# Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor

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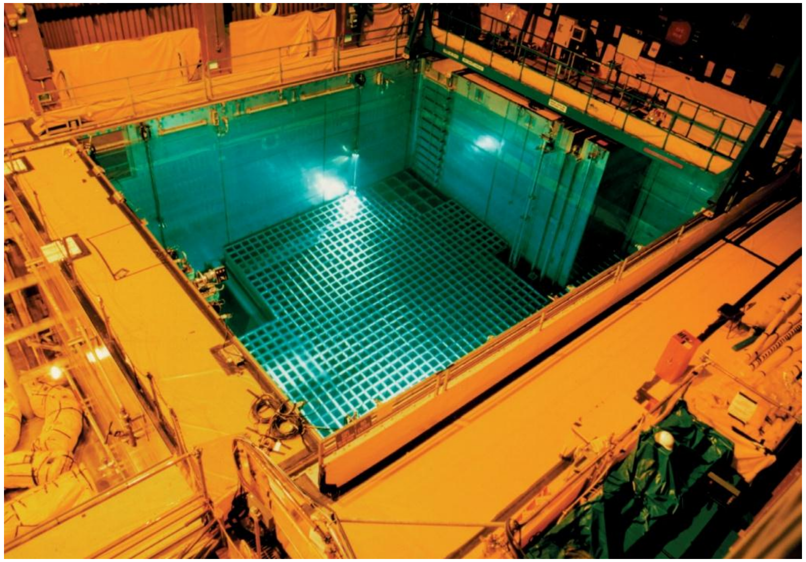
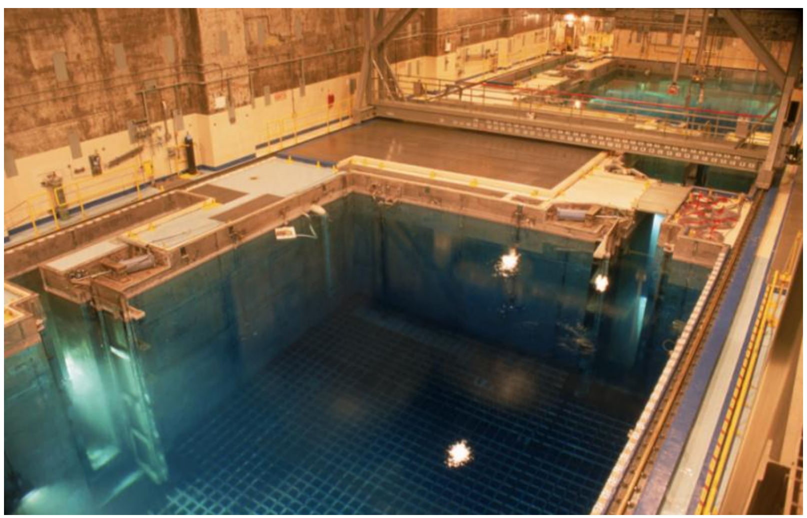
# U.S. Spent Fuel Pools





- Spent fuel rods stored in spent fuel pools (SFPs) under at least 20 feet of water
- Typically ~1/3 of fuel in reactor replaced with fresh fuel every 18 to 24 months
- Spent fuel stored in pools for a minimum of 5 years

# Purpose of Spent Fuel Pool Study

- After Fukushima accident, NRC received numerous requests to require licensees to expeditiously move spent fuel from pools to dry storage casks
- The spent fuel pool study's (SFPS) primary objective was to determine if accelerated transfer of spent fuel from the spent fuel pool to dry cask storage significantly reduces risks to public health and safety
- The study updates publicly available consequence estimates of a postulated beyond-design-basis earthquake affecting a SFP under high-density and low-density loading conditions.
- Results published in NUREG-2161

# Technical Approach

- Two conditions considered:
  - Representative of the current situation for the reference plant (i.e., high-density loading and a relatively full SFP)
  - Representative of expedited movement of older fuel to a dry cask storage facility (i.e., low-density loading)
- Elements of the study include
  - Seismic and structural assessments based on available information to define initial and boundary conditions
  - SCALE analysis of reactor building dose rates
  - MELCOR accident progression analysis (effectiveness of mitigation, fission product release, etc.)
  - Emergency planning assessment
  - MACCS2 offsite consequence analysis (land contamination and health effects)
  - Probabilistic considerations
  - Human reliability analysis of mitigation measures (Note: Performed after interim report was completed)

# Seismic/Structural Results

- Past SFP risk studies indicate that seismic hazard is the most prominent contributor to SFP fuel uncovery
- A severe seismic event (1 in 60,000 per year) was chosen to challenge SFP integrity
  - Assessment of location and size of failure, and its likelihood
- More severe than representative plant's SSE (and most US plants' SSEs)
- No liner tearing and no leaking (90% likelihood)
- Liner tearing spreading along the base of the wall (5% likelihood)
  - Moderate damage state (moderate leak)
- Liner tearing localized near the liner backup plates (5% likelihood)
  - Low damage state (small leak)

No leakage of water near the bottom of the walls was reported for 20 SFPs affected by two major recent earthquakes in Japan

- Consistent with low likelihood of leakage estimated for this study

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# Likelihood of Release

**Seismic event**

- Initiating event frequency of 1 in 60,000 years (**1.7E-5**/yr)

**Loss of normal SFP cooling (as modeled)**

- Assumed to be **84%** (based on station blackout probability given a 0.7g seismic event)

1.7E-5 x 0.84 = 1.4E-5

**SFP damage**

- Leak probability of **10%** given a 0.7g seismic event

1.7E-5 x 0.84 x 0.1 = 1.4E-6

**Radionuclide release**

- Fraction of operating cycle when fuel is susceptible to ignition in the event of pool leak
  - **8% without** credit for 10 CFR 50.54(hh)(2) mitigation

