Full-Scope Site Level 3 PRA

Advisory Committee on Reactor Safeguards
Reliability and PRA Subcommittee

October 4, 2017
(Open Session)

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Outline

- Open Session
  - Project status overview
  - Documentation – NUREG, Part 1
- Closed Session
  - Level 2 PRA
    - Internal events and floods
    - Internal fires, seismic events, high winds
    - Shutdown
  - Level 3 PRA – internal events and floods
  - Spent fuel pool PRA

Level 3 PRA Project Status Overview

October 4, 2017

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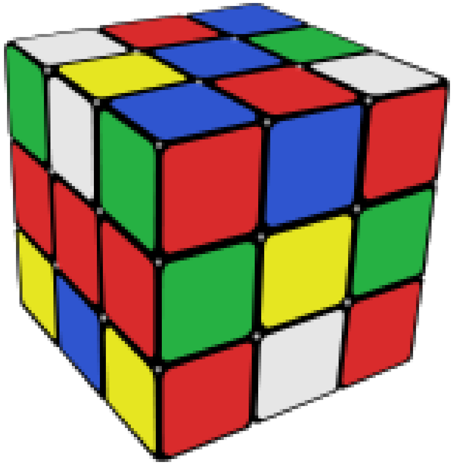
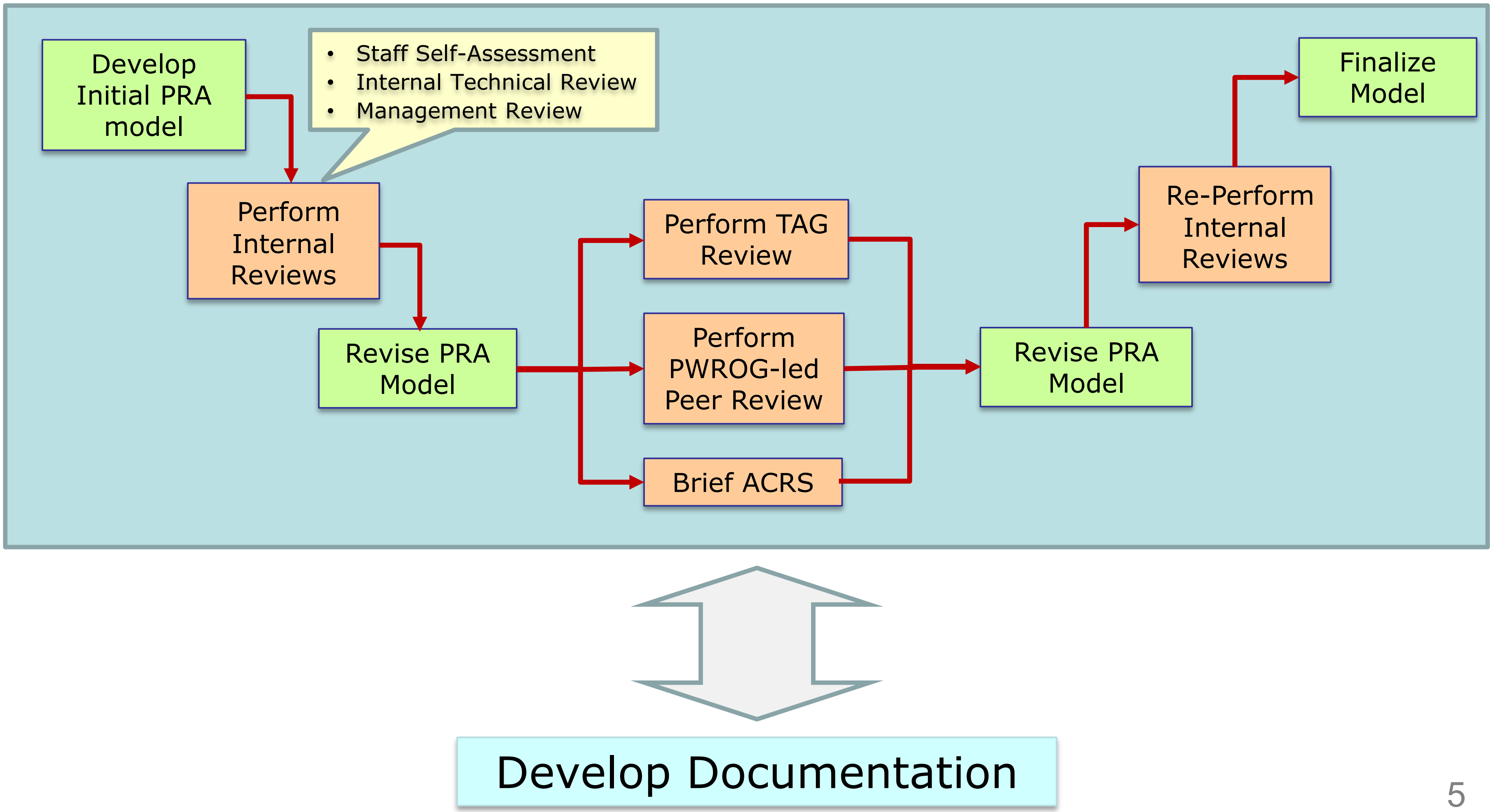
Outline of Presentation

