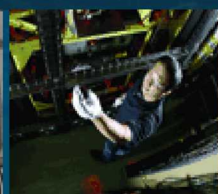
SAND2018-9358PE

MELCOR Application to PRA – Various Best Practices



SAND2018-????

PRESENTED BY

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U.S.NRC

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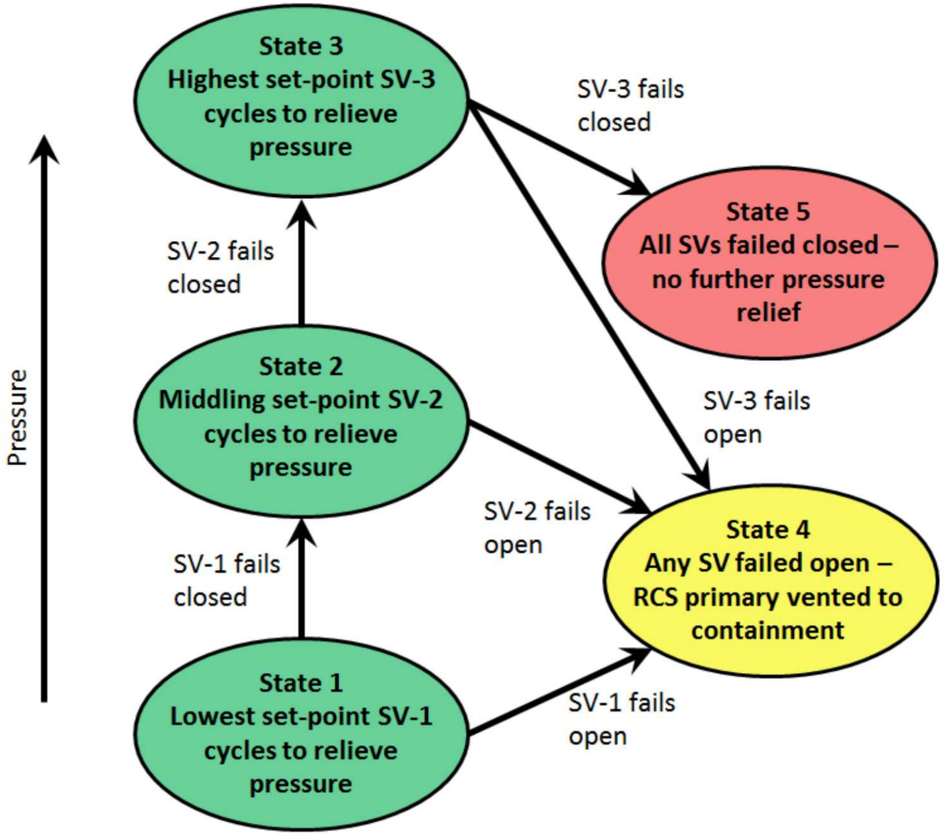
Presentation Objectives

- Present various current SNL best modeling practices important to applying MELCOR in a PRA including practices associated with:
  - PWR safety valve failure
  - PWR hot leg natural circulation
  - Consequential steam generator tube ruptures (C-SGTRs)
  - Hydrogen ignition
  - ISLOCA
  - Dynamic PRAs

Pressurizer Safety Valve (SV) Modeling

- Important what the 3 parallel SVs do as a system more so than what any valve does individually
- Failure to open (FTO) and failure to close (FTC) possibilities
- If a FTO occurs or a FTC but in a mostly closed position, pressure relief transitions from the affected valve to the next set-point valve (State 1 to State 2 for example)
- If a FTC occurs, the RCS vents unregulated to containment (State 2 to State 4 for example)
- Should all 3 valves FTO, State 5 (no relief) develops



Possible transitions in the 3-SV pressurizer pressure relief system considering both FTO and FTC valve conditions

SV Failure Modeling

- FTO, FTC, failure upon passing liquid and overheating failure have all been all considered but only stochastic FTC emerges as a viable failure mode
- FTO has been discounted due to exceedingly low probability
- Identified by nuclear valve testing specialists has been that passing water isn't necessarily threating to an SV but that passing fluid (liquid or gas) that is hot or cold is (hot or cold being relative to the design temperature of the valve)
- Hot or cold fluid passing through the SVs has not been observed in PWR STSBO calculations
- Accordingly, failure upon passing water and overheating failure have been discounted

SV Failure Modeling (2)

- Probabilities obtained from Table 20, "Failure probabilities for PWR code safety valves (behavior after scrams)", in NUREG/CR-7037 inform the characterizations of stochastic SV failure
- Table 20 reports on SV operation subsequent to actual scram events for both main steam system (MSS) and reactor coolant system (RCS) valves but little information on RCS valves is presented
- Information on initial, subsequent demand and recovered function reported

Demand	# Failures	# Demands
Initial	16	621
Subsequent	0	223

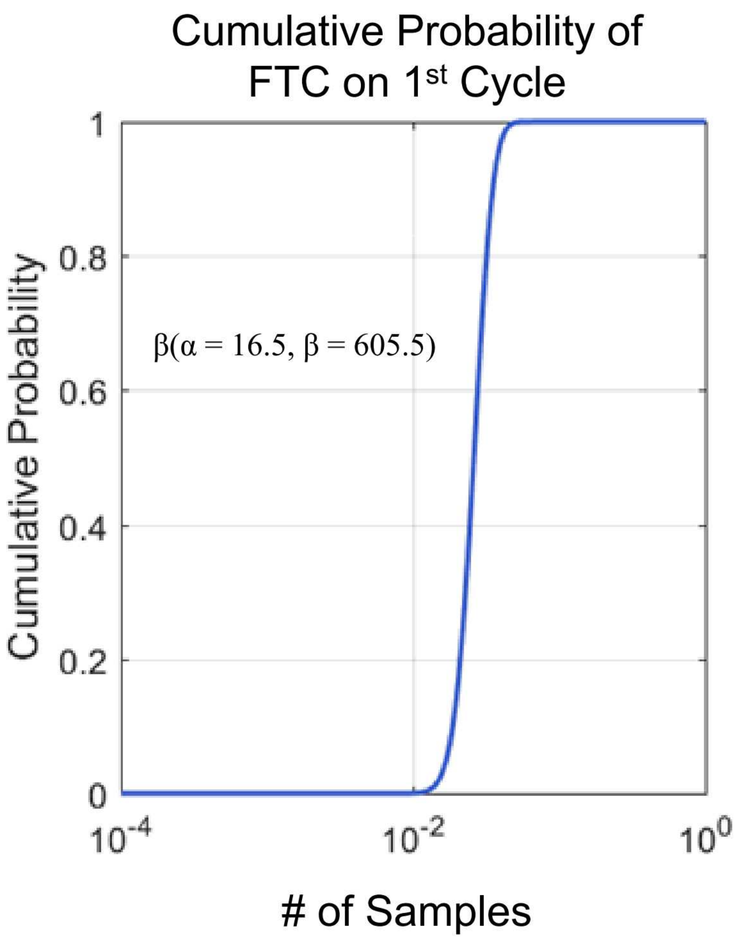
SV Failure Modeling (3)

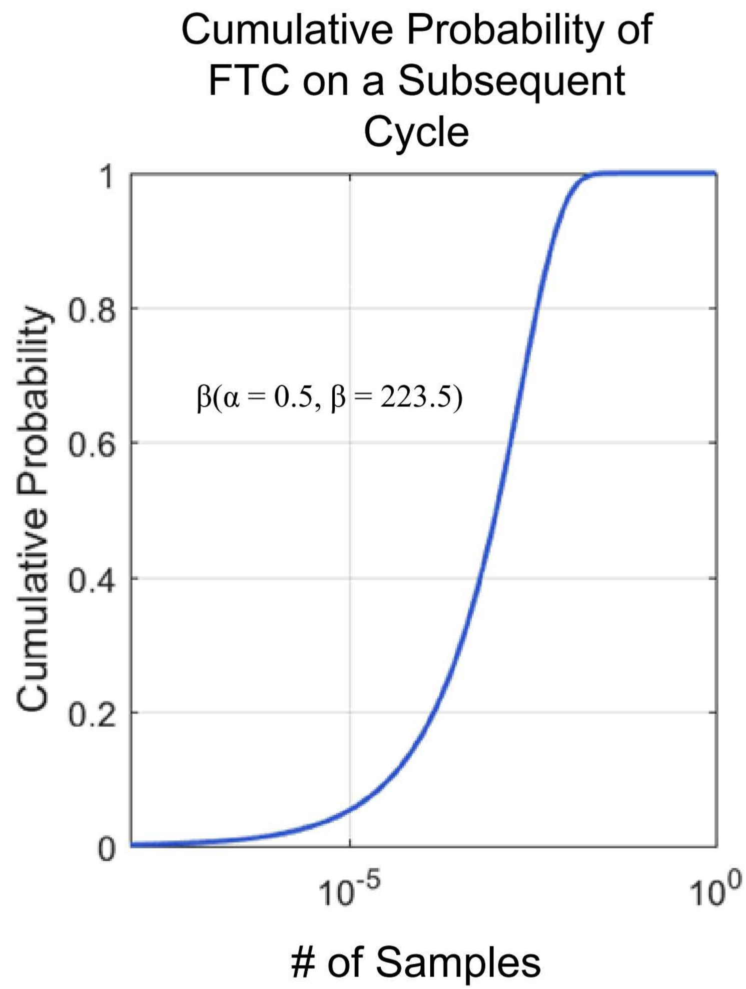
- FTC probability differs largely between initial and subsequent demands, i.e., if an SV operates per design on initial demand it will likely operate per design on all subsequent demands
- The assumption is made in that MSS and RCS SVs are alike enough in construct and servicing that their failure data can be jointly considered
- Recovered valve function, e.g., a previously stuck-open valve closing when pressure reduces, was not taken to be successful valve operation

SV Failure Modeling (4)

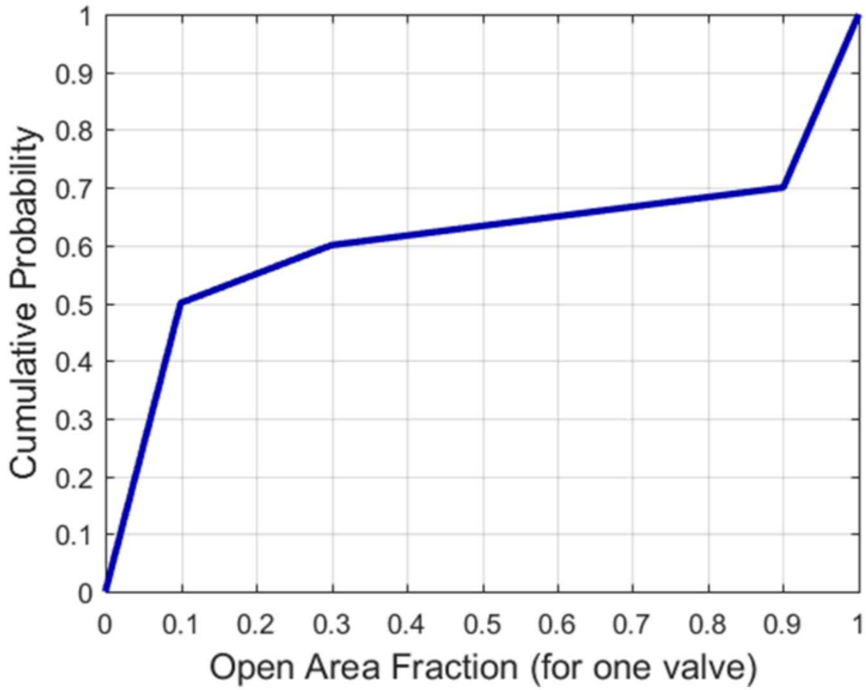
- NUREG/CR-7037 (Table 22) reports on failure rates in SV testing but the rates differ markedly from the rates evidenced by actual plant events suggesting (to the UA analysts) that aspects of the testing were inconsistent with actual conditions experienced by an installed valve, and as such, the testing data was not considered applicable
- Discussions with nuclear valve testing specialists and close examination of Licensee Event Reports indicate that an SV will likely be in either a weeping (mostly closed) or mostly open position following a failure to close

SOARCA UA SV FTC Sampling – Cycles to Failure



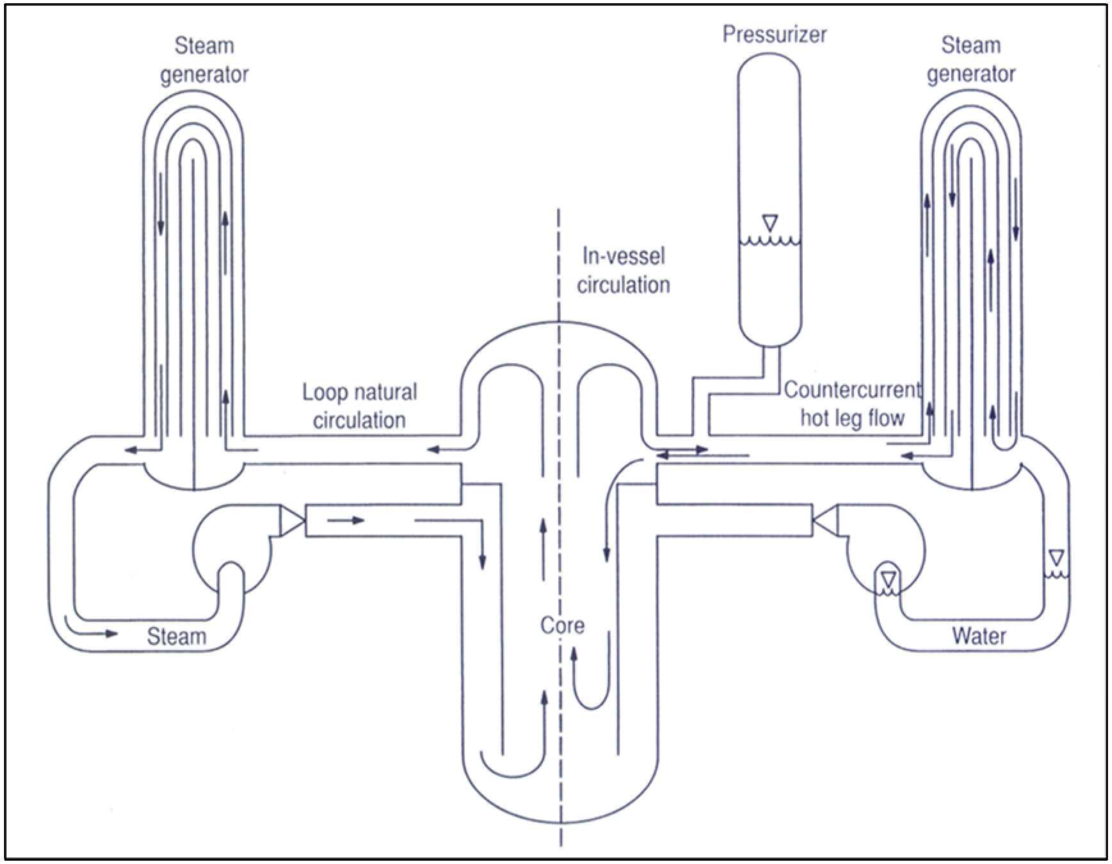


SOARCA UA SV FTC Sampling – Valve Position After FTC



Hot Leg Natural Circulation Modeling

- NRC and industry have studied SGTR vulnerabilities and potential consequences for decades because of the containment bypass situation an SGTR could cause in the case of core damage
- During a severe accident, natural convection would establish where hot gases circulate between the reactor core and the steam generators
- The gasses would heat the steam generator tubes increasing their vulnerability to rupture



