ACCIDENT DOSE CONSEQUENCES

Computer Based Training Module Available on NANTeL

ABSTRACT

This CBT is a self-paced, detailed, comprehensive, nuclear industry generic overview of the overall purpose, terminology and objectives of Dose Consequence Assessment for postulated accidents. The course discusses key parameters (inputs, assumptions, methodology design features and acceptance criteria) for assessment of various types of hypothetical accidents for both PWR and BWR sites (e.g., rod ejection or rod drop accidents, main steam line or reactor coolant failures, fuel handling, etc.). The module has undergone one round of revision to address ownership issues and feedback via NANTeL and other sources to make it more effective and seamless for the learners. The final exam was revised to add the open book resource documents link and reformat selected questions to improve clarity based on exam analysis and feedback.

INTENDED AUDIENCE

1. Experienced nuclear plant mechanical, nuclear, and safety analysis engineers who are developing expertise in Accident Dose Consequences
2. Site engineering Managers or Supervisors

DURATION

- 2.5 hours
- An additional 8-12 hours for reading materials provided within the CBT

TERMINAL LEARNING OBJECTIVES

1. Describe the overall purposes, terminology, objectives, and pertinent regulations and guidance of dose consequences assessment for postulated accidents.
2. Identify the general elements and principles of an accident dose consequences assessment e.g., radioactivity inventories and boundaries and their assumed failures, mitigating features and actions.
3. Describe the key parameters (inputs, assumptions, methodology, design features and acceptance criteria) of radiological consequences assessments for the following postulated accidents:
   - Main Steam Line Failures Outside Containment (PWR)
   - Feedwater System Pipe Breaks Inside and Outside Containment (PWR)
   - Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break (PWR)
   - Spectrum of Rod Ejection Accidents (PWR)
   - Spectrum of Rod Drop Accidents (BWR)
   - Failure of Small Lines Carrying Primary Coolant Outside Containment
   - Steam Generator Tube Failure (PWR)
   - Main Steam Line Failures Outside Containment (BWR)
   - Loss-of-Coolant Accidents
   - Fuel Handling Accident/Spent Fuel Cask Drop Accidents
   - Radioactive Waste Process Accidents
   - Different accidents and emergent accident issues, including consideration of already-analyzed bounding events, and application of 10CFR50.59 principles

KEY INDUSTRY DOCUMENTS

1. BTP 11-6, "Postulated Radioactive Releases Due to Liquid-Containing Tank Failures"
2. RG 1.3, "Assumptions for Evaluation of BWR LOCA"
3. RG 1.4, "Assumption for PWR LOCA Radiological Consequences"
4. RG 1.11, "Instrument Lines Penetrating Primary Containment"
5. RG 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors"
6. RG 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for BWR and PWR"
7. RG 1.24, "Assumptions for Evaluating PWR Gas Storage Tank Failure"
8. Technical Specification Task Force (TSTF) Traveler 51 (TSTF-51-A, Rev. 2)
9. Regulatory Guide 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants"
10. RG 1.70, "Initiating Events Analyzed In SAR Section 15"
11. RG 1.77, "Assumptions for Evaluating PWR Control Rod Ejection"
12. NRC RSICC Code Package CCC-681 RATAF
13. Generic Letter 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal"
14. NUREG-0016 (BWR)/NUREG-0017 (PWR), "Calculation of Releases of Radioactive Materials in Gaseous Liquid Effluents"
15. NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants"
16. NUREG-0737, "Clarification of TMI Action Plan Requirements"
17. NUREG/CR-6604 Supplement 01, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal and Dose Estimation"
18. NUREG/CR-6604 Supplement 02, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal and Dose Estimation"
19. RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants"
20. RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"
21. RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants"
22. RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors"
23. RG 1.98, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor"
24. RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants"
25. Oak Ridge National Laboratory - Radiation Safety Information Computational Center under the name RATAF, package C-00681
26. NUREG-0800, Standard Review Plan Branch Technical Position (BTP) 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure"
27. 10 CFR 50.34 "Contents of Applications Technical Information"
28. 10 CFR 50.67, "Accident Source Term"
29. 10 CFR 100.11, "Determination of Exclusion Area LPZ and Population Center Distance"
30. NRC NUREG/CR-6604, "Simplified Model for RADTRAD Estimation"
31. SRP 15.3.3 – 15.3.4, "RCP Rotor Seizure and Shaft Break"
32. SRP 15.4.8, "Appendix A PWR Control Rod Ejection Rad Consequences"
33. SRP 15.4.9, "Appendix A BWR Control Rod Drop Rad Consequences"
34. SRP 15.6.2, "Rad Consequences of Small Primary Coolant Lines Failure Outside Containment"
35. SRP 15.6.3, "Rad Consequences of PWR SG Tube Failure SGTR"
36. SRP 15.6.4, "Rad Consequences of BWR MSLB Outside Containment"
37. SRP 15.6.5, "Appendix A Rad Consequences DBA LOCA Including Containment Leakage Contribution"
38. SRP 15.6.5, "Appendix B Rad Consequences DBA LOCA Leakage From ESF Components Outside Containment"
39. SRP 15.6.5, "Appendix D Rad Consequences of BWR DBA LOCA MSIV Leakage Control System Leakage"

40. SRP 15.7.4, "Rad Consequences of Fuel Handling Accidents"
41. SRP 15.7.5, "Spent Fuel Cask Drop Accidents"
42. Safety Guide 5: Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors
43. TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites"