U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Probabilistic Risk Assessment – Selected Research Activities

December 6, 2012

Presented at a meeting of the Pressurized Water Reactor Owners' Group Risk Management Subcommittee

Gary DeMoss and Donald Helton
Division of Risk Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Outline

- Background on PRA Research
- Overview of Selected Research Activities
  – Common Cause Failure (CCF) Treatment in Event & Condition Assessments (ECA)
  – Seismically-Induced Fires & Floods
  – Vogtle Level 3 PRA
  – Confirmatory Success Criteria Analyses
- Recent Publications

Overall PRA Research Goals

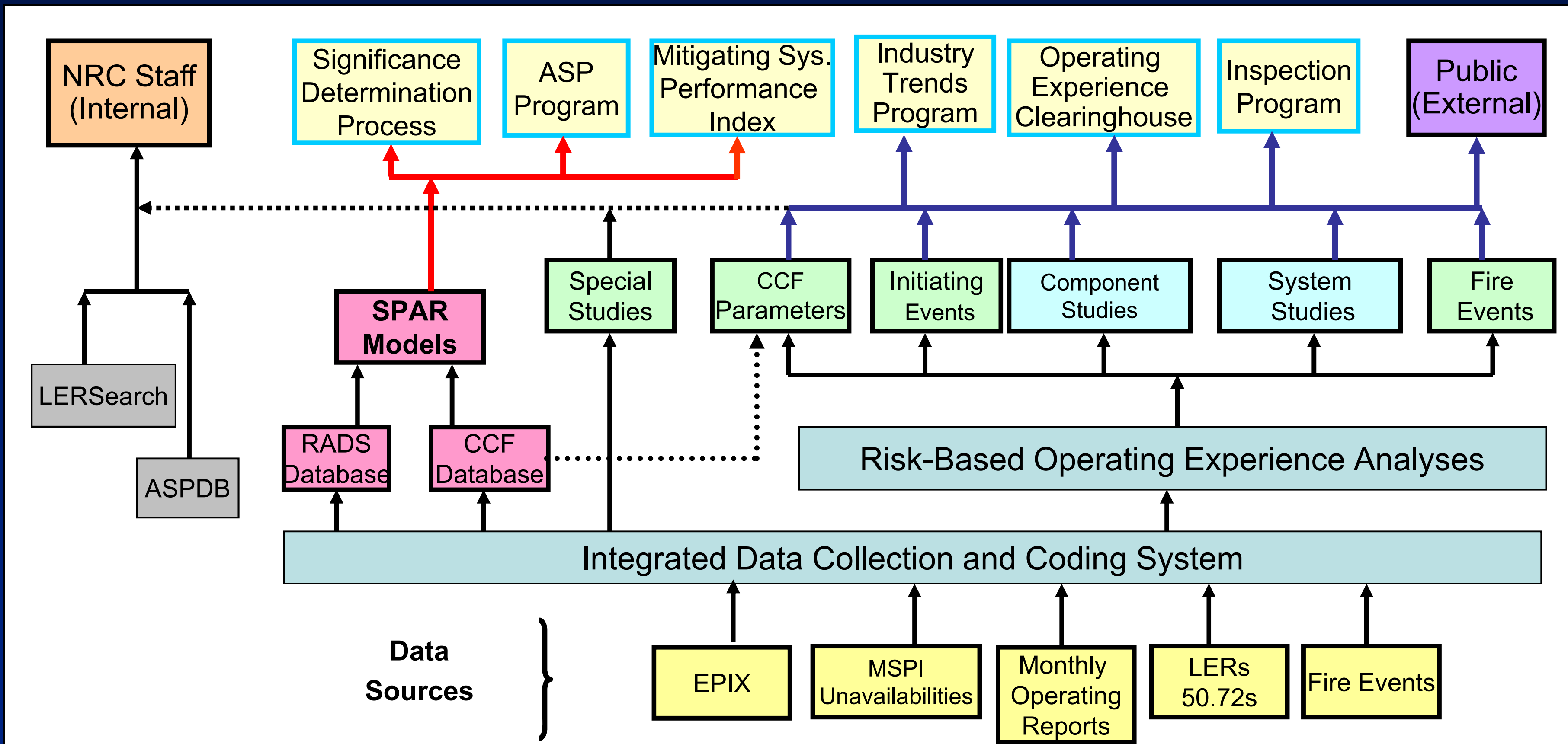
- Support the reactor oversight and operating experience programs
- Using risk-informed approaches to improve the effectiveness and efficiency of regulation
- Expand PRA infrastructure to encompass new and advanced reactor concepts and designs
- Support continuous advancement in PRA state-of-the-art and state-of-practice