Hot Leg Natural Circulation Modeling

- NRC and industry performed 1/7th scale experimental tests [EPRI Report NP 6324 D, 1989 and EPRI Report TR 102815, 1993]
- Tests studied by NRC with CFD [NUREG-1781]
- Modeling parameters were developed by NRC to characterize the natural circulation for lumped parameter codes like MELCOR [NUREG-1922]
- SNL developed MELCOR countercurrent flow model to manage the phenomenology in the hot legs under the subject conditions
- MELCOR model addresses fundamental aspects but cannot alone establish understood steam generator tube bundle flows
- Proactive flow manipulations are additionally required

MELCOR Countercurrent Flow Model

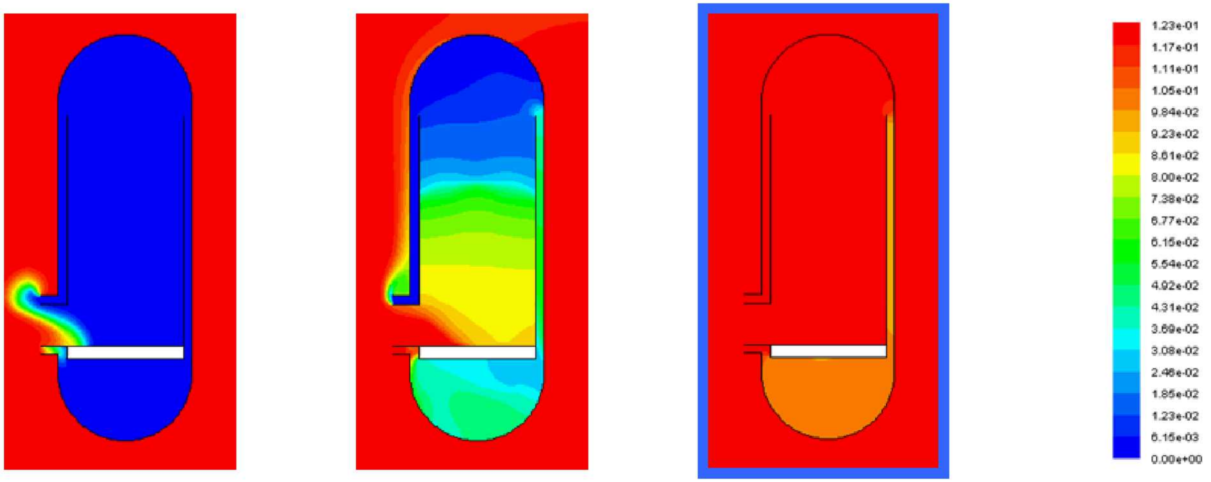
Next 12 slides…

Motivation

- HTGR accident with breach of pressure boundary will involve air ingress
  - Original scenario was that air entered by diffusion in stable density gradient (hot helium over cold air)
    - Timescale: many hours
  - Computational Fluid Dynamic (CFD) simulation shows that air enters via stratified counter-current flow in breach and circulation within vessel
    - Timescale: a few minutes
- Calculational model required for these accidents
  - Could have wider applicability

Air Ingress in HTGR

Snapshots of simulation (air mole fraction) from “NRC/INL Meeting on Methods for VHTRs”, E. Kim, C. Oh, R. Schultz, INL (2008)



1.0 sec **16.0 sec** **256.0 sec**

Characteristics of Model

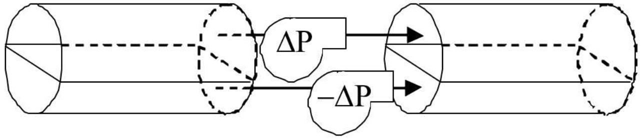
- Physical mechanism
  - Flows in two counter-current gas streams limited by exchange of momentum (drag) between them
- Can occur in other situations
  - Flows through openings between rooms
  - Natural circulation in PWR hot legs
- Correlations exist based on Froude number

$$Fr = \frac{Q}{\sqrt{\ell^5 g \Delta\rho / \overline{\rho}}}$$

  - Form derivable from concept of momentum exchange
  - Epstein-Kenton, Journal of Heat Transfer, 1989
- Not “natural” (or easy) to directly impose such a correlation in MELCOR

Historical Capability for PWR Hot Leg Case

- Old concept used in MELCOR for many years
  - Split path in two to allow counter-current flow
  - Account for momentum exchange in flow equations
    - Implement as “pumps” with ΔP calculated from relative velocities



    - Use control functions to determine pump ΔP

$$\Delta P = -\frac{2fL}{D_h}\bar{\rho}(v_1 - v_2)|v_1 - v_2| = -C\bar{\rho}(v_1 - v_2)|v_1 - v_2|$$

- Current “state of the art” uses controllers on pumps to match a Froude correlation

New CCF Model

- Pump approach "internalized"
  - We have coded an appropriate ∆P calculation
    - Specific input was added to couple two flow paths
      - Input parameters based on correlations (with defaults)
      - Flexible enough for a variety of applications
    - Form of ∆P has been generalized to better match published Epstein-Kenton correlations
  - Terms added directly to flow equations
    - Increased stability because of implicit numerics (CF-based models are inherently explicit)
- Can be tested in a variety of ways
  - Compare analytic results with correlations
  - Compare calculated results with correlations
  - Compare calculated results with CFD

New Input for Counter-Current Flow Model

- Input added as a table, FL_CCF
  - Define name and number for each instance
  - Identify two flow paths that together model one “real” path
    - Areas should sum to total area, but need not be equal
    - Junction openings should have correct total, not overlap
  - Define characteristic length (height) and discharge coeff.

```
! Coupling of two flow paths by momentum exchange
FL_INPUT
. . .
FL_ID   'Upper'    1
. . .
FL_ID   ‘Lower'    2
. . .
FL_CCF  1 !  Name   num   FL1      FL2     CharLen  C_D
        1   'Test'  123  'Upper'  'Lower  0.7366   0.0358
```

PWR Hot Leg Natural Circulation Test Case

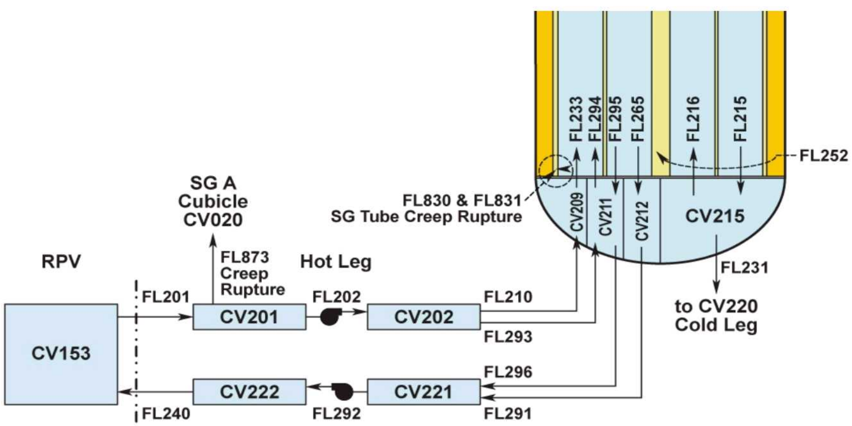
- One loop (hot leg and steam generator) of a PWR
- Fast-running
- Constant properties in vessel upper plenum
- Nodalization shown on next slide
  - “State of the art”, tuned to CFD results
    - QUICK-CF pumps used to limit counter-current flow in HL
    - Specified velocity paths used to enforce flow splits for mixing in SG lower plenum
    - QUICK-CF pumps used to enforce ratio of steam generator circulating flow to hot leg flow

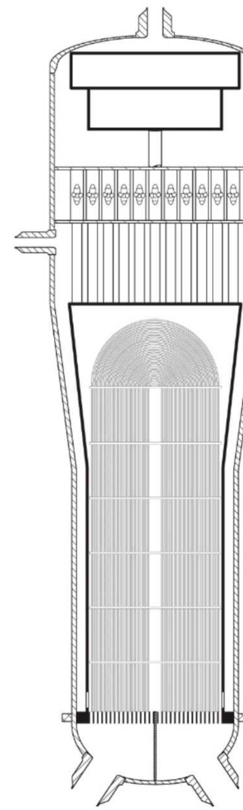
Hot Leg / Steam Generator Nodalization

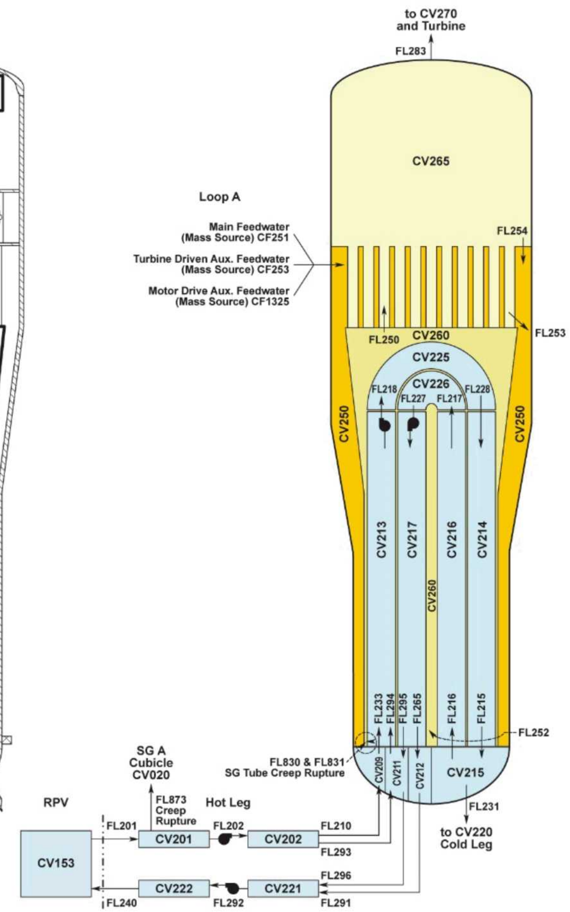
Pumps in FL202 and FL292 impose Froude correlation

Pumps in FL 218 and FL227 enforce circulating flow ratio

Velocity in FL293 and FL295 forced equal to FL210 and FL265, respectively





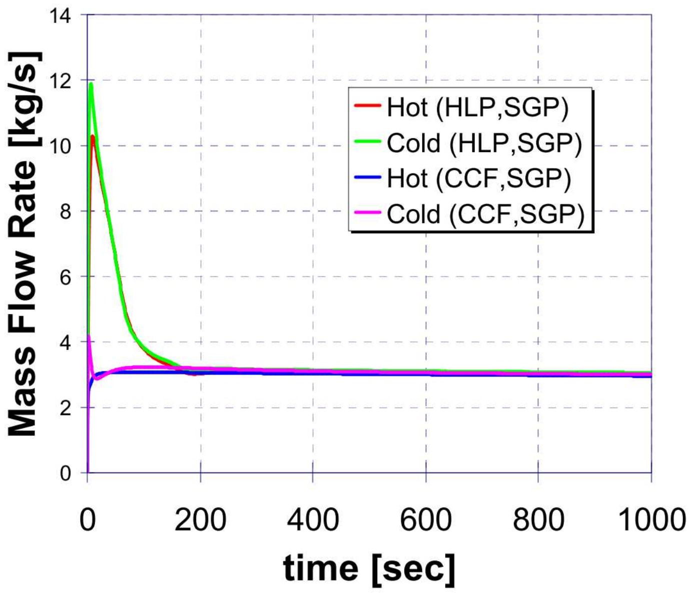


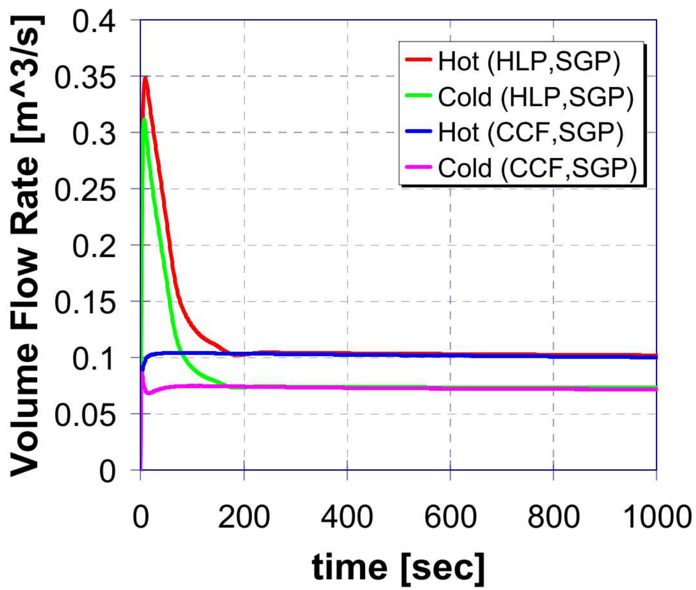
PWR Natural Circulation Calculation

- Calculation run "as delivered"
  - Initial transient (after initial overshoot of flows) led to almost-steady natural circulation
    - Transient a result of very *un*steady initial state
- Rerun, replacing hot leg pumps by CCF model
  - Discharge coefficient "tuned" for agreement: $C_D = 0.065$ compared to $C_D = 0.394$ in previous example
- Will discuss further after presenting results

PWR Natural Circulation Calculation Results

- Results labeled by models in use
  - HotLegPump, SteamGeneratorPump, Counter-CurrentFlow
  - Calculation "as delivered" is "(HLP ,SGP)"
  - Calculation with CCF model is "(CCF,SGP)"