1.7E-5 x 0.84 x 0.1 x 0.08 = **1.1E-7/yr or lower** (by about a factor of twenty with credit for 10 CFR 50.54(hh)(2) mitigation)

# MELCOR Spent Fuel Pool Modeling

- Rack component
- Thermal radiation modeling
- Implementation of additional fuel rod components to represent edge rods and a sub-grid radiation model based on PWR tests
- Air oxidation modeling
- Hydraulic resistance model
  - NUREG/CR-7144, "Laminar Hydraulic Analysis of a Commercial Pressurized Water Reactor Fuel Assembly," (Jan. 2013)
- Integral Zirconium Fire Experiments
  - Conducted at Sandia National Laboratories for BWR and PWR fuel assemblies under complete loss of coolant accidents
  - NUREG/CR-7143, "Characterization of Thermal-Hydraulic and Ignition Phenomena in Prototypic, Full-Length Boiling Water Reactor Spent Fuel Pool Assemblies After a Postulated Complete Loss-of-Coolant Accident," (Mar. 2013)
  - OECD/NEA Sandia Fuel Project (PWR fuel). Tests completed in 2012. Ongoing MELCOR model development and code assessment

**Convective Heat Transfer Surfaces:**

Ring 1 — Clad, Canister / Water Rods, Rack

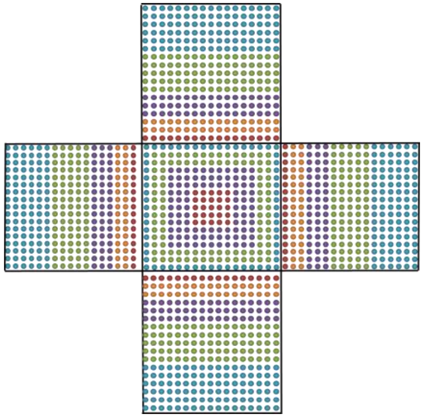
Ring 2 — Clad, Canister / Water Rods, Racks

**Radiative Heat Transfer Flow Path:**

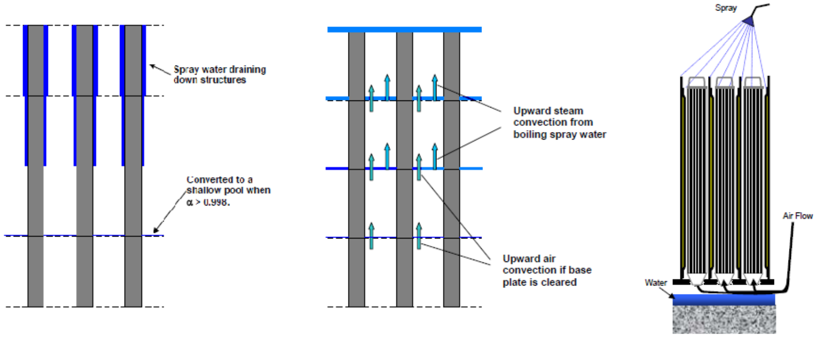
Ring 1: Fuel → Clad → Canister / Water Rods → Rack → Ring 2: Canister / Water Rods → Clad → Fuel

Canister / Water Rods ↓ Racks

Multiple fuel rod components in the center assembly (Ring 1) and four peripheral assemblies (Ring 2)

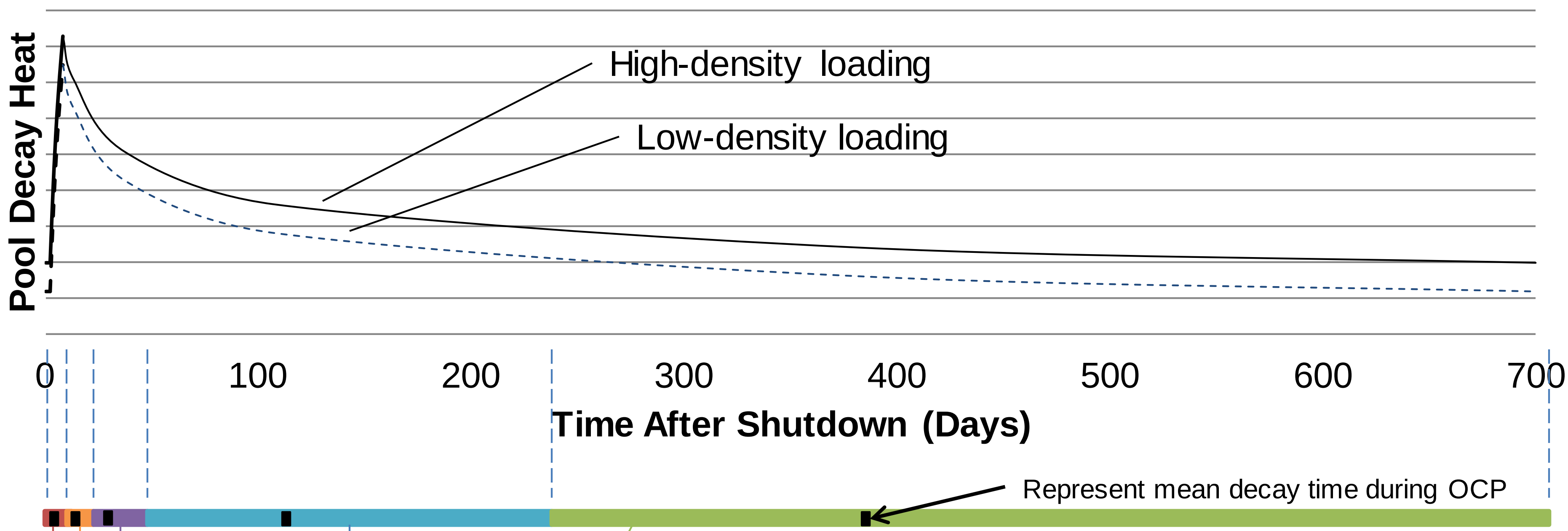


## Integral Spray Model



# MELCOR Analysis

Pool Decay Heat and Operating Cycle Phases (OCPs)




