unavailable for one refueling cycle because of inoperable alternate minimum flow valves	0.006	04/03/1991	PWR
Turkey Point 3	Turbine load loss with trip; control rod drive auto insert fails; manual reactor trip; power-operated relief valve sticks open	0.001	12/27/1986	PWR
Catawba 1	CVCS system leak (130 gpm) from the component cooling water/CVCS heat exchanger joint (i.e., small-break loss-of-coolant accident)	0.003	06/13/1986	PWR
Davis-Besse	Loss of feedwater; scram; operator error fails emergency feedwater; power-operated relief valve fails open	0.01	06/09/1985	PWR
Hatch 1	Heating, ventilation, and air conditioning (HVAC) water shorts panel; safety relief valve fails open; high-pressure coolant injection fails; reactor core isolation cooling unavailable	0.002	05/15/1985	BWR
La Salle 1	Operator error causes scram; reactor core isolation cooling unavailable; residual heat removal unavailable	0.002	09/21/1984	BWR
Salem 1	Trip with automatic reactor trip capability failed	0.005	02/25/1983	PWR

Other Precursor Activities

- Past Workshops
  - Annapolis, MD, 1992 (NUREG/CP-0124)
  - Madison, WI, 1995 (Bier, 1998)
  - Brussels, Belgium, 2001 (NEA, 2003)
  - Washington, DC, 2003 (Phimister, 2004)
- PSA-Based Event Analysis (PSAEA)
  - Annual international workshops led by Belgium
  - Exchange results and experiences

Significance Determination Process

- Part of Reactor Oversight Program
- Determines significance of findings
  - Characterize performance deficiency
  - Use review panel (if required)
  - Obtain licensee perspective
  - Finalize
- Differences from ASP
  - Supports fault finding and response
  - Focuses on a single performance deficiency (i.e., not necessarily the combined effect of anomalies)
  - Results are broad categories (colors)

ΔCDF < 1E-6 ΔLERF < 1E-7
1E-6 < ΔCDF < 1E-5 1E-7 < ΔLERF < 1E-6
1E-5 < ΔCDF < 1E-4 1E-6 < ΔLERF < 1E-5
ΔCDF > 1E-4 ΔLERF > 1E-5

CDF = Core damage frequency
LERF = Large early release frequency

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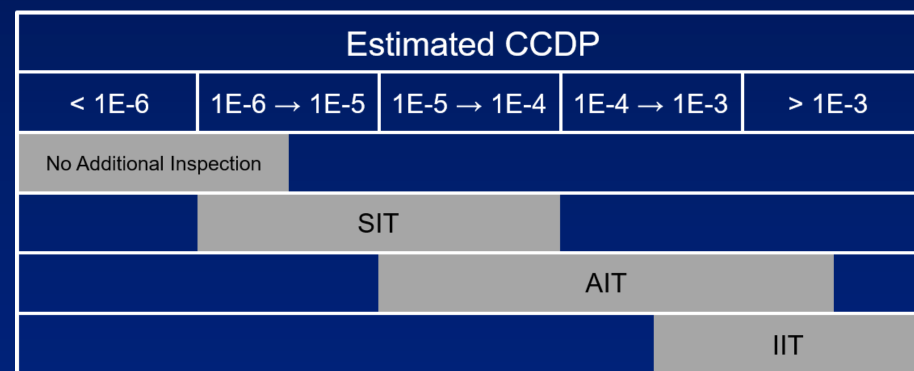
Incident Investigation

- Management Directive (MD) 8.3
- Determines NRC response to an incident
  - No additional inspection
  - Special Inspection Team (SIT)
  - Augmented Inspection Team (AIT)
  - Incident Inspection Team (IIT)
- Differences from ASP
  - Quick turnaround
  - Determines level of reactive inspection



Adapted from U.S. Nuclear Regulatory Commission, "NRC Incident Investigation Program," *Management Directive 8.3*, June 25, 2014. (ADAMS ML13175A294)

Example: Robinson Fire (3/28/2010)

- Electrical fault causes fire and subsequent reactor trip with losses of main feedwater (MFW) and reactor coolant pump (RCP) seal injection/cooling (*LER 261-10-002*)
- Incident Response (MD 8.3)
  - CCDP = 4×10^{-5} => augmented inspection
  - Initial evaluation recommended a special inspection; loss of RCP seal injection/cooling not known at the time
- Significance Determination Process (SDP)
  - Two White findings: licensee performance deficiencies involving inadequate training and procedures.
  - Five Green findings
- Accident Sequence Precursor (ASP)
  - CCDP = 4×10^{-4}
  - Non-recoverable loss of MFW modeled with RCP seal injection diverted away from RCP seals (unknown to operators) and component cooling water (CCW) isolated via return isolation valve (recovered by operators).