Status of Use of CCF Model PWR HL NC

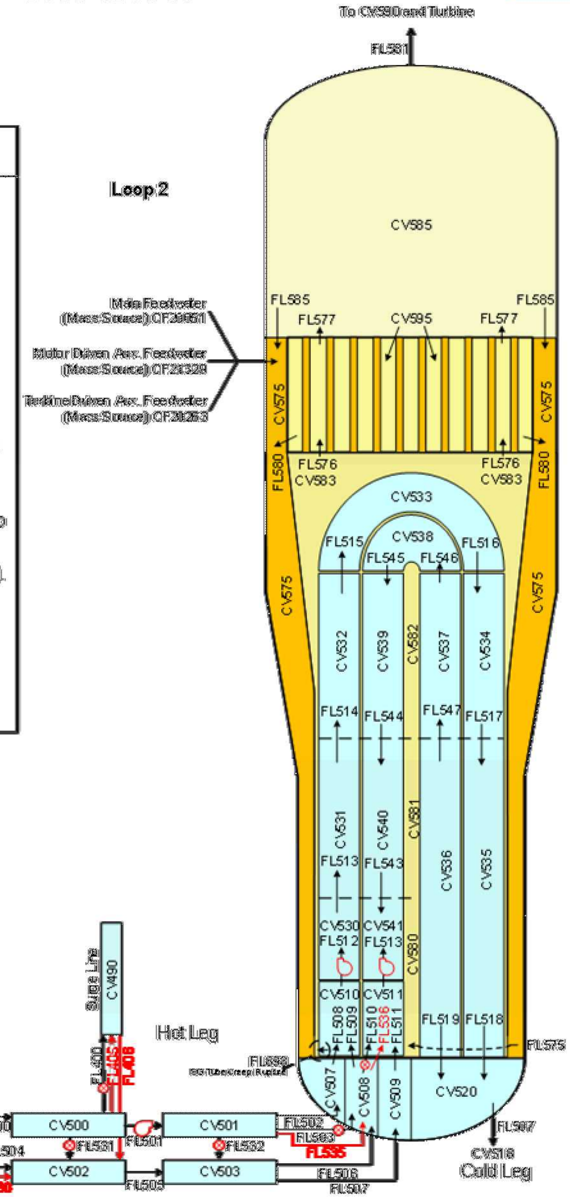
- Model works well for applications like HTGR air ingression and containment flows
- Application to PWRs is not alone sufficient
  - There is a significant difference from situation for which the model was formulated
  - Model derivation assumed that net buoyancy force driving circulation can be evaluated from densities in two volumes
  - Not the case in PWR hot leg natural circulation: significant additional buoyancy results from hot gas in rising section of steam generator tubes
- Current "state of the art" PWR calculations impose Froude correlation based on same two densities
  - Discharge coefficient determined from CFD calculations that include buoyancy in steam generator tubes
  - Questions arise about application under other conditions, just as with CCF model

Current Best Modeling of Hot Leg Natural Circulation Utilizing MELCOR Countercurrent Flow Model

- Hot leg and steam generator nodalization supportive of hot-leg countercurrent natural circulation
- Natural circulation, consistent with CFD analyses of NUREG-1922, accomplished through use of MELCOR countercurrent flow model (just described) and with proactive flow path openings/closings and momentum additions

Natural Circulation Flow Path Definitions
(identified in red on diagram)

*Natural circulation entry conditions defined as (a) hot leg CVs <5% water, (b) >10 K super heat in hot leg, and (c) recirculation pumps tripped.
*Maintain natural circulation flow paths when (a) hot leg CVs <25% full of water, (b) pumps are off, (c) no major creep rupture failures, and (d) loop seal flow is <20% of HL flow. (CF5664)
*FL504, FL531 and FL532 are open and FL530 is closed during non-natural circulation conditions. FL504, FL531 and FL532 are closed and FL530 is open in natural circulation conditions.
*FL503 and FL510 are open and FL535 and FL536 are closed in non-natural circulation conditions. FL503 and FL510 are closed and FL535 and FL536 are open in natural circulation conditions.
*FL535/FL502 and FL536/FL511 are adjusted to give mixing ratio of 20%/80%.
*FL501 pressure drop adjusted to give $C_D = 0.12$ (from FLUENT).
*FL512 and FL513 pressure drop adjusted to give Tube_flow/HL_flow (M_{ratio}) = 2.
*FL400 is open and FL405 and FL406 are closed during non-natural circulation conditions. FL400 is closed and FL405 and FL406 are open in natural circulation conditions.



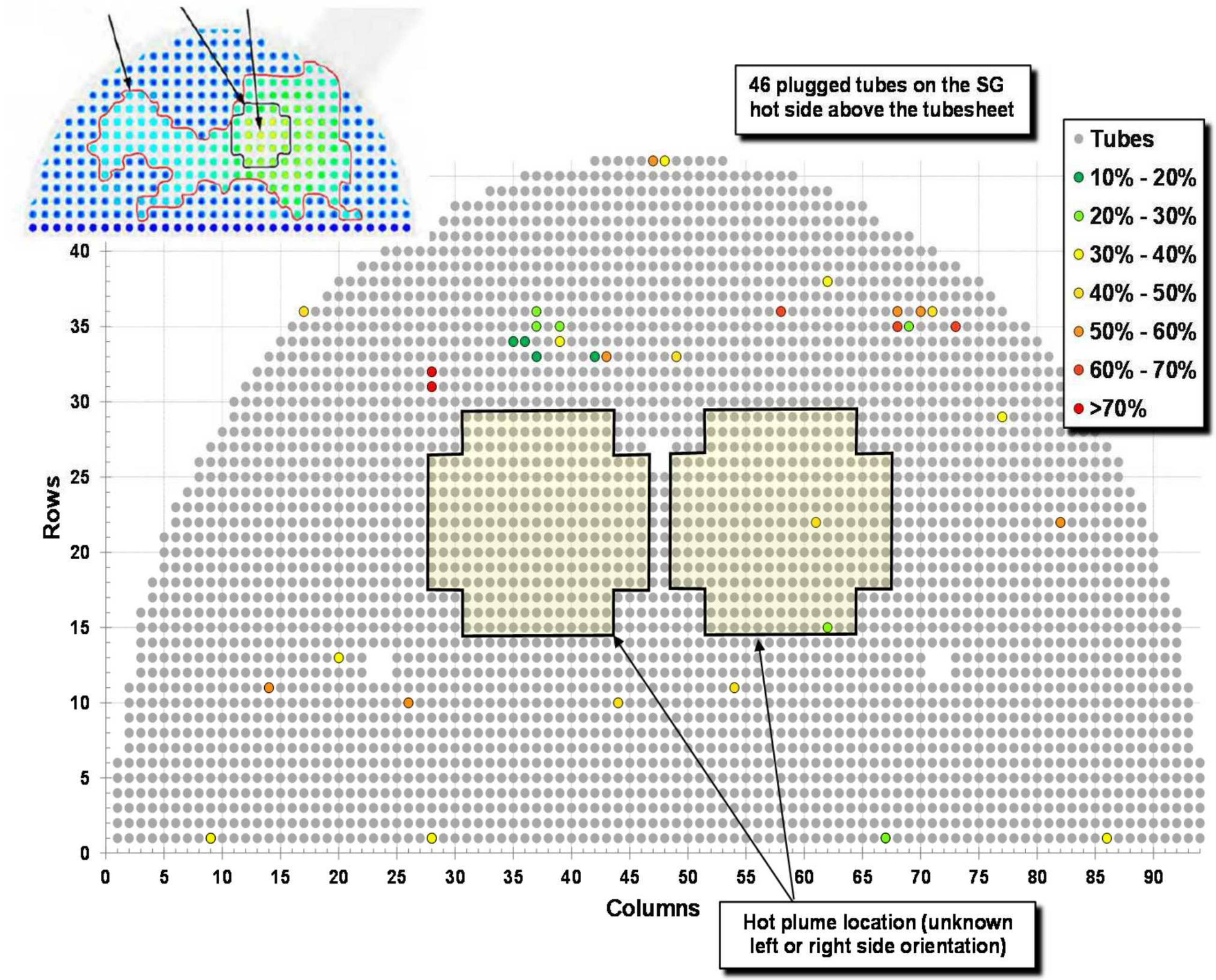
C-SGTR

- Overheating and stress from pressure differential could combine to threaten steam generator tubes
- Overheating could happen in a severe accident
- Stress can be compounded by flaws such as wall thinning or dings
- Industry data, in conjunction with current SNL/NRC SOARCA work, suggests that flaws are common enough and large enough that a consequential SGTR occurring during a severe accident is a viable concern
- Current SNL/NRC SOARCA uncertainty work suggests the threat of an SGTR is not limited to just the hottest tubes in a tube bundle but, dependent upon flawed degree, could exist for even the coolest tubes

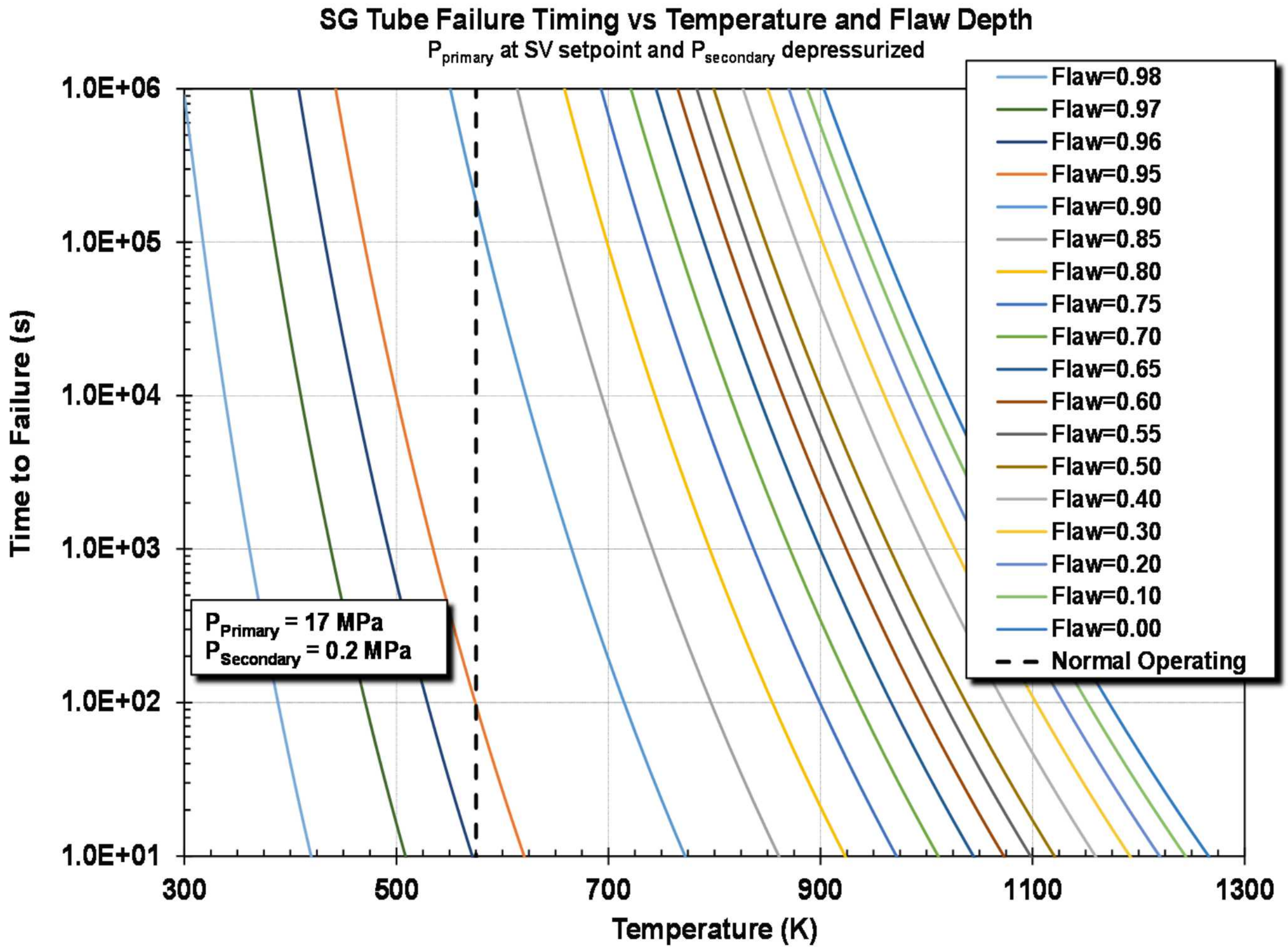
Steam Generator Tube Flaw Prevalence and Degree

- Tube flaws, including dings, are expressed as effective reductions in wall thickness
- Tube thinning due to wear at anti-vibration bars relates directly to reduced wall thickness
- Such wear is well understood, well monitored and well managed by plugging tubes
- Less manageable are dings to tubes from loose parts or maintenance activities
- The minimum possible effective thickness of a tube in an operating steam generator can be determined by considering simply the hoop stress given the pressure difference between the primary and secondary sides of the RCS - the minimum thickness needed to keep stress below the yield stress of the tube material is the thinnest a tube could be lest it rupture at rated conditions (e.g., 25% of the manufactured thickness for Surry tubes)
- NRC research has shown that a normalized flaw depth of of 0.3 to 0.4, i.e., a reduction in wall thickness of 60% to 70% would be needed for a tube to rupture in a severe accident situation
- Accordingly only flaw depths greater than 0.3 are considered in SNL's current best practice

Flaw Depths in the Steam Generator Tubes that Have Been Plugged in the Two Surry Units



Steam Generator Tube Rupture Timings at Rated Pressure Versus Temperature and Flaw depth



Steam Generator Hottest Tube Representation

- The hot leg natural circulation modeling described above captures the average temperature of "upward" (hotter) and "return" (cooler) flowing steam generator tubes but not the temperature of the hottest tube
- To capture the temperature of the hottest tube for worst-case creep damage consideration, a representation of a single steam generator tube is defined but separately from the balance of the RCS representation
- The single tube representation has boundary conditions applied to it that account for the localized hottest flow temperatures that would exist in a tube bundle relative to hot leg inlet and steam generator return flow temperatures
- Localized hottest flow temperatures are taken to be those suggested by the generally accepted CFD calculations of hot leg natural circulation conditions referenced above
- Average hot leg inlet and steam generator return flow temperatures are taken to be those calculated by MELCOR

Steam Generator Hottest Tube Representation (2)

- Pressure, gas constituency, flow velocity, flow temperature and convective heat transfer coefficient are supplied as boundary conditions to the tube side of single tube model
- The heat structure representing the single SG tube is coupled to the boiler of the steam generator
- The temperature of the heat structure is used in determining the creep damage associated with the hottest tube in the steam generator
- From consideration of the hot leg natural circulation research identified above, the temperature of the flow supplied to the single tube model (the hottest tube inlet temperature) was taken from:

$$Tn = (Tht - Tct) / (Th - Tct)$$

where:

Tn	= Normalized temperature (nominally 0.43)
Tht	= Hottest tube inlet temperature
Th	= Hot leg inlet temperature
Tct	= Cold-tube return flow temperature

Distribution of Flaws in a SG Tube Bundle

- For prediction of the tube failure characteristics, there are three distinct regions of interest [NUREG-1922]:
  - The hottest tube region,
  - The broader hot up-flow region, and
  - The remainder of the steam generator
- In the case of Surry, the fractions of the total flaws in a steam generator residing in the different regions are given by:
  - $X_{hot} = (0.41)*(0.13)*(0.61) = 0.032$
  - $X_{upflow} = (0.41)*(1 - 0.13)*(0.61) = 0.22$
  - $X_{cold} = 1 – (X_{hot} + X_{upflow}) = 0.75$

where,

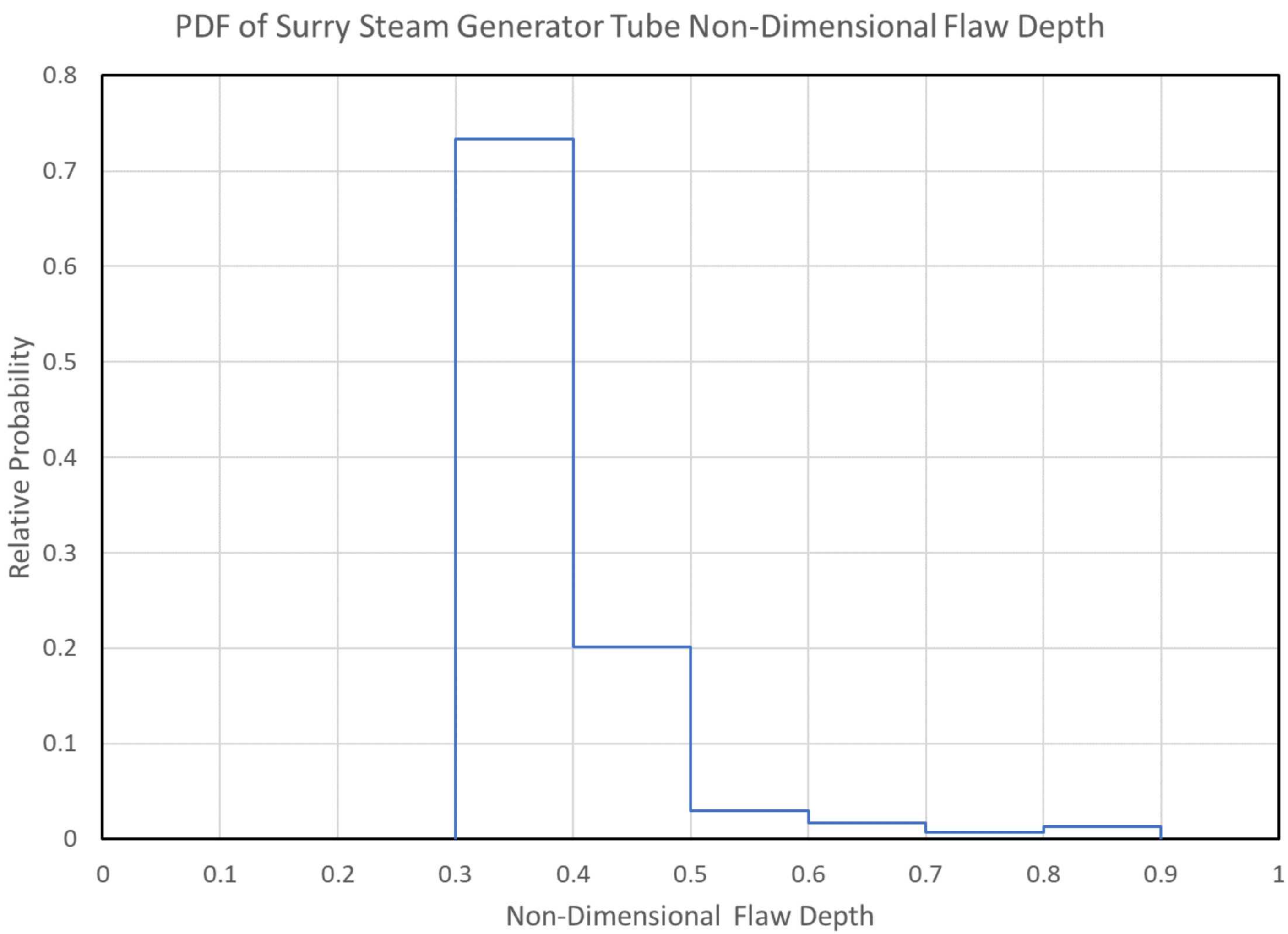
  - 41% of the steam generator tubes are in up-flow during natural circulation
  - 13% of the up-flow tubes are hottest tubes
  - Surry ISI reports show 61% of the flaws are on the inlet side of the SG

Uncertain Treatment of Flaw Depth

- In the case of Surry, the table below presents flaw number and depth per SG
- By the equations on the previous slide:
  - 0.14 flawed tubes reside in the hottest region
  - 0.94 flawed tubes reside in the remainder of the up-flow region
  - 3.20 flawed tubes reside in the balance of the tube bundle
- For each UA realization and for each SG, five flaw samples were randomly made from the distribution on the next slide (associated with the table below):
  - The maximum of three of the samples was used for the cold region flaw depth as only the most severely flawed tube in this region was be modeled
  - Another of the samples was used for the up-flow region
  - The remaining sample was used for the hottest region 14% of the time (in 14 of every 100 realizations) such that there was no flaw in the hottest region 86% of the time

Non-dimensional Flaw Depth	NUREG-2195 Total Flawed Tubes per SG	Surry Data Total Flawed Tubes per SG	Surry Data & NUREG-2195
0.3 to 0.4	3.126	Not included	3.126
0.4 to 0.5	0.858	Not included	0.858
0.5 to 0.6	Not included	0.125	0.125
0.6 to 0.7	Not included	0.069	0.069
0.7 to 0.8	Not included	0.028	0.028
0.8 to 0.9	Not included	0.056	0.056
Total			4.26

Given a Flawed Tube in a Surry SG – The Probability of the Non-Dimensional Depth of the Flaw

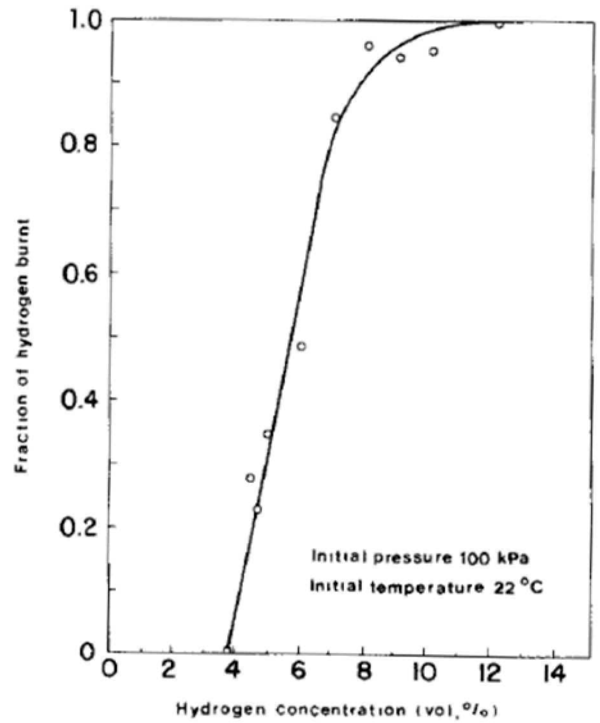


Deflagration Ignition Sources

- Various ignition sources have been considered in the SOARCA work:
- A hot plume (> 0.1 m/s and > 847 K) issuing from:
  - An RCS breach at a hot leg nozzle (Sequoyah and Surry)
  - An RCS breach in the pressurizer surge line (Sequoyah and Surry)
  - The PRT through a broken burst disk (Sequoyah and Surry)
- Core debris on the containment floor (Sequoyah and Surry)
- 10% mole fraction H_2 (Peach Bottom)
- An occasional momentary spark somewhere in containment (some Sequoyah) characterized by:
  - A 1-second duration
  - A half-hour frequency
  - Appearance in one randomly-chosen control volume on each occasion

Combustibility Considerations

- Combustibility is dependent upon relative concentrations of fuel, oxidizer and diluent gasses measured as mole fractions (= ratios of partial pressure to total pressure in an ideal gas mixture)
- Too little fuel or oxidizer or too much diluent prohibits burns
- The strength of a burn depends on the amount of fuel available to burn that actually burns, i.e., combustion completeness is important
- Lesser values of LFL relate to lesser accumulations of fuel at ignition and hence lesser strength burns
- Combustion completeness is a function of fuel concentration at ignition
- While it might seem, for example, that ignition of H_2 at a mole fraction of 0.08 would involve twice the H_2 in a burn than ignition at 0.04 would (volume and pressure being the same), 0.08 would actually involve 33 times more H_2 because 0.08 would burn down to 0.00032 while 0.04 would burn down to 0.03762

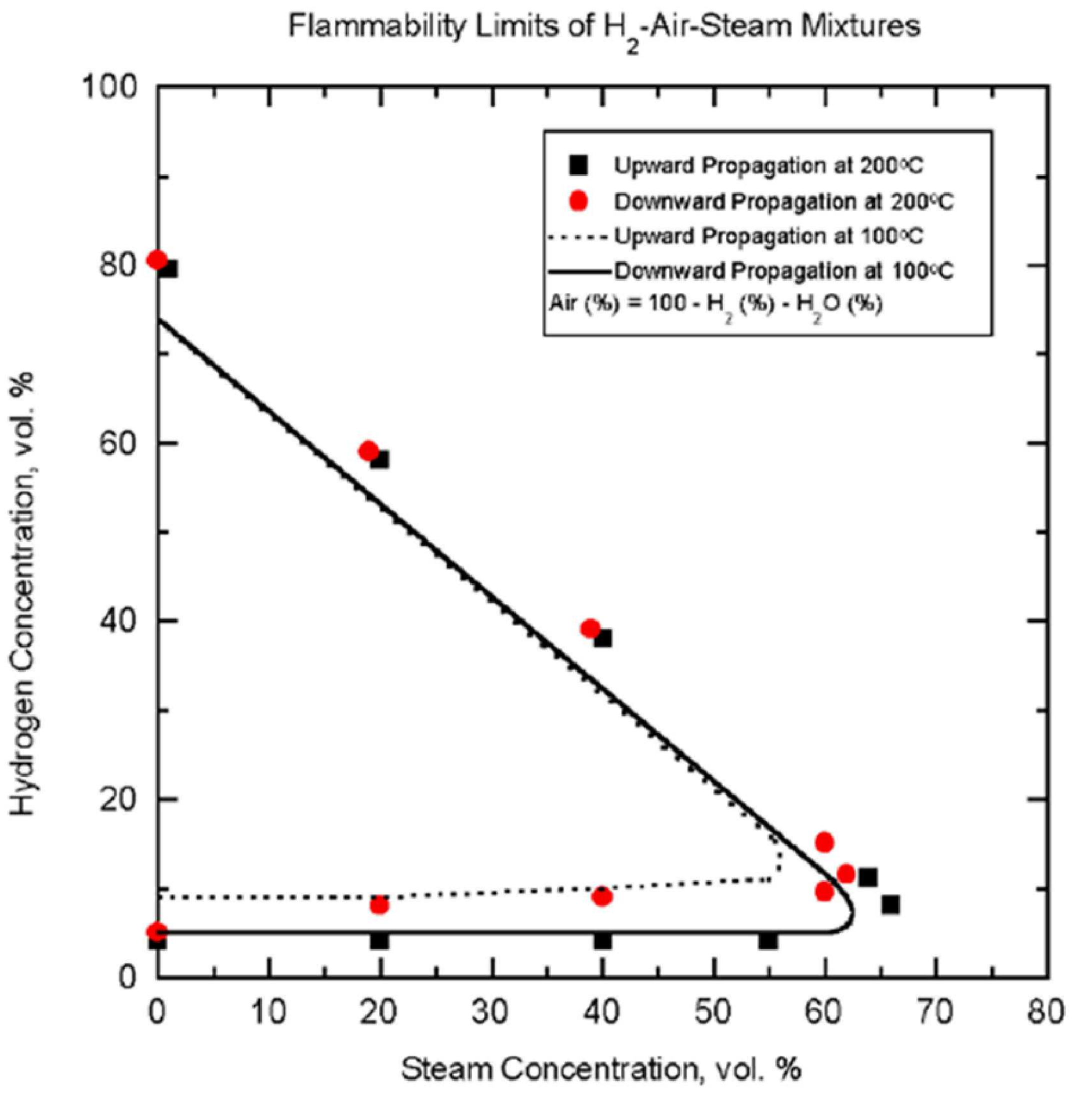


Combustible Mixture Criteria

- For the origination of a burn, the SOARCA Sequoyah and Surry models expands upon MELCOR's default fixed criteria of > 0.10 H_2, > 0.05 O_2 and < 0.55 H_2O to include LFL variability per work of Kumar
- For the propagation of a burn from one control volume to another, MELCOR's default fixed criteria of > 0.04 H_2, > 0.06 H_2, and > 0.09 H_2, for upward, lateral and downward propagation, respectively, > 0.05 O_2 and < 0.55 H_2O, were maintained (propagation directionality being user-defined in flow path descriptions)
- Provision added for variably defining H_2 LFL in consideration of the direction a burn would need to propagate from its origin: 0.04, 0.06 or 0.09 dependent upon propagation being upward (e.g., from a floor), lateral (e.g., in a horizontal duct) or downward (e.g., from a ceiling), respectively

Combustible Mixture Criteria (2)