- Reactor, at-power, internal events and floods
- Reactor, at-power, internal fires and seismic events
- Reactor, at-power, high winds and other hazards
- Reactor, low power and shutdown
- Spent fuel pool
- Dry cask storage
- Integrated site
- Path Forward

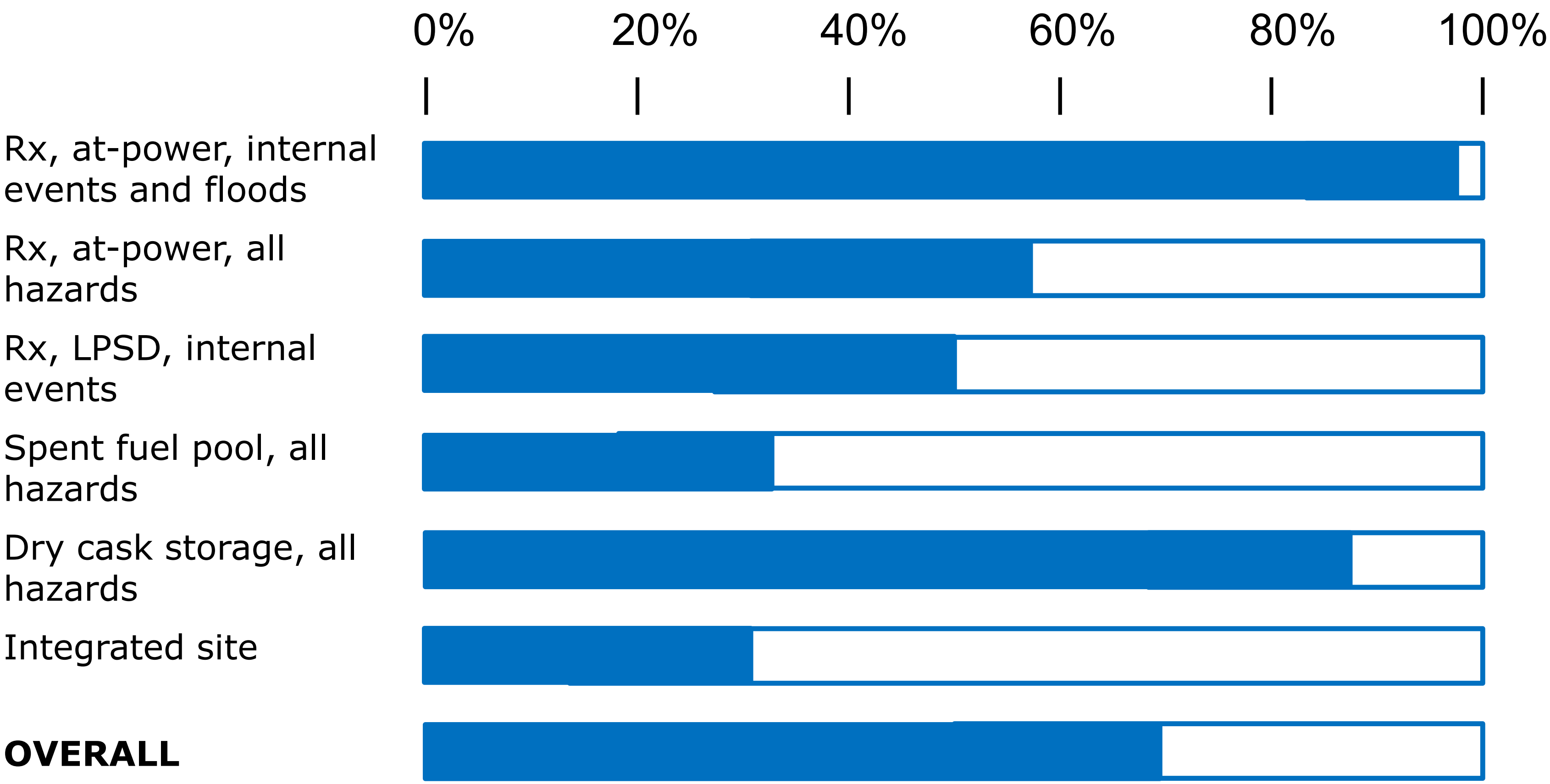


Generic Process for PRA Model Development



Project Status

Combined status of model development, project reviews, and project documentation



Reactor, At-Power, Internal Events and Floods

- Completed ASME/ANS PRA standard-based peer review of Level 1, 2, and 3 PRAs, led by PWR Owners Group
- Completed substantive update to Level 1, 2 and 3 PRAs to address peer review and other comments
  - Level 1 internal flood report nearing completion
  - Level 2 internal event and flood PRA undergoing internal technical review
  - Level 3 internal event and flood PRA report being finalized (prior to internal technical review)
- Completed expert elicitation for interfacing systems LOCA

Reactor, At-Power, Internal Fires and Seismic Events

- Completed initial revision of Level 1 fire and seismic PRA models and documentation based on new input from SNC
- Both models and documentation have been updated to incorporate internal technical review comments
- Revised fire PRA is undergoing project management review; revised seismic PRA is in the queue for project management review
- Level 2 modeling for internal fires and seismic events is on-going
  - Leveraging internal event Level 2 PRA
  - Hazard-specific adjustments made to bridge tree and plant damage state (PDS) modeling
  - Working on impacts to system performance, human reliability analysis (HRA), and containment event tree

Reactor, At-Power, High Winds and Other Hazards

- Completed ASME/ANS PRA standard-based peer review, led by PWROG
- Completed substantive update to "Other Hazards" report to address peer review and other comments
  - Currently undergoing final project management review
- Performed substantial update of high wind PRA to address peer review and other comments, as well as incorporate additional information obtained from high wind walkdown and follow-on analyses
