Confirmatory Thermal-Hydraulic Analysis to Support Success Criteria in NRC's PRA Models

Suzanne Dennis*, Shawn Campbell, Don Helton
U.S. Nuclear Regulatory Commission, Rockville, MD, USA

Abstract: The success criteria (SC) for system performance and operator timing in the NRC's standardized plant analysis risk (SPAR) models are largely based on the SC used in the associated U.S. nuclear industry PRA models. PRA models have used a variety of methods to determine SC, including conservative design-basis analyses and more realistic best-estimate methods, as well as varying modeling assumptions. Consequently, in some situations plants that should behave similarly from an accident sequence standpoint have different SC for specific scenarios. In addition, concerns periodically arise when reviewing licensee sequence timing and SC analyses in the course of performing event or condition risk assessments that could be better resolved with an up-dated set of thermal-hydraulic SC calculations. The U.S. Nuclear Regulatory Commission's (NRC's) Office of Nuclear Regulatory Research is investigating particular success criteria and sequence timing issues of interest for BWR/4 Mark 1 reactors. This paper will present a summary of NRC's work on updated success criteria.

Keywords: PRA, Success Criteria, NRC.

1. INTRODUCTION

The success criteria (SC) for system performance and operator timing in the NRC's standardized plant analysis risk (SPAR) models are largely based on the SC used in the associated U.S. nuclear industry PRA models, although in some cases they are based on other sources such as past NRC studies. PRA models have used a variety of methods to determine SC, including conservative design basis analyses and more realistic best estimate methods, as well as varying modeling assumptions. Consequently, in some situations, plants that should behave similarly from an accident sequence standpoint have different SC for specific scenarios. In addition, concerns periodically arise when reviewing licensee sequence timing and SC analyses in the course of performing event or condition risk assessments that could be better resolved with an updated set of thermal hydraulic SC calculations. For these reasons, this project investigates particular success criteria and sequence timing issues of interest for BWR/4 Mark 1 reactors.

This project is a continuation of work previously documented for other plant type and scenario pairings in NUREG-1953, "Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models—Surry and Peach Bottom" [1], NUREG/CR-7177, "Compendium of Analyses to Investigate Select Level 1 Probabilistic Risk Assessment End State Definition and Success Criteria Modeling Issues" [2], and NUREG-2187, "Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models—Byron Unit 1" [3].

For this paper a summary of the planned calculations are presented. The paper describes the selection of SC issues that are being investigated for the study. From a spectrum of possible issues, and in consultation with the NRC's risk analysts, four issues were selected, as follows:

- success criteria for situations with degraded high-pressure injection & relief valve criterion for non-ATWS
- mitigating strategies (namely FLEX support guidelines) applied to loss-of-ac-power and other scenarios
- emergency core cooling system (ECCS) injection following containment failure or venting
- safe and stable end-state considerations

These issues will be investigated using a plant-specific MELCOR model for a representative plant.

* suzanne.dennis@nrc.gov

2. DEGRADED HIGH-PRESSURE INJECTION & RELIEF VALVE CRITERION FOR NON-ATWS

Following certain initiating events, coincident with degraded high-pressure injection capabilities, operators will use alternate high-pressure injection capabilities to maintain RPV level. If there is insufficient capability to maintain RPV level above a specified level band or insufficient blowdown capacity in the suppression pool and a low-pressure injection system is available, operators will manually initiate the automatic depressurization system (the automatic function having been inhibited very early in the emergency operating procedures for non-ATWS conditions). This initiation will open up multiple safety relief valves in order to depressurize the reactor pressure vessel and allow low-pressure injection. For the relevant (non-ATWS) PRA sequences, assumptions are made regarding what high-pressure capabilities are needed to maintain level, when operator action is required, and how many ADS valves must open, in order to reach conditions where low-pressure injection sources (e.g., low-pressure core spray) in conjunction with any available high-pressure injection sources (e.g., control rod drive hydraulic system [CRDHS]) can provide adequate inventory control and decay heat removal prior to core damage. The relevant success criteria in many PRA models originated from design-basis analyses, and in the case of many models, has been refined over time to remove conservatism. However, there are a number of related modeling assumptions (e.g., water level representation used for the operator cue for manual actuation) and scenario definition characteristics (e.g., amount of credit for CRDHS), that when combined with the accepted variability in computational modeling and user effect, can result in different analyses predicting different requirements for substantively similar designs/conditions. For this reason, the sequence timing and success criteria assumptions for ADS relief valve criteria for non-ATWS sequences periodically becomes an important aspect of an event or condition assessment.