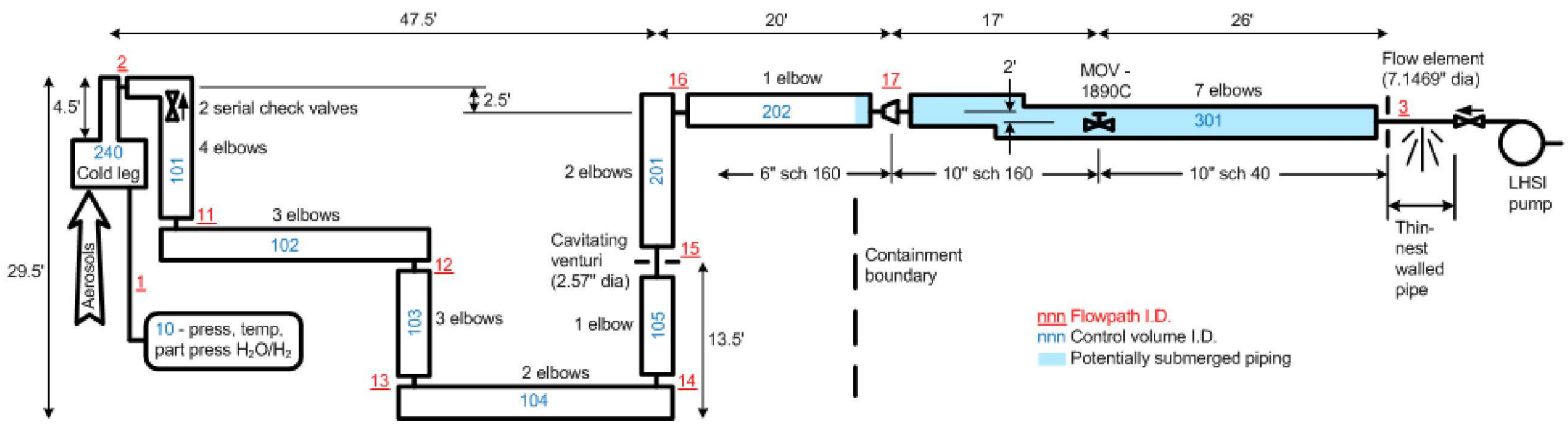
- Additionally define sufficient fuel (H_2 and or CO) to be a function of diluent (e.g., H_2O or CO_2) concentration
- Further include a temperature dependence of 0.005 and 0.01 less H_2 per 100°C for upward and downward propagation, respectively
- Maintain MELCOR default of 0.05 O_2 necessary for a burn



ISLOCA Modeling Illustration

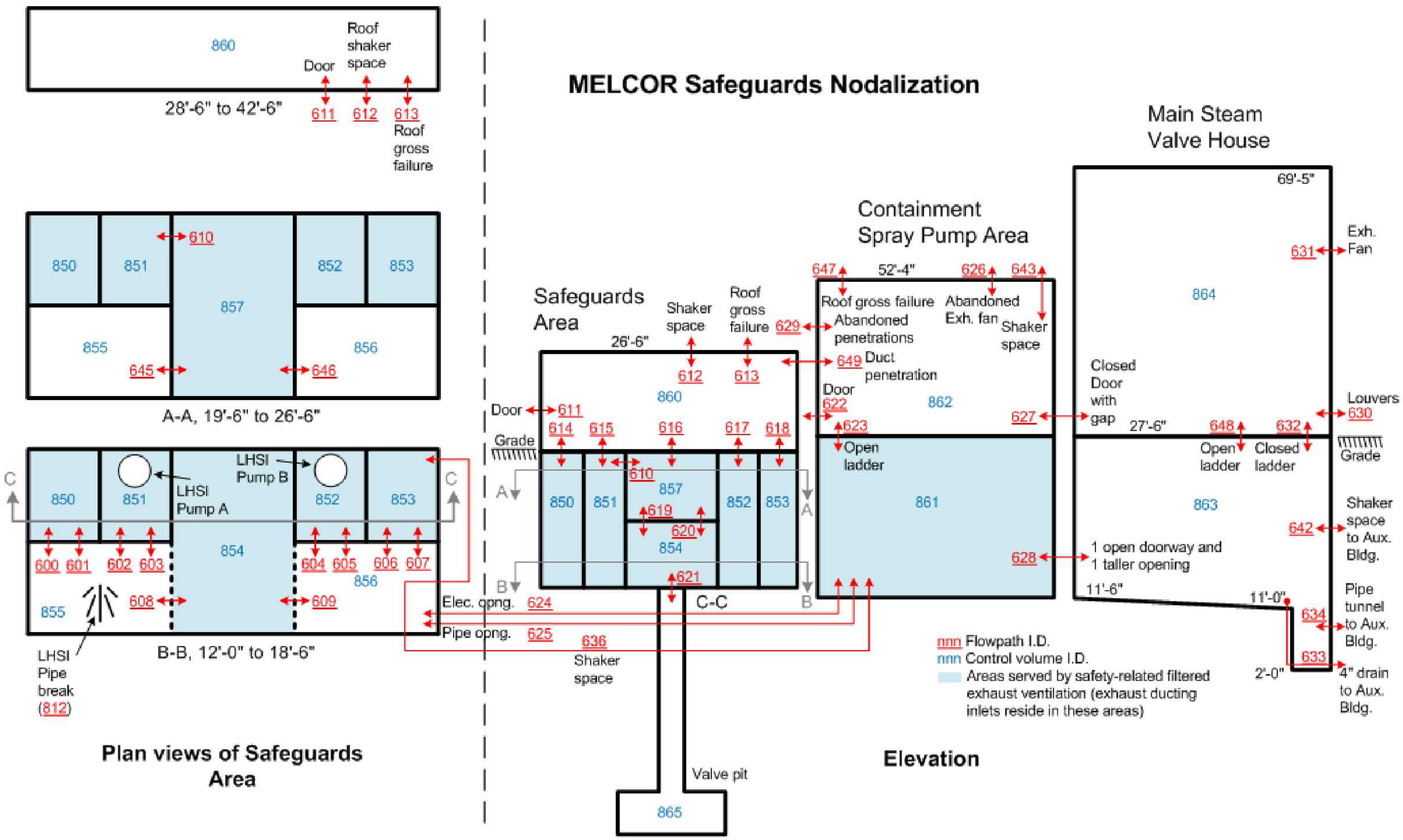
- ISLOCA scenario documented in NUREG/CR-7110, Volume 2, Rev. 0, SOARCA, Surry Integrated Analysis
- Important similarities surely exist between Surry and other PWR ISLOCA scenarios but important differences also surely exist
- Limiting flow area in the faulted injection piping is key
- Auxiliary building flooding response is key
- Function of safety class auxiliary building ventilation/filtration is key
- Faulted injection piping geometry is key (especially with respect to mitigation strategy)

LHSI Piping Nodalization

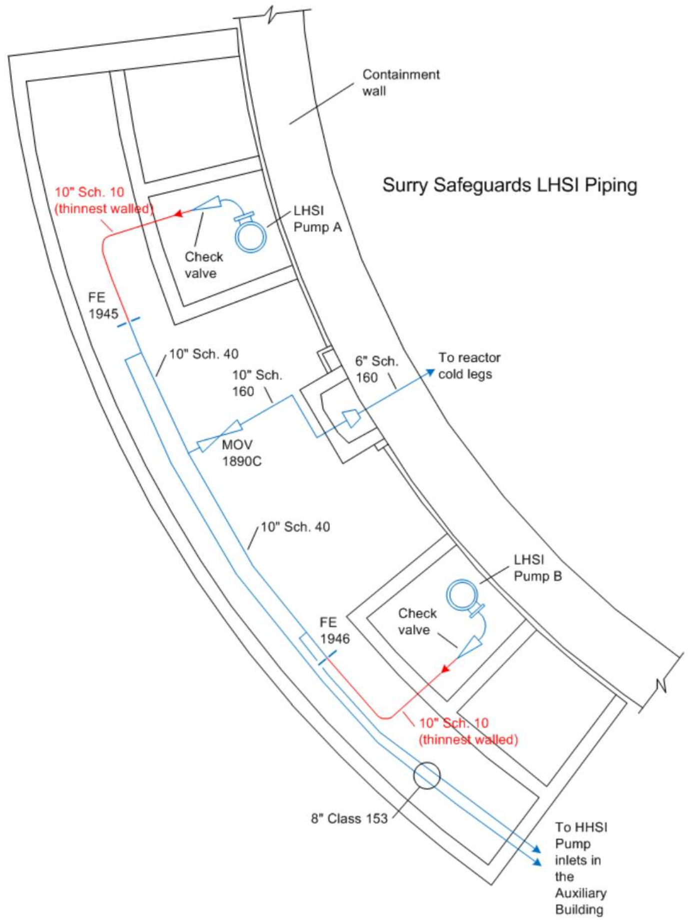


MELCOR Representation of Low Head Safety Injection Line

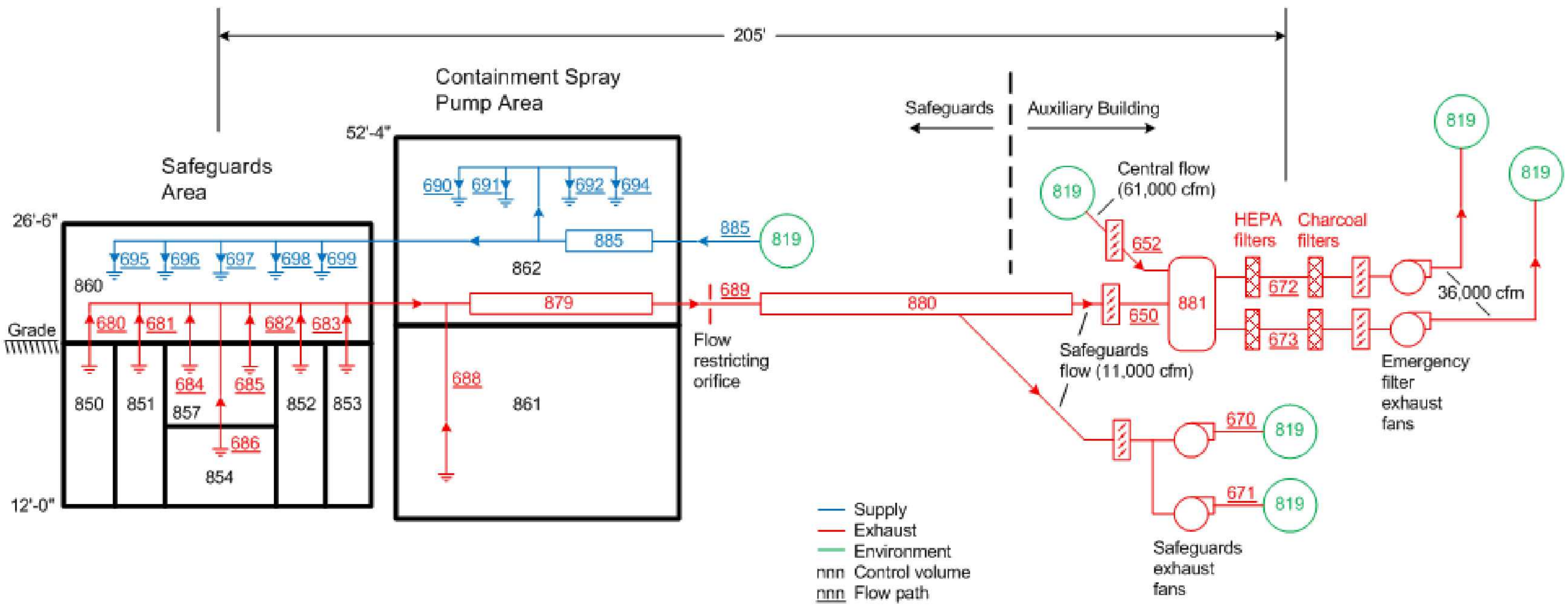
Safeguards Nodalization



Safeguards LHSI Piping



Safeguards Ventilation Nodalization



MELCOR Representation of Safeguards Ventilation

Assumptions (I)

Assumption	Basis	Affect
Both serial check valves serving a safety injection line fail open such that their disks are displaced entirely from their seats.	Conservative	The rate of RCS leakage is likely greater in the MELCOR calculation than it could be at the plant.
Timely operator response to conserve RWST inventory as demonstrated in the Surry simulator including: • Stopping LHSI Pump A 6 min and 17 sec into the accident • Stopping LHSI Pump B 15 min and 44 sec into the accident • Isolating LHSI pump suctions 16 min and 18 sec into the accident • Stopping 2 of the 3 total HHSI pumps	Response times demonstrated in the Surry simulator.	Without these operator actions, fission product releases to the environment would initiate dramatically earlier in the MELCOR calculation.
A submerged break location in the Safeguards Area.	Slightly more likely given the lengths of vulnerable piping above and below the anticipated pool level.	Releases to the environment would be somewhat higher in the MELCOR calculation without the benefit of pool scrubbing in the Safeguards Area.
Close fitting flexible metallic flashing covering the shaker space between the Safeguards Area and the Containment Spray Pump Area.	Difficult to quantify.	The tightness of the flashing determines whether the LHSI piping in the Safeguards Area is submerged or not. Considerably more Cs and I in the form of CsI is deposited and retained in the piping if it is submerged. Tightness defined keeps the piping submerged for the course of the calculation.
No switchover to the unaffected unit's RWST.	Conservative	Core damage could be delayed if the operators were to align HHSI to the unaffected unit's RWST once the affected unit's RWST was exhausted.
No throttling of HHSI to the minimum required to remove decay heat.	Conservative	Time to core damage could be extended dramatically if only enough water were injected to remove decay heat (the water being boiled off and exhausted out the ISLOCA pipe break).

