Tabletop Exercises on Industry's Proposed Prioritization Process

NRC Public Meeting
December 18-19, 2013

Tabletop Exercises on the Prioritization Process

1. SAMG & EOP Integration (NTTF Recommendation #8)
2. Extended Loss of AC Power and Associated FLEX
3. NTTF 2.1 Flooding Hazard Reevaluation
4. Cyber Security
5. Reliable Spent Fuel Pool Instrumentation

Tabletop Exercise #1
SAMG & EOP Integration
(NTTF Recommendation #8)

Ground Rules for Tabletop Exercise #1

Tabletop Exercise #1: Prioritization of SAMG & EOP Integration (NTTF Recommendation #8)	
Dates (tentative)	1. Generic characterization and assessment, December 18-19, 2013 2. Plant-specific prioritization, TBD
Location	NRC Rockville offices, Three White Flint North, Room 6A28
Time	TBD
Objective of tabletop	a) To exercise the draft guidance documents, b) To assess whether the process is *structured, robust, transparent, and straightforward,* and c) To identify areas for improvement in the guidance.
Scope of exercise	1. For the generic assessment, assess the issues associated with the NRC Proposed Rule and Regulatory Basis regarding Onsite Emergency Response Capabilities. Develop the generic characterization which will serve as input to the plant-specific evaluation(s). 2. Plant-specific prioritization, TBD
Industry guidance documents	1. Draft NEI *Guidelines for Prioritization and Scheduling Implementation* 2. Draft NEI *Guidance for Generic Assessment Expert Team (Initiative for Improving Nuclear Safety and Regulatory Efficiency)* 3. Draft NEI *Guidance for Integrated Decision Making Panel (Initiative for Improving Nuclear Safety and Regulatory Efficiency)*
Reference documents	1. Proposed Rule, Docket # NRC-2012-0031, Onsite Emergency Response Capabilities, 78 FR 68774, November 15, 2013. 2. USNRC, Onsite Emergency Response Capabilities, Regulatory Basis to Address Nuclear Regulatory Commission Near-Term Task Force Recommendation 8, October 1, 2013, (noticed as 78 FR 63901, October 25, 2013). 3. ACRS Subcommittee on Plant Operations and Fire Protection, transcripts of meeting on February 6, 2013 (ML13063A403). 4. NEI Anthony R. Pietrangelo comment on Draft Regulatory Basis, March 19, 2013 (ML13079A822). 5. NEI slides, Industry Perspective on NRC NTTF Recommendation 8 Proposed Rule and Regulatory Basis, November 19, 2013 (ML13330B717). 6. BWROG & PWROG slides, Update on Owners' Groups Activities – NTTF Recommendation 8, November 19, 2013 (ML13330B714).
Pre-tabletop activities	1. Participants should review the Proposed Rule and Regulatory Basis. 2. Participants should review the various industry documents including the status of Owners Group activities and comments on the Proposed Rule and Draft Regulatory Basis.
Tabletop activities	1. Discuss the tabletop exercise objective and desired outcomes 2. Use the guidance documents to characterize and prioritize the particular issue, proposed rule regarding Onsite Emergency Response Capabilities 3. Briefly document the results 4. Identify specific improvements to the guidance documents
Preliminary conclusions to draw from tabletop exercise	Determine a) whether guidance is adequately *structured, robust, transparent, and straightforward,* and b) whether the aids including tables, figures, screening questions, matrices, and worked examples are useful.
Lessons-learned	A list of the major lessons learned from the tabletop exercise should be carried forward to enhance the guidance documents and to improve future tabletop activities

Aspects of SAMG & EOP Integration to Evaluate

- Existing level of risk
- Benefits
- Defense-in-depth impact

SAMG & EOP Integration

From the Proposed Rule notice 78 FR 68774, Nov. 15, 2013:

1) Have strategies and guidance for mitigating the consequences of severe accidents
2) Integrate event and accident mitigating procedures
3) Identify command and control roles, responsibilities, and authorities during the progression of an event or accident
4) Conduct related drills, exercises or both
5) Provide training, and
6) Incorporate severe accident situations in written examinations and operating tests for all types of operators.

SAMG & EOP Integration (cont.)

From the Regulatory Basis for Onsite Emergency Response Capabilities, Oct. 1, 2013:

1. Accident mitigating procedures (EOPs, SAMGs, EDMGs, and supporting guidelines) were developed via separate initiatives. There is no regulatory requirement for a comprehensive strategy that ensures that these procedures work together as an integrated approach for responding to an event that progresses past design basis assumptions.