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Operating Experience

- Accident Sequence Program (ASP)
  - Purpose is to identify accident precursors and to factor operating experience insights into the regulatory process.
  - Input to NRC performance measures reported in the annual performance and accountability report to Congress and Industry Trends Program
  - Determination of safety significance of events and regulatory issues
  - Program has been screening and analyzing events since 1979
- RES Data Collection, Analysis, and Trending Programs
  - Data sources are integrated into collections and databases
  - Analysis feeds the Industry Trends and other programs, as well as quantification of SPAR models

RES Data Collection, Analysis, and Trending Programs



Reactor Oversight Support – SPAR and SAPHIRE

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

CCF Treatment in ECA – ECA Philosophy

- Observed equipment failures, deficiencies, human errors, and outages mapped into PRA model
  - Provides numerical estimate of risk significance of event
  - Can be retrospective or prospective
  - Example NRC applications: Significance Determination Process (SDP) and ASP
- "Failure memory" approach to quantification:
  - Basic events in PRA associated with equipment failure set to True (1.0)
  - Basic events associated with equipment that operated successfully or was unchallenged are not modified (remain 'nominal' failure probabilities)
    - If successes are credited, then CDF or CCDP equals zero
  - Other basic event probabilities (e.g., CCF, human failure) conditioned on salient event characteristics

CCF Treatment in ECA – ECA Philosophy (2)

- Examples of treatment of successes:
  - No loss of offsite power or hurricane season: nominal LOOP frequency unchanged for condition assessment
  - Pump operated for 24 hours during the event: nominal pump FTR probability unchanged
  - Valve opened during required T.S. LCO test: nominal valve FTO probability unchanged
- Examples of conditioning on salient event characteristics:
  - Offsite power non-recovery probability adjusted (decreased or increased) based on observed failures
  - CCF probability adjusted (increased) based on observed performance deficiency that caused a failure in a common cause component group (CCCG)

CCF Treatment in ECA - Basics

Deficiency → Dependence → CCF Potential

- ECA estimates risk significance of deficiencies and associated component failure(s)
- Dependence exists between redundant components because of similar components, environments, and human interactions
- Failure of a redundant component is more likely given an observed deficiency that resulted in a single failure
- CCF is means to quantify risk impact of dependencies that are NOT explicitly modeled in the PRA
  - Conditional CCF probability captures risk significance of deficiencies that have the potential to fail redundant components within a defined CCCG
  - Base PRA does not reflect an actual deficiency; only the likelihood of one occurring (an observed deficiency increases plant risk)
- Deficiencies increases the potential for CCF
  - $P(CCF \mid Deficiency) \neq P(CCF)_{Base}$ or 0
  - If deficiency can affect redundant components (e.g., deficient procedures or control of maintenance) a dependency exists
  - Despite increased potential for CCF, individual failure is the most likely outcome (95-99% of the cases)
  - Independence unlikely unless deficiency was clearly outside PRA component boundary

CCF Treatment in ECA – ECA Guidance

- CCF potential evaluated at level of deficiency
- All deficiencies have the <u>potential</u> for CCF and therefore the probability of CCF cannot be eliminated in ECA
  - Consistent with failure memory approach
  - Avoid arguments that dive down to piece-part level
    - Differences can always be found at low enough level, but these differences may not be meaningful with respect to identified deficiency and dependence
    - Crediting "success" for CCF for piece parts on the redundant train(s) is inconsistent with the failure memory approach
  - Credit for programmatic actions to mitigate CCF potential (i.e., success) should be applied qualitatively and separately from the numerical risk results
- NRC uses existing Alpha Factor Model and CCCG boundaries in SPAR models
  - Explicitly model deficiency observed outside CCCG, as appropriate
  - Use industry-wide α parameters
  - Use SAPHIRE code for CCF probability adjustment
- Development of deficiency-level factors not ruled out
  - Follow accepted modeling and data analysis practices

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

CCF Treatment in ECA – Path Forward

- Currently evaluating failure data using INL-based tools and databases (including EPIX)
  - Bayesian parameters
  - Un-modeled dependencies
  - Recency of data
  - http://nrcoe.inel.gov/resultsdb/
- Grant with University of Maryland's Center for Risk and Reliability to explore cause-based CCF approaches
- Welcome external stakeholder involvement in future research to advance state-of-practice in CCF modeling

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Seismically-Induced Fires & Floods - Background

