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

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Teleconference

Suzanne Dennis
301-415-7000
Suzanne.Dennis@nrc.gov

Acronyms

- ac – alternating current
- ADS – automatic depressurization system
- ANS – American Nuclear Society
- ATWS – anticipated transient without SCRAM
- CRDHS – control rod drive hydraulic system
- DAEC – Duane Arnold Energy Center
- ECCS – emergency core cooling system
- EDG – emergency diesel generator
- FSG – FLEX Support Guideline
- HPCI – high-pressure coolant injection
- LOOP – loss of offsite power
- LPI – low pressure injection
- MELCOR – not an acronym
- MSIV – main steam isolation valves
- NRC – Nuclear Regulatory Commission
- PRA – probabilistic risk assessment
- RCIC – reactor core isolation cooling
- RPV – reactor pressure vessel
- SDP – Significance Determination Process
- SPAR – Standardized Plant Analysis Risk
- SRV – safety relief valve
- SSC – system, structure, or component

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, specifically the plans for upcoming MELCOR accident analyses;
  - 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

- Introduction
- Background on this project
- Current status and non-proprietary aspects of the analysis plans
- Discussion
- Public comments

- Transition to Closed Portion
- Proprietary aspects of the analysis plans
- Discussion
- Wrap-up

Background

- 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
- 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

Background (2)

- Past work:
  - NUREG-1953: Surry and Peach Bottom success criteria
  - NUREG/CR-7177: investigation of broad success criteria modeling assumptions
  - NUREG-2187: Byron Unit 1 success criteria
  - Has both confirmed existing SPAR modeling assumptions and supported specific SPAR modeling changes
- The approach:
  - 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, 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

MELCOR

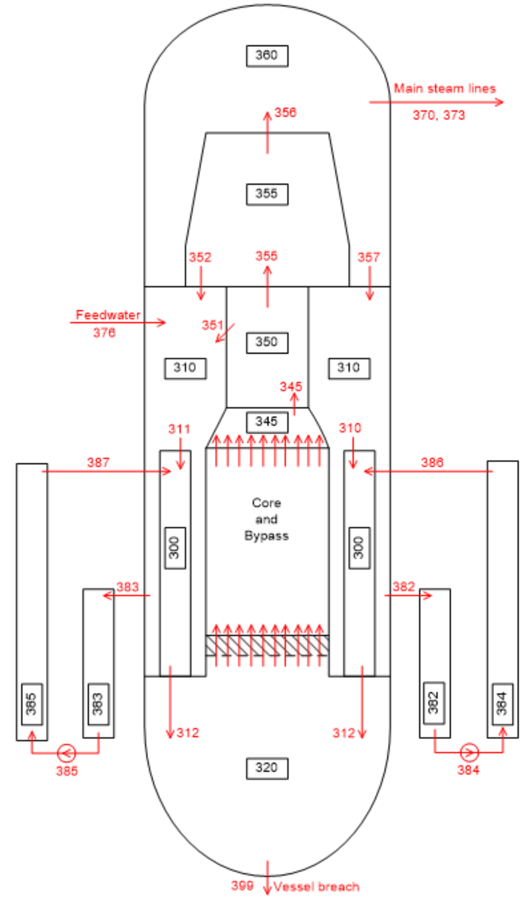
- MELCOR is a fully integrated, engineering-level computer code that models the progression of accidents in light water reactors.
- MELCOR simulates the thermal-hydraulic and post-core damage behavior of the plant, in terms of the major SSCs and actions that affect this response
- MELCOR model is somewhat like the software used to support NPP simulator functionality, except that in the case of typical MELCOR models there is 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

MELCOR (2)

- Phenomena modeled by MELCOR include:
  - the thermal-hydraulic response of the primary reactor coolant system, the reactor cavity, the containment, and surrounding buildings;
  - core uncovery, fuel heatup, cladding oxidation, fuel degradation, and core melting and relocation;
  - thermal and mechanical loading and failure of the lower head;
  - core-concrete attack;
  - hydrogen production, transport, and combustion;
  - fission product release, transport, and deposition;
  - and the impact of engineered safety features (e.g., containment sprays, containment fan coolers, and filters) on thermal-hydraulic and radionuclide behavior
- Only the capabilities relevant prior to the onset of core damage are utilized for this project

