US NRC Probabilistic Risk Assessment Confirmatory Success Criteria Analysis Using the Duane Arnold Site

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Acronyms

- ac – alternating current
- ADS – Automatic Depressurization System
- ATWS – Anticipated Transient Without SCRAM
- BWR – Boiling Water Reactor
- CRDHS – Control Rod Drive Hydraulic System
- DAEC – Duane Arnold Energy Center
- ECCS – Emergency Core Cooling System
- EDG – Emergency Diesel Generator
- ESF – Engineered Safeguard Features
- FSG – FLEX Support Guideline
- HPCI – High-pressure Coolant Injection
- LOOP – Loss-of-offsite power
- MELCOR – not an acronym
- MSIV – Main Steam Isolation Valves
- NRC – Nuclear Regulatory Commission
- NSSS – Nuclear Steam Supply System
- PRA – Probabilistic Risk Assessment
- RCIC – Reactor Core Isolation Cooling
- RHR – Residual Heat Removal System
- RPS – Reactor Protection System
- RPV – Reactor Pressure Vessel
- SDP – Significance Determination Process
- SPAR – Standardized Plant Analysis Risk models
- SRV – Safety Relief Valve
- TRACE – TRAC/RELAP5 Advanced Computational Engine

Purpose and desired outcome

- Purpose:
  - To discuss substantive aspects of work that the US NRC has initiated in the area of confirmatory success criteria analysis, as part of its normal activities to refine and improve its independent Standardized Plant Analysis Risk (SPAR) models using the Duane Arnold Energy Center (DAEC) as the reference plant;
  - To provide an opportunity for the NRC to describe the plans for this project, and for NextEra and other interested parties to provide feedback;
  - To communicate that this activity is not directly associated with an existing regulatory issue or decision for DAEC.
- Desired outcome:
  - Public awareness of the project;
  - Input on how the work should be framed and executed;
  - Identification of any related activities or interactions that might not be on the team's radar.

Meeting outline

- Introductions
- Background on this project
  - ❖ Opportunity for questions and feedback
- More details about what we’re doing and where we’re heading
  - ❖ Opportunity for questions and feedback
- Discuss the 1st and 2nd topics to be investigated
  - ❖ Opportunity for questions and feedback
- Discuss the 3rd and 4th topics to be investigated
  - ❖ Opportunity for questions and feedback
- Wrap-up

Background on this project

- Objectives:
  - To support confirmation/changes to the independent SPAR models
  - To provide off-the-shelf analyses for NRC risk analysts to consult
  - To foster in-house expertise and knowledge transfer
- Led by the Office of Nuclear Regulatory Research at the request of the Office of Nuclear Reactor Regulation
  – Contractor involvement in the MELCOR model development and the SPAR model implementation
- To support NRC risk analysts, but not part of a specific regulatory action
- PRA scope:
  - Level 1 – i.e., beyond-design-basis scenarios up to the time of core damage
  - Equipment/system performance requirements to support system analysis (i.e., fault tree modeling) and sequence timing to support human reliability analysis
    - Does not deal directly with issues related to reliability modeling (e.g., common cause failure, human performance) – these are covered under other NRC projects

Background on this project (2)

- Past work:
  - NUREG-1953 – Selected issues for 3-loop Westinghouse and General Electric BWR Mark I plants
  - NUREG/CR-7177 – Selected success criteria modeling issues (e.g., core damage surrogate selection, effect of specific modeling assumptions)
  - NUREG-2187 – Selected issues for 4-loop Westinghouse plants with large, dry containments
- The above work has both confirmed existing SPAR modeling assumptions and supported specific SPAR modeling changes

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Questions or feedback at this point?

Plans for project scope

- The current DAEC SPAR model includes Level 1 at-power internal event, internal flood, internal fire, tornado, and seismic initiators;
- The approach here is to:
  - perform plant-specific thermal-hydraulic (MELCOR) analysis for Duane Arnold,
  - apply the findings to the Duane Arnold SPAR model, and
  - extend the insights of the analyses to other plants' SPAR models (leveraging expertise at NRC and Idaho National Labs), considering the important differences in design and operation that exist between the plants.
- The project will rely on the best available information in developing the MELCOR thermal-hydraulic model and performing the thermal-hydraulic analysis, including:
  - Licensing information previously submitted to the NRC (e.g., the Updated Final Safety Analysis Report; Order EA-12-049 and Order EA-13-109 Overall Integrated Plan and Updates);
  - Generic BWR technology design and operation information;
  - Information provided by the plant voluntarily for the conduct of this project.