2. SAMGs and additional supporting guidelines are not required by NRC regulations. SAMGs exist as an industry commitment, and the supporting guidelines are in the development process.

SAMG & EOP Integration (cont.)

3. Licensees are not required to clearly identify roles, responsibilities and lines of decisionmaking authority (i.e., command and control structures) or processes for transferring responsibilities and authorities as conditions change in severe accident scenarios.
4. The NRC's regulations do not identify the training and qualifications necessary for key personnel relied upon to implement severe accident mitigating strategies.
5. Current regulations governing exercises do not require licensees to demonstrate implementation of all the procedures groups designed to address beyond design basis events during exercises.

SAMG & EOP Integration (cont.)

From BWROG & PWROG slides, Update on Owners' Groups Activities – NTTF Recommendation 8, November 19, 2013:

Owners' Group vs. Licensee Responsibilities - Examples

Owners' Group	Licensee
• Enhance generic SAMG guidance	• Revise site-specific SAMGs to incorporate enhanced guidance
• Develop generic FLEX support guidelines	• Develop site-specific FLEX Support guidelines
• Develop generic industry guidance for command and control	• Update command and control structure for onsite accident management

SAMG & EOP Integration (cont.)

Procedure Integration

- Integration of accident mitigation capabilities across appropriate procedures and guidelines:
  - Ensure no gaps in procedure guidance from event initiation through SAMG
  - Using capabilities across all procedures and guidelines
- Clear transitions between procedures and guidelines

SAMG & EOP Integration (cont.)

Command and Control Structure

- Command and Control structure is different for onsite versus offsite activities:
  - Command and Control for onsite activities may remain in the main control room or be transferred to the Technical Support Center
  - Command and Control for offsite activities may be transferred to the Emergency Operations Facility (EOF)
- An effective Command and Control structure should be based on "key functions" rather than locations:
  - Any transfer of Command and Control from one person to another should be clearly identified

SAMG & EOP Integration (cont.)

Programmatic Controls

- Elements common to all accident management procedures/guidance (EOP, AOP, FSG, EDMG & SAMG):
  a) Start with technical basis development:
     - Generic or plant specific technical bases
  b) Process for development of procedures and guidance:
     - Plant specific implementation
  c) Verification and Validation of procedures and guidance:
     - Objectives, methods, and disposition of findings
     - Transitions
  d) Maintenance of procedures and guidance:
     - Document control, ongoing assessment, and updating for new information

SAMG & EOP Integration (cont.)

Programmatic Controls

- Programmatic control elements that should remain different between EOP, AOP, FSG, EDMG or SAMG:
  - Level of detail and rigor of technical bases and guidance
  - Requirements in regulations and best practices
  - Methods available for Verification and Validation

SAMG & EOP Integration (cont.)

From ACRS Subcommittee meeting transcripts of 2/6/13:

Industry-wide cost estimates:

– $16 Million for procedure upgrades

– $17.75 Million for training program implementation

SAMG & EOP – Step 1

Does the proposed activity or issue:

1. ☐ YES ☐ NO Result in an impact on the frequency of occurrence of a risk significant accident initiator?

Justification:

2. ☐ YES ☐ NO Result in an impact in the availability, reliability, or capability of SSCs and personnel relied upon to mitigate a risk significant transient, accident, or natural hazard?

Justification:

3. ☐ YES ☐ NO Result in an impact in the consequences of a risk significant accident sequence?

Justification:

SAMG & EOP – Step 1 (cont.)

4. ☐ YES ☐ NO Result in an impact in the capability of a fission product barrier?

Justification:

5. ☐ YES ☐ NO Result in an impact in defense-in-depth capability?

Examples include:

a. Strengthen balance of accident prevention and mitigation
b. Reduce reliance on programmatic activities
c. Reduce probability of common-cause failures

Justification:

If ALL the responses are NO and Confidence is sufficient, issue or activity has NO IMPACT and screens out (DROP) STOP.

If any of the responses is YES, then continue to **Step 2** or develop a plan.

Step 2 (More than Minimal Assessment):

Does the proposed activity or issue:

1. ☐ YES ☐ NO Result in more than a minimal decrease in frequency of occurrence of a risk significant accident initiator?