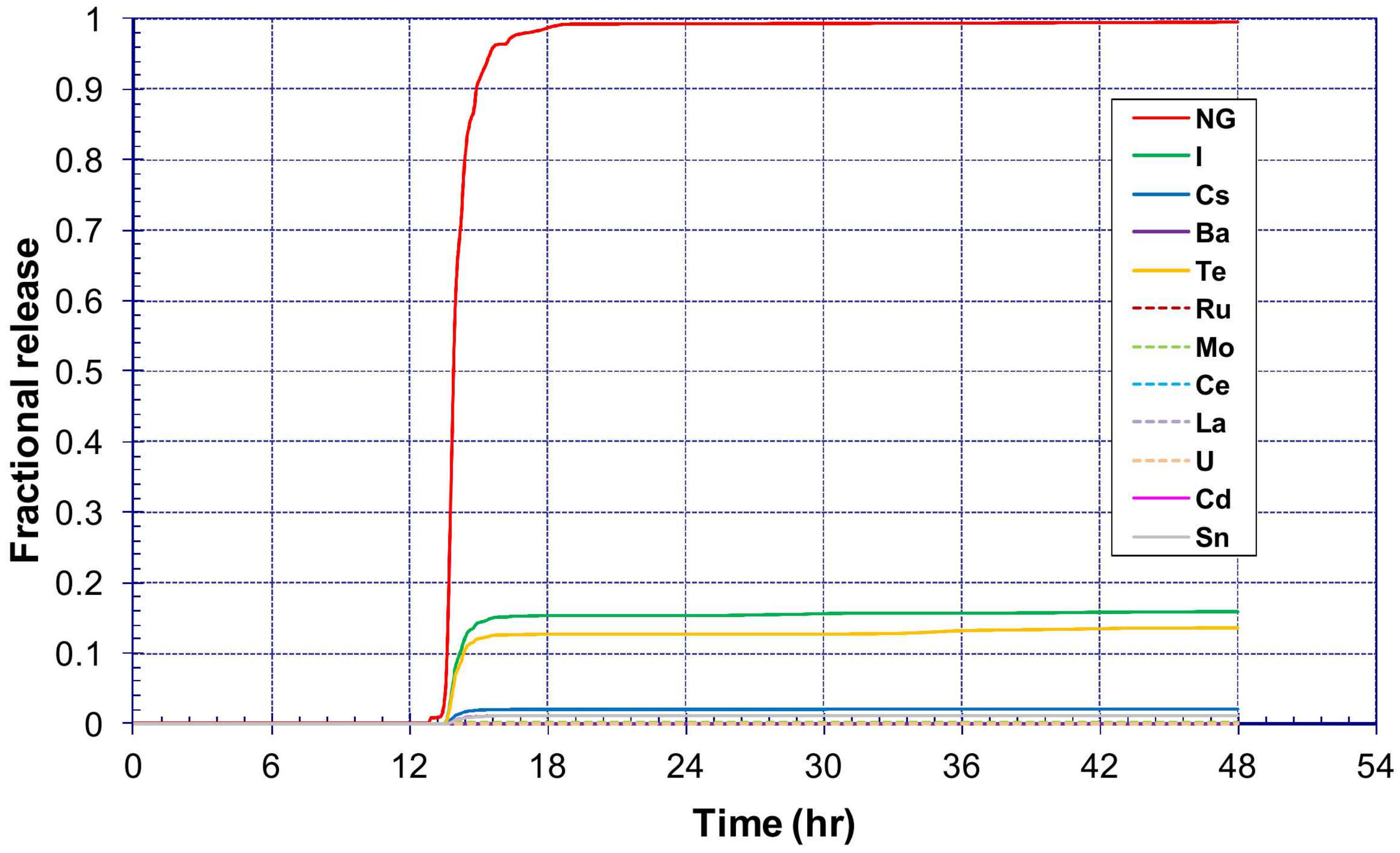
Assumptions (2)

Assumption	Basis	Affect
Efficient aerosol deposition in LHSI piping by turbulent deposition and impaction (substantiated by LACE testing).	Best-estimate phenomenology	Without deposition in the LHSI piping, releases to the environment in the MELCOR calculation would be much greater.
No resuspension of radionuclides deposited in the LHSI piping (revaporization is accounted for).	Difficult phenomena to model	Resuspension would increase releases to the environment.
No effect from deposited materials in the LHSI piping on the flow in the piping.	Difficult to represent	The depositions of material in the LHSI piping accomplished by MELCOR are large to the point that flow in the piping would be affected.
Aerosols generated from core concrete interaction deposit by turbulent deposition and impaction identically to how uranium aerosols do.	Unquantified influence	If concrete aerosols deposited less readily than uranium aerosols, more concrete aerosols would migrate to the exhaust ventilation HEPA filters in the MELCOR calculation, possibly overloading the filters. Overloading the filters would trip the exhaust fans.
The 11.7 kg of CsOH initialized in the MELCOR reactor core has the vapor pressure characteristics of CsMoO4.	SOARCA default	The CsOH in the MELCOR calculation may be transporting less readily than it would physically.
No damage to Safeguards exhaust system from H2 /CO burns in the Safeguards Area.	Difficult to quantify	Damage to the exhaust system would increase releases to the environment.
No loading on Safeguards Area exhaust ventilation HEPA filters from smoke produced by fire in the Safeguards buildings.	Difficult to quantify	Substantial smoke loading on the filters would cause the fans of the exhaust system to shut down. Overheating of the fission products deposited in the filters would be a threat given the loss of flow through the filters.
Complete mixing in the HEPA filter inlet plenum of hot air pulled from the Safeguards Area with cool air from other areas.	Difficult to quantify	The temperature of the air flow pulled from Safeguards in the MELCOR calculation is severe. If this hot flow were not to mix well with much larger cool flows from elsewhere in the Auxiliary Building, the filters would fail on excessive temperature. A forced flow to the environment laden with fission product aerosols would ensue.
Operators accomplish RHR entry (in the mitigated scenario) before the HHSI pump motors in the Auxiliary Building basement flood.	Envisioned successful mitigative strategy	This timing would need to be accomplished for the RHR-entry mitigation strategy to be successful.
Gamma absorption fractions of 0.55 and 0.33 for 6" and 10" pipe, respectfully	Gamma ray attenuation of steel	23% and 33% of the decay power generated by fission products deposited in 6" and 10" injection piping, respectively, escapes the piping

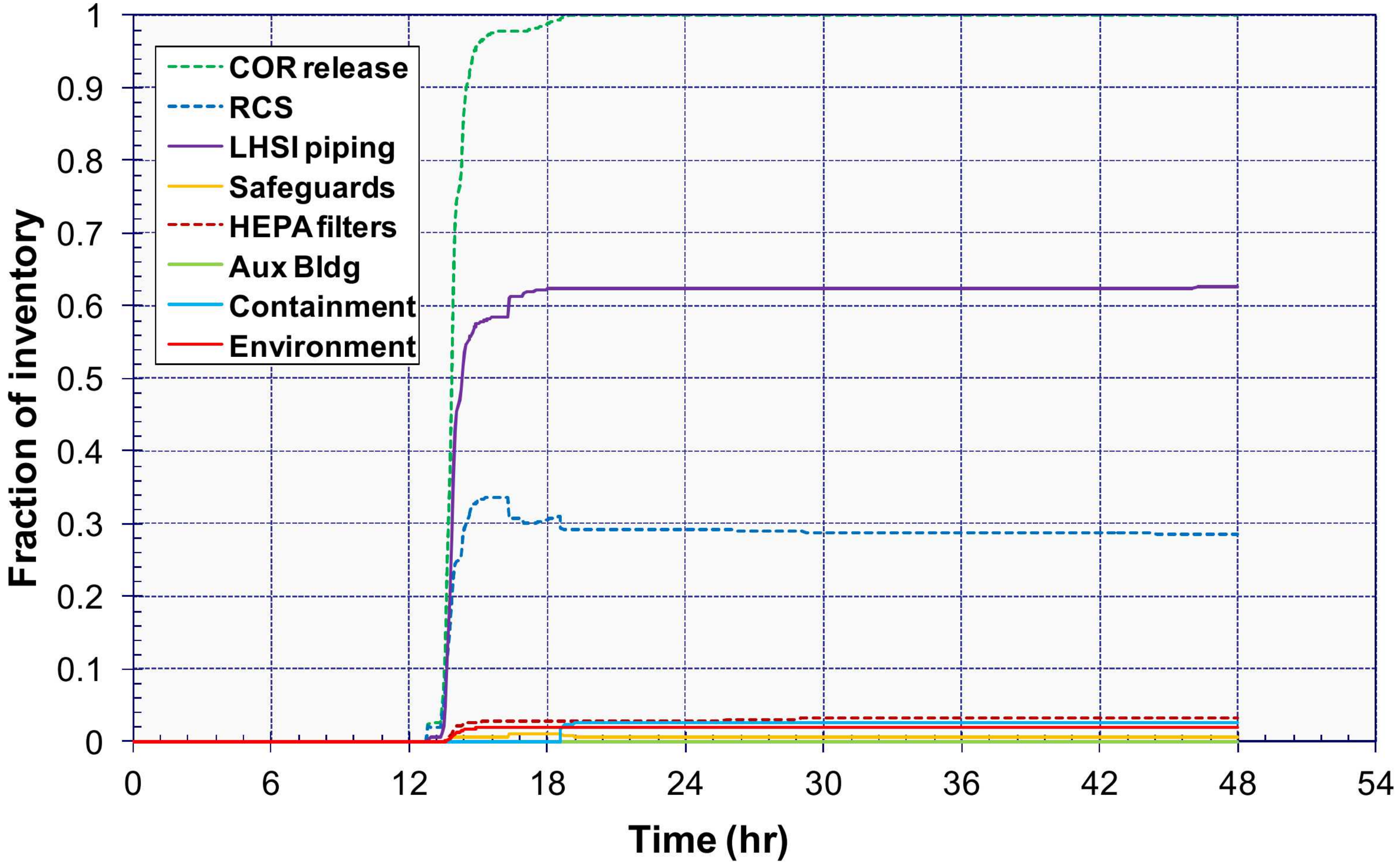
Event Timing

Event	hh:mm:ss
LHSI check valves fail	00:00:00
LHSI piping ruptures in Safeguards Area (outside Containment)	00:00:00+
Safeguards personnel door opens	0.16 s
Safeguards roof flashing tears	0.36 s
SCRAM	00:00:22
ECCS initiates	00:00:26
Safeguards filtered exhaust ventilation system starts	00:00:26
LHSI isolation valve MOV 1890C motor floods (valve inoperable)	00:02:41
MSVH/Aux. Bldg. pipe tunnel opens (penetration sealant dislodges)	00:04:13
Operators stop LHSI Pump A	00:06:17
Operators secure 1 of 3 HHSI pumps	00:15:00
Operators stop LHSI Pump B	00:15:44
Operators isolate LHSI pump suctions (RWST spillage to Safeguards ends)	00:16:18
Accumulators begin discharging	00:28:27
Operators begin cooldown	01:00:00
Accumulators exhausted	01:12:11
Operators secure 2 of 3 HHSI pumps	01:45:00
RWST exhausted, HHSI ends	06:12:--
Water level at TAF	10:15:--
First fuel rod gap release	12:49:06
First hydrogen burn	13:29:28
Release of 1% of core inventory of iodine to environment	13:39:--
Safeguards roof fails grossly (from hydrogen burn)	13:54:20
Reactor lower head fails	18:34:16
Safeguards exhaust ventilation fans trip on low inlet press	No trip
Safeguards exhaust ventilation filters fail on excessive temperature	No failure

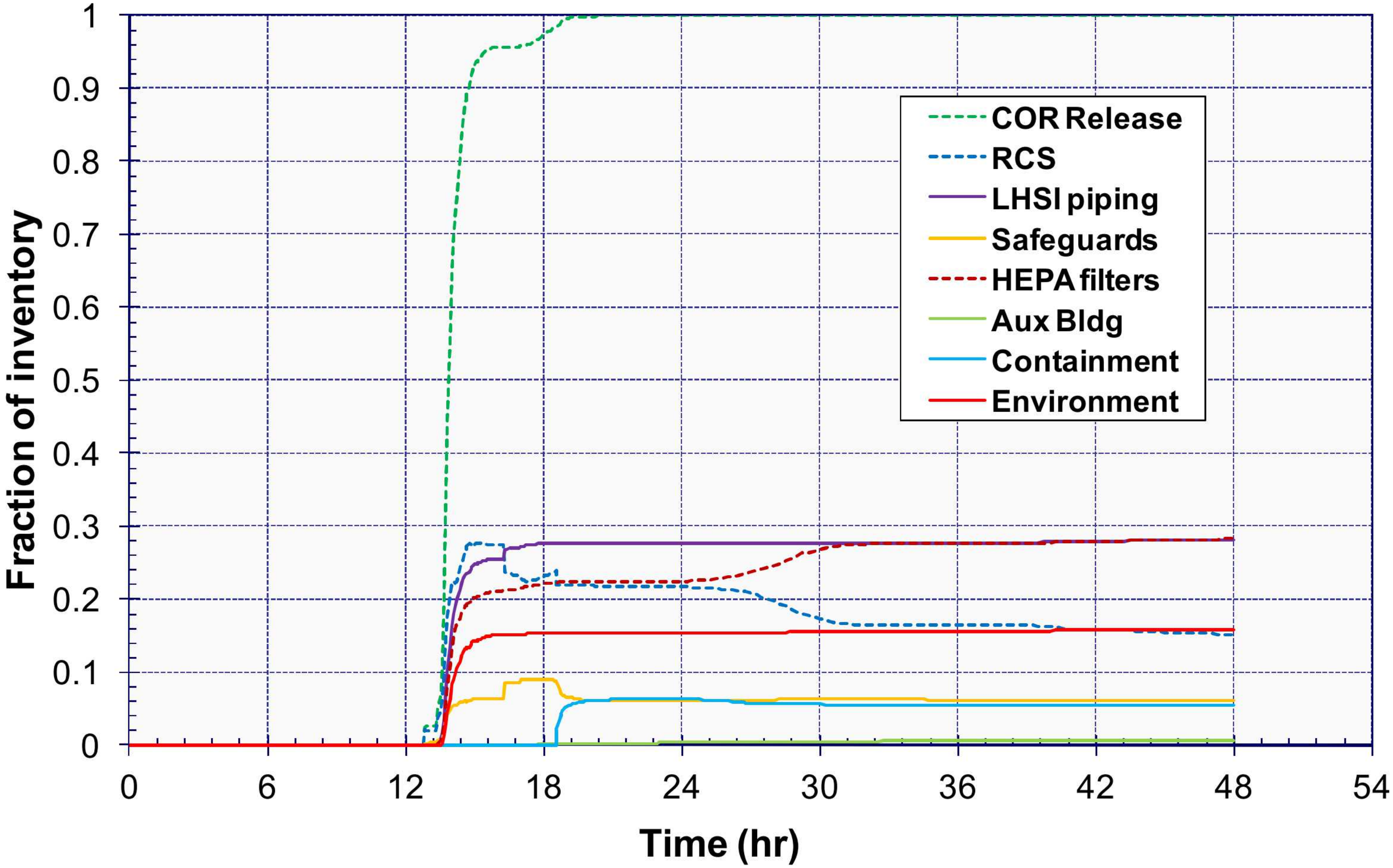
Fission Product Release to the Environment