- Seismic events have the potential to cause:
  - multiple failures of safety-related SSCs;
  - induce separate fires or flooding events in multiple locations at the site; and
  - degrade the capability of plant SSCs intended to mitigate the effects of fires and floods.
- The NTTF recommended, <u>as part of the longer term review known as "Tier 3"</u>, evaluation of potential enhancements to the capability to prevent or mitigate seismically induced fires and floods
- Commission agreed with Tier 3 Prioritization, but directed the staff to initiate development of PRA method to evaluate potential enhancements as part of Tier 1 activities

Seismically-Induced Fires & Floods - Background

- PRA Method Challenges:
  - hazard definition & characterization
  - seismic fragilities for SSCs, including fire protection components
  - modeling concurrent and subsequent initiating events
  - treatment of systems interactions
  - human reliability analysis methodologies suitable for seismically induced hazards
  - multiunit risk considerations

Seismically-Induced Fires & Floods - Status

- Staff developed an initial plan for PRA method development in SECY 12-0025.
- Key Considerations
  - Limited number of staff with required knowledge, skills, and abilities
  - No current consensus state-of-practice methods exist for seismically induced fires and floods for NPPs
  - ASME/ANS Joint Committee on Nuclear Risk Management recently formed a working group to address multiple concurrent events
  - Other Tier 1 activities will provide substantial information relevant to this issue

Seismically-Induced Fires & Floods - Assessment

- Results from several Tier 1 recommendations will better inform the this issue (e.g., 2.1 – Seismic and flooding hazard evaluation, 4.2 – Mitigation strategies)
- More efficient to wait until sufficient information becomes available from these efforts.
- Some work can be done now:
  - Standards development organization engagement
  - Assess results from NTTF Recommendations 2.1, 4.2, 5.1, 7.1 and other activities
  - Continue PRA method development activities

Vogtle Level 3 PRA - Objectives

- Develop a Level 3 PRA, generally based on current state of practice methods, tools, and data,* that (1) reflects technical advances since completion of the NUREG-1150 studies, and (2) addresses scope considerations that were not previously considered (e.g., multi-unit risk)
- Extract new insights to enhance regulatory decisionmaking and to help focus limited agency resources on issues most directly related to the agency's mission to protect public health and safety
- Enhance NRC staff's PRA capability and expertise and improve documentation practices to make PRA information more accessible, retrievable, and understandable
- Obtain insight into the technical feasibility and cost of developing new Level 3 PRAs

* "State-of-practice" methods, tools, and data are those that are routinely used by the NRC and licensees or have acceptance in the PRA technical community.

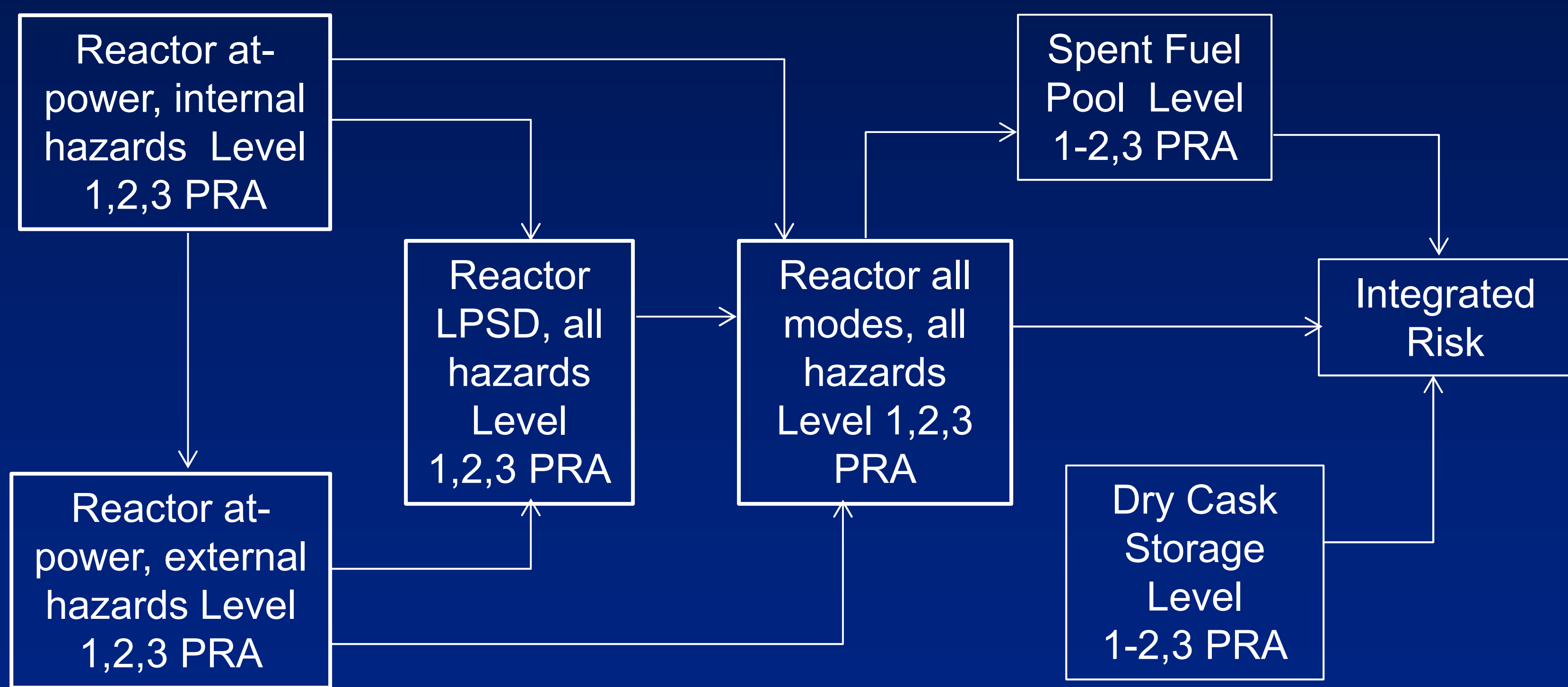
U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Vogtle Level 3 PRA - Scope