OCP #5: Remainder of the operating cycle (66%)

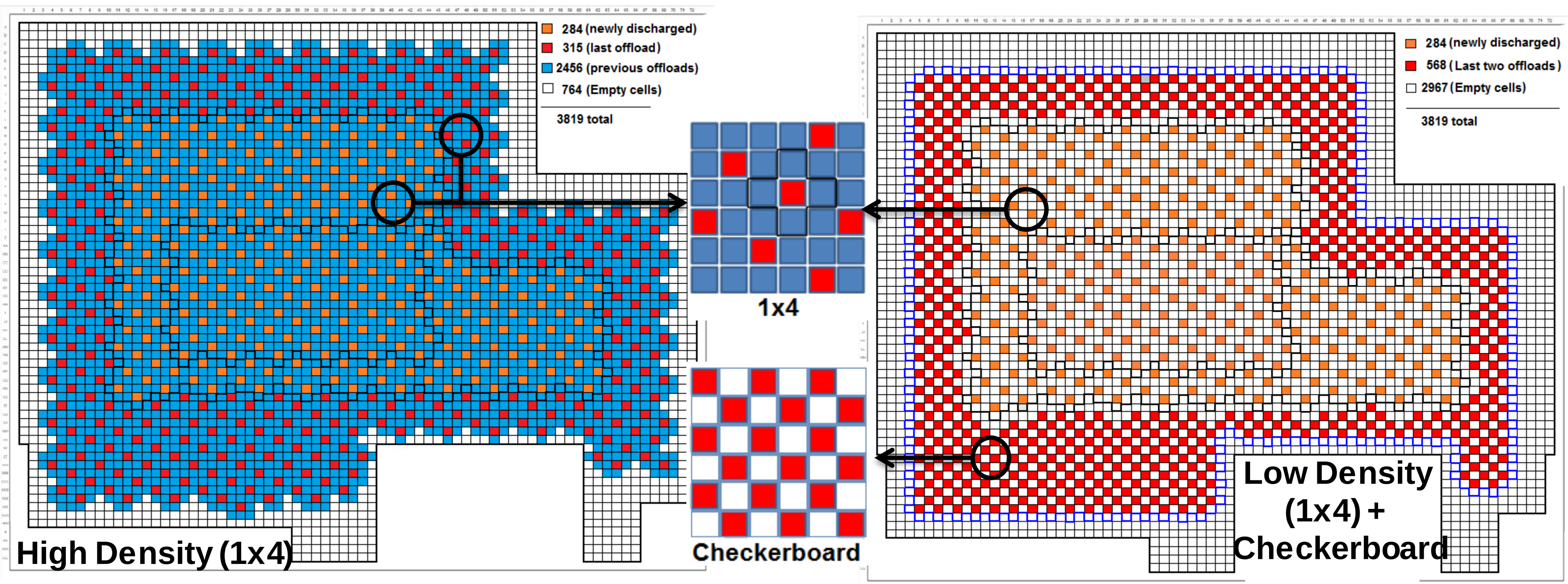
OCP #4: From 60 to 240 days (26% of OC)

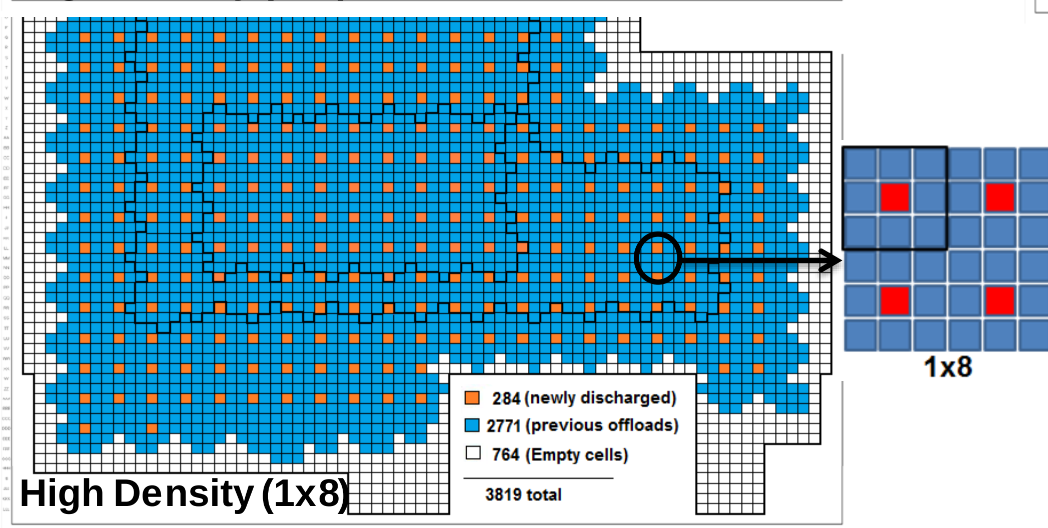
OCP #3: Post-outage (25 to 60 days)(5% of OC)

OCP #2: Latter half of outage (8 to 25 days)(2% of OC)

OCP #1: Defueling (2 to 8 days)(1% of OC)

