The workshop brought together experts to gain a better understanding of what are the sources of uncertainty, how they are manifested in the PRA, and what are their potential significance to the PRA model and results. The workshop specifically addressed uncertainties associated with risk assessments for internal fires, seismic events, low power and shutdown conditions, and for the Level 2 portion of a PRA. Four parallel technical sessions were held the first day where the subject matter experts presented their perspectives followed by an open discussion among the experts and the workshop participants. The open discussion focused on delving into the details of each source regarding identifying where was the lack of knowledge, differentiating whether the identified source was actually a lack of knowledge or an assumption or approximation made to simplify the PRA model, understanding if the source was dependent on how the PRA results were being used or independent of the PRA use, the basis for the significance of the source of uncertainty, whether the possibility existed to develop a consensus model, and what work (e.g., research, develop guidance) could be done to improve our knowledge and thereby reduce the impact of the source. Moreover, each source was also considered with regard to operating light water reactors (LWRs) and new advanced LWRs. A summary of the key issues discussed for each technical topic is provided below.

1. **Fire Summary –**
    a. Plant Partitioning
        i. Fire partition is not rated and not getting credit
        ii. Assuming failure rates are the same as for a rated barrier
    b. Equipment selection
        i. Example: release of radioactive water and it had a potential impact on operator action
        ii. There is a generic list of Multiple Spurious Operations (MSOs), and maybe this type of impact should be added and considered in the expert panel process
    c. Fire scenario selection, most of the model uncertainty is here
        i. Worry about incomplete scenario development. Potentially have results that don't reflect the level of detail that you need
        ii. Control room abandonment don't do a very detailed assessment of taking control from the safe shutdown panel
            1. Spread and control model for fires in control rooms need to be verified.
        iii. There may be a time delay after the equipment fire target damage
            1. Need to develop method to account for delay.
            2. Capability Category II does not include this. Standard committee is going to consider this in light of delay.
        iv. Fire growth time can vary and may not be independent of heat release. Not doing a very good job of correlation between the parameters.
            1. Correlation between heat release keep rate and growth needs to be looked at.
            2. Appendix E of NUREG/CR-6850 needs to be looked at, verified and updated. Establishes severity factor curves.
        v. Severity factor are based on conservative estimates (essentially failure of first target). New models are out there and being verified, but need some additional models.
        vi. Damage criteria is available as temperature criteria for cables. Significance of damage criteria is variable.

      vii. Fire wrap degradation needs to be identified.
- 1. Impact on scenario is large, the failure rate is low thus low impact on overall results.

     viii. NUREG-1824 (verification of tools) cautions that tool may not be appropriate. There is no fire model perfect for any particular situation and the result can be substantially different. Much higher uncertainty than thermal-hydraulic.

      ix. Choosing between two models or deciding which is most appropriate. Even the best available tool is going to probably fall short.

       x. NUREG/CR-6850 may be incorrect that fire spreads from cabinet to cabinet based on GE-Hitachi data.

      xi. Smoke damage has high uncertainty on what role it plays in core damage. Smoke damage may not be important, but recent evidence indicates smoke may have impacted a breaker (longer term impact).
- 1. No model to predict smoke damage to other systems
- 2. Not sure it is really a phenomenological problem, so low

     xii. Adjusting fire brigade manual suppression times. What is the time that should be credited in the suppression curve.
- 1. The Fire model is reflecting an unrealistic damage zone which is an effect of manual suppression versus damage control.
- 2. Negative impact of fire brigade (deenergize the wrong equipment, spray the wrong material, maybe more like a completeness issue). Error of commission.

     xiii. Analyst has choice in how to setup model and physical layouts
- 1. Example: is the cable modeled as a hunk of copper, or steel, etc.

d. Ignition frequencies

e. Circuit failure (CF) Likelihood
- i. CF probabilities (looks like parameter uncertainty) which is a state-of-knowledge uncertainty.
  - 1. This one may belong more as a parameter.
  - 2. This would be an apparent place to do research.
  - 3. Deriving the parameters is based on model (like the expert panel) and therefore the parameter may be influenced by this sub-model uncertainty.
  - 4. NUREG-1855 needs a strict definition of model uncertainty (limiting consensus models that will ultimately be manifested as a parameter uncertainty).

f. Human Reliability Analysis (HRA)
- i. Time-lines for human error probabilities (HEPs) (related to actions taken with respect to fire)
  - 1. No simulator runs.
  - 2. Better guidance to establish time-lines for these types of action. Recovery actions outside control room. No good model for abandonment scenarios and ultimate shutdown.

g. Seismic fire interactions -- new area
- i. Example: turbine blade ejection leading to fire and flooding
  - 1. Should analyze as an initiator in fire or flood area of PRA.

