State-of-the-Art Reactor Consequence Analyses (SOARCA) Project

Accident Analysis

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Sandia is a multiprogram laboratory operated by Sandia Corporation, a Lockheed Martin Company, for the United States Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

SOARCA Objectives

- Perform a state-of-the-art, realistic evaluation of severe accident progression, radiological releases and offsite consequences for important accident sequences
  - Phenomenologically based, consistent, integral analyses of radiological source terms
- Provide a more realistic assessment of potential offsite consequences to replace previous consequence analyses
  - 1982 Siting Study

SOARCA Accident Progression Modeling Approach

- Full power operation
- Plant-specific sequences with a CDF$\geq 10^{-6}$ (CDF$\geq 10^{-7}$ for bypass events)
- External events included
- Consideration of all mitigative measures, including B.5.b
- Sensitivity analyses to assess the effectiveness of different safety measures
- State-of-the-art accident progression modeling based on 25 years of research to provide a best-estimate for accident progression, containment performance, time of release and fission product behavior

1982 Siting Study

- Evaluated potential consequences relevant to generic siting criteria
- Used hypothesized, generalized, source term categories
  - Based on limited knowledge and bounding rationale
  - Uncoupled from specific plant design or specific sequences
- Consequences dominated by
  - Source term magnitude and timing
  - Population density
  - Emergency response

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Radiological Source Terms

- 1982 Siting Study results were dominated by the SST1 source term
  - Loss of safety features
  - Large FP release from core
  - Severe early reactor and containment failure or bypass
- 1982 SST1 characterization (magnitude, timing and frequency) reflected then state of understanding and modeling
  - Early containment failure modes contemporaneously cited included alpha mode (steam explosion) failure, direct containment heating, hydrogen combustion
- Research and plant improvements over 25 years have dramatically altered our view of the early failure modes

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Severe Accident Improvements

- Research/plant improvements provided bases to conclude that some presumed early containment failure modes have been shown to be
  - negligible/highly improbable
    - In-vessel steam explosion and alpha mode failure
      - **SERG, Sizewell PRA, Experiments (FARO, KROTOS, TROI)**
    - direct containment heating due to high pressure melt ejection
      - **DCH Issue Resolution, experiments at SNL, ANL, Purdue**
  - or can be prevented by accident management
    - BWR Mark I liner melt through
    - Hydrogen control systems
- For large dry concrete containments, increased containment leakage is failure mode (vs catastrophic failure of the containment)

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Preliminary SOARCA Findings

- No sequences could be identified which resemble the characteristics of the dominant sequence from the 1982 study sequences
  - Sequences which were identified have lower frequencies than that assigned to SST1 in 1982 study
- All sequences identified could be prevented or significantly mitigated by existing or recently developed plant improvements
  - Important to realistically treat plant features/capabilities and include in probabilistic assessments
  - Confirmed by MELCOR analyses and served as the basis for evaluating plant/operator response including the TSC

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Preliminary SOARCA Findings

- Containment failure or bypass sequences are still identified in some plant specific PRA but even in those instances severity of conditions are significantly reduced
  - Reactor vessel lower head failure delayed even for the most severe (and most remote) of sequences (~ 7- 8 hrs) and much delayed for more likely severe sequences (~20+ hrs)
  - Bypass events are delayed beyond timing of SST1, bypass events also reflect scrubbed releases due to submergence of break (consistent, mechanistic modeling) or fission product deposition in the system piping
- These conditions while identified as important in current/past PRA, may now be considered to be more amenable to mitigation because of timing (revealed by integral analyses) and plant capabilities

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Preliminary SOARCA Findings

- Without those mitigation strategies, sensitivity studies indicate a radiological release fraction which is significantly smaller than earlier studies.
- Unmitigated sensitivities also result in a delayed release

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Peach Bottom Atomic Power Station
Emergency (B.5.b) Equipment

- Portable power source for SRVs and level indication
- Manual operation of RCIC without dc power
- Portable diesel driven pump (250 psi, 500 gpm) to makeup to RCS, drywell, CST, Hotwell, etc. and provide external spray
- Portable air supply to operate containment vent valves
- Off-site pumper truck can be used in place of portable diesel driven pump

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Peach Bottom Atomic Power Station
Long-term Station Blackout Without Mitigation

Without B.5.b mitigation

– Accident progression

Core uncovery in 9 hrs
Core damage in 10 hrs
RPV and containment failure in 20 hrs, start of radioactive release, (liner melt-through or containment head flange leakage)
Time between start of evacuation and radioactive release: ~17 hrs