- In scope:
  - Reactor, SFP, (future) dry cask storage
  - All modes, all hazards
  - Multi-unit risk
- Not in scope:
  - aqueous transport and dispersion of radioactive materials
  - effects of aging on structure, system, and component reliability
  - consequential (linked) multiple initiating events (e.g., seismically induced fires and floods)
  - digital instrumentation and control, including software

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Vogtle Level 3 PRA - Approach



Vogtle Level 3 PRA – Points of Interest

- Computational tools:
  - SAPHIRE – PRA model
  - MELCOR - Accident progression modeling
  - Finite element modeling using LS-DYNA
  - Offsite consequence analysis using MACCS2
- Licensee model:
  - Internal hazards Level 1 PRA
  - Internal events (and major internal flood scenarios) Level 2 PRA; internal fires LERF PRA
  - Ongoing seismic modeling work

Vogtle Level 3 PRA - Activities

- Programmatic:
  - Technical Analysis Approach Plan
  - Public meeting – 11/28/12
  - ACRS Subcommittee – 12/4/12
- Approach:
  - Obtaining and reviewing information from SNC
  - Developing the Level 1 internal events model
  - Assembling necessary supporting pieces for future aspects, for example:
    - Developing reactor MELCOR model for Level 2 PRA and Level 1 PRA success criteria
    - Coordinating with licensee on supporting analyses for the seismic PRA

Confirmatory Success Criteria Analyses

- Use of MELCOR to explore selected Level 1 success criteria issues:
  - Enhance in-house expertise
  - Improve basis, or support change, for some SPAR success criteria
  - Provide additional analytical results for NRC risk analysts to use when working on SDP / ASP issues
- Previously completed analysis for Surry and Peach Bottom documented in NUREG-1953
  - Led to some changes in 3-loop Westinghouse high-head SPAR model (e.g., bleed and feed venting requirements) and BWR Mark 1 models (e.g., CRD credit)

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Success Criteria Analyses – NUREG/CR

- Impending NUREG/CR on underlying elements:
  - Analytical approximations to SBO time to core damage considering fuel heatup (rather than TAF)
  - One-off sensitivity analyses for 4 scenarios covering PWR/BWR, at-power/shutdown
  - Quantitative comparison of timing effects when selecting core damage surrogates
  - Investigation of timing to arrest fuel heatup for recovery actions
  - MELCOR/MAAP4 comparison for 2 LoMFW scenarios (using EPRI TR-1023032 MAAP4 results)
- To be issued for public comment in January 2013

Success Criteria Analyses – Ongoing work

- Current analysis – Byron Unit 1 (high head 4-loop, large/dry Westinghouse)
  - SBLOCA time to depressurize for low-head recirculation
  - SBLOCA time to depressurize for condensate feed
  - SBLOCA bleed & feed
  - Loss of main feedwater– bleed & feed
  - SGTR with failure to isolate and cooldown
  - MLOCA injection success criteria
  - MLOCA time to depressurize for low-head recirculation
  - Loss of shutdown cooling

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Success Criteria Analyses - Byron Results