h. Shutdown fire could be even higher

2. **Seismic – Seismic PRA consists of three main parts:** Hazard Analysis, Fragility Analysis, Plant Response Model
    a. Hazard analysis (three items corresponding to High Level Requirements (HLRs) from ASME/ANS PRA Standard).
        i. Seismic Source Characterization
            1. What models can be used to categorize the earthquake source.
            2. Medium uncertainty, there is some progress in this area in the referenced reports.
            3. Reports help resolve this area, and update the previous source geometry and earthquake recurrence.
        ii. Ground motion characterization
            1. High uncertainty.
            2. Probably the largest uncertainty in the hazards analysis.
            3. The Ground Motion Prediction Equation models have high uncertainty associate with them.
            4. There is some development underway for Next Generation Attenuation for central and eastern North America.
            5. New model won't necessarily reduce the uncertainty until the data on earthquakes is updated. Resolution here means getting a handle on the uncertainty. This is really treated as a parameter uncertainty in the PRA because ground motion characterization is an input to the PRA in the form of parameter distributions.
            6. NUREG-1855 might be missing discussion on how we disposition issues that have huge uncertainty in risk-informed decision making.
        iii. Site response
            1. Role of hazard and fragility analysis.
            2. High uncertainty for soil sites, low for rock sites.
            3. Data lacking for sites, techniques to evaluate site response is not standardized).
            4. Need to standardize techniques across plants for site response. Need guidance to characterize hazard and its uncertainty. Have a uniform approach to characterizing uncertainty.
    b. Fragility analysis (four items corresponding to HLRs)
        i. Soil-structure interaction
            1. High uncertainty significance.
        ii. Conservative assumption of structural failures
            1. Biases of PRA results.
            2. Medium uncertainty.
            3. Done to make analysis more efficient, but may mask contributions of fragility of one structure, system and component (SSC) with regard to another.
            4. More detailed but costly PRA could resolve this.

   iii. Inadequate fragility test data
     1. Test data is important in obtaining plant specific fragility. Extrapolation models may be source of model uncertainty.
     2. Medium uncertainty
     3. Fragility testing is rarely done, but more testing at high cost could be done.
     4. This is example of sub-model uncertainty manifested as parameter uncertainty in seismic PRA.
   iv. Plant-specific loss of offsite power fragility
     1. Loss of offsite power fragility is significant.
     2. Medium uncertainty.
     3. Better plant specific analysis might remove unneeded conservatism in estimate.
 c. Plant response model
   i. Treatment of human error under seismic conditions
     1. Approach is crude.
     2. High uncertainty.
     3. A few actions can have a large impact on seismic PRA.
     4. Improve human failure rate model in seismic conditions. Maybe adapt fire HRA model with different stresses.
   ii. Items of Unknown Significance (Medium uncertainty)
     1. Seismic induced fire and flood
       a. They are qualitative and their significance is unknown.

3. Low Power and Shutdown (LPSD) –
 a. Many of the sources of uncertainty were screened out because it wasn't a model uncertainty but was refinement related, i.e., level of detail issues. The uncertainty could be reduced by increasing the level of detail in the LPSD PRA.
 b. Experience in LPSD is far less than at-power and as a community we are lacking in expertise. Lacking detailed regulatory review in this area.
 c. Need useful guidance to address when we should consider the level of discretization for definition of initiating events and accident sequences.
   i. Accumulate the experience analysts have had.
   ii. No community good practice.
 d. Completeness issue
   i. Examples given in presentations.
 e. Refinement or completeness
   i. Assumption of equipment failing at time of demand. Analyze all failures to occur at t=0. Could different ordering of scenarios lead to other likely scenarios?
   ii. Since many LPSD cutsets involve multiple HEPs, the specific ordering of the top events in the event tree can lead to different scenarios.

f. Parameter uncertainties
   i. Are data applicable and complete for accident precursor?
   ii. Duration of plant operating states (POSs).
   iii. Advantage to have centralized database for LPSD.
   iv. Many precursors that lead to a loss and/or interruption of residual heat removal (RHR) may not reportable be under 10CFR50.72 and 10CFR50.73 (e.g., loss of reactor coolant system inventory that is isolated before the RHR function is lost or interrupted). Thus, there are completeness issues for shutdown precursor data.
g. Model or completeness
   i. Minimum mission time of 24 hours.
   ii. Medium uncertainty.
   iii. What is the SSS and how do you get there?
h. Lack of experience
   i. Medium uncertainty.
   ii. Lack of community experience at different level of detail and the lack of detail reviews, particularly from the regulator. However, the regulator performs detailed risk reviews of shutdown performance deficiencies identified under the reactor oversight process. NRO reviews quantified shutdown PRAs that cover internal events, shutdown fires, shutdown floods, shutdown high winds, and seismic margins.
   iii. Limited cases run for LPSD compared to at-power, are codes applicable, large early release frequency (LERF). Applicability of thermal hydraulic codes for certain reactor coolant system configurations is an issue.
   iv. LERF is evaluated in Reactor Oversight Project for LPSD using NUREG/CR-6595, Rev 1, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events' ."
i. HRA
   i. High uncertainty.
   ii. LPSD does not have a lot of automatic responses, relies on operator responses to initiating events. HRA methods may not apply to all POSs.
   iii. High variability from HRA analyst-to-analyst
   iv. Treatment of Dependencies between human failure events
      1. Some of the LPSD scenarios are a much longer time.
      2. Can we construct a lower limit on a joint HEP for a sequence? Same question applies to at-power.
      3. Ongoing NRC/EPRI project to improve HRA modeling using a caused based approach.
   v. Need criteria for feasibility of operator actions
      1. Example: Temperature has to be less than boiling in the vessel.
      2. Likely variable treatment from analyst-to-analyst.
      3. Regarding the feasibility for operator actions especially outside the control room - such as containment closure before reactor coolant system boiling. Feasibility depends on environmental factors and logistic factors such as steam, noise, high radiation, fog, accessibility of SSCs (need scaffolding?), adequacy of pre-staging, etc.