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IAEA/NEA International Nuclear and Radiological Event Scale (INES)

- Tool for communicating event safety significance to the public
- Logarithmic scale for severity
- Considers impacts on
  - People and the environment
  - Radiological barriers and control
  - Defence in depth
- Voluntary use by Member States
- Not a notification or reporting system for emergency response

Level	Description
7	Major Accident
6	Serious Accident
5	Accident with Wider Consequences
4	Accident with Local Consequences
3	Serious Incident
2	Incident
1	Anomaly

International Atomic Energy Agency, "INES: The International Nuclear And Radiological Event Scale, User's Manual, 2008 Edition," © IAEA, 2013. https://www-pub.iaea.org/books/IAEABooks/10508/INES-The-International-Nuclear-and-Radiological-Event-Scale-User-s-Manual-2008-Edition

Other (Potential) Uses of Retrospective PRA

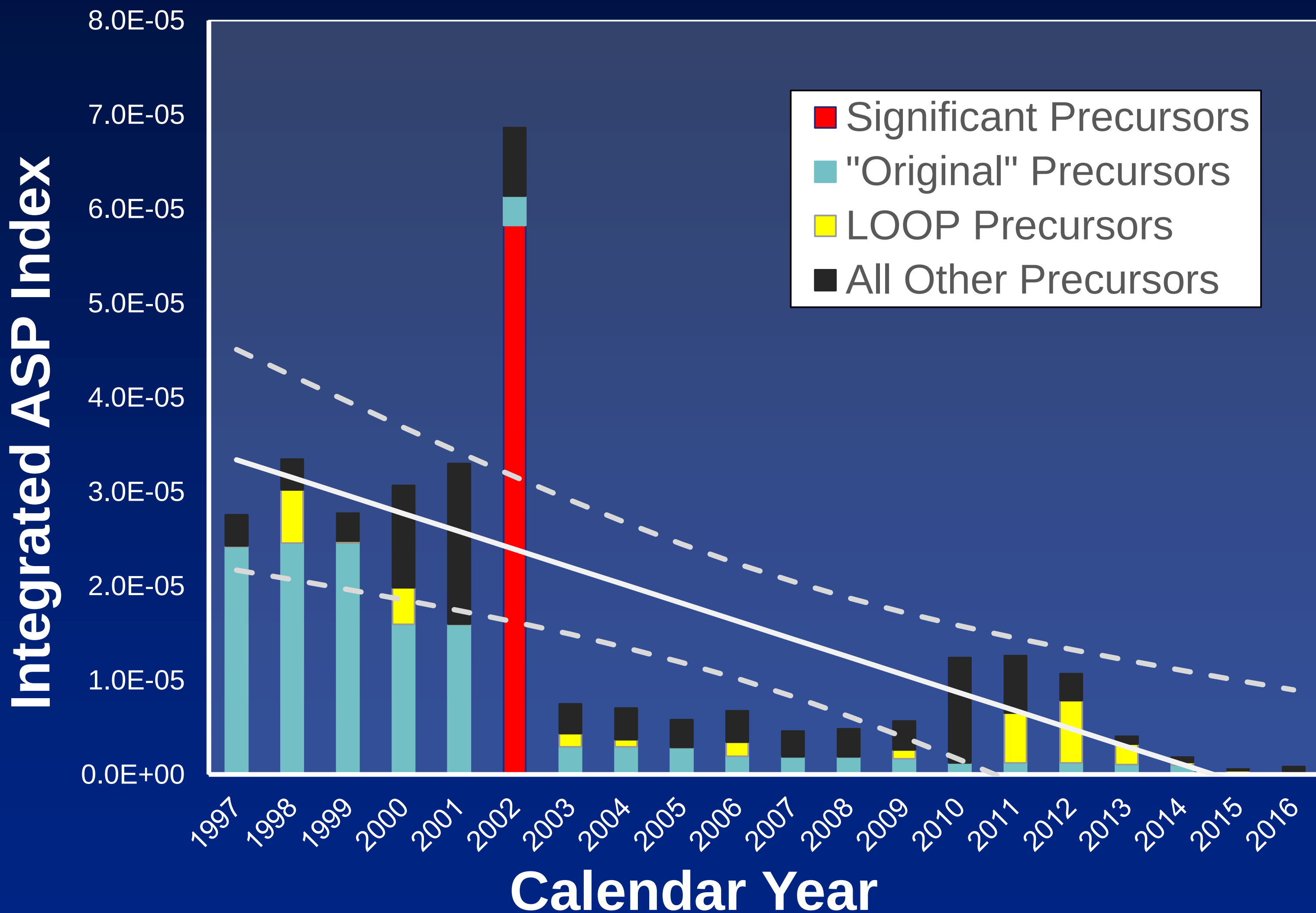
- Fleet health index
- Alternative method to estimate average CDF