  - Currently undergoing internal technical review

Reactor, Low Power and Shutdown

- Completed initial LPSD Level 1 PRA model for internal events
  - Currently incorporating feedback from internal technical review
- Work continuing on LPSD Level 2 PRA
  - Completed bridge tree and PDS modeling and quantification
  - Completed MELCOR analyses
  - Working on containment event tree and HRA
- Performed a Phenomena Identification and Ranking Technique (PIRT) expert elicitation to identify ranked list of focus areas for LPSD PRA
  - Contractor report completed (contains proprietary information)
  - Work initiated on a NUREG/CR (for public release)

Spent Fuel Pool PRA

- Level 1 analysis is nearly complete for most of the initiating events under consideration
- Continuing work includes:
  - Human reliability analysis: method has been defined and is being exercised for the events of interest
  - Accident progression analysis: preliminary results are under investigation
  - Documentation is ongoing

Dry Cask Storage PRA

- Completed initial Level 1/2/3 model and documentation for all hazards
- Revised consequence analysis to be Vogtle-specific
- Completed internal technical review (NMSS)
- Currently undergoing project management review

Integrated Site PRA

- Developed an approach for an integrated site PRA model using single-source PRA model results and risk insights to prioritize the systematic identification and modeling of multi-source accident scenarios and inter-source dependencies
- To provide additional confidence that potentially important multi-source accident scenarios are not missed, this approach is coupled with the use of systematic techniques to search for and prioritize potential multi-source accident scenarios that may not be captured by relying only on results and insights from individual single-source PRA models.
- Completed pilot applications of the approach for:
  - Reactor Units 1 & 2, at-power, internal events, Level 1 PRA
  - Reactor Units 1 & 2, at-power, internal events and floods, Level 2 PRA
  - Reactor Units 1 & 2, at-power, seismic events, Level 1 PRA