High-level issues to be investigated

- Intent is not to comprehensively confirm all success criteria in the chosen plant's SPAR model:
  - Focus on issues of known importance:
    - Issues that were central to past risk analyses
    - Issues that are expected to be central to future risk analyses
    - Changes in plant design and operation
  - Develop a basis for confirming or modifying that issue's treatment in a set of SPAR models.
- Since the focus is on issues important to event and conditions assessment, the issues may or may not be important to the estimated baseline plant risk.
- Four specific topics:
  1. Success criteria for depressurization during non-ATWS scenarios
  2. FLEX Support Guidelines (FSGs) applied to loss-of-ac-power and other scenarios
  3. ECCS injection following containment failure
  4. Safe and stable end-state considerations

MELCOR model development

- MELCOR is a “systems level” thermal-hydraulic and severe accident modeling code developed by Sandia National Labs under NRC sponsorship (melcor.sandia.gov)
- A MELCOR model is similar in some respects to the software used to support NPP simulator functionality, except that in the case of typical MELCOR models there is much more capability in modeling the response after fuel heatup and less capability with respect to modeling support systems, normal operation, and the human-machine interface
- The model will be developed based on:
  - Existing MELCOR and TRACE models for similar plants
  - The aforementioned generic and plant-specific BWR information
- Modeled plant features will include:
  - Structures: primary and secondary containment, condensate storage tank
  - Systems: NSSS, RCIC, HPCI, CRDHS, RHR, ADS, RPS, and ESF
  - Components: SRVs, MSIVs, flow restrictors, vessel internals
  - Phenomena: drywell leakage, suppression pool heatup and thermal stratification
  - Operator actions: manual RPV depressurization, containment venting
- Shakedown, and limited validation, will be performed
  - For example, comparison against generic BWR or plant-specific Order EA-12-049 (mitigating strategies) and Order EA-13-109 (hardened containment venting system) analysis

Plant selection

- NRC's risk analysts were canvassed for input on issues relative to various NPP design types
  - Recall that recent work has focused on 4-loop Westinghouse plants
- Within the BWR/3 and BWR/4 design classes, the suite of plants were considered with regard to:
  - Thermal power level
  - SPAR internal events station blackout contribution
  - SPAR model scope
  - Design and operational considerations
    - For example, similarity of cross-tying capabilities, number of trains of emergency power
  - Utility activeness in risk-informed activities
- Ultimately, DAEC was chosen as a plant that provided a good balance of the various considerations
- NRC approached NextEra (owner/operator of DAEC) about voluntarily supporting the project, and NextEra/DAEC has agreed

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Questions or feedback at this point?

Non-ATWS SRV/ADS analysis

- Topic to be explored:
  - The success criteria for automatic depressurization during non-ATWS PRA sequences (e.g., minimum # of SRVs required) has been a point of departure between NRC and licensee PRA models during a couple of recent SDPs
  - These criteria factor into most internal events initiators, for sequences such as:
    - Successful reactor shutdown, 4160V ac power available, SRVs successfully reseat (if demanded), high-pressure injection fails, manual reactor depressurization fails, and core damage occurs
- Example modeling assumptions of interest:
  - Number of SRVs that actuate
  - SRV discharge path characteristics that affect flow rates and depressurization
  - Credit for CRDHS flow prior to and following depressurization
  - Source, and achieved flow, of low-pressure injection
  - Manual actions taken prior to depressurization to stabilize pressure below the relevant set-point(s)
  - Automatic, as opposed to manual, initiation of ADS (i.e., failure to inhibit early automatic actuation)
  - Timing of manual actuation (and associated uncertainty in the thermal-hydraulic model's estimation of the water level cue)
  - Delayed closure of MSIVs and manual reactor trip (for relevant initiators)

Non-ATWS SRV/ADS analysis (2)

- Preliminary sequences of interest:
  - A transient sequence with success of high-pressure injection, depressurization, and low-pressure injection (among other successes and failures), where core damage does not occur
  - A transient sequence with failure of high-pressure injection, failure of depressurization, and failure of CRDHS (among other successes and failures), where core damage occurs
  - A small loss-of-coolant accident sequence with failure of high-pressure injection, and success of depressurization and low-pressure injection (among other successes and failures), where core damage does not occur
- A handful of variations (based on the list on the preceding slide) would be run on each sequence.

