# L3PRA: Updating NRC's Level 3 PRA Insights and Capabilities

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***Abstract***

The U.S. Nuclear Regulatory Commission (NRC) has recently started a full-scope, Level 3 probabilistic risk assessment (PRA) study intended to address all relevant site radiological sources (including the spent fuel pool), internal and external initiating event hazards, and modes of operation for a 2-unit, Westinghouse four-loop pressurized water reactor station with a large, dry containment. This paper provides a brief overview of the project objectives and approach, identifies a number of key questions and issues (many of which were recognized prior to the Fukushima accident), and describes the current plans and status of the study.

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**Abstract**

The U.S. Nuclear Regulatory Commission (NRC) has recently started a full-scope, Level 3 probabilistic risk assessment (PRA) study intended to address all relevant site radiological sources (including the spent fuel pool), internal and external initiating event hazards, and modes of operation for a 2-unit, Westinghouse four-loop pressurized water reactor station with a large, dry containment. This paper provides a brief overview of the project objectives and approach, identifies a number of key questions and issues (many of which were recognized prior to the Fukushima accident), and describes the current plans and status of the study.

## 1. INTRODUCTION

In response to Commission direction following the submission of SECY-11-0089, "Options for Proceeding with Future Level 3 Probabilistic Risk Assessment (PRA) Activities" [1], the U.S. Nuclear Regulatory Commission (NRC) has recently started a full-scope, Level 3 PRA study intended to address all relevant site radiological sources (including the spent fuel pool), internal and external initiating event hazards, and modes of operation for a 2-unit, Westinghouse four-loop pressurized water reactor station with a large, dry containment.

The overall purpose of this "L3PRA" study is to support the NRC's mission to protect public health and safety by addressing both important changes that have taken place since the NUREG-1150 PRAs (performed in the mid-to-late 1980s) [2] and past risk study scope limitations that limit insights regarding potentially important issues. The post-1150 changes include: (a) modifications to enhance NPP operational performance, safety, and security (e.g., the development and implementation of risk-informed regulations; improved operational, maintenance, and training practices; implementation of severe accident management guidelines – SAMGs; and implementation of extensive damage mitigation guidelines – EDMGs); (b) significantly improved understanding and modeling of severe accident phenomena; and (c) advances in PRA technology (e.g., improved methods, models, analytical tools, and data through research and operating experience). Examples of scope limitations in past risk studies include lack of consideration of multi-unit site effects and ex-reactor vessel radiological sources (e.g., spent fuel pools, dry storage casks).

Although the specific regulatory uses of the study have not been specified, Ref. [2] identifies a number of potential future uses of the L3PRA study, and Level 3 PRA studies in general. These potential uses include: (a) confirm the acceptability of the NRC's current use of PRA in risk-informed regulatory decision making (e.g., the use of Level 1 and limited-scope Level 2 reactor PRAs to support regulatory applications and the use of Regulatory Guide – RG – 1.174 [3] subsidiary numerical objectives based on the reactor-specific risk metrics core damage frequency – CDF – and large early release frequency – LERF); (b) verify or revise regulatory requirements and guidance, particularly those based on NUREG-1150 information (e.g., RG 1.174 and the regulatory analysis guidelines [4] and technical evaluation handbook [5] used by the NRC staff to evaluate the costs and benefits of

proposed backfits [6]); (c) support specific risk-informed regulatory applications (e.g., provide the technical basis for risk-informing the regulation of spent fuel storage and handling, siting, and emergency preparedness, and focus the NRC's Reactor Oversight Process); (d) develop and pilot test PRA technology, standards, and guidance; (e) prioritize generic safety issues and nuclear safety research programs; (f) develop in-house PRA technical capability and support PRA knowledge management and risk communication activities; and (g) support future risk-informed licensing of new and advanced reactor designs (e.g., resolving issues with small modular reactor (SMR) designs, using risk insights to enhance the safety focus of SMR reviews, and modifying risk-informed regulatory guidance for new reactors.

Bearing these potential future uses in mind while recognizing the limitations inherent in the study of a single site, the specific objectives of the L3PRA study are to:

(1) develop a Level 3 PRA, generally based on current state of practice, that (a) reflects technical advances since the last NRC-sponsored Level 3 PRAs were completed over 20 years ago, and (b) addresses scope considerations that were not previously considered;
(2) extract new insights to enhance regulatory decision making and to help focus limited NRC resources on issues most directly related to the NRC's mission to protect public health and safety;
(3) enhance the NRC's PRA staff capability and expertise, and improve documentation practices to make PRA information more accessible, retrievable, and understandable; and
(4) demonstrate the technical feasibility and evaluate the realistic cost of developing new Level 3 PRAs.