Justification:

2. ☐ YES ☐ NO Result in more than a minimal improvement in the availability, reliability, or capability of SSCs and personnel relied upon to mitigate a risk significant transient, accident, or natural hazard?

Justification:

3. ☐ YES ☐ NO Result in more than a minimal decrease in the consequences of a risk significant accident sequence?

Justification:

Step 2 (cont.):

4. ☐ YES ☐ NO Result in more than a minimal improvement in the capability of a fission product barrier?

Justification:

5. ☐ YES ☐ NO Result in more than a minimal improvement in defense-in-depth capability? Examples include:
   a. Strengthen balance of accident prevention and mitigation
   b. Reduce reliance on programmatic activities
   c. Reduce probability of common-cause failures

Justification:

If ALL the responses are NO and Confidence is sufficient, issue or activity screens to MINIMAL IMPACT.

If any of the responses is YES, then continue to **Step 3** or develop a plan.

Guidance for Step 2, Question 1:

Accident Initiator Categories (Representative)	Risk Significant?	More than Minimal Decrease ?
Transients initiated by frontline systems		
Transients initiated by support systems		
Primary system integrity loss (e.g. SGTR, RCP seal LOCA, LOCA)		
Secondary system integrity loss		
Internal flooding		
Internal fires		
Earthquakes		
External flooding		
Tornados and High Winds		
Other External Hazards		

Guidance for Step 2, Question 2:

Considerations	Potential Action Effect?	More than Minimal Improvement?
Changes in maintenance, training		
Changes in specific SSCs (e.g., installing a more reliable component)		
Changes in materials		
Equipment replacements to address age related degradation		
Changes in redundancy and diversity		
Additional of equipment		
Changes in operating practices		

Step 3A Qualitative Assessment

Matrix by Current Risk and Potential Impact

UB is upper bound of the risk range; Mid is "mid-range" (0.3 times UB); LB is factor of 10 lower than UB

Current Risk associated with Issue *Note: Address the specific issue first; then assess impacts on other risk contributors potentially impacted*	**Potential Impact of Action (Reduction in Risk)**					
	None	Very Small/Minimal	Small	Medium	High	Comments
	0%	0-25%	25-50%	50% to 90%	>90%	Can adjust these initial ranges as appropriate
	Outcome Note: Quantitative values are delta CDF/LERF					
Green (VL) LB	<VL/Green	<VL/Green	<VL/Green	<VL/Green	<VL/Green	No change
Green (VL) Mid	VL/Green	VL/Green	VL/Green	VL/Green	VL/Green	No change
Green (VL) UB	VL/Green	VL/Green	VL/Green	VL/Green	VL/Green	Maximum reduction is 1E-6/1E-7
White (L) LB	VL/Green	VL/Green	VL/Green	VL/Green	VL/Green	
White (L) Mid	VL/Green	VL/Green	L/White	L/White	L/White	White above 25% Category
White (L) UB	VL/Green	L/White	L/White	L/White	L/White	Maximum reduction is 1E-5/1E-6
Yellow (M) LB	VL/Green	L/White	L/White	L/White	L/White	
Yellow (M) Mid	VL/Green	L/White	M/Yellow	M/Yellow	M/Yellow	Only change is 25% Category
Yellow (M) UB	VL/Green	M/Yellow	M/Yellow	M/Yellow	M/Yellow	Maximum reduction is 1E-4/1E-5
Red (H) LB	?	M/Yellow	M/Yellow	M/Yellow	M/Yellow	Addressed by upper bound Yellow
Red (H) Mid	?	H/Red	H/Red	H/Red	H/Red	
Red (H) UB	?	H/Red	H/Red	H/Red	H/Red	

Step 3A Worksheet

Existing Level of Risk

From the Regulatory Basis Document:

- In SECY-11-0137 the NRC staff determined that licensee procedures and guidelines already existed for severe accidents and are available for operator use, therefore no imminent hazard was identified.

Step 3A Worksheet (cont.)

Potential Risk Reduction

From Regulatory Basis Document:

- In the State-of-the-Art Reactor Consequence Analysis (SOARCA) each accident was modeled twice, once assuming the accident progressed unmitigated and again assuming that mitigation (including use of EOPs, SAMGs, and EDMGs) was successfully implemented. The results confirmed that time would be available to implement the accident mitigating procedures, and that, if successfully implemented, the accident mitigating procedures and guidelines would be effective in preventing core damage or significantly reducing radiological releases. Further, the analysis determined that implementation of the additional equipment associated with EDMGs would be especially helpful in counteracting station blackout scenarios.