To investigate this issue, the approach is to quantitatively address the variability around a point estimate that would arise from reasonable alterations to the boundary conditions and underlying modeling for this particular scenario. Factors of interest in this regard (based on previous examinations) include items that would have both a positive and negative influence on core heatup:

- Number of safety/relief valves participating in the depressurization or degraded performance of one or more valves;
- SRV discharge path characteristics that affect flow rate and depressurization;
- Failure-to-run of high-pressure injection, as opposed to failure-to-start;
- Credit for CRDHS flow prior to and following depressurization, including:
    - Normal post-trip flow;
    - Enhanced flow using one or both trains;
- Credit for additional alternate injection from Standby Liquid Control;
- Source and achieved flow of low-pressure injection;
- Manual actions taken prior to ADS to stabilize pressure/level and/or to pursue a normal plant cooldown;
- Automatic, as opposed to manual, initiation of ADS (i.e., failure to inhibit automatic actuation);
- Timing of manual actuation (i.e., variation within the allowable level band);
- Amount of recirculation pump seal leakage.

3. MITIGATING STRATEGIES (NAMELY FLEX SUPPORT GUIDELINES) APPLIED TO LOSS-OF-AC-POWER AND OTHER SCENARIOS

Following the severe accidents of March 2011 at the Fukushima Daiichi site in Japan, the U.S. NRC issued several new regulatory requirements, including Order EA-12-049, “Order to Modify Licenses with regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events” [4]. This order required all U.S. nuclear power plants to implement strategies that allow them to cope without their permanent electrical power sources for an indefinite amount of time. The associated strategies must keep the reactor core and spent fuel stored in pools cool, as well as protect the containment. The mitigation strategies use a combination of already-installed equipment (e.g., steam-powered pumps), additional portable equipment that is stored on-site, and equipment that can be flown in or trucked in from one of two regional support centers.

To facilitate implementation of the above order, the NRC issued Interim Staff Guidance in the form of JLD-ISG-12-01 [5]. The guidance states, in part:

> "The NRC staff considers that the development, implementation, and maintenance of strategies and guidance in conformance with the guidelines provided in NEI 12-06, Revision 4, are an acceptable means of meeting the requirements of Order EA-12-049, subject to the exceptions, additions, and clarifications in the enclosure to this ISG. However, NRC endorsement of NEI 12-06, Revision 2, does not imply NRC endorsement of references listed in NEI 12-06, Revision 4."

NEI 12-06, Revision 2 [6], in turn, provides development, implementation and maintenance guidance for the strategies and equipment, including the FLEX Support Guidelines (FSGs) which serve as a new set of guidance governing response to declared ELAP events.

These strategies and equipment are designed for use in postulated accidents where an extended loss of all ac power (ELAP) is declared during the course of responding to a station blackout, and so this project will seek to provide confirmatory information with respect to the success criteria and sequence timing assumptions associated with potential licensee use in risk-informed licensing and oversight submittals. There have also been instances where licensees have sought credit for these strategies and equipment in non-ELAP scenarios (loss-of-ac power scenarios more generally, or otherwise). For this reason, this project will also seek to develop similar confirmatory information for other scenarios of interest.

This series of cases investigates what PRA functions the FLEX equipment and strategies can satisfy and what limitations need to be placed on failure or success of these equipment and strategies. For these cases, uncertainties of interest are:

- Time of loss-of-ac power (i.e., emergency diesel generator failure-to-run);
- Time of battery depletion;
- Time of ELAP declaration;
- Time of reactor core isolation cooling (RCIC) loss (if other than upon battery depletion);
- Suppression pool conditions:
  - heatup (i.e., net positive suction head or bearing over-temperature)
  - pressure (i.e., high turbine exhaust pressure)
  - level (i.e., insufficient suction)
- RCIC turbine flooding due to reactor pressure vessel (RPV) over-fill or insufficient steam due to RPV under-fill;
- RCIC delivered flow;
- Availability of high pressure coolant injection;
- Number of relief valves actuating during depressurization and timing of action (also the subject of Section 2);
- Amount of recirculation seal leakage;
- Flow rate achieved by ac-independent injection, and timing of injection;
- Timing and nature of containment venting (also investigated as part of ECCS injection following containment failure or venting);
- Effect of containment venting/failure on late injection (also the subject of Section 4).

4. EMERGENCY CORE COOLING SYSTEM (ECCS) INJECTION FOLLOWING CONTAINMENT FAILURE OR VENTING

The evaluation of ECCS injection following containment failure is discussed further below. Many BWR PRAs credit coolant injection following containment venting and containment failure caused by the slow over-pressurization of containment resulting from a loss of containment heat removal. The key characteristic of these sequences is the failure of containment (or the venting of containment) before core damage occurs. These sequences often involve a loss of ac power. Although historically the SPAR models have not given credit for injection following containment failure; recently some of the new revisions to the SPAR models include some credit for late (post-containment failure) injection.