j. Conclusion:
   i. Level of detail prevalent in concerns in LPSD standard.
   ii. Matter of how much resources putting in to address them.
   iii. Reflection of the immaturity of the LPSD PRA. The methods are similar to at-power, but do not know how far to go to get a realistic analysis.
   iv. International review of LPSD might lend some of that regulatory review.
   v. A realistic model has to reflect the tear-down and reassembly of each plant.
   vi. Same techniques for at-power will probably roll over to LPSD.
   vii. Model structure on hot standby and low power need to be sorted out before really addressing uncertainty.
   viii. Fire modeling at shutdown is difficult for "on average." When fire and LPSD come together, approximations are major.
   ix. Typical outages are not consistent across all plants. Operating states will not match.

4. **Level 2 –**
   a. Scope:
      i. Full Level 2 (not just LERF)
      ii. Considered accident management strategy development.
      iii. Limited to at-power conditions.
      iv. Attempted to account for new reactor designs.
   b. Level 1-2 interface
      i. Loss of information on plant damage state (PDS) (not necessarily model uncertainty, just impact results)
         1. Is the selection process conservative or non-conservative bias?
         2. High uncertainty.
         3. Level of detail issue, and not necessarily agreed that it's model uncertainty.
         4. Hard to check sensitivity issues without redoing a significant portion of the PRA.
         5. Concern would be addressed by a properly performed peer review
         6. Damage states experience and attributes are well resolved for operating reactors. Attributes in defining PDSs are more uncertain for new reactors due to lack of knowledge.

      ii. Partial or degraded performance not credited in Level 1
         1. High uncertainty.
         2. Need better understanding of partial injection model (below Level 1 success criteria).
         3. May influence Level 1 data collection.
   c. Containment capacity
      i. Methods for containment response from dynamic loads
         1. High model uncertainty.
         2. Is there an improved tool to analyze containment and establish containment failure probabilities?

      ii. Containment capacity
1. Seismic induced leakage that could impact the accident progression.
2. Medium uncertainty.
3. Should be able to factor this into seismic PRA.

     iii. Quasi-steady failure threshold correlation for failure pressure and leak rate
1. Medium uncertainty.
2. Leak before break assumption could generate different fission product release in Level 2.
3. Perform sensitivity study. Impact Level 3 analysis.

     iv. Aging impacts on threshold failure
1. Low based on NUREG/CR-6920.

d. Severe accident progression (most uncertainties here)

      i. Recovery of degraded core.
1. Do we trust MAAP and MELCOR CD recovery? Recriticality could be an issue. These are impact accident management strategies.
2. High uncertainty.
3. Need focused sensitivity studies.

     ii. Onset of fuel relocation
1. Affect source terms and hydrogen production.
2. Medium uncertainty.
3. Good benchmarking in codes.
4. Looking at SOARCA uncertainty analysis.

     iii. Treatment of natural circulation and how this impacts loop seal clearing.

     iv. Thermally induced mechanisms inducing failure of reactor coolant system (examples in presentations).
1. High uncertainty.
2. Sensitivity studies could be important for new reactor designs.

      v. Lower head failure modes and ex-vessel cooling
1. High uncertainty.
2. Assumptions, or what the deterministic codes predict, really impact what the accident sequence progression is.
3. Sensitivity study would be helpful. Need more experimental work.

     vi. Energetic containment challenges
1. Ability to mechanistically treat energetic structure/SSC challenges.
2. Longer term response to these failure modes and what other impacts exist on SSCs.
3. High uncertainty.

     vii. Core concrete interactions
1. Late releases and land contamination could be important.

e. Probabilistic treatment
   i. Similar issues exist in LPSD.
   ii. Generic source of model uncertainty.
   iii. Equipment and instrument survivability for beyond core damage events
      1. What can be credited.
      2. Data for random failure under degraded conditions not covered.
      3. Conservative is not credit things that lack data.
      4. More realistic risk profile needs to consider these in Level 2 model development.
      5. High uncertainty.
   iv. Model-parameter and phenomenological correlation
      1. Random selection of parameters should not reflect unrealistic combinations of parameters.
      2. Medium.
   v. Passive system reliability
      1. High uncertainty.
      2. Could be focused research and experiments.
f. Source term analysis
   i. Release groups could be more important for long term health effects
   ii. Release models from Capability Category I (of ANS Level 2 PRA Standard).
      1. High uncertainty.
   iii. All models base fission product retention in primary and long term evolution in containment and aux building.
      1. Truncate runs at certain times after failure.
      2. Not always at safe stable state.
      3. Issue of when you truncate the sequence run can be a high source of model uncertainty.