Step 3A Worksheet (cont.)

From Pietrangelo (NEI) letter to NRC on 3/19/13:

- Deferral of this rulemaking would in no way slow industry efforts to develop and implement improvements to the SAMGs…The SAMG changes in progress will incorporate the FLEX and EDMG equipment to further integrate the response capabilities.

- Severe accident response cannot be mechanistically prescribed in detailed procedures. The transition from emergency operating "procedures" to FLEX support and severe accident "guidance" is deliberate and necessary to preserve the capability to address accident scenarios with necessary flexibility to deal with uncertainties in accident progression.

Step 3A Worksheet (cont.)

From Pietrangelo (cont.):

- As noted in NEI's ANPR comments, licensees already have in place emergency response command and control structures that reflect the emergency preparedness planning standards of 10 CFR 50.47(b), the requirements of 10 CFR 50, Appendix E, and related guidance (e.g., NUREG-0654). These structures also incorporate the decisionmaking capabilities necessary to implement SAMGs as recommended by the appropriate Owners Group. Separate, event-based command and control structures would unnecessarily complicate response efforts, and potentially introduce confusion among licensee and offsite authority responders.

- The operator training and qualification expectations need to be carefully considered and balanced with respect to the very remote likelihood of a beyond design-basis event versus the potential dilution of operator focus from more risk significant plant transients and accidents.

Step 3A Worksheet (cont.)

Current risk level:

Potential % reduction in risk:

Priority:

Tabletop Exercise #2
Extended Loss of AC Power (ELAP) and Associated FLEX

Ground Rules for Tabletop Exercise #2

Tabletop Exercise #2: Prioritization of Extended Loss of AC Power and Associated FLEX Aspects	
Dates (tentative)	1. Generic characterization and assessment, December 18-19, 2013 2. Plant-specific prioritization, TBD
Location	NRC Rockville offices, Three White Flint North, Room 6A28
Time	TBD
Objective of tabletop	a) To exercise the draft guidance documents, b) To assess whether the process is *structured, robust, transparent, and straightforward,* c) To identify areas for improvement in the guidance.
Scope of exercise	1. For the generic assessment, assess the issues associated with the NRC Order EA-12-049 regarding, specifically, extended loss of AC power mitigation strategies, and the associated NEI 12-06 FLEX implementation. Develop the generic characterization which will serve as input to the plant-specific evaluation(s).
Industry guidance documents	1. Draft NEI *Guidelines for Prioritization and Scheduling Implementation* 2. Draft NEI *Guidance for Generic Assessment Expert Team (Initiative for Improving Nuclear Safety and Regulatory Efficiency)* 3. Draft NEI *Guidance for Integrated Decision Making Panel (Initiative for Improving Nuclear Safety and Regulatory Efficiency)*
Reference documents	1. ACRS Letter Report, "Proposed Rulemaking on Station Blackout Mitigation Strategies," June 17, 2013 (ML13161A247) 2. NRC Order EA-12-049, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," (ML12054A735) 3. NEI 12-06 Rev. 0, "Diverse and Flexible Coping Strategies (Flex) Implementation Guide" 4. JLD-ISG-2012-01, "Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," (ML12229A174) 5. NRC-2011-0299-0065, "Station Blackout Mitigation Strategies, Regulatory Basis Document" 6. NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program" 7. Emergency AC power and Alternate AC (AAC) power design information and any relevant Station Blackout (SBO) risk insights from the PRA due to externally initiated events (plant-specific only)
Pre-tabletop activities	1. Participants should have an overview of the qualitative and quantitative risk insights in NUREG-1742 to become familiar with aspects of the existing level of risk associated with beyond-design-basis SBO due to externally initiated events. Additional risk insights are available from NRC's Regulatory Basis Document, NRC-2011-0299-0065. 2. Participants should review the NRC Order EA-12-049, NEI 12-06 Rev. 0 FLEX guidance, and NRC Interim Staff Guidance JLD-ISG-2012-01 to understand the scope, requirements, and guidance for SBO mitigation strategies. Additional discussion can be found in the ACRS Letter Report of June 17, 2013. 3. For the plant-specific prioritization, participants should have information regarding plant-specific emergency AC power and AAC power design, including implementation of relevant aspects of NEI 12-06 on FLEX.
Tabletop activities	1. Discuss the tabletop exercise objective and desired outcomes 2. Use the guidance documents to characterize and prioritize the particular issue, Extended Loss of AC Power and the associated FLEX aspects per NEI 12-06, using Steps 1, 2 and 3A 3. Briefly document the results 4. Identify specific improvements to the guidance documents
Preliminary conclusions to draw from tabletop exercise	Determine a) whether guidance is adequately *structured, robust, transparent, and straightforward,* and b) whether the aids including tables, figures, screening questions, matrices, and worked examples are useful.
Lessons-learned	A list of the major lessons learned from the tabletop exercise should be carried forward to enhance the guidance documents and to improve future tabletop activities