# SFP Loading (OCP2/3/4/5)



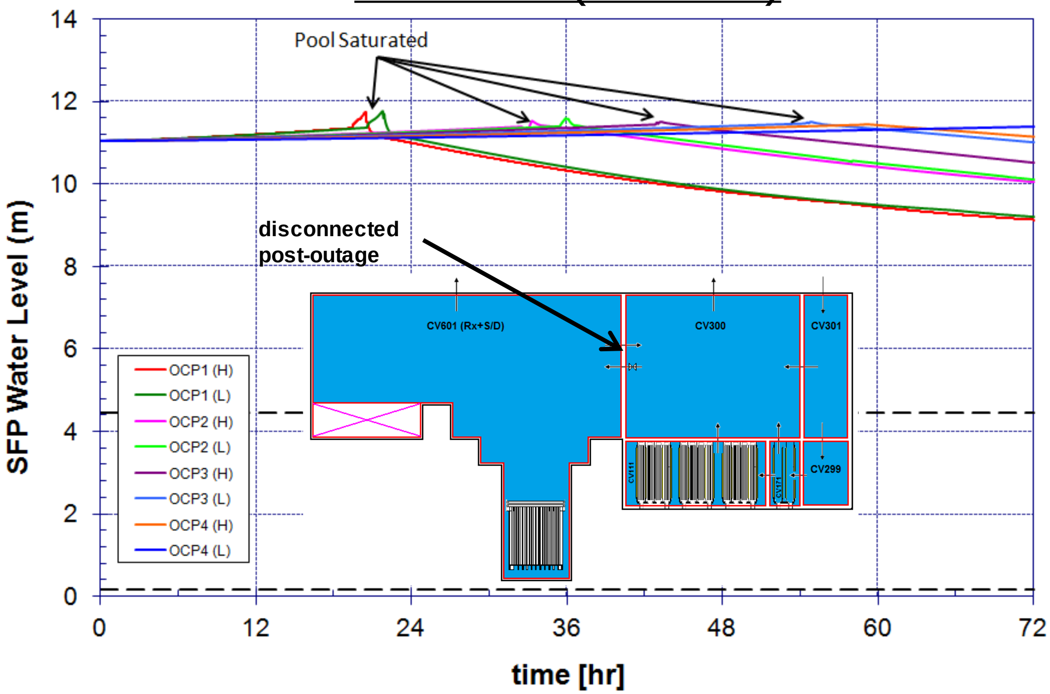




- Newly discharged 1x4 (or 1x8)
- Previous 2 offloads (fuel < 5 years) checkerboard for low density (due to limitation of available cells)
- Blue cells represent older fuel
- White cells represent empty locations for full core offload capability (and after removal of older fuel in low density case)

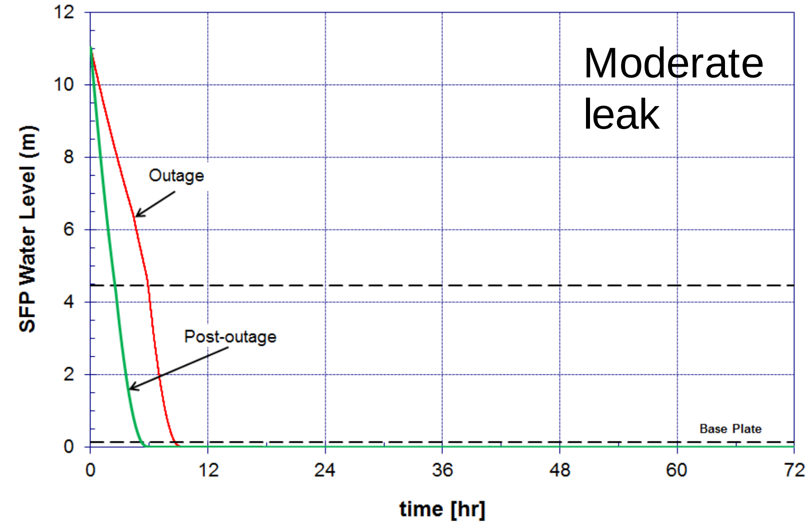
# MELCOR RESULTS (1)

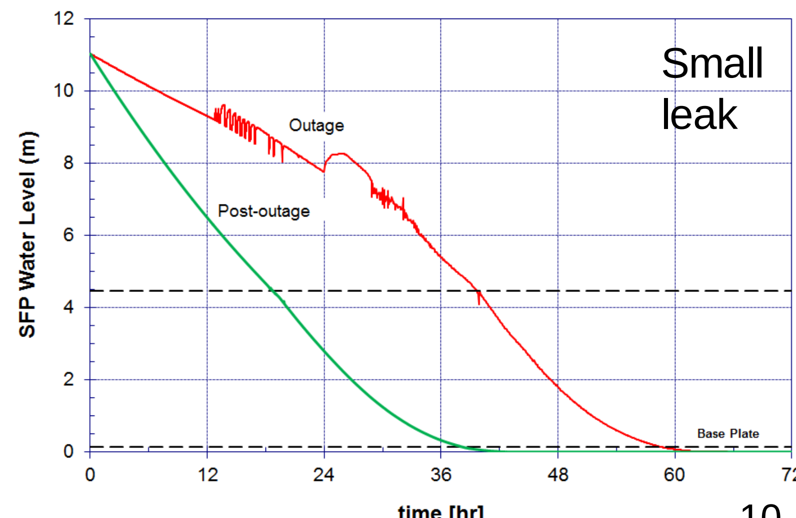
## Unsuccessful deployment of 50.54(hh)(2) mitigation

