Stainless Steel Canister Degradation:

Discussion of Consequence Analysis Approach

Rick Reid
Program Manager

Shannon Chu
Senior Technical Leader

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Overview

- EPRI Used Fuel and HLW Program Project for 2016
- Current Project Objective: Identify and evaluate currently available analyses and recommend additional analyses that may still be needed to quantify the consequences associated with confinement boundary penetration of stainless steel canisters
  - Status: Effort is just beginning, several reports have been identified, a few have been partially reviewed, no evaluations are complete
- Ultimate Goal: Provide appropriate consequence information for consideration during the development and implementation of aging management plans for dry cask storage systems (DCSSs)

Considerations for this Project

- Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) are not applicable to DCSSs
- Need to select risk parameter of interest
  - Probability of confinement breach
  - Quantity of radionuclide release
  - Radiation dose (to workers and to public)
  - Latent cancer fatalities
- Need to define scope of interest
  - Storage duration
  - *Normal vs. Off-Normal vs. Accident Conditions*
  - *Include comparisons of risk from breach to risk associated with inspection, repair, and mitigation*

Risk Parameter: Radionuclide Release or Dose

- Canister confinement breach frequency not recommended
  - This parameter does not account for consequence to public or workers
- Quantity of radionuclide release
  - May be provided on a per canister basis
  - Providing a generic radionuclide release (per canister) result would assist plants in developing site specific dose information
- Radiation dose (to workers and to public)
  - Worker dose and site boundary dose results can be compared to established regulatory limits
  - Providing a site boundary TEDE for limiting geographical/meteorological conditions and maximum number of canisters anticipated at one ISFSI could provide easily relatable risk perspective for the industry
- Latent cancer fatalities not recommended

Initially Identified Reference Categories

- PRA Scoping Study
  - EPRI considered dry cask storage risk in 2002
- Risk-Informed References
  - Provide context and basis for considering consequence (to public or workers) of canister breach
- Qualitative Discussions of Failure Risk and Consequences for Dry Cask Storage
  - Provide discussion of damage mechanisms and progression
- Dry Cask System PRA Reports
  - Results = Latent Cancer Fatalities
- Dry Cask System Evaluations
  - Results = Dose

EPRI 1003011 Dry Cask Storage Probabilistic Risk Assessment Scoping Study (2002)

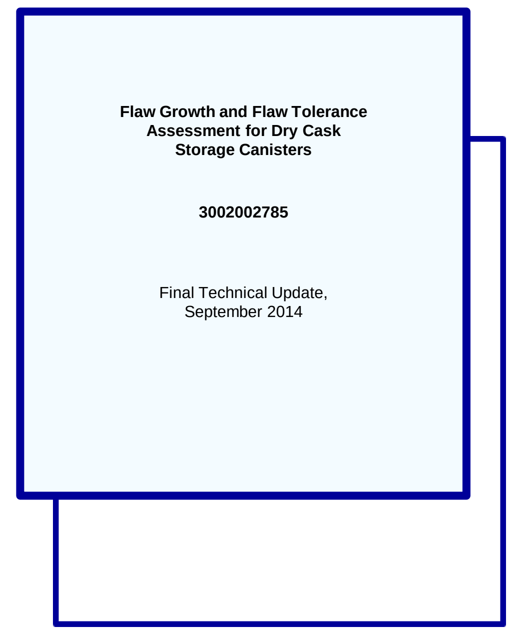
- Initiating Events
  - **Aging Related Degradation Mechanisms are NOT included**
- Accident Scenarios
- Human Error Interface
- Systems Analysis
- Data Development
- Structural Evaluation
- Thermal Hydraulic Analysis
- **Radionuclide Release and Consequence Evaluations**
  - Report suggests dose at a certain distance as PRA end state
- PRA Computer Modeling and Quantification

Risk-Informed References

- Basis Document for Large Early Release Frequency (LERF) Significance Determination Process (SDP), November 2002. (ML30240484)
- An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis, May 2011.(Reg-Guide 1.174)
- Assessment of Age-Related Degradation of Structures and Passive Components for U.S. Nuclear Power Plants, July 2000. (ML003751396)
- Suggestions?