Key Upcoming Milestones

- Complete updated reactor, at-power, other hazards report (October 2017)
- Dry cask storage, Level 1, 2, and 3 PRA ready for technical adequacy review (October 2017)
- Reactor, at-power, Level 1, internal fire PRA ready for technical adequacy review (November 2017)
- Reactor, at-power, Level 1, seismic event PRA ready for technical adequacy review (December 2017)
- Complete updated reactor, at-power, Level 2, internal event and flood PRA (December 2017)
- Reactor, LPSD, Level 1, internal event PRA ready for technical adequacy review (December 2017)

Acknowledgements

- SNC
- PWR Owners Group
- Westinghouse
- EPRI
- NSIR, NRO, NRR, NMSS, Regions, TTC
- National Laboratories (INL, SNL, PNNL, BNL)
- Commercial Contractors (ERI, ARA, IESS)
- ACRS

Acronyms and Definitions

ANS	American Nuclear Society
ARA	Applied Research Associates
ASME	American Society of Mechanical Engineers
BNL	Brookhaven National Laboratory
EPRI	Electric Power Research Institute
ERI	Energy Research, Inc.
HRA	Human reliability analysis
IESS	Innovative Engineering & Safety Solutions, LLC
INL	Idaho National Laboratory
LOCA	Loss of coolant accident
LPSD	Low power and shutdown
PDS	Plant damage state
PIRT	Phenomena Identification and Ranking Technique
PNNL	Pacific Northwest National Laboratories
PRA	Probabilistic risk assessment
PWR	Pressurized-water reactor
PWROG	PWR Owners Group
SNC	Southern Nuclear Operating Company
SNL	Sandia National Laboratories
TAG	Technical Advisory Group

Level 3 PRA Project Draft Report – Part 1

Advisory Committee on Reactor Safeguards
Reliability and PRA Subcommittee

October 4, 2017
(Open Session)

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NUREG Report

- User friendly
- Accessible
- Retrievable
- Understandable
- Informative

Goals and Challenges

- Contains sufficient information to understand:
  - Design and operation of the plant
  - The technical approach
  - Major assumptions
  - Major results
  - Major insights and perspectives
  - Potential uses
  - Potential future work
- Major challenges
  - The level of detail of information in the report recognizing concern regarding propriety information
  - The significant amount of information – what to and not to include – so as not to overwhelm the reader but remain informative
  - How to represent the information in an efficient, effective, and understandable manner for a "four dimensional" PRA model that addresses multiple sources, multiple hazards, multiple operating states, and all three PRA levels