By constraining the study to the current PRA state-of-practice, the intent is to limit PRA method development work to only a few topic areas where such work is judged to be needed to ensure meeting the project objectives.

The remainder of this paper provides a brief overview of the L3PRA study's technical approach, identifies a number of key questions and issues, and describes the current plans and status of the study.

## 2. TECHNICAL APPROACH AND CHALLENGES

This section describes, at a high level, the technical approach to be followed for the L3PRA study. A more detailed project plan will be developed after assessing the extent of information and models currently available for the analysis site, and identifying the scope and nature of technical work to complete the study.

### 2.1 Technical approach philosophy

As indicated in Section 1, the L3PRA study will generally be based on current state-of-practice methods, tools, and data. "State-of-practice" methods, tools, and data refer to those that are routinely used by the NRC and licensees and/or have acceptance in the PRA technical community.

The state-of-practice methods to be used will be primarily identified based on the results of an earlier scoping study (performed by NRC staff in several technical working groups, as discussed in SECY-11-0089 [1]) and through additional interactions targeting the NRC's experts in each technical area.

After a list of proposed methods or approaches has been compiled, it will be provided to an internal NRC Technical Advisory Group (TAG) for review and comment.

### 2.2 Tools and models

The NRC envisions using the following in-house tools and models for performing the L3PRA study:

- Systems Analysis Programs for Hands-on Integrated Reliability Evaluation (SAPHIRE) [7] – Version 8 has increased capability for handling large complex models and will be used to analyze both internal and external hazards and all plant operating states.
- MELCOR Severe Accident Analysis Code [8] – will be used to determine system success criteria, accident sequence timing, severe accident phenomenology, and fission product behavior.
- MELCOR Accident Consequence Code System, Version 2 (MACCS2) [9] – considers atmospheric transport and deposition under time-variant meteorology, short- and long-term mitigation actions and exposure pathways, deterministic and stochastic health effects, and economic costs in the evaluation of accident consequences.
- WINMACCS– interfaces with MACCS2, addresses emergency preparedness (EP) response and population movement.
- The NRC's Standardized Plant Analysis Risk (SPAR) model [10] for the plant being studied – addresses the likelihood of reactor core damage using a standard set of event trees for each plant design class and standardized input data for initiating event frequencies, equipment performance, and human performance. As part of the L3PRA study, the set of SPAR model event trees and fault trees for the plant being studied will be expanded, as appropriate.

Note that the MELCOR, MACCS2, and WINMACCS codes were used in performance of the NRC's State-of-the-Art Reactor Consequence Analyses (SOARCA) project [11]. The NRC initiated the SOARCA project to develop best estimates of the offsite radiological health consequences for a limited set of potential severe reactor scenarios. The project, which began in 2007, combined up-to-date information about the project plants' layout and operations with local population data and emergency preparedness plans. This information was then analyzed using state-of-the-art computer codes that incorporate decades of research into severe reactor accidents. The L3PRA study will take advantage of the modeling advances that occurred as part of the SOARCA project, as well as other current and recent research related to these codes.

### 2.3 Key challenges and gaps in PRA technology

The L3PRA project team has identified gaps in PRA technology and other challenges that will be addressed to the extent practical in the study. Figure 1 illustrates these gaps and challenges in terms of the scope elements of a full-scope, site-wide Level 3 PRA (the top white boxes) and the principal tasks for each scope element. The methods for addressing these tasks are categorized in Figure 1 using the following color scheme:

<u>Green</u>: A consensus method is available that requires no modification (e.g., the fault tree approach for system reliability analysis and the parameter estimation approach for independent component failures).

| Internal events | Internal floods | Internal fires | Seismic and other external hazards | Low power and shutdown | Severe accident progression | Consequence analysis |
|---|---|---|---|---|---|---|
| Initiating event analysis (Y) | Identification and screening analysis (G) | Screening analysis (G) | Screening/ bounding analysis (G) | Initiating event analysis (G) | Plant damage state analysis (G) | Source term binning (G) |
| Accident sequence analysis (Y) | Plant response analysis | Plant response analysis | Plant response analysis | Accident sequence analysis (G) | Accident progression and source term analysis | Modeling of emergency preparedness response and development of MACCS2 input deck |
| Success criteria analysis (G) | | | | Success criteria analysis (G) | | |
| Systems analysis (G) | | | | Systems analysis (G) | | |
| Data analysis (Y) | Flood frequency analysis (G) | Fire initiation analysis (Y) | Event frequency analysis* (G) | Data analysis (O) | | |
| | | Fire damage analysis (G) | Fragility analysis (O) | | | |
| Human reliability analysis (G) | Human reliability analysis (O) | Human reliability analysis (G) | Human reliability analysis (O) | Human reliability analysis (O) | Human reliability analysis (Y) (O) | Human reliability analysis (G) (O) |
| Quantification (G) | Quantification (G) | Quantification (G) | Quantification (G) | Quantification (G) | Quantification (G) | Consequence calculation (G) |
| | | | | | | Risk integration (R) |
| Uncertainty analysis (G) | Uncertainty analysis (G) | Uncertainty analysis (G) | Uncertainty analysis (G) | Uncertainty analysis (G) | Uncertainty analysis (O) | Uncertainty analysis (O) |