FLEX Support Guideline strategies for loss-of-ac power and other scenarios

- Topic to be explored:
  - New capabilities are being put in to place under Order EA-12-049 and Order EA-13-109, relating to mitigating strategies and hardened containment venting in response to Fukushima lessons-learned
  - Other NRC activities address regulatory licensing and oversight aspects of these new capabilities, and other NRC activities address when and how these capabilities will be considered in risk-informed activities
  - **The current activity is focused solely on scoping success criteria and sequence timing issues that may be informative for future NRC risk modeling**
  - NRC is aware that some licensees are crediting this equipment in some specific risk-informed licensee activities, and welcomes input on success criteria-related challenges that have been identified
- Example modeling assumptions of interest:
  - Time of loss-of-ac (e.g., diesel generator failure-to-run) and time of battery depletion
  - Time of RCIC failure and effect of the mode of failure (e.g., flow terminates versus fails with flow as-is)
  - Estimate of suppression pool temperature (including RCIC net positive suction head and bearing temperature effects)
  - Number of relief valves actuating during depressurization and timing of action (also relates to earlier topic)
  - Flow rate achieved by ac-independent injection, and timing of injection
  - Timing and containment venting or failure (also relates to later topic)
  - Effect of containment venting/failure on late injection (also relates to later topic)

FLEX Support Guideline strategies for loss-of-ac power and other scenarios (2)

- Preliminary sequences of interest:
  - A grid-related LOOP with failure of EDGs, success of RCIC prior to battery depletion, failure to align a portable diesel generator to supply power to station battery chargers, successful manual depressurization, successful alternate injection using a FLEX pump, successful containment venting, and successful late injection (amongst other successes and failures), where core damage is averted
  - A loss of main feedwater with success of high-pressure injection, failure of suppression pool cooling, success of manual depressurization, failure of CRDHS and low-pressure injection, success of alternate low-pressure injection using a FLEX pump, successful containment venting, and successful late injection (amongst other successes and failures), where core damage is averted
- As before, a handful of variations (based on the list on the preceding slide) would be run on each sequence.

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Questions or feedback at this point?

ECCS injection following containment failure

- Topic to be explored:
  - RPV coolant injection following containment venting or containment failure caused by the slow over-pressurization of containment, resulting from a loss of containment heat removal (historically known as “TW” sequences”)
    - Many licensee BWR PRAs credit such injection
  - Some potential complications affect the degree to which credit is warranted:
    - High containment pressure (prior to venting/failure) impacts on back-pressure of turbine-driven pumps and ADS valves
    - Equipment survivability if containment failure or venting discharges to the reactor building
    - Suppression pool boiling following rapid depressurization, and subsequent degradation of ECCS pumps drawing from the suppression pool
    - Direct damage to ECCS injection or suction lines caused by containment failure

ECCS injection following containment failure (2)

- Example modeling assumptions of interest:
  - The leakage path from primary containment to the reactor building or environment
  - The extent of “normal leakage” or containment isolation impairment at the time of the initiator and resulting containment isolation signal
  - The timing (and associated pressure) of venting
  - The vent path used
  - At what point the vent path is closed
  - The response of the SRVs and ECCS pumps to the elevated pressure and the depressurization
- The current thought is to leverage the sequences already used for the FSG investigation above.

Safe and stable end-state

- Topic to be explored:
  - Typically, PRAs have used a 24-hour mission time for component reliability and sequence truncation; sequences are sometimes carried beyond 24 hours if a cliff-edge is known to exist
  - The ASME/ANS PRA Standard (Requirement AS-A2) states, “For each modeled initiating event, IDENTIFY the key safety functions that are necessary to reach a safe, stable state and prevent core damage”
  - The point here is to scope what additional success criteria-related requirements would be needed to extend typical PRA sequences to a longer period of time (e.g., 48 or 72 hours)
- Example modeling assumptions of interest:
  - Room heatup for long-term equipment operation (i.e., the potential that equipment performance will degrade due to environmental conditions)
  - The leakage path from primary containment to the reactor building or environment
  - The extent of “normal leakage” of containment isolation impairment at the time of the initiator and resulting containment isolation signal;
  - The initial volumes of water in the condensate storage tank and suppression pool;
  - Thermal-hydraulic uncertainties affecting the rate of containment pressurization
- Again, the current thought is to leverage the sequences already used for the FSG investigation above.

Going forward

- Near-term effort will be focused on:
  - Coordinating with DAEC to receive additional plant-specific information
  - Fleshing out the specific calculations to be performed
  - Developing the plant-specific MELCOR modeling, and performing model shakedown/validation
- We plan to hold another public meeting toward the turn of the calendar year, once analysis is imminent
- The analysis will be documented in a NUREG

Questions or feedback – final opportunity during this meeting?

If you think of something after the fact, please feel free to email me at Donald.Helton@nrc.gov