Aspects of ELAP and FLEX to Evaluate

- Existing level of risk
  - CDF and LERF
- Benefits and potential reduction in risk (ΔCDF and ΔLERF)
- Defense-in-depth impact

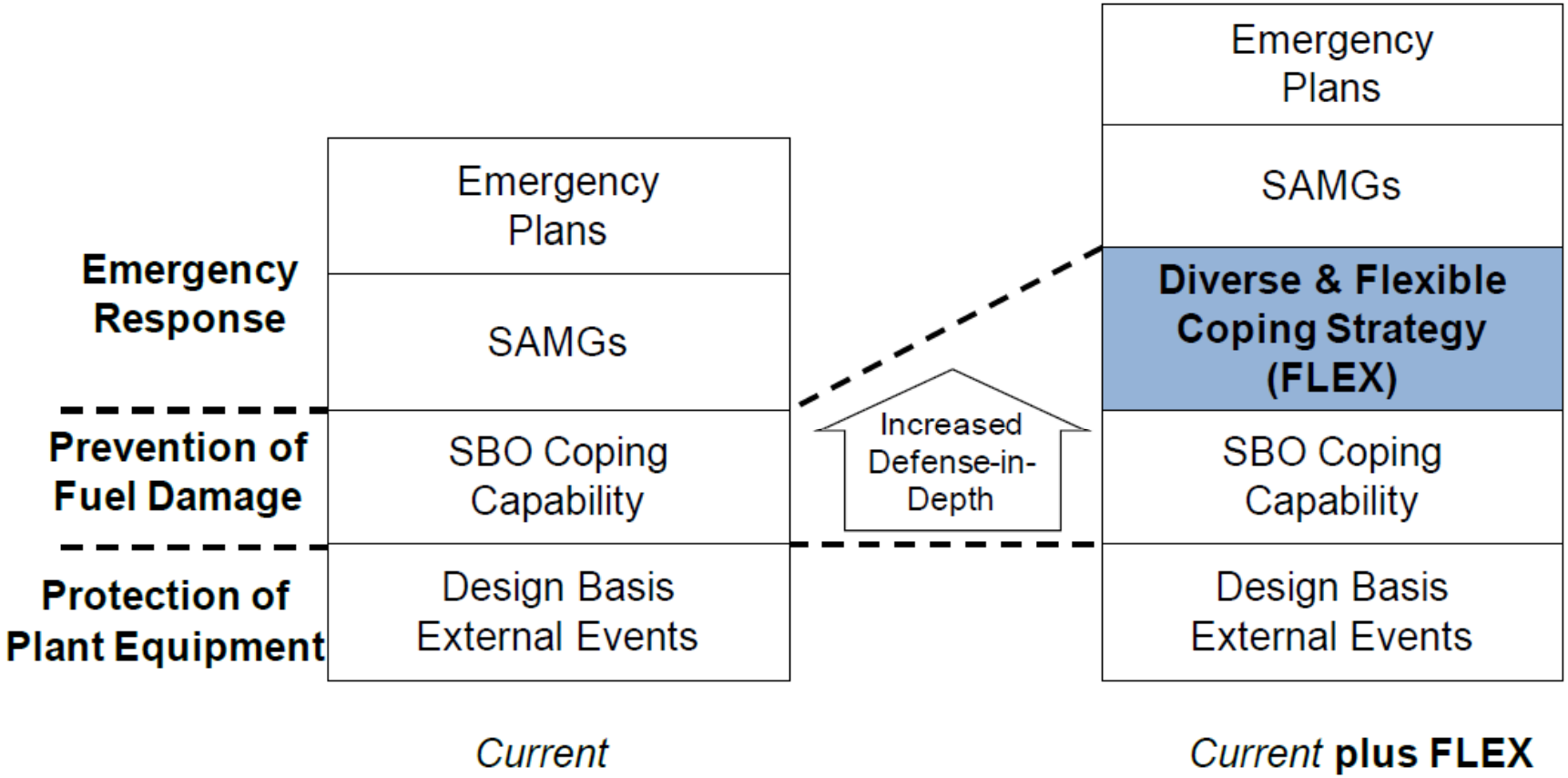
ELAP and FLEX

From the NRC Regulatory Basis Document for Station Blackout Mitigation Strategies:

- Three-phase approach for mitigating beyond-design-basis external events that lead to an extended loss of AC power and loss of normal access to the ultimate heat sink (UHS) condition.
  1. Use of installed equipment and resources
  2. Provision of sufficient, portable, onsite equipment and consumables to maintain or restore critical functions
  3. Obtain sufficient offsite resources to sustain those functions indefinitely.

From NEI 12-06

Figure 1-1
FLEX Enhances Defense-in-Depth



NEI 12-06 (Cont.)

- This capability will address both an ELAP (i.e., loss of off-site power, emergency diesel generators and any alternate AC source but not the loss of AC power to buses fed by station batteries through inverters) and a loss of UHS which could arise following external events that are within the existing design basis with additional failures and conditions that could arise from a beyond-design-basis external event.

- The loss of off-site power is generally attributed to damage to the grid and/or on-site power transmission equipment that is essentially unrecoverable in the near-term.

NEI 12-06 (Cont.)

- The use of portable equipment to charge batteries or locally energize equipment may be needed under ELAP/LUHS conditions. Appropriate electrical isolations and interactions should be addressed in procedures/guidance.

- Sufficient equipment to address all functions at all units on-site, plus one additional spare, i.e., an N+1 capability, where "N" is the number of units on-site.

- Longer term strategies may be developed to prolong Phase 1 coping that will allow greater reliance on permanently installed, bunkered or hardened ac power supplies that are adequately protected from external events.

ELAP and FLEX – Step 1

Does the proposed activity or issue:

1. ☐ YES ☐ NO Result in an impact on the frequency of occurrence of a risk significant accident initiator?

Justification:

2. ☐ YES ☐ NO Result in an impact in the availability, reliability, or capability of SSCs and personnel relied upon to mitigate a risk significant transient, accident, or natural hazard?

Justification:

3. ☐ YES ☐ NO Result in an impact in the consequences of a risk significant accident sequence?

Justification:

ELAP and FLEX – Step 1 (cont.)

4. ☐ YES ☐ NO Result in an impact in the capability of a fission product barrier?

Justification:

5. ☐ YES ☐ NO Result in an impact in defense-in-depth capability?

Examples include:

a. Strengthen balance of accident prevention and mitigation
b. Reduce reliance on programmatic activities
c. Reduce probability of common-cause failures

Justification:

If ALL the responses are NO and Confidence is sufficient, issue or activity has NO IMPACT and screens out (DROP) STOP.

If any of the responses is YES, then continue to **Step 2** or develop a plan.

Step 2 (More than Minimal Assessment):

Does the proposed activity or issue:

1. ☐ YES ☐ NO Result in more than a minimal decrease in frequency of occurrence of a risk significant accident initiator?

Justification:

2. ☐ YES ☐ NO Result in more than a minimal improvement in the availability, reliability, or capability of SSCs and personnel relied upon to mitigate a risk significant transient, accident, or natural hazard?

Justification:

3. ☐ YES ☐ NO Result in more than a minimal decrease in the consequences of a risk significant accident sequence?

Justification:

Step 2 (cont.):

4. ☐ YES ☐ NO Result in more than a minimal improvement in the capability of a fission product barrier?

Justification:

5. ☐ YES ☐ NO Result in more than a minimal improvement in defense-in-depth capability? Examples include:
   a. Strengthen balance of accident prevention and mitigation
   b. Reduce reliance on programmatic activities
   c. Reduce probability of common-cause failures

Justification:

If ALL the responses are NO and Confidence is sufficient, issue or activity screens to MINIMAL IMPACT.

If any of the responses is YES, then continue to **Step 3** or develop a plan.