**Spent fuel pool** (O)

- Initiating event analysis
- Accident sequence analysis
- Success criteria analysis
- Systems analysis
- Data analysis
- Human reliability analysis
- Quantification
- Uncertainty analysis

**Dry cask storage** (O)

- Initiating event analysis
- Accident sequence analysis
- Success criteria analysis
- Systems analysis
- Data analysis
- Human reliability analysis
- Quantification
- Uncertainty analysis

**Site risk** (R)

- Initiating event analysis
- Equipment and operator dependency analysis
- Model integration
- Risk metrics
- Uncertainty analysis

*Event frequency analysis is categorized as "Orange" for external flooding, only. For seismic and other "non-flooding" external hazards, this element is characterized as "Green."

**Figure 1. Level 3 PRA Project Scope Elements and Principal Tasks**

Yellow: Methods exist, but limited effort is required to either improve them or to select among several consensus approaches (e.g., common-cause failure modeling).

Orange: No method has been developed and/or demonstrated in an integrated PRA application, but existing methods or approaches could be adapted with moderate effort (e.g., human reliability analysis for actions following a seismic event or core damage). This category also includes elements for which methods exist, but substantial effort may be required to implement them (e.g., seismic fragility analysis if it is necessary to estimate seismic fragilities for a large number of structures, systems, and components as part of the L3PRA study).

Red: New method development is necessary, which could require significant effort (e.g., addressing multi-unit risk).

Approaches will need to be developed or improved to address technical elements in the "non-green" categories, above. In improving or developing these methods, the L3PRA study will strive to be realistic, avoiding excessively conservative assumptions or analytical simplifications. However, the desire for realism will need to be balanced against resource and schedule limitations. Accordingly, not all aspects of the study will necessarily receive the same level of analytical rigor. Consistent with the philosophy and practice of PRA, the level of analytical rigor will be a function of risk significance, thereby ensuring that conservative assumptions and modeling limitations do not result in inappropriate risk insights.

Based on the number of "red" and "orange" entries in Figure 1, the greatest challenges for the L3PRA study are posed by the current limits in the modeling of multi-unit site risk (as opposed to single unit risk), in spent fuel PRA technology (i.e., for spent fuel pools and dry storage casks), and in human reliability analysis (HRA) for other than internal events and internal fires. These challenges are briefly discussed below. The general approach to addressing these challenges will be to primarily rely on existing research and the collective expertise of the project team (including the TAG and contractors), and to perform limited new research only for a few specific technical areas (e.g., multi-unit risk). Specific research activities, either past or current, that are expected to contribute to the resolution of these challenges are identified in the following discussions.

### *2.3.1 Modeling of site risk including multiple units and spent fuel sources*

In order to evaluate the risk of the entire site, the study needs to address all site radiological sources. As discussed in SECY-11-0089, because the Commission's safety goals, Quantitative Health Objectives (QHOs), and subsidiary numerical objectives have traditionally been applied on a per reactor basis, most PRAs developed to date do not explicitly consider multi-unit accidents in which initiating events lead to reactor core damage in multiple units at the same site. Current PRA models, therefore, typically do not identify and address dependencies between systems at multi-unit sites, particularly those with highly complex support system dependencies involving systems and subsystems that are shared by multiple units. Such dependencies are also not addressed as they pertain to spent fuel pools and dry storage casks.