Table of Contents

Section 1 -- Introduction

1.1 Background

- History of project

1.2 Objective

- Stated objectives from SECY papers

1.3 Scope

- Issues included and not included
- Compared to NUREG-1150
- PRA elements

1.4 Assumptions & Limitations

- High level across the project

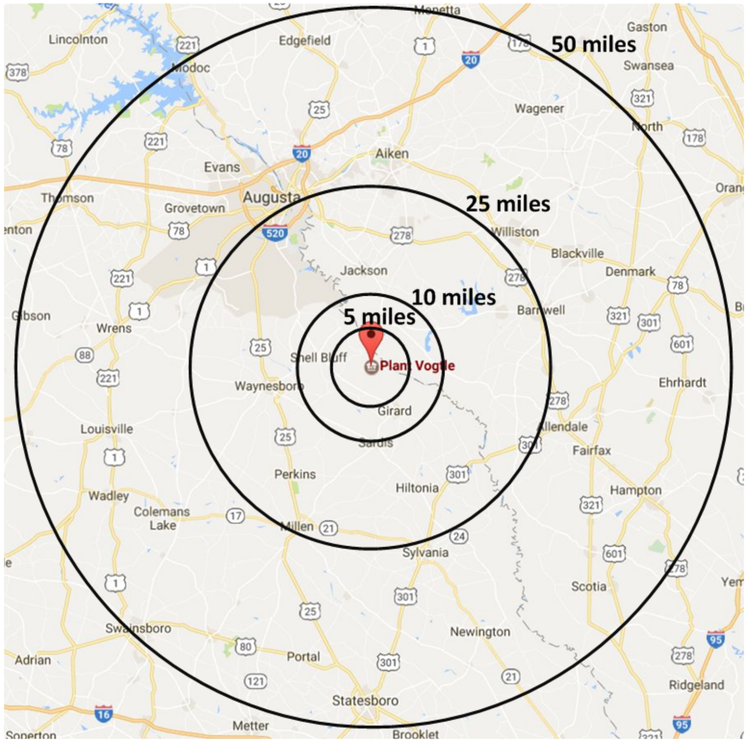
1.5 Document Structure

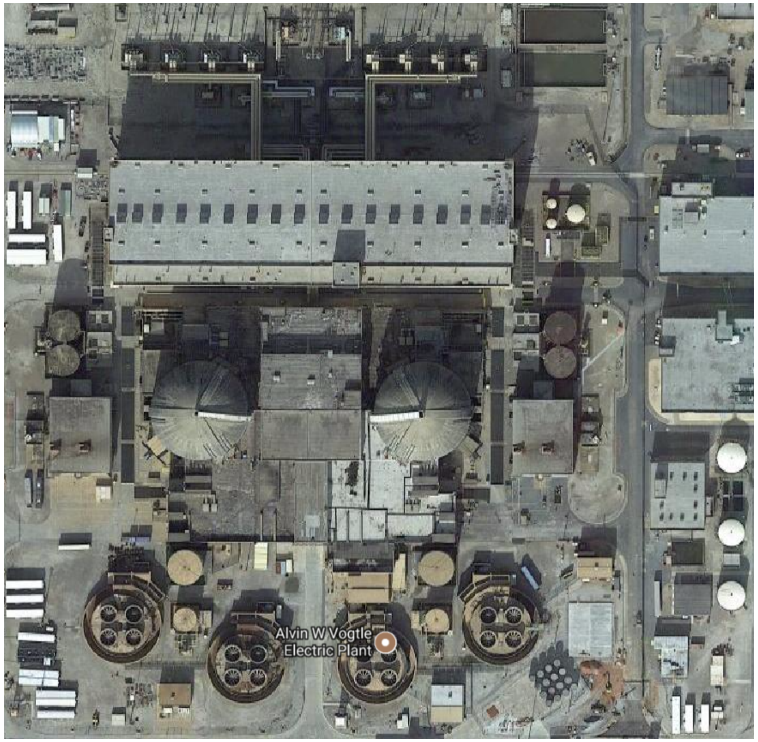
Section 2 – Summary of Plant Design and Operation

- Description of site, reactors, spent fuel pools, dry cask storage
- Brief description provided for each structure and system modeled
  - Purpose and function
  - Configuration
  - Actuation
  - Success criteria
  - Dependencies
- Simplified schematic provided for structures and systems
- Dependency diagram provided
- No actual system layout provided nor plant-specific labeling

Section 2.1 – Vogtle Site

- High level description of plant site and location





Section 2.2/2.3 – Reactor Plant Design

- Includes descriptions, schematics and dependency diagrams

Front Line Systems	Support Systems
Accumulators	AC and DC electrical
High pressure injection/recirculation	Nuclear service cooling water
Low pressure injection/recirculation	Component cooling water
Primary operated relief valves	Auxiliary component cooling water
Residual heat removal	Circulating Water
Main feedwater	Turbine plant closed cooling water
Auxiliary feedwater	Turbine plant cooling water
Reactor protection	Instrument air
Containment spray	
Containment cooling	
Containment isolation	

Section 2.4 – Spent Fuel Pool Storage

2.4.1 Overview

- High level discussion of spent fuel pool (SFP) structure and associated systems

2.4.2 Spent Fuel Pool Cranes

- Cranes used to move fuel assemblies within the pool and for transporting new fuel containers

2.4.3 Spent Fuel Pool Cooling and Purification System

- System removes the decay heat from the SFP

2.4.4 Auxiliary and Fuel Handling Building Heat, Ventilation, and Air Conditioning

- System provides ventilation and filtration and maintains suitable atmosphere for personnel and equipment