Guidance for Step 2, Question 1:

Accident Initiator Categories (Representative)	Risk Significant?	More than Minimal Decrease ?
Transients initiated by frontline systems		
Transients initiated by support systems		
Primary system integrity loss (e.g. SGTR, RCP seal LOCA, LOCA)		
Secondary system integrity loss		
Internal flooding		
Internal fires		
Earthquakes		
External flooding		
Tornados and High Winds		
Other External Hazards		

Guidance for Step 2, Question 2:

Considerations	Potential Action Effect?	More than Minimal Improvement?
Changes in maintenance, training		
Changes in specific SSCs (e.g., installing a more reliable component)		
Changes in materials		
Equipment replacements to address age related degradation		
Changes in redundancy and diversity		
Additional of equipment		
Changes in operating practices		

Step 3A Qualitative Assessment

Matrix by Current Risk and Potential Impact

UB is upper bound of the risk range; Mid is "mid-range" (0.3 times UB); LB is factor of 10 lower than UB

Current Risk associated with Issue *Note: Address the specific issue first; then assess impacts on other risk contributors potentially impacted*	**Potential Impact of Action (Reduction in Risk)**					
	None	Very Small/Minimal	Small	Medium	High	Comments
	0%	0-25%	25-50%	50% to 90%	>90%	Can adjust these initial ranges as appropriate
	Outcome Note: Quantitative values are delta CDF/LERF					
Green (VL) LB	<VL/Green	<VL/Green	<VL/Green	<VL/Green	<VL/Green	No change
Green (VL) Mid	VL/Green	VL/Green	VL/Green	VL/Green	VL/Green	No change
Green (VL) UB	VL/Green	VL/Green	VL/Green	VL/Green	VL/Green	Maximum reduction is 1E-6/1E-7
White (L) LB	VL/Green	VL/Green	VL/Green	VL/Green	VL/Green	
White (L) Mid	VL/Green	VL/Green	L/White	L/White	L/White	White above 25% Category
White (L) UB	VL/Green	L/White	L/White	L/White	L/White	Maximum reduction is 1E-5/1E-6
Yellow (M) LB	VL/Green	L/White	L/White	L/White	L/White	
Yellow (M) Mid	VL/Green	L/White	M/Yellow	M/Yellow	M/Yellow	Only change is 25% Category
Yellow (M) UB	VL/Green	M/Yellow	M/Yellow	M/Yellow	M/Yellow	Maximum reduction is 1E-4/1E-5
Red (H) LB	?	M/Yellow	M/Yellow	M/Yellow	M/Yellow	Addressed by upper bound Yellow
Red (H) Mid	?	H/Red	H/Red	H/Red	H/Red	
Red (H) UB	?	H/Red	H/Red	H/Red	H/Red	

Step 3A Worksheet

Existing Level of Risk

From Regulatory Basis Document :

- NUREG-1776 concluded that 10 CFR 50.63 resulted in a substantial reduction in the mean CDF associated with station blackout events from 4.2E-05 /RY to 1E-05 /RY.

- NUREG/CR-6890 concluded that SBO risk was at an average CDF of 3E-06 per reactor critical year, with a range from 6.6E-07 to 5.3E-06. The analysis considered risk associated with internal events only; as such, seismic, flood and fire events were excluded. However, the effects of severe weather events resulting in a LOOP were considered.

Step 3A Worksheet (cont.)

- The staff has concluded that the remaining known risk associated with station blackout is below a level warranting action under the adequate protection provisions of 10 CFR 50.109(a)(4). However, this conclusion is subject to change based on the feedback and lessons-learned from implementation of EA-12-049, NTTF Recommendation 2.3, and NTTF Recommendation 2.1 activities.

From NEI 12-06:

- Beyond-design-basis seismic events have been extensively studied in seismic margin assessments (SMAs) and seismic PRAs (SPRAs). These studies have demonstrated that an ELAP is a dominant contributor to seismic risk.

Step 3A Worksheet (cont.)

From NRC Generic Letter 2011-XX: Seismic Risk Evaluations for Operating Reactors, Appendix D, Seismic Core Damage Frequencies, 2008 USGS weakest link (ML100270756):

U.S. average seismic CDF ~ 2E-5 /yr

Step 3A Worksheet (cont.)

From NUREG-1742, IPEEE Insights:

About half of the contributors listed in Table 2.2 for seismic failure involve the failure of the electrical systems, which includes the failure of the offsite power (17% of all contributors listed in Table 2.2); the failure of various components of the electrical system (17%), such as MCCs, load centers, switchgear, and relays; the failure of the emergency diesel generator (EDG) (8%); and the failure of the dc batteries (5%).

Step 3A Worksheet (cont.)