Qualitative Discussions of Failure Risk and Consequences for Dry Cask Storage

- EPRI's Failure Modes and Effects Analysis
  - Qualitative discussion of failure effects on fuel and cladding, potential accident scenarios following confinement loss
- EPRI's Flaw Growth and Flaw Tolerance Assessment
  - Estimated time frames for loss of helium pressurization and air ingress
- Extended Storage and Transportation Evaluation of Drying Adequacy (CNWRA and NRC, June 2013)
  - Provides discussion relevant to potential consequences of the combination of inadequate drying and through-wall CISCC


Flaw Growth and Flaw Tolerance Assessment for Dry Cask Storage Canisters

3002002785

Final Technical Update, September 2014

Dry Cask System PRA Reports

- NUREG-1864 A Pilot Probabilistic Risk Assessment Of a Dry Cask Storage System At a Nuclear Power Plant (2007)
- EPRI1009691 Probabilistic Risk Assessment (PRA) of Bolted Storage Casks (2004)

Dry Cask System Evaluations

- Holtec MPC-68 and SNC TranStor Canisters
- Best-Estimate Offsite Dose from Dry Storage Cask Leakage
- Radiological Impact of Clad and Containment Failures in At-Reactor Spent Fuel Storage Facilities
- Additional Vendor Dose Evaluations (FSAR and Proprietary)
- Additional Sources of Source Term Data
  - Standard Review Plan for Transportation Packages for Spent Nuclear Fuel (NUREG-1617)
    - Provides recommendation for release fractions and specific activities for the contributors to the releasable source term for packages designed to transport irradiated fuel rods
  - Bolted Cask Monitoring Evaluations

Dry Cask System Evaluations – Holtec and SNC TranStor

- Accident Dose Calculations at 500m and 3219m Downwind for QA CATEGORY I Canister Leakage Under Hypothetical Accident Conditions for the Holtec MPC-68 and SNC TranStor Canisters, Revision 1, May 1999 (ML010330302)
  - Inputs and Assumptions
    - Assumes a 30-day accident release with continuous wind direction
    - BWR fuel, providing the largest radionuclide inventory - bounding dose estimates
    - Number of BWR assemblies per canister (68); internal volume (5.99 m3); leak rate under hypothetical accident conditions (1.58E-5 cm3/s)
    - Relies on assumed release fractions (for TranStor used release fractions of 1.0 for gases and 0.1 for all other radionuclides)
    - Relies on references for deposition and inhalation models and dose conversion factors
    - Boundaries of interest: 500 m (identified as the nearest distance from a canister to the site owner controlled area fence) and 3,219 m (identified as the distance to the nearest resident)
  - Results for Holtec
    - 13.4 mrem/y Total Effective Dose Equivalent (TEDE) at 500 m downwind; 0.653 mrem/y TEDE at 3,219 m downwind
  - Results for SNC TranStor
    - 2.7 mrem/y TEDE at 500 m downwind; 0.131 mrem/y TEDE at 3219 m downwind

Dry Cask System Evaluations- HI-STORM

- Best-Estimate Offsite Dose from Dry Storage Cask Leakage, June 2000. (ML031540409)
  – Utilizes RADTRAD (Radionuclide Transport and Removal and Dose Estimation) reactor accident analysis code
  – Report provides a table of values for fission product release, confinement leak rate, and other key parameters defined for normal, off-normal, and accident scenarios
  – Results include 30 day accident TEDE of 0.096 mrem and normal (365 day) dose of < 0.12 mrem
  – Analyses account for deposition of radionuclides within cask containment, offsite dose dramatically reduced compared to Safety Analyses Report (SAR) results which assumed radioactive material remained airborne and available for leakage for 30 day accident duration

2016 Project Approach and Deliverable

- Provide evaluation of key references
- Develop deliverable summary report documenting evaluation in CISCC context
- Make recommendations for continued effort in 2017 and beyond

Possible Recommended Consequence Analysis Steps

- Determine quantity of radioactive material that could be released from a single canister
- Define scenarios to be evaluated (static storage, events during storage, etc.)
- Define defect morphology(s) and size(s)
- Determine release factions for radioculides by physical form (gaseous, particulate fission products, particulate activation products)
- Define dispersion mechanisms and patterns
- Define dose pathways (direct, inhalation, ingestion, etc.)
- Identify receptors and location(s)
- Determine dose and/or other selected metrics

Additional Analyses to Consider

- **Comparisons to dose associated with inspection, mitigation, repair, and repackaging**
- Combination of through-wall cracking and accident scenario
- Radionuclide release – dose relationship
  - Conservatively define bounding ISFSI site with limiting geographical/meteorological conditions and maximum number of canisters anticipated
  - Work backwards from dose limit using bounding ISFSI site to get acceptable release quantity for comparison to per canister release results
    - Iterations on assumptions for canister heat load, crack dimensions, and failed fuel fraction
  - EPRI report (NP-2716) from 1982 offers such an approach for Skagit/Hanford nuclear site
- Other?

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