Section 2.5 – Dry Cask Storage

- Dry Cask Storage (DCS) System
- Multipurpose Canister (MPC)
- Transfer and Storage Overpacks
- Dry Cask Storage Operating Stages
- Dry Cask Storage Process
- SFPs and Cask Loading Pit
- Cask Washdown Area
- Cask Transfer Facility
- Independent Spent Fuel Storage Installations
- Vertical Cask Transporter
- Alternate Cooling Water System
- Supplemental Cooling System
- Forced Helium Dehydration System
- Automated Welding System
- Low Profile Transporter
- Mating Device
- Other Plant Dry Cask Storage Supporting Systems

Section 3 – Summary of Approach

- Section 3.1 – Overall Approach
- Section 3.2 – Technical Analyses
- Section 3.3 – Reactor Risk Model
- Section 3.4 – SFP Risk Model
- Section 3.5 – DCS Risk Model
- Section 3.6 – Site Risk Model
- Section 3.7 – Other Hazards

Section 3.1 – Overall Approach

- Basic approach
  - Separate models for each source (reactor, SFP, DCS)
  - For reactor, started with internal events and expanded
  - For SFP and DCS, a single integrated model was constructed that addressed the risk from significant hazards

Section 3.2 – Technical Analysis

- For each technical element
  - Purpose/objectives of analysis
  - Major steps associated with analysis
  - Output/products of the analysis
- Technical elements
  - Plant Familiarization
  - Screening analyses
  - Initiating event analyses
  - Structural analyses
  - Human reliability analyses
  - Quantification analyses
  - Consequence analyses
  - Hazard and fragility analyses
  - Uncertainty analyses
  - Systems analyses
  - Accident progression analyses
  - Parameter estimation analyses
  - Source term analyses

Section 3.3 – Reactor Risk Model

- Organized by plant operating state, risk level, and hazard
- Level 1, at-power conditions
  - Internal events model based on SNC model that was converted to SAPHIRE
  - Expanded to address other hazards while leveraging the work performed by SNC
  - Where work on particular technical element was needed, followed guidance in Section 3.2
- Level 2 & 3, at-power conditions
  - Based on guidance in Section 3.2
- LPSD – Level 1, 2 and 3
  - Ranked risk significance of plant outage types, plant operating states and initiating event categories to focus analysis

Section 3.4 – SFP Risk Model

- Single integrated Level 1 and Level 2 model was constructed
- Prioritization scheme developed to focus the SFP PRA model
  - Speed of the accident
  - Amount of sloshing
  - Significance of the hazard
- SFP model involved seismic hazard with fuel uncovery from sloshing
- Model followed the technical elements as described in Section 3.2

Section 3.5 – DCS Risk Model

- Single integrated Level 1 and Level 2 model was constructed
- Level 1 and Level 2 model based on NUREG-1864 and expanded
  - Modeled in detail all known hypothetical hazards/events that had the potential to challenge systems and result in radionuclide release
  - Screened hazards/events based on previous experience
- Level 3 model followed the guidance in Section 3.2

Section 3.6 – Site Risk Model

- Assumed risk dominated by dependencies among risk sources and significant contributors from individual risk sources
- Developed scheme to logically combine important accident scenarios from the individual radiological sources
- Only evaluating consequences
- Used a systematic scheme to identify and prioritize potential scenarios that might be missed by solely relying on results and insights from the individual single-source models

Section 3.7 – Reactor: Other Hazard Risk Models

- Over 30 other hazards identified, examples

Aircraft	Coastal erosion	Damn failure	Fog
High temperature	Landslide	Meteor	Pipeline accident
Soil shrink-swell	Storm surge	Transportation	Volcanic

- Developed criteria for screening
  1. The hazard does not result in a plant trip (manual or automatic) or a controlled manual plant shutdown while at power and does not impact any SSCs that are required for accident mitigation from at-power transients or accidents.
  2. The hazard cannot occur close enough to the plant to affect it.
  3. The hazard is included in the definition of another analyzed hazard.
  4. The hazard has a significantly lower mean frequency of occurrence than another hazard.
  5. The current design-basis hazard has a mean frequency less than 1×10^{-5} per year, and the mean value of the conditional core damage probability is assessed to be less than 1×10^{-1}.

- All other hazards were screened from detailed analysis

NUREG REPORT Part 1-- Status

- Initial draft is complete
- Starting the review process:
  - Internal reviews – staff review then management review
  - TAG review
- Need to decide when to initiate “public review”
  - When entire NUREG is written or in pieces?