– Offsite radioactive release is relatively small

1 – 4 % release of volatiles, except noble gases
Release is much less severe than 1982 Siting Study

– Accident progression timing and emergency evacuation significantly reduce potential consequences

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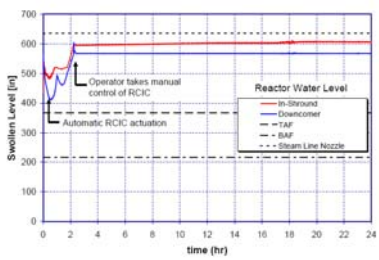
Peach Bottom Atomic Power Station
Long-term Station Blackout With Mitigation

Swollen Vessel Water Level Response



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Preliminary Findings Summary

- B.5.b measures have potential to prevent or significantly delay core damage
- Without B.5.b mitigative measures
  - Releases are significantly lower than 1982 study
  - Releases can be significantly delayed
- Accident progression timing (long time to core damage and containment failure) and mitigative measures significantly reduce the potential for core damage and/or containment failure

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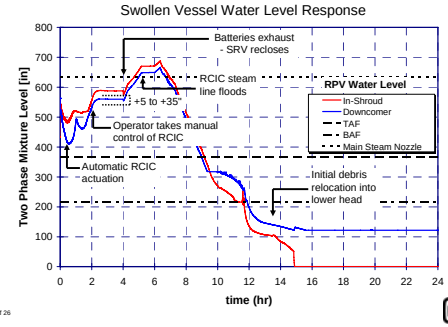
Peach Bottom Atomic Power Station
Long-term Station Blackout Without Mitigation

Swollen Vessel Water Level Response



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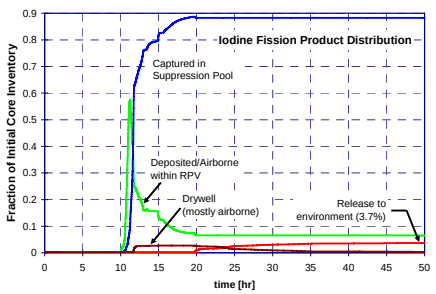
Long-term Station Blackout Without Mitigation



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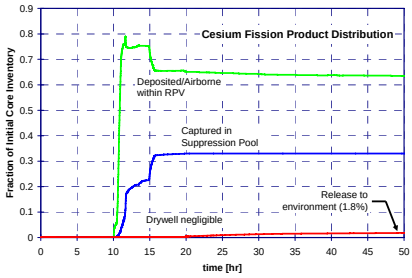
Long-term Station Blackout Without Mitigation



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Surry Nuclear Station
Emergency (B.5.b) Equipment/Procedures

- 2 diesel-driven high-pressure skid-mounted pumps for injecting into the RCS
- 1 diesel-driven low-pressure skid-mounted pump for injecting into steam generators or containment
- Portable power supply for restoring indication
- Portable air bottles to operate SG PORVs
- Manual operation of TDAFW
- Spray nozzle (located on site fire truck) for scrubbing fission product release

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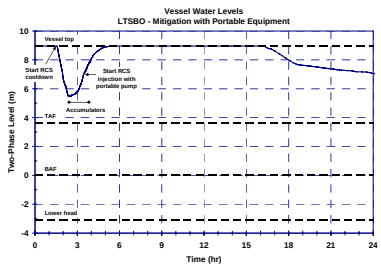
Surry Power Station
Long-term Station Blackout With Mitigation

Swollen Vessel Water Level Response



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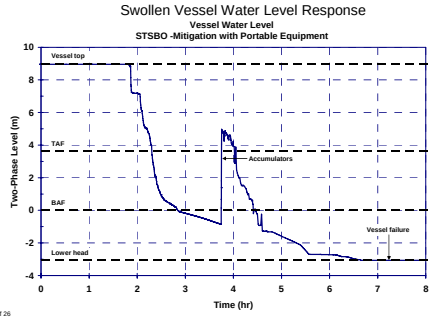
Surry Power Station
Short-term Station Blackout With Mitigation (Emerg. CS)

Swollen Vessel Water Level Response



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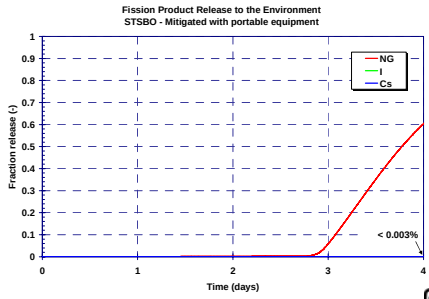
Surry Power Station
Short-term Station Blackout With Mitigation (Emerg. CS)