### No Leak (Boil-Off)




### Drain-down



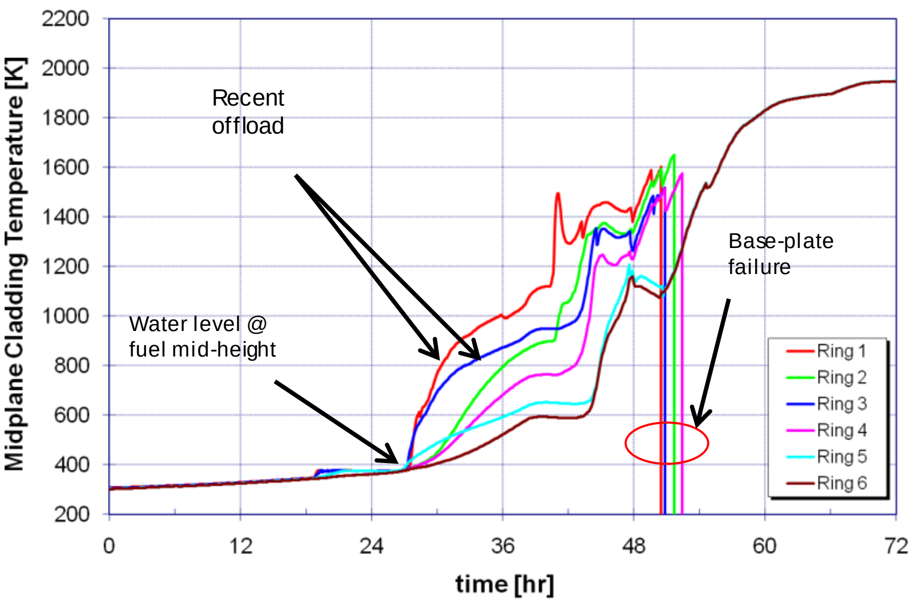



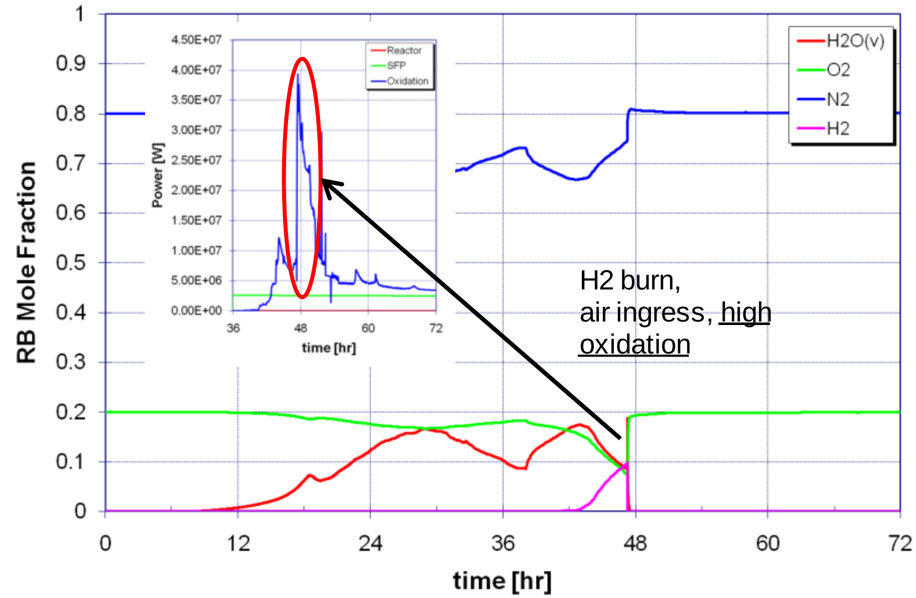

- No fuel uncovery in 3 days (~7 days for OCP1)
- Moderate leak drain-down time (~ 9 hrs OCP1/2; ~ 6 hrs OCP3/4/5)
- Small leak drain-down time (~ 62 hrs OCP1/2; ~ 42 hrs OCP3/4/5)

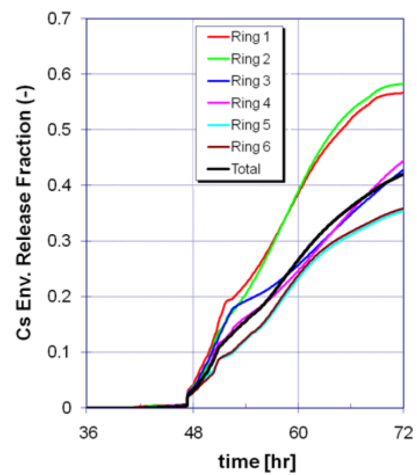
# MELCOR RESULTS (2)