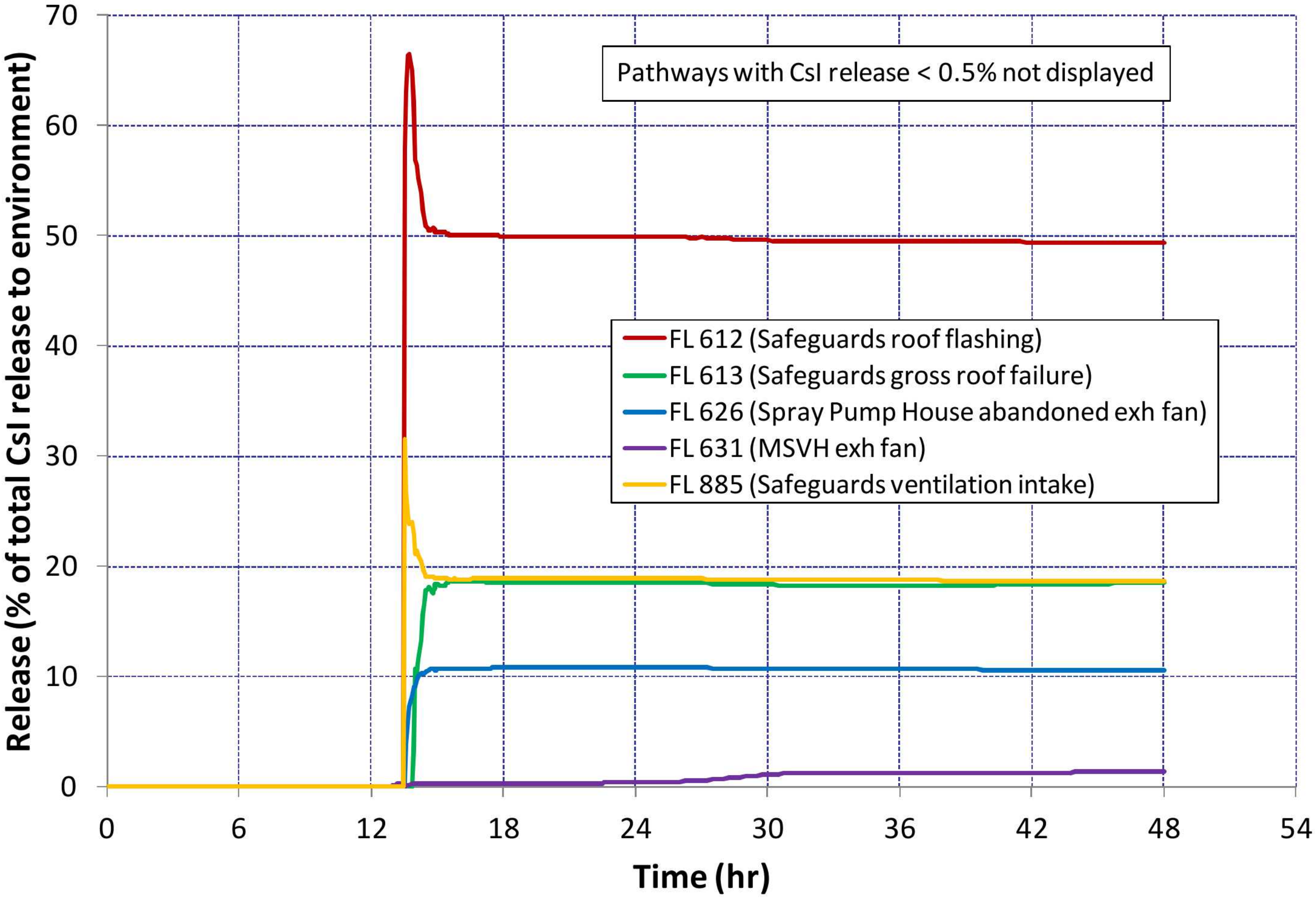
Cesium Distribution



Iodine Distribution



CsI Releases to Environment by Available Pathways

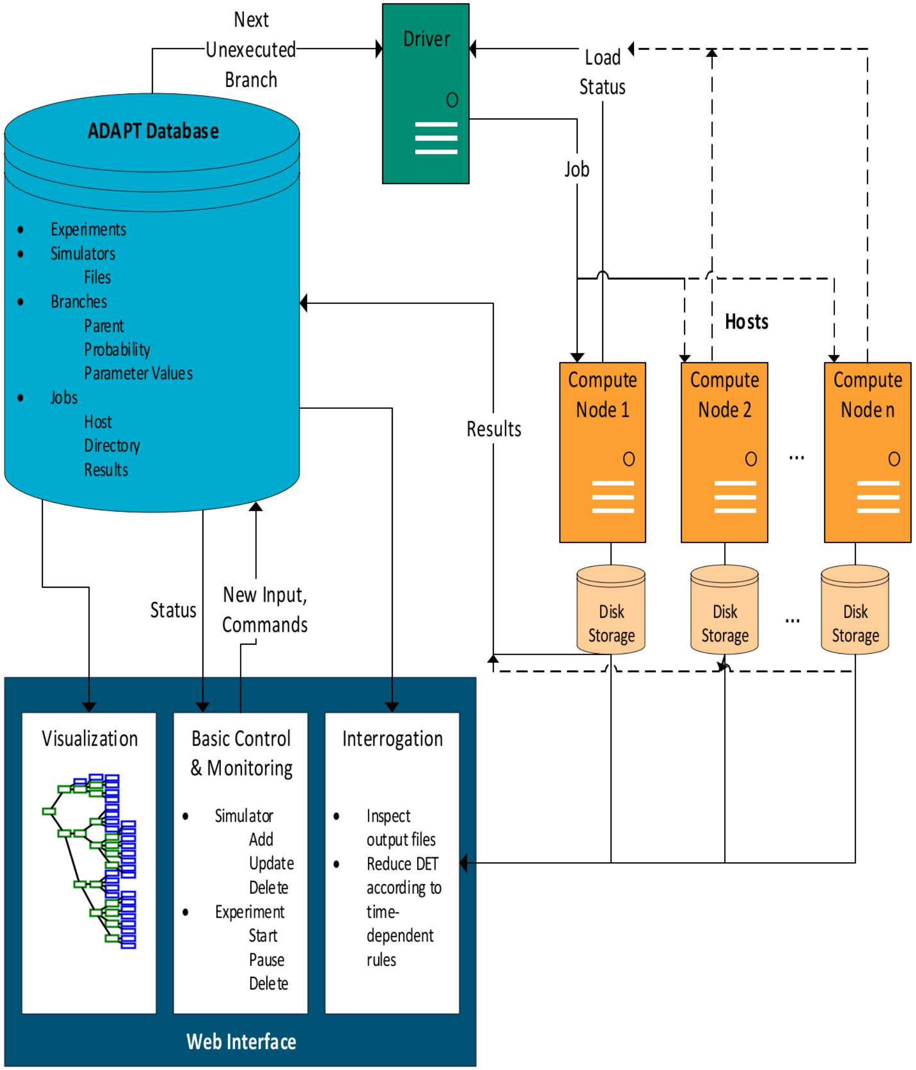


Dynamic Probabilistic Risk Assessment (PRA)

- Traditional PRA requires analysts to assume order of events
  - Does not explicitly account for timing of events
    - Will an event have different effects on incident progression based on its timing?
  - Uncertainties in event ordering increase with incident complexity and time
- Dynamic PRA is driven by time-resolving models of the relevant phenomena
  - Events occur according to physically-meaningful rules
    - E.g., hydrogen igniter success is queried only when a combustible mixture has accumulated
  - Events may re-occur as appropriate (e.g., valve failure query on cycling)
  - Dynamic event trees (DETs) are easily incorporated into a traditional PRA

ADAPT Approach

- DET driver developed for/by SNL (2006-present)
  - Tracks DET database, launches jobs, and presents results
  - Jobs may be run on local machines up to HPCs
  - Supports linking multiple simulator codes
  - Calculates figures of merit using time-dependent output data
- Simulator- and domain-agnostic
  - Simulators must meet a short list of requirements
    - Capable of restarting from saved state with new input
  - Simulator interactions performed via signal files rather than shared memory
    - Traceability
    - Portability over diverse computational hosts



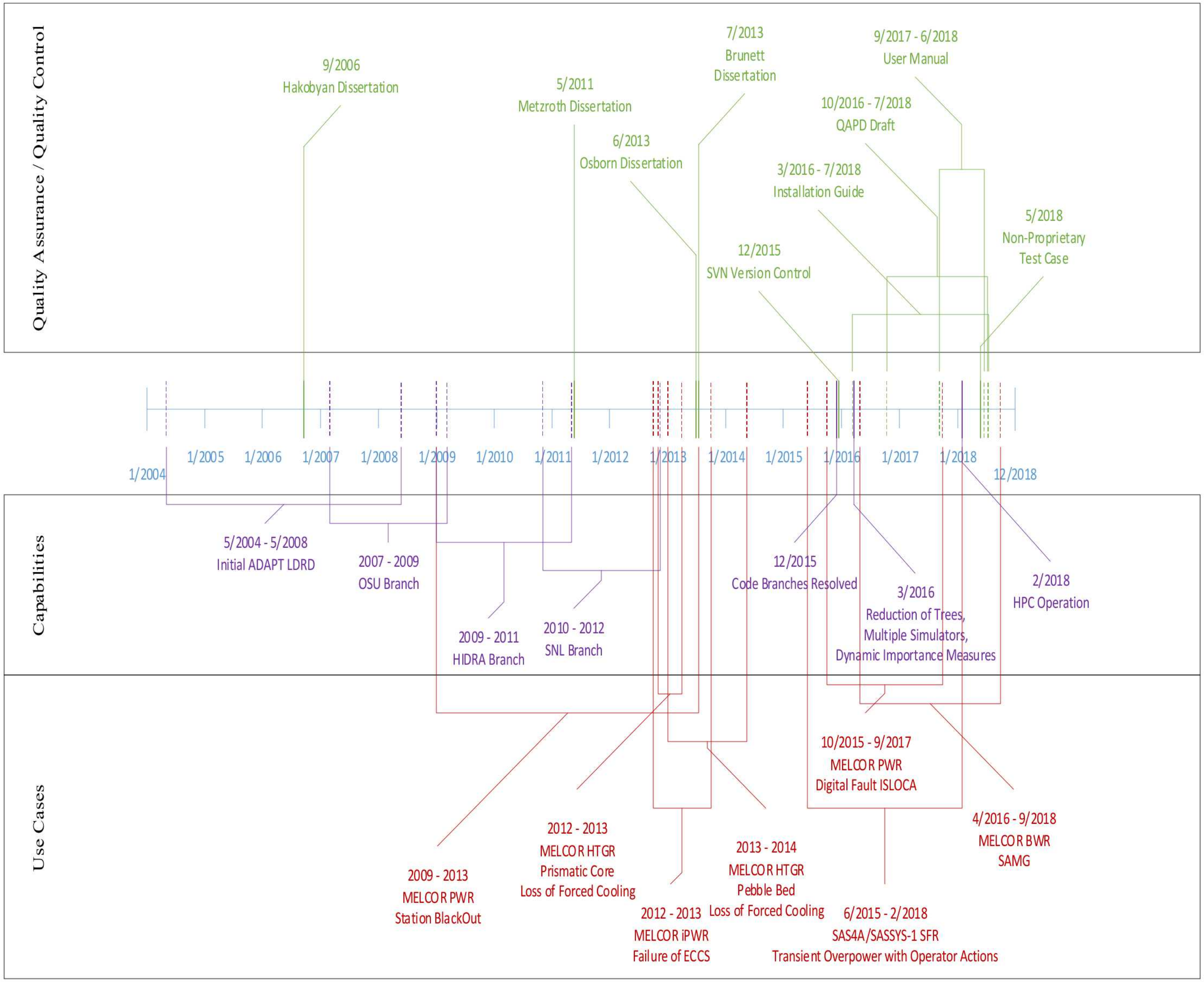
ADAPT Applications

Years	System	Incident	Simulator(s)
2006-2011	PWR	SBO	MELCOR
2009	SFR	Aircraft Crash	RELAP5
2013	PWR	SBO	MELCOR
2013-2014	PWR	SBO	MELCOR
2014	HTGR	LOFC	MELCOR
2015-2017	PWR	SBO	MAAP4
2015-2017	SFR	TOP	SAS4A/SASSYS-1
2015-2018	PWR	ISLOCA	MELCOR, RADTRAD
2015-2018	BWR	SBO	MELCOR
2016-2018	SNF Cask	Derailment	STAGE, RADTRAN

PWR: Pressurized Water Reactor
SFR: Sodium-cooled Fast Reactor
HTGR: High Temperature Gas-cooled Reactor
BWR: Boiling Water Reactor
SNF: Spent Nuclear Fuel

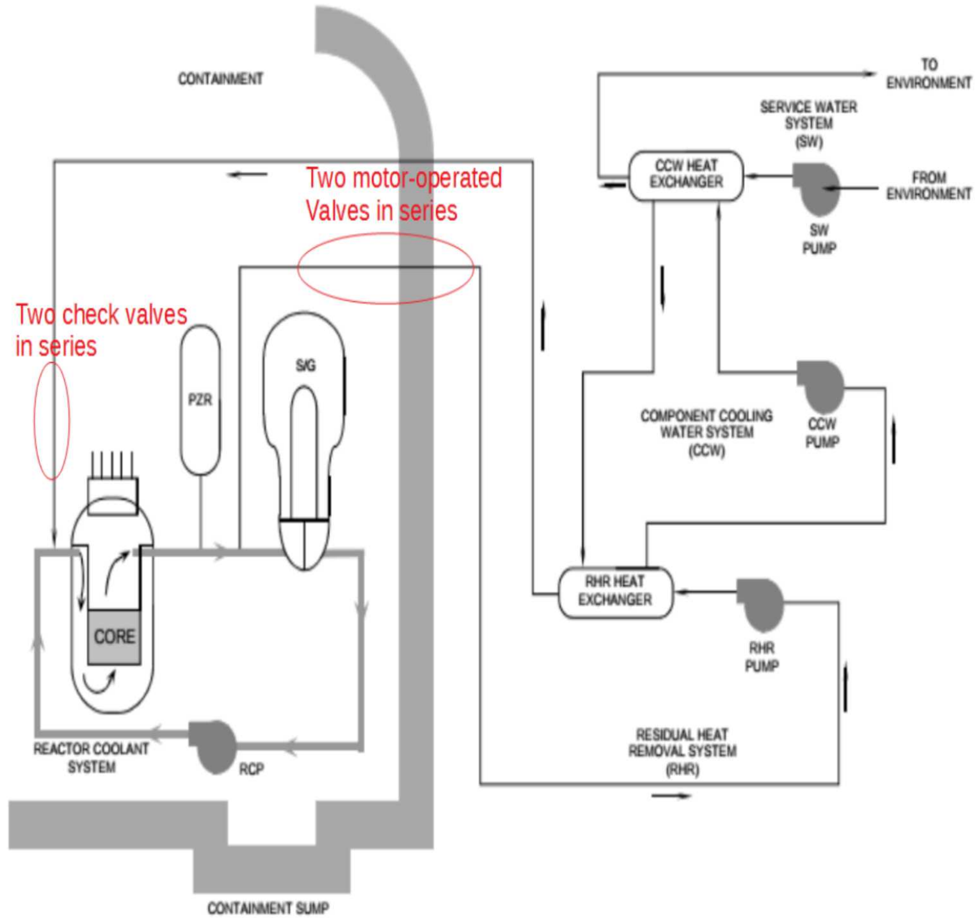
SBO: Station Blackout
LOFC: Loss of Forced Cooling
TOP: Transient Overpower
ISLOCA: Interfacing System Loss of Coolant Accident