- All results are <u>preliminary</u>
- E.g., LoMFW (LoDCB + independent DD-AFW failure)
  - Preliminary calculations support change in bleed and feed venting from 1 Charging*2 PORVs to 1 Charging*1 PORV for Byron (evaluation of applicability to other plants not performed yet)
  - Confirm existing treatment of SI venting capability (2 PORVs)
- E.g., Spontaneous SGTR with failure to isolate / cooldown
  - Provides range of time-to-overfill depending on leak rate
  - Provides updated timing estimates for recovery actions
- Will be documented in a NUREG to be issued for public comment in Spring 2013

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Some Recently-Issued Technical Reports

- NUREG-1921, “EPRI/NRC-RES Fire Human Reliability Analysis Guidelines”
- NUREG-2117, “Practical Implementation Guidelines for SSHAC Level 3 and 4 Hazard Studies,” Revision 1
- NUREG-2122, “Glossary of Risk-Related Terms in Support of Risk-Informed Decision Making” [ADAMS # ML121570620] - Draft
- NUREG-2125, “Spent Fuel Transportation Risk Assessment”
- NUREG-2128, “Electrical Cable Test Results and Analysis During Fire Exposure (ELECTRA-FIRE), A Consolidation of Three Major Fire-Induced Circuit and Cable Failure Experiments Performed Between 2001 and 2011,” Draft Report for Comment
- NUREG-2150, “A Proposed Risk Management Framework”
- NUREG/CR-7120, “Radionuclide Behavior in Soils and Soil-to-Plant Concentration Ratios for Assessing Food Chain Pathways“
- NUREG/CR-7123, “A Literature Review of the Effects of Smoke from a Fire on Electrical Equipment”

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Some Recently-Issued SECY Papers

- SECY-12-0110, “Consideration of Economic Consequences within the U.S. Nuclear Regulatory Commission’s Regulatory Framework”
- SECY-12-0123, “Update on Staff Plans to Apply the Full-Scope Site Level 3 PRA Project Results to the NRC’s Regulatory Framework”
- SECY-12-0133, “Status of the Accident Sequence Precursor Program and the Standardized Plant Analysis Risk Models”

Acronym List

- ABWR – Advanced Boiling Water Reactor
- ACRS – Advisory Committee on Reactor Safeguards
- ADAMS – Agencywide Document Access and Management System
- ALWR – Advanced Light-Water Reactor
- ANS – American Nuclear Society
- ASME – American Society of Mechanical Engineers
- ASP – Accident Sequence Precursor
- ASPDB – Accident Sequence Precursor Data Base
- BWR – Boiling Water Reactor
- CCCG – Common Cause Component Group
- CCF – Common Cause Failure
- CCDP – Conditional Core Damage Probability
- CDF – Core Damage Frequency
- CRD – Control Rod Drive (Hydraulic System)
- DD-AFW – Diesel-Drive Auxiliary Feedwater
- ECA – Event & Condition Assessment
- EPIX - Equipment Performance and Information Exchange
- EPRI – Electric Power Research Institute
- FTO – Fails to Open
- FTR – Fails to Run
- GE – General Electric
- INL – Idaho National Laboratory
- LCO – Limiting Condition of Operation
- LER – Licensee Event Report
- LERF – Large Early Release Frequency
- LoDCB – Loss of DC Bus
- LoMFW – Loss of Main Feedwater
- LOOP – Loss of Offsite Power
- LPSD – Low Power and Shutdown
- MAAP – Modular Accident Analysis Program
- MACCS2 – MELCOR Accident Consequence Code System 2
- MLOCA – Medium Loss of Coolant Accident
- MSPI – Mitigating System Performance Index
- NPP – Nuclear Power Plant
- NRC – Nuclear Regulatory Commission
- NTTF – Near Term Task Force
- P(a|b) – Probability that a occurs, given b occurs
- PORV – Pilot- (or Power) Operated Relief Valve
- PRA – Probabilistic Risk Analysis
- PWR – Pressurized Water Reactor
- RADS - Reliability and Availability Database System
- SAPHIRE - Systems Analysis Programs for Hands-on Integrated Reliability Evaluations
- SBLOCA – Small Loss of Coolant Accident
- SDP – Significance Determination Process
- SECY – a.k.a., Commission Paper
- SFP – Spent Fuel Pool
- SGTR – Steam Generator Tube Rupture
- SI – Safety Injection
- SNC – Southern Nuclear Company
- SPAR – Standardized Plant Analysis Risk
- SSCs – Structures, Systems and Components
- TAF – Top of Active Fuel
- TS – Technical Specification
- US-APWR – U.S. Advanced P0ressurized-Water Reactor