Unsuccessful deployment of 50.54(hh)(2) mitigation

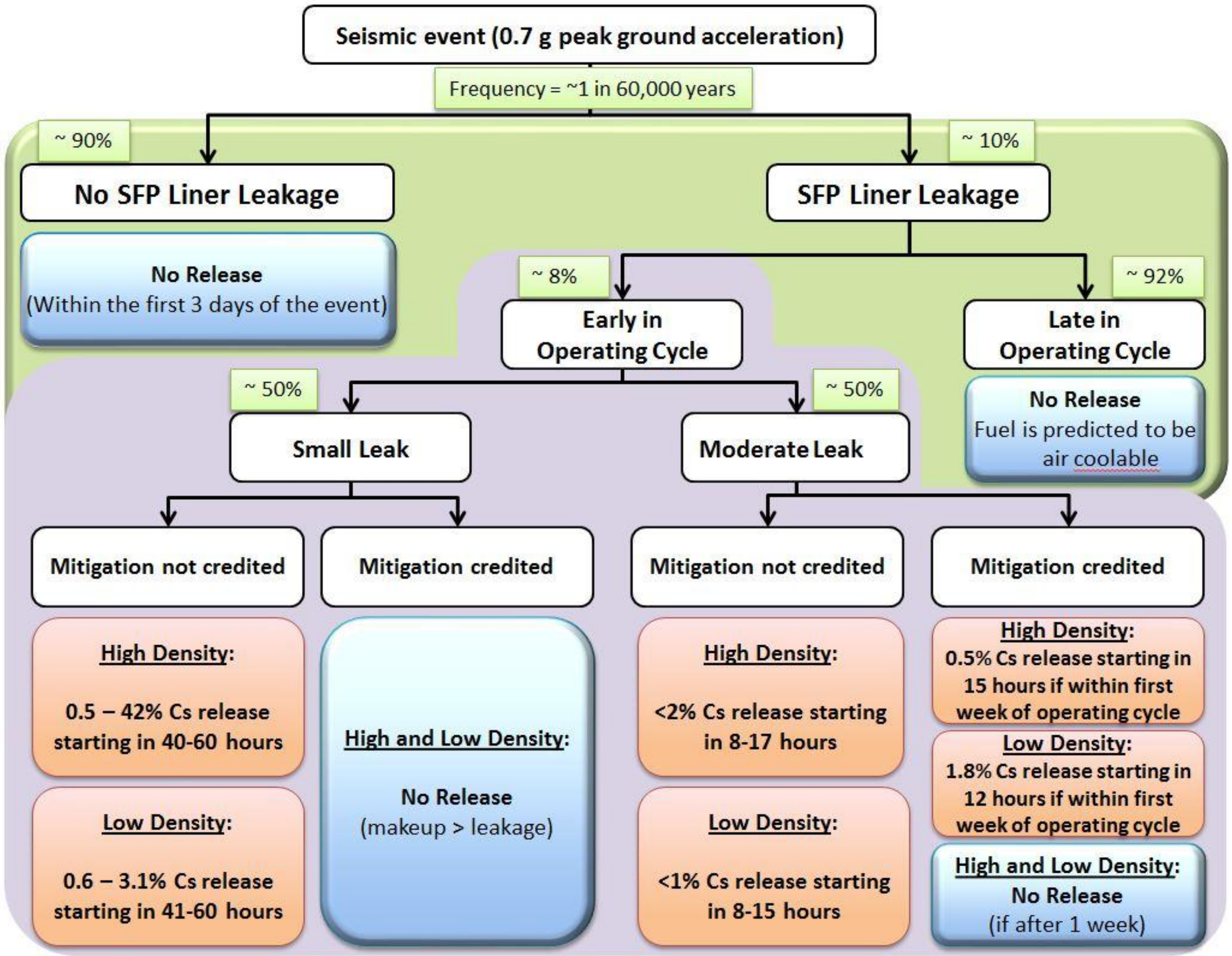
Post-Outage High Density Small Leak (OCP3)