MELCOR model

- Model was developed by Energy Research Inc.
- NRC reviewed and commented on model
- Model quality assurance review and model testing have been performed
- Model has been handed off to the NRC from ERI



High-level issues to be investigated

- Intent is not to comprehensively confirm all success criteria in the chosen plant's SPAR model
- Four specific performance topics were chosen:
  - Success criteria for depressurization during non-ATWS scenarios
  - FLEX Support Guidelines (FSGs) applied to loss-of-ac-power and other scenarios
  - ECCS injection following containment failure or venting
  - Safe and stable end-state considerations

Non-ATWS SRV/ADS analysis

- 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
- 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

Non-ATWS SRV/ADS analysis (2)

- Preliminary modeling assumptions
  - Number of SRVs that actuate
    - One, two, or three
  - Credit for CRDHS flow prior to and following depressurization
    - Nominal or maximized
  - Manual actions taken prior to depressurization to stabilize pressure below the relevant set-point(s)
    - Closure of MSIVs or traditional plant cooldown
  - Timing of manual actuation
- Additional sensitivity studies will be run for other variables, including
  - Failure of HPCI
  - Higher recirculation pump leakage
  - Reduced LPI flow
  - Automatic initiation of ADS

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

- New capabilities have been 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
- The current activity is focused solely on scoping success criteria and sequence timing issues that may be informative for future NRC risk modeling
- 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

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

- Preliminary modeling assumptions
  - Time of loss-of-ac
    - Zero or two hours
  - Time of battery depletion, if at all
    - Four hours, eight hours, and indefinite operation (with successful use of portable pump)
    - Time of RCIC failure
    - Four hours or eight hours
  - Flow rate achieved by ac-independent injection, and timing of injection
    - Nominal flow and plus or minus 25%
  - Timing of containment venting or failure
    - Procedurally-driven venting
- Additional sensitivity studies will be run for other variables, including
  - Changes in recirculation seal leakage
  - Reduced RCIC flow
  - Failure of alternate injection

ECCS injection following containment failure or venting

- 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
- The current thought is to leverage the sequences already used for the FSG investigation above.
- Preliminary modeling assumptions
  - The timing (and associated pressure) of venting
    - Proceduralized pressures
  - The vent path used
    - Drywell or hard pipe vent
  - At what point the vent path is closed
    - Based on accident progression and intent of vent (e.g., to maintain pressure or temperature)
  - The response of the SRVs and ECCS pumps to the elevated pressure and the depressurization
    - Functional, degraded, or non-functional

ECCS injection following containment failure or venting (2)

- Additional sensitivity studies will be run for other variables, including
  - Size of rupture area in event of containment failure
  - Reduction of alternate injection flow
  - SRV failure

Safe and stable end-state

- 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"
- Goal - 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)
- Again, the current thought is to leverage the sequences already used for the FSG investigation above.

Safe and stable end-state (2)

- Preliminary modeling assumptions
  - The initial volume of water in the condensate storage tank
    - High normal or low normal
  - The initial volume of water in the suppression pool
    - High normal or low normal
  - Recirculation pump seal leakage
    - Normal, low, or high
  - Decay heat formula
    - Scaled from another BWR/4 MELCOR model or ANS curve
  - CRDHS flow
    - Nominal or maximized
  - MELCOR thermal-hydraulic coefficients
- Additional sensitivity studies will be run for other variables, including:
  - SRV failure
  - Decreased RCIC flow
  - Reduction in alternate injection flow

Going forward

- Near-term effort will be focused on:
  - Continuing model shakedown/validation
  - Performing MELCOR calculations and analysis
  - Performing peer review
- The analysis will be documented in a NUREG

QUESTIONS OR FEEDBACK?

CLOSED PORTION