There are a number of concerns regarding emergency coolant injection performance during the time leading up to and immediately after containment failure (or venting). These issues are primarily associated with accident sequences that include failure of long-term heat removal or anticipated transients without scram (ATWS) where heat removal is simply inadequate for the heat being generated. The progression of these sequences includes the effects of high pressure inside containment and then the consequences of subsequent containment failure or venting. Specifically, as the containment atmosphere pressurizes, there is the potential that some injection systems might cease working because of increased back pressure on the turbine steam exhaust and/or the automatic depressurization system (ADS) valves being forced closed by the high ambient pressure. Additional concerns arise when the containment fails, or is vented. In this case, there is the potential that the severely adverse environment produced in the reactor building as a result of containment failure (or venting, depending on the configuration of the vent path used) could fail needed safety equipment. In addition, at the time of containment failure (or venting), the rapid depressurization of the suppression pool water could generate boiling in the suppression pool, and ECCS pumps not designed for two-phase flow could fail. Finally, rupture of containment could directly affect continued ECCS operation, if injection or suction lines were damaged. Each of these mechanisms has the potential to result in failure of some or all coolant injection and lead to core damage.

An additional issue that has received attention in recent years concerns the reliance on containment overpressure when assessing the operability of emergency coolant injection during a postulated design basis accident (a.k.a., containment accident pressure). This issue is not considered further per se, in that the focus is on the response of the system during the actual predicted conditions (e.g., operation of ECCS when the containment pressure is elevated). However, the same basic considerations apply here once containment has been vented or has failed, or if a containment isolation failure prevented containment pressurization. The uncertainties of interest for late injection following containment venting or containment failure are:

- 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 safety/relief valves and ECCS pumps to the elevated pressure and the depressurization.

5. SAFE AND STABLE END-STATE CONSIDERATIONS

The ASME/ANS PRA Standard [7] defines a safe and stable state as "a plant condition, following an initiating event, in which [reactor coolant system] RCS conditions are controllable at or near desired values." 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." Requirement SC-A5 elaborates by requiring (for Capability Category II/III) that for sequences where stable plant conditions are not achieved at 24 hours, additional evaluations must be performed. Examples of appropriate evaluation techniques include assigning an appropriate plant damage state for the sequence, extending the mission time until an acceptable end-state is reached, or modeling additional system recovery or operator actions. Only in the definition of "success path" does the standard provide a later backstop time (that being 72 hours), and the success path concept is only invoked in the Seismic Margins assessment (Section 10). Meanwhile, NUREG-2122 [8] defines safe stable state as "Condition of the reactor in which the necessary safety functions are achieved," and goes on to state, "In a PRA, safe stable states are represented by success paths in modeling of accident sequences. A safe stable state implies that the plant conditions are controllable within the success criteria for maintenance of safety functions."

Historically, Level 1 PRA models (including the SPAR models) have typically assumed a mission time of 24 hours, unless core damage was imminent at that time. The analysis in this portion of the report scopes the additional operator actions or system functionality that would be required to extend the sequence duration to a longer period of time (e.g., 48 or 72 hours). Examples of common events of interest in this regard are refill of the condensate storage tank (CST), recovery of suppression pool cooling, alignment of additional alternative RPV injection water sources, and additional containment venting operations.

The uncertainties of interest explored for this issue are:

- Room heatup concerns for long-term equipment operation (e.g., the potential that equipment performance will degrade or operators will be unable to access equipment due to environmental conditions);
- 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 initial volumes of water in the CST and suppression pool;
- Thermal-hydraulic uncertainties affecting the rate of containment pressurization;
- Decay heat formulation in the MELCOR model (the default adopted from a different plant versus the built-in ANS curve);
- Recirculation pump seal leakage.

6. CONCLUSION

The analyses and supporting calculations will be performed in the four categories as described above. The analyses and conclusions from this work will be made publicly available in an NRC NUREG report. The results of these calculations will be considered in future revisions of NRC's SPAR models and the risk-informed activities supported by these models.

References

[1] U.S. Nuclear Regulatory Commission, *Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models—Surry and Peach Bottom (NUREG-1953),* Washington, D.C., 2011.

[2] U.S. Nuclear Regulatory Commission, *Compendium of Analyses to Investigate Select Level 1 Probabilistic Risk Assessment End-State Definition and Success Criteria Modeling Issues (NUREG/CR-7177,* Washington, D.C., 2014.

[3] U.S. Nuclear Regulatory Commission, *Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models—Byron Unit 1 (NUREG-2187),* Washington, D.C., 2016.

[4] U.S. Nuclear Regulatory Commission, *Order to Modify Licenses with regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (EA-12-049),* 2012.

[5] U.S. Nuclear Regulatory Commission, *Compliance with Order EA-12-049, Order Modifying Licenses with regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events, Revision 1 (JLD-ISG-2012-01),* Washington, D.C., 2016.

[6] Nuclear Energy Institute, *Diverse and Flexible Coping Strategies (FLEX) Implementation Guide, NEI 12-06, Revision 2,* Washington, D.C., 2015.

[7] ASME, American Nuclear Society, *Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications (ASME/ANS RA-Sa-2009),* New York, NY, 2009.

[8] U.S. Nuclear Regulatory Commission, *Glossary of Risk-Related Terms in Support of Risk-Informed Decisionmaking (NUREG-2122),* Washington, D.C., 2013.