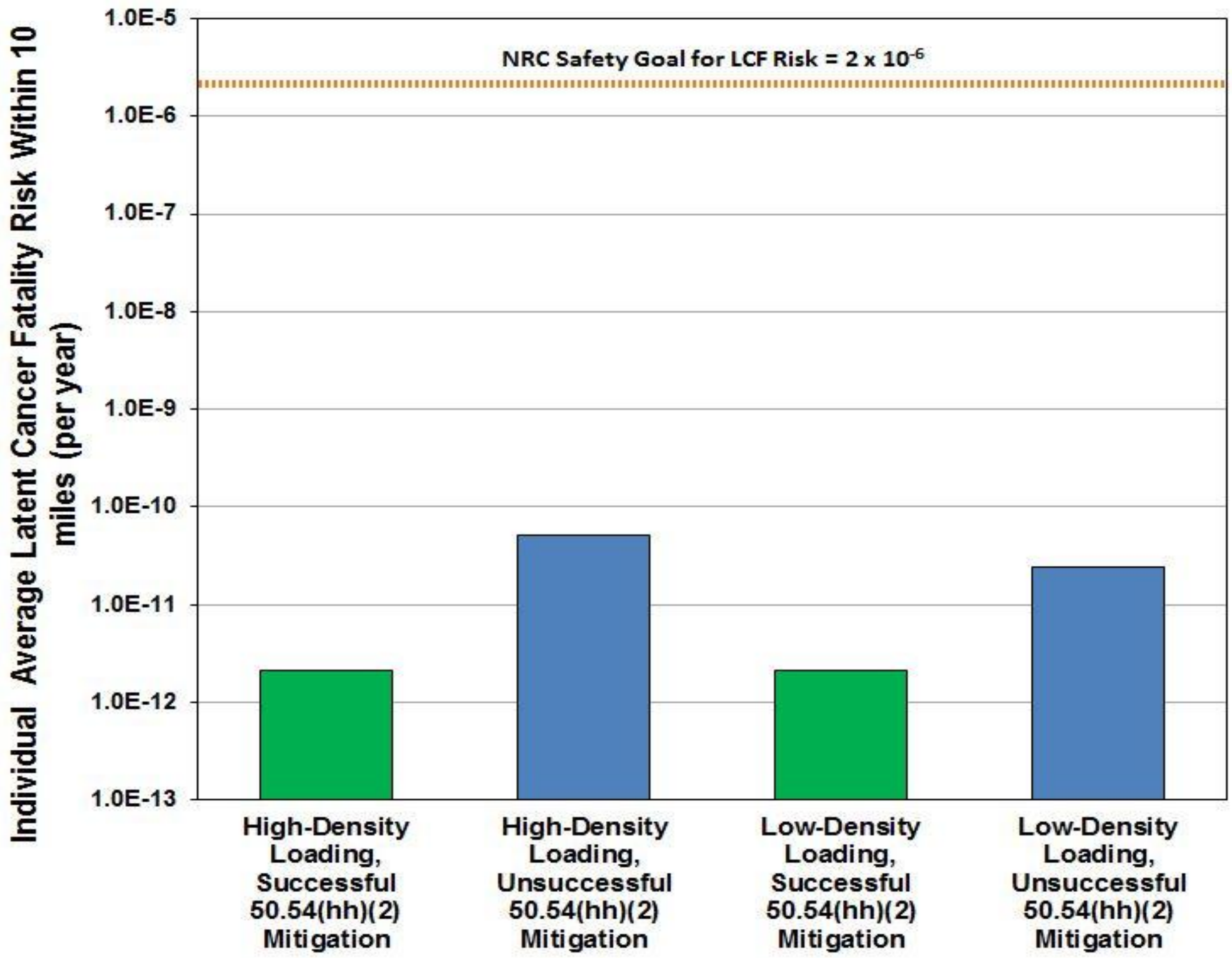



# MELCOR Analysis



# MACCS Consequence Results



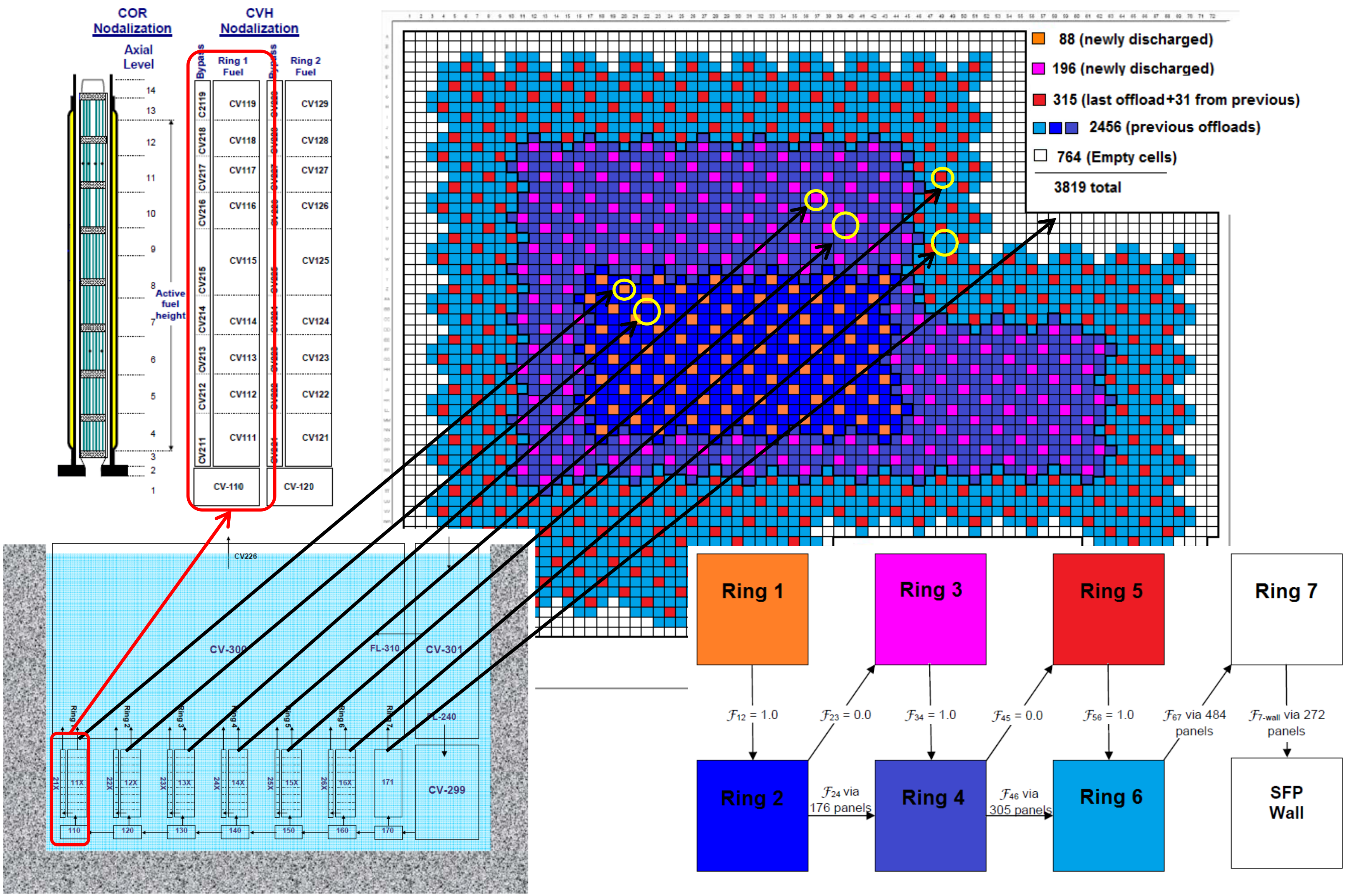

**Comparison of Population-Weighted Average Individual Latent Cancer Fatality Risk Results for this Study to the NRC Safety Goal (plotted on logarithmic scale)**

# Summary

- Past SFP risk studies have shown that storage of spent fuel in a high density configuration is safe and risk is appropriately low
- Results consistent with past studies' conclusions that SFPs are robust and not expected to leak as a result of a seismic event
- Likelihood of release is 1 in 10,000,000 years or lower for the reference plant
- Spent fuel is only susceptible to a release within a few months after defueling; after that it is coolable by air
- Successful mitigation generally prevented potential releases
- In the very unlikely event a release occurs, no early fatalities were predicted for any of the scenarios and individual latent cancer fatality risk is low
- A more favorable loading pattern or improvements to mitigation strategies significantly reduced potential releases