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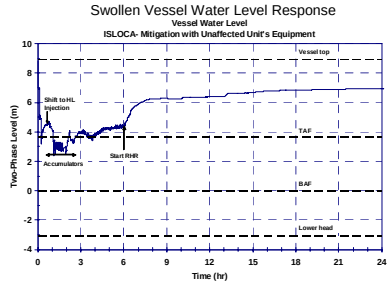
Surry Power Station
ISLOCA With Mitigation

Swollen Vessel Water Level Response



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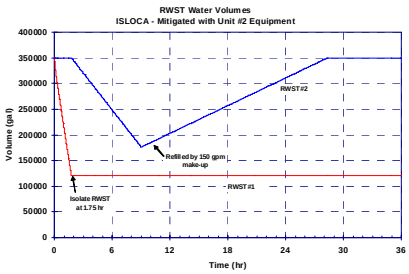
Surry Power Station
ISLOCA With Mitigation

ISLOCA mitigated using Second Unit RWST



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Mitigative Measures Sensitivity Analysis

Without mitigative measures

- Long term SBO
  - Core damage at 16 hrs
  - Containment failure at 45 hrs (increased containment leakage)
  - Public evacuation begins at 2.5 hrs
- Short term SBO
  - Core damage at 3 hrs
  - Containment failure at 25 hrs
  - Public evacuation begins at 2.5 hrs
- ISLOCA
  - Release scrubbed in flooded Aux building room
  - Non-mitigated analysis ongoing
- SGTR
  - Unsuccessful mitigation not considered credible
  - >40 hrs to core damage and offsite release

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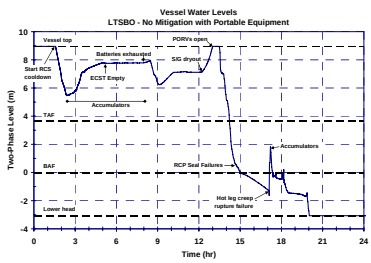
Surry Power Station
Long-term Station Blackout Without Mitigation

Swollen Vessel Water Level Response



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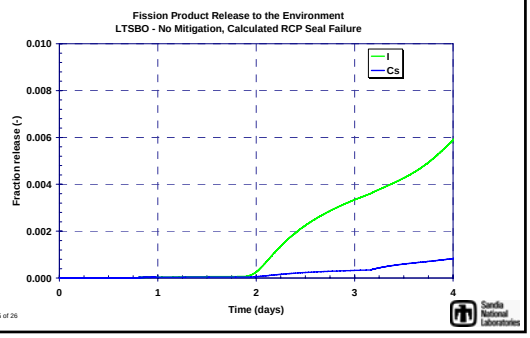
Surry Power Station
Long-term Station Blackout Without Mitigation



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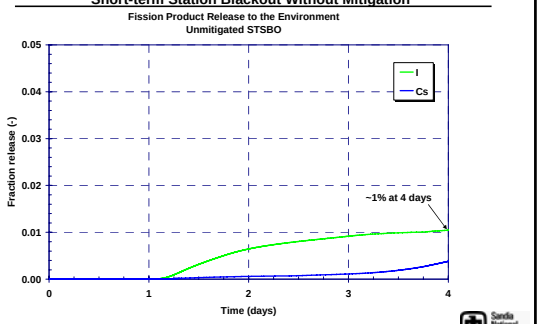

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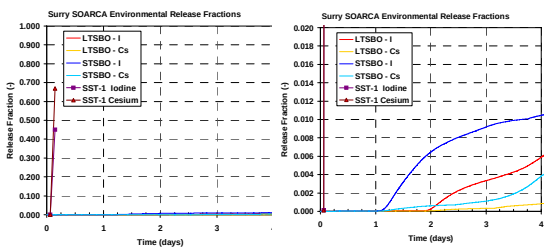
Surry Station Blackouts Compared to SST-1



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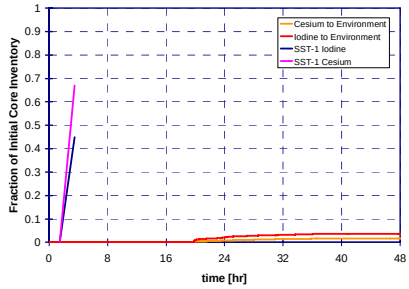
Peach Bottom Long Term Station Blackout Compared to SST-1



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Summary

- SOARCA study completing evaluation of Surry and Peach Bottom plants
- Releases for unmitigated accident vastly reduced and delayed in time compared to SST-1
- Mitigation shown to capable of terminating accidents
- Sequoyah analysis getting underway
- Uncertainty analysis and peer review planned

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