Integrated ASP Index (IAI)

- Concept
  - Use numerical results of ASP analyses to indicate fleet performance
  - Increases with number of precursors
  - Increases with severity of precursors
- Definition

$$IAI = \frac{1}{T_{CY}}\left(\sum_{j=1}^{M_I} CCDP_j + \sum_{k=1}^{M_C} \Delta CDP_k\right)$$

T_{CY} = total calendar years
M_I = # initiating event precursors
M_C = # degraded condition precursors
$CCDP$ = conditional core damage probability
ΔCDP = change in core damage probability



Adapted from: I. Gifford, C. Hunter, and J. Nakoski, "U.S. Nuclear Regulatory Commission Accident Sequence Precursor Program: 2016 Annual Report," May 2017. (ML17153A366)

Relationship with Fleet CDF?

$$CDF_{fleet} = \sum_{i=1}^{N_{fleet}} CDF_i$$

A simple estimator, following Apostolakis and Mosleh (1979):

$$\widehat{CDF_{fleet}} = \frac{1}{T}\left[\sum_{i=1}^{N_{fleet}} \sum_{j=1}^{N_{event,i}} X_{ij}\right] = \frac{1}{T}\left[\sum_{k=1}^{N_{event}} X_k\right]$$

$$X_k = \begin{cases} 1 & Plant\ k\ has\ a\ CD\ accident \\ 0 & otherwise \end{cases}$$

$$E\left[\widehat{CDF_{fleet}}\right] = \frac{1}{T}\sum_{k=1}^{N_{event}} E[X_k] = \frac{1}{T}\sum_{k=1}^{N_{event}} P\{X_k = 1\}$$

- Addresses aleatory uncertainty
- Same mathematical foundation as basic PRA (Barlow and Proschan, 1965)

An Alternative to "Standard" PRA?

- Concept: use statistical estimates of CDF with CCDPs serving as data
  - Proposed in early days of precursor analysis (1980s)
  - Possibly reviving as part of statistical approaches using actual accidents (e.g., TMI-2, Chernobyl, Fukushima)
- Some earlier technical challenges have been addressed (e.g., more detailed models)
- Continuing technical challenges include:
  - Model limitations (shared with prospective PRA)
  - Specifying the analysis conditions (the "givens"): "failure memory" modeling, neglect of hazard variations
  - Incorporating full set of knowledge built into PRAs (e.g., risk from scenarios not involved in actual incident)

Comments

- Retrospective PRA is an extremely valuable source of information generally overlooked by the broader PRA community
  - PRA-oriented, structured view of actual events
  - Prioritization of issues needing attention
- Current programmatic challenges to retrospective PRA analyses include:
  - Resources spent on arguments over modeling and analysis results
  - Questions of added value given existing OpE programs (e.g., NRC OpE Clearinghouse)
  - Potential for increased polarization (which is the “right” approach, vs. what we can learn from different approaches)

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NRC OpE Clearinghouse

Inputs → **OpE Program** → **Products**

Inputs

Domestic OpE: Industry

- Daily Event Reports *
- Plant Status Reports *
- Licensee Event Reports *
- Part 21 Reports *
- INPO Reports

Domestic OpE: NRC

- Inspection Findings *
- Preliminary Notifications *
- Regional Project Calls
- Construction Experience
- Studies/Trends

International OpE

- Incident Reporting System (IRS)
- International Nuclear Event Scale (INES)
- Bilateral Exchanges

OpE Clearinghouse

- Screening
- Communication
- Evaluation
- Application

↓

Storage

Products

Influencing Agency programs

- Inspection *
- Licensing *

Informing Stakeholders

- Generic Communications *
- OpE Briefings
- COMMunications
- Periodic OpE Newsletter
- OpE Notes
- Notable OpE
- Tech Review Group Report

Taking Regulatory Actions

- Rulemaking *
- Information Request *

* Available on the public NRC Web Page