# BACKUP SLIDES

# High-Density Post-Outage SFP MELCOR Model

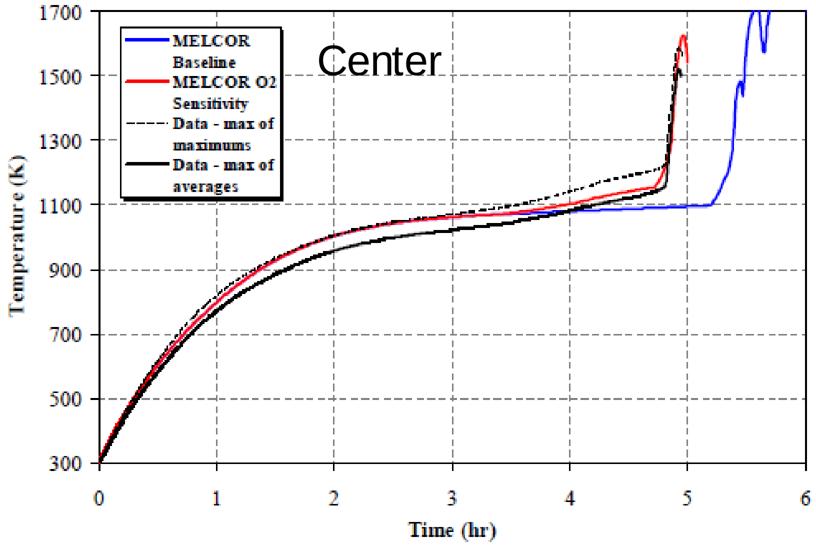


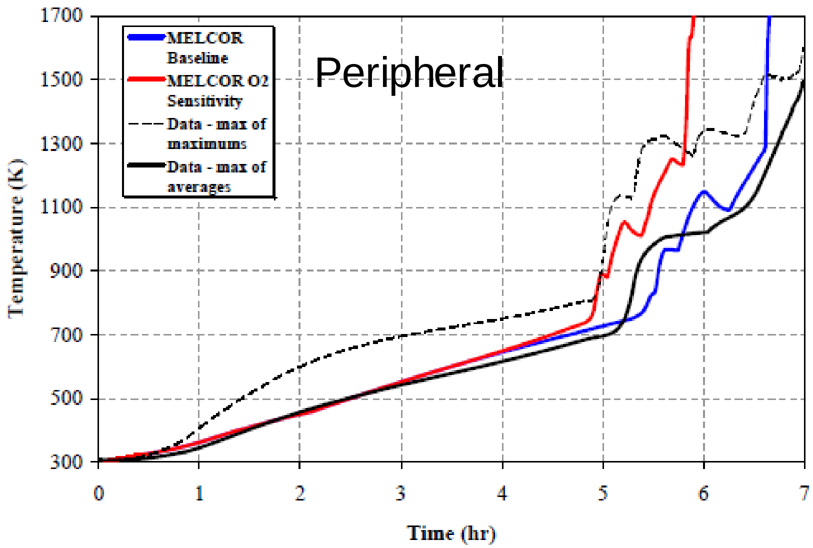
# MELCOR RESULTS (CONSIDERATION OF UNCERTAINTY)

| Sensitivity | Results |
|---|---|
| Hydrogen combustion ignition criteria | Reduced from 10% to 7% given inherent uncertainties in this parameter. The cesium release of 50% is much higher than the base case of 1.6% for the moderate leak in OCP2 |
| 1x8 fuel loading pattern | Very favorable results, e.g., cesium releases for OCP2 reduced significantly and no releases predicted for OCP3 |
| Uniform fuel loading pattern during outage | In general, higher releases are predicted. Effectiveness of mitigation is impacted (some release for OCP2) |
| Multiunit or concurrent accident effects | Loss of the reactor building due to a hydrogen deflagration from a concurrent reactor event can have a positive or negative impact depending on the timing of the explosion |
| Molten core concrete interaction | Certain radionuclide species (Ce/La) can become more volatile in the presences of sparging ablation gases with releases higher by orders of magnitude. Cs release is increased by 40% |
| Radiative heat transfer | Ring to ring radiative heat transfer modeling does not significantly impact results. |
| Time truncation | Increasing the simulation time from 3 days to 4 days has a modest effect on the predicted releases (25% increase in Cs release for unmitigated OCP3 small leak) |
| Reactor building leakage | Leakage area has no significant impact on the accident progression and cannot prevent hydrogen deflagration for large release scenarios |

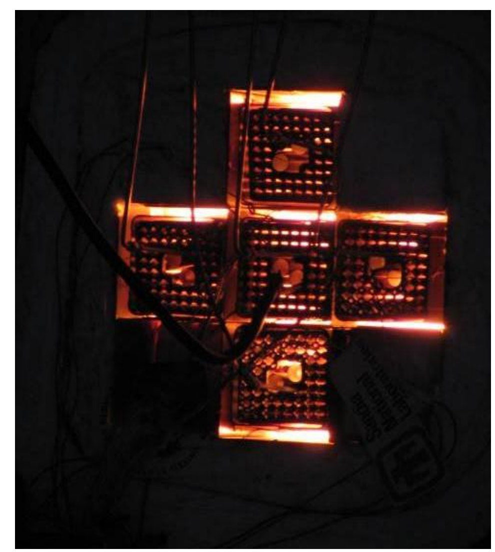
# BWR Assembly (1X4) Ignition Test

## (NUREG/CR-7143)




- "Hot" center assembly in 1 $\times$ 4 arrangement
- Equivalent of 15 day-old fuel surrounded by background assemblies (cold neighbor BC)
- Strong radial heat transfer to un-powered peripheral assemblies
- Investigated sensitivity to reaction kinetics
- MELCOR model results showed good comparison
- More confidence in MELCOR BWR whole pool calculations and better analysis of different storage arrangements



# Expedited Transfer of Spent Fuel to Dry Cask Storage

- The insights from this analysis informed a broader regulatory analysis of the SFPs at all U.S. operating nuclear reactors as part of Japan Lessons-learned Tier 3 plan.
- The regulatory analysis concluded that expedited transfer does not result in a substantial increase in safety, and does not pass a cost/benefit analysis.

- Staff recommended that the Commission not pursue additional studies on this issue
- Commission agreed with staff recommendation and directed staff to provide additional information.