Insights from MELCOR Independent Confirmatory Analyses for New Reactor Design Certification

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PSA 2017 – International Topical Meeting on Probabilistic Safety Assessment and Analysis
September 24 - 28, 2017

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New Reactor Licensing

- Two-step process for licensing new reactors
  - Vendor submits to NRC a safety analysis for its design
    - Design certification (DC)
  - Utility submits to NRC a safety analysis to build and operate a certified design at a specific site
    - Combined license (COL)

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Safety Analysis Report

- Chapter 15 contains deterministic analyses of hypothetical or maximum credible accidents
- Chapter 19 contains probabilistic and deterministic analyses of severe accidents
  - Includes accidents more severe than in Chapter 15
  - Assumes failure of multiple safety systems
  - Demonstrates the containment is robust in the event of a core melt accident

Probabilistic and Deterministic Analyses of Severe Accidents

- Vendor performs severe accident simulations
  - Quantify branch points and source terms for PRA event tree
  - Demonstrate containment robustness for the more likely severe accident scenarios
  - Uses an integrated severe accident model (e.g., Modular Accident Analysis Program (MAAP))
  - Includes simulations for a range of severe accident scenarios

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Probabilistic and Deterministic Analyses of Severe Accidents (continued)

- NRC performs independent confirmatory analysis
  - Obtain insights on the vendor’s analysis results
  - Understand the proposed design to assess its conformance to NRC regulations
  - Determine if the vendor’s analyses reflect design descriptions in the Safety Analysis Report
  - Validate that the vendor’s approach and assumptions meet NRC regulations
  - Efficient way for NRC to evaluate vendor’s analyses

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NRC Approach for Independent Confirmatory Analysis

- Perform MELCOR simulations for a sample of the more likely severe accident scenarios
- Compare MELCOR predictions with vendor's predictions
  - Thermal hydraulics (time to core damage)
  - Accident progression (time to lower head failure)
  - Fission product release (cesium release to environment)

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Scenario Selection

- For large light-water reactors
  - At-power accidents – scenarios selected include Station Blackout with and without new reactor severe accident design features
  - Shutdown accidents – mid-loop accident with open RCS, assuming: failure of decay heat removal, no safety injection, closed and opened containment

MELCOR Model

- Single integrated model of the plant as described in the Safety Analysis Report
  - Plant design data (RCS and containment geometry)
  - Plant systems
  - Timing of operator actions
- Use best-estimate systems and phenomenological modeling and assumptions
  - Uses realistic values for input – not intentionally biased in a conservative or non-conservative fashion
- Physics models and spatial nodalization are detailed
- Physics models validated by experiments

NRC MELCOR Model Adapted for Shutdown Accidents

- Recently, NRC performed a MELCOR simulation for an accident occurring when the reactor is shut down for refueling (mid-loop accident)
- NRC modified MELCOR at-power deck to enable the simulation
  - Shifted decay heat curve by 4 days
  - Simulated nozzle dams by blocking flow paths connecting the steam generators to the RCS
  - Added a flow path to represent an open pressurizer manway
  - Isolated the accumulators from the RCS
  - Set initial RCS pressure, temperature, and water level to mid-loop conditions

Outcomes
Midloop Accidents

- The staff assessed possible modeling differences that could account for differences between the staff's and the applicant's predictions
- Surge line flooding/countercurrent flow limitation
  - Water held up in the pressurizer can reduce the time to core uncovery
- Pressurizer compartment
  - Whether the removable access wall is installed can affect aerosol deposition in containment
- Installed steam generator nozzle dams and open steam generator manways
  - Modeling of the configuration of the RCS can affect accident progression and source term

Outcomes
Midloop Accidents (continued)

- Nozzle dam failure following start of core damage
  - Nozzle dams can fail when subjected to increased steam pressure resulting from relocation of core debris into residual water in the reactor vessel lower plenum
- Timing regarding when equipment hatch is open
  - Having the equipment hatch open throughout the scenario can affect the source term
- A code bug was identified as a result of the staff's detailed review of the applicant's predictions of aerosol transport and deposition
- Subsequent to staff review, the applicant revised its calculations for mid-loop accidents

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Outcomes
At-Power Accidents

- The staff assessed possible modeling differences that could account for differences between the staff's and the applicant's predictions
- Accumulators
  - Not modeling accumulators could lead to faster accident progression and lower amounts of steam in containment
- Hot leg creep rupture
  - High temperature creep rupture of ex-vessel piping occurs as a result of hydrogen and superheated steam from an overheating core circulating through the RCS in high-pressure scenarios
  - Ex-vessel piping rupture can affect accident progression and source term

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Outcomes
At-Power Accidents (continued)

- Reactor coolant pump seal leakage and seal failure
  - Seal leakage and seal failure can affect accident progression and source term as a result of introducing additional holes in the RCS
- Assumed timing of operator actions
  - Operators may open valves a) to depressurize the RCS to effect feed-and-bleed and b) to flood the cavity
  - Opening valves puts the RCS and containment into a different configuration affecting severe accident simulations
- Subsequent to staff review, the applicant performed sensitivity analyses to demonstrate that the risk analysis was insensitive to these modeling differences

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Conclusion

- Once the differences between the applicant's analyses and the staff's confirmatory calculations are reconciled, the staff has reasonable assurance that:
- the applicant's Level 2 PRA is technically adequate
- the design has features for the prevention and mitigation of severe accidents
- the design has an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products