ADAPT Timeline



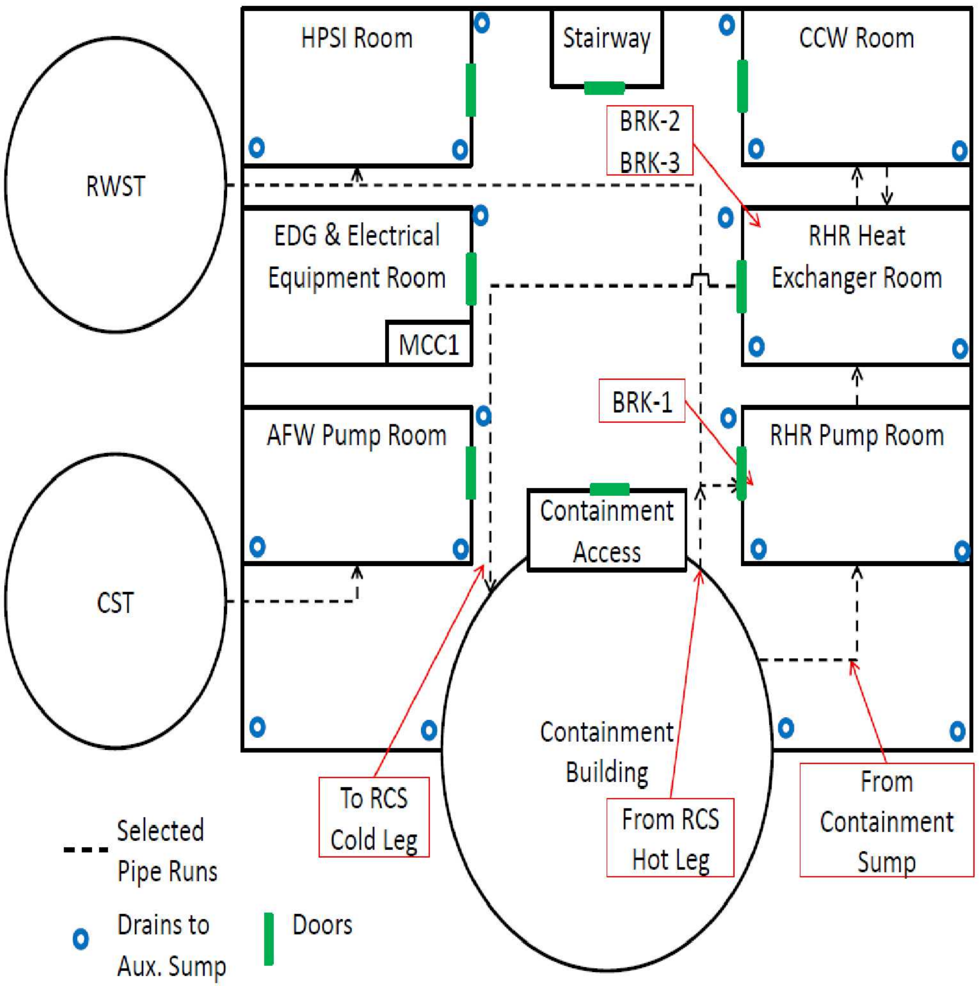
Example Analysis – PWR ISLOCA

- Three loop PWR
- Initiating event: Failure of digital valve controllers
  - Motor-operated valves (MOVs) open inappropriately on residual heat removal (RHR) suction line
  - Challenges integrity of RHR and component cooling water (CCW) components
- Complex, fast-evolving accident
  - Possibility for large breaks
  - Flooding may impede mitigative actions
  - Shared support systems may be damaged



Example Analysis – PWR ISLOCA

- Each break requires manual action in auxiliary building to mitigate
- Suction line break (BRK-1)
  - Flood into auxiliary building
  - Disables a safe shutdown pathway
- Heat exchanger (HX) tube break (BRK-2)
  - Overpressurizes CCW until isolated
- HX shell break (BRK-3)
  - Flood into auxiliary building
  - CCW system break



Example Analysis – PWR ISLOCA

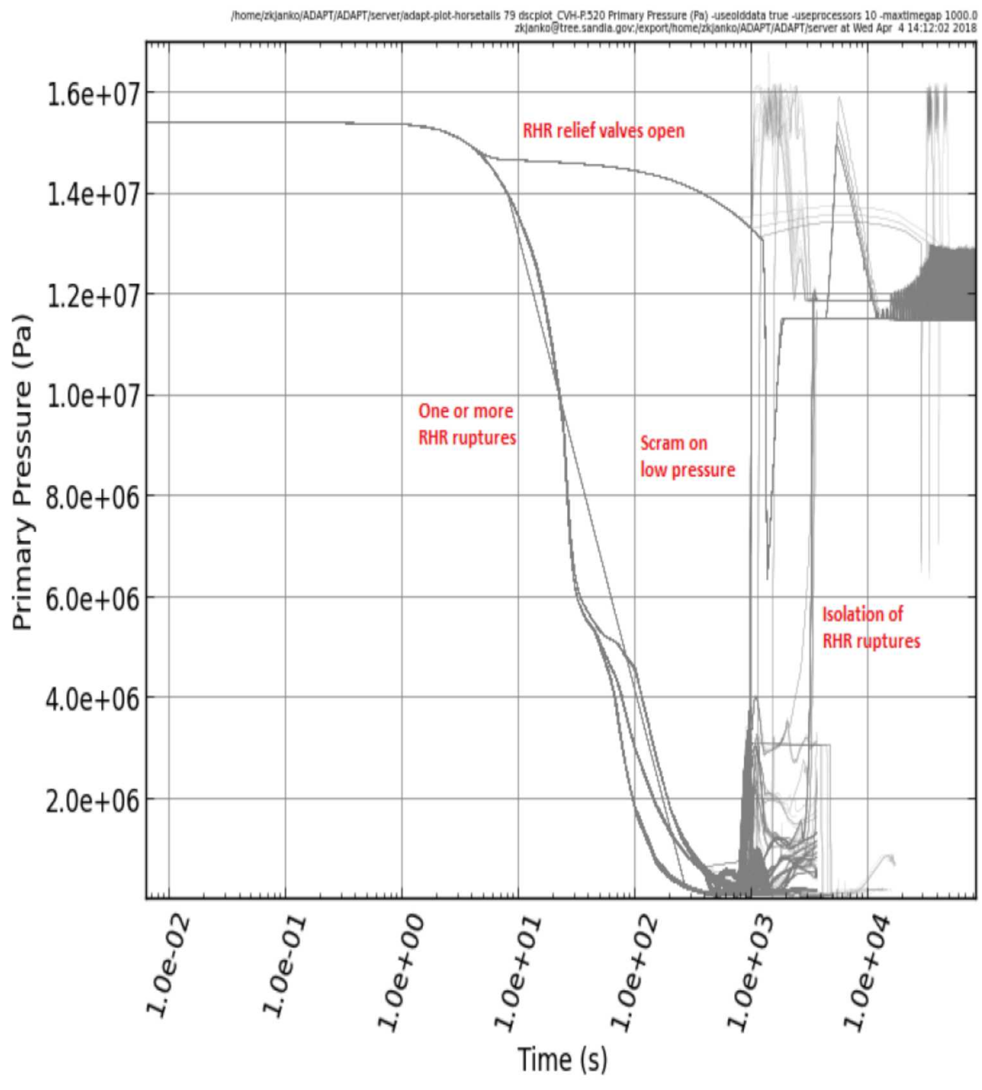
- ADAPT case handles diverse uncertainties
  - RHR component overpressure capacities
  - Operator mitigation action success and timing
    - May be delayed or fail due to flooding or high radiation dose
  - Operator dose tolerance
  - Status of doors in auxiliary building lower level
    - Closed, sealed doors impede flow of water and radionuclides
- Ending conditions
  - 10% of fuel damaged
  - 24 hours of simulation time

Example Analysis – PWR ISLOCA

- Coupled-code analysis
  - MELCOR calculates general accident progression
    - Thermal-hydraulics, core degradation, radionuclide transport
  - RADTRAD calculates doses in auxiliary building
  - Iterative process
    - MELCOR reaches mitigation action completion time
    - MELCOR passes auxiliary building source term to RADTRAD
    - RADTRAD calculates dose rate and integrates over mitigation action completion time
    - RADTRAD passes dose to MELCOR
    - MELCOR continues with either success or failure of action, depending on sampled dose tolerance

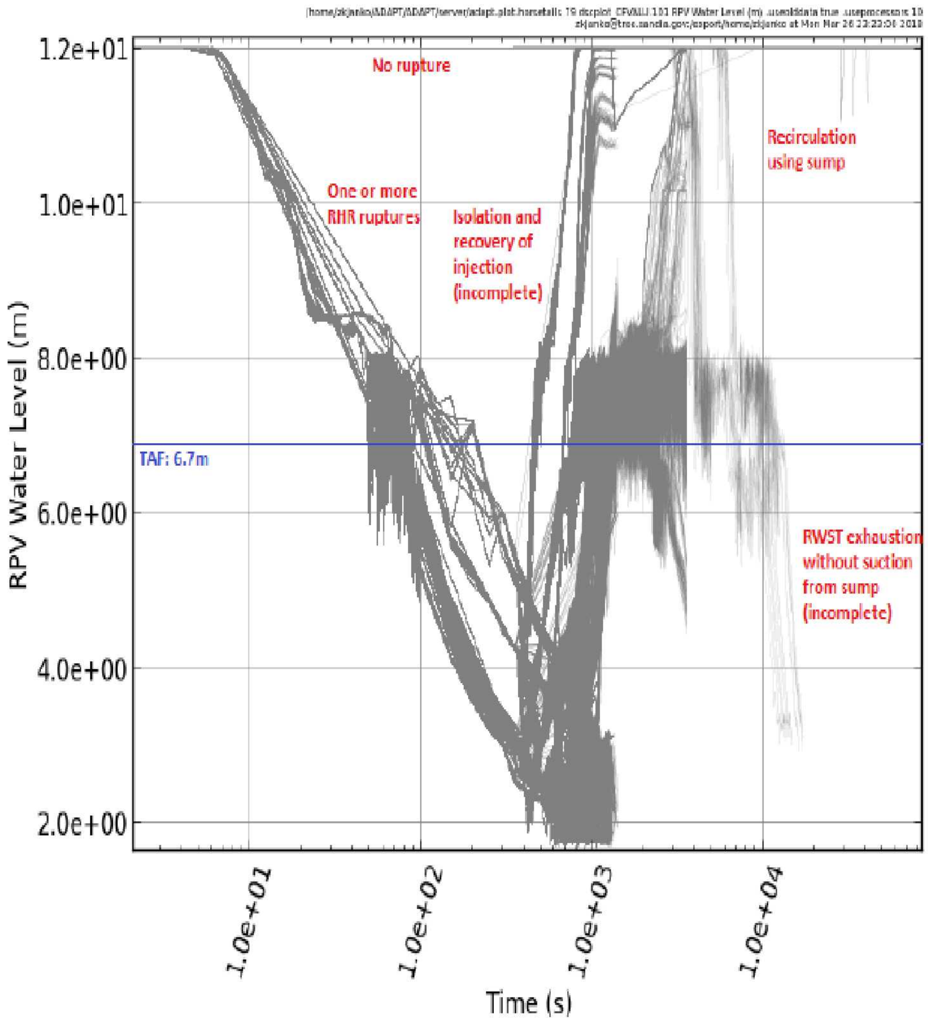
Example Analysis – PWR ISLOCA

- Primary pressure falls quickly with any RHR rupture
- Typical ISLOCA procedures emphasize rapid depressurization into containment
  - Open a pathway from RCS through PORVs
  - Offers preferable alternative to the rupture
  - This ISLOCA outpaces baseline depressurization action timing



Example Analysis – PWR ISLOCA

- Makeup capabilities may be recovered through mitigative actions
  - Isolate leaks
  - Switch to alternative pathways



Example Analysis – PWR ISLOCA

- Isolation activities reduce fuel oxidation and Cs releases
- Interesting results for door status and PORV blowdown
  - Sealed door contains water in a room that may require access for mitigative actions
  - PORV blowdown occurs too late to help in this scenario
    - Already blown down by earliest assumed action time
    - Allows RCS inventory to boil off earlier
    - Allows radionuclides to leave RCS earlier

Branching Parameter	Core Intact Fraction	Hydrogen Generation	Peak Containment Pressure	Environmental Cs Release Fraction
RHR HX Room Door Closed	1.0	0.99	1.0	1.2
RHR Pump Room Door Closed	1.0	3.2	1.0	11
RHR HX Tube Isolation	1.0	0.98	1.0	0.32
RHR HX Shell Isolation	1.0	0.69	1.0	0.56
RWST Isolation	1.0	0.097	1.0	0.074
PORV Blowdown	1.0	3.2	1.0	3.7

Example Analysis – PWR ISLOCA

- Earlier mitigation action timing generally favorable
  - If earliest sampled time not achievable, generally little sensitivity in Cs releases between next two values
- Highest containment pressure peak for middle sampled success timing of RHR HX tube isolation
  - Not necessarily desirable or undesirable
- (… of all

Branching Condition	Value	Hydrogen Generation	Environmental Cs Release Fraction	Peak Containment Pressure
RHR HX Tube Isolation Timing	393.0 s	$1.0 * 10^{-4}$	$1.9 * 10^{-4}$	$2.1 * 10^{-3}$
	608.0 s	$1.3 * 10^{-2}$	$4.9 * 10^{-2}$	$6.4 * 10^{4}$
	1050.0 s	0.16	$8.4 * 10^{-2}$	$3.0 * 10^{-2}$
	10^{20} s	$4.6 * 10^{9}$	$1.4 * 10^{10}$	$2.6 * 10^{-4}$
RHR HX Shell Isolation Timing	393.0 s	$5.6 * 10^{-4}$	$2.3 * 10^{-19}$	$2.2 * 10^{-3}$
	608.0 s	$6.4 * 10^{-2}$	$2.9 * 10^{-3}$	$2.3 * 10^{-2}$
	1050.0 s	$6.9 * 10^{-2}$	$5.4 * 10^{-2}$	$3.2 * 10^{-2}$
	10^{20} s	$1.2 * 10^{10}$	$4.4 * 10^{10}$	$2.4 * 10^{10}$

Summary

- DPRA can give additional insight to complex event progressions
  - What physical parameters are impactful?
  - How does the timing of human interaction affect the outcome?
- ADAPT is a flexible DET generation and analysis platform
  - Diverse physical systems
    - Nuclear power plant safety
    - Shipping cask safety/security/safeguards
    - Limited only by availability of appropriate simulators
  - Easily adaptable to various computational environments
    - No proprietary software dependencies
    - SSH/SCP connections or HPC submission using the Slurm job scheduler
  - Extensible data analysis tools
    - Dynamic importance measure platform
    - Reduction of DETs
    - Scalable from hundreds to 1M+ branches