To understand the contribution of these multi-unit and non-reactor effects to the overall site risk, PRA models (and perhaps methods) need to be enhanced to address the following:

- Initiating events common to multiple reactors and/or spent fuel pools and dry casks

- Common or dependent equipment and operator actions between multiple reactors and/or spent fuel pools and dry casks (including timing and sequencing effects)
- Shared stacks, ventilation systems, or other pathways for combustible gases
- Effects of core damage, radiological release, and mitigation actions on operator response (including control room habitability)
- Screening of potentially large numbers of non-contributing site-wide scenarios
- Integrated models for all site radiological sources, including consideration of model end-states, risk metrics, and mission times
- Integrated uncertainty analysis for overall site risk

It is anticipated that addressing the multi-unit and non-reactor effects on overall site risk will be one of the most complex challenges of the project. Accordingly, one of the initial project tasks will be to research the risk impacts and related issues associated with the operation of multiple nuclear power plant units at a single site.

*2.3.2 Spent fuel PRA technology*

As discussed in SECY-11-0089, process areas not related to reactor core operations that can contribute to nuclear site accident risk include those associated with onsite nuclear spent fuel handling and storage. Principal risk-related studies that have previously been performed in these areas include a study of the spent fuel pool accident risk at decommissioning nuclear power plants (NUREG-1738 [12]), a dry cask storage PRA (NUREG-1864 [13] ), an industry PRA of bolted storage casks (EPRI TR-1009691 [14]), and a number of NRC dry cask storage and transportation security assessments. (NRC has also sponsored work assessing the effect of spent fuel pool draindown events on reactors [15].) Although these and other risk-related studies have addressed various aspects of the risk of accidents involving spent fuel pools and dry cask storage, they involved very focused applications. The challenge for the L3PRA project is to apply the enhanced technology for modeling the risk of accidents involving spent fuel that has been developed since these earlier studies, in a manner that enables a meaningful comparison and relative ranking of these process area risk contributors as part of a comprehensive site Level 3 PRA. Example areas requiring additional focus include: success criteria determination, HRA, accident phenomenological analysis, and source term analysis. Some of these areas are expected to be addressed to some degree as part of a current NRC study that addresses the consequences of a beyond-design-basis earthquake on a spent fuel pool for a selected boiling water reactor [16].

*2.3.3 Human reliability analysis for other than internal events and internal fires*

Currently, state-of-practice HRA methods exist for addressing operator performance in Level 1 internal events PRA and in internal fire PRA. NRC is also currently developing an improved HRA approach in response to Commission direction [17], and aspects of this new approach will be used to the extent that they are available consistent with the schedule for the L3PRA project. However, as discussed in SECY-11-0089, state-of-practice methods do not currently exist for post-core damage and external events HRA modeling, and such modeling is beyond the scope of the approach being pursued in response to SRM-M061020.

Post-core damage HRA modeling primarily involves operator actions incorporated into SAMGs and EDMGs mentioned earlier in this paper. Since the operator actions addressed in both the SAMGs and EDMGs are “knowledge-based” (as opposed to “rule-based”), decision makers responding to the

severe accident need to use their knowledge and problem solving skills to identify an appropriate course of action, and this is being done under unfamiliar conditions. In addition, many of the SAMG and EDMG strategies to mitigate the effects of one problem result in adverse effects on another problem. The decision makers must therefore make risk-benefit decisions when considering different strategies. Since the most appropriate response to a given condition cannot be determined in advance, the definition of what constitutes a failure and the identification of post-core damage human failure events or recovery actions that can be credited in the PRA model presents a unique challenge.

In addition to addressing challenges associated with the modeling of SAMGs and EDMGs, state-of-practice methods do not currently exist for addressing operator performance in response to various external events (e.g., seismic events and external flooding) or when the reactor is at low power or shut down. Therefore, these areas will require further investigation. As indicated earlier, current NRC research in response to SRM-M061020 may be helpful.

## 3. L3PRA – CURRENT PLANS AND STATUS

An initial high-level project plan for the L3PRA project was submitted to the Commission in March 2012 [18]. Subsequent activities have involved establishing a communication protocol to control and manage the flow of information between the NRC and the analysis site, assembling the L3PRA project team of staff and contractors, and preparing a detailed project plan and schedule that includes documentation of the proposed technical approach. Once the project technical approach is completed, and information becomes available from the analysis site, the technical work of the project will commence.

## 4. CONCLUDING REMARKS

There have been significant improvements in PRA technology and understanding and modeling of severe accident phenomena, as well as enhancements in NPP operational performance, safety, and security, since the NUREG-1150 PRAs were performed in the mid-to-late 1980s. It is also recognized that there is a need to consider site-wide risks, that is, the risks from the storage of spent fuel (in pools or dry casks) and the intercoupled effects of having more than one reactor unit on a site. Incorporating these improvements and site-wide considerations into a full-scope Level 3 PRA could yield new insights to enhance regulatory decision making, recognizing the limitations inherent in the study of a single site.

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