From NRC Handout on Filtering Strategies Rulemaking 12/12/13:

Draft Site-Specific ELAP Frequencies

Site	Containment Type	EPS Class	SBO Coping Time	Weather Related ELAP	Seismic ELAP
Browns Ferry	Mark I	4	4	2.0E-07	1.4E-07
Brunswick	Mark I	2	4	3.6E-06	2.1E-06
Columbia	Mark II	2	4	3.6E-06	1.8E-06
Cooper	Mark I	2	4	3.6E-06	7.4E-07
Dresden	Mark I	4	4	2.0E-07	3.6E-07
Duane Arnold	Mark I	2	4	3.6E-06	4.5E-07
Fermi	Mark I	4	4	2.0E-07	1.6E-07
FitzPatrick	Mark I	4	4	2.0E-07	1.5E-07
Hatch	Mark I	3	4	1.8E-06	9.9E-08
Hope Creek	Mark I	3	4	1.8E-06	4.5E-07
La Salle	Mark II	3	4	1.8E-06	8.0E-07
Limerick	Mark II	4	4	2.0E-07	7.2E-07
Monticello	Mark I	2	4	3.6E-06	3.7E-07
Nine Mile Point	Mark I/II	2	4	3.6E-06	3.7E-07
Oyster Creek	Mark I	2	4	3.6E-06	6.7E-07
Peach Bottom	Mark I	3	8	1.4E-06	8.1E-07
Pilgrim	Mark I	2	8	2.8E-06	2.0E-06
Quad Cities	Mark I	4	4	2.0E-07	1.4E-07
Susquehanna	Mark II	3	4	1.8E-06	4.8E-07
Vermont Yankee	Mark I	2	8	2.8E-06	6.2E-07

Step 3A Worksheet (cont.)

Potential Risk Reduction

From Regulatory Basis Document:

- EA-12-049 requires strategies with increased capability to implement protective actions concurrently at multiple units at a site. The strategies (currently being implemented) are intended to add multiple ways to maintain or restore core cooling, containment, and SFP cooling capabilities in order to improve the defense-in-depth of licensed nuclear power reactors.

- In addition, the strategies and guidance enhance the safety and preparedness capabilities established following September 11, 2001, and made generically applicable in 10 CFR 50.54(hh)(2). In order to address the potential for more widespread effects of beyond-design-basis external events, EA-12-049 requires strategies with increased capability to implement protective actions concurrently at multiple units at a site.

Step 3A Worksheet (cont.)

From ACRS letter report of June 17, 2013 (ML13161A247):

- The mitigating strategies requirements would follow an approach similar to Order EA-12-049. The order and the new rule are intended to provide a substantial increase in defense-in-depth protection against station blackouts that extend beyond a plant's coping time with current equipment and strategies. Many of the mitigating strategies will depend on portable equipment that is not permanently installed. Neither NEI 12-06 nor JLD-ISG-2012-01 provide sufficient guidance on the evaluation of the feasibility and reliability of the manual actions needed to implement the mitigating strategies. The guidance for the new mitigating strategies rule should address this issue. NUREG-1852 discusses relevant methods for the evaluation of manual actions.

Step 3A Worksheet (cont.)

Current risk level:

Potential % reduction in risk:

Priority:

Tabletop Exercise #3
NTTF 2.1 Flooding Hazard Reevaluation

Ground Rules for Tabletop Exercise #3

Tabletop Exercise #3: Prioritization of NTTF Recommendation 2.1Flooding Hazard	
Dates (tentative)	1. Generic characterization and assessment, December 18-19, 2013 2. Plant-specific prioritization, TBD
Location	NRC Rockville offices, Three White Flint North, Room 6A28
Time	TBD
Objective of tabletop	a) To exercise the draft guidance documents, b) To assess whether the process is *structured, robust, transparent, and straightforward,* c) To identify areas for improvement in the guidance.
Scope of exercise	1. For the generic assessment, assess the issues associated with flooding hazard reevaluations per Recommendation 2.1 of the Near-Term Task Force (NTTF) review of insights from the Fukushima Dai-ichi accident. Develop the generic characterization which will serve as input to the plant-specific evaluation(s).
Industry guidance documents	1. Draft NEI *Guidelines for Prioritization and Scheduling Implementation* 2. Draft NEI *Guidance for Generic Assessment Expert Team (Initiative for Improving Nuclear Safety and Regulatory Efficiency)* 3. Draft NEI *Guidance for Integrated Decision Making Panel (Initiative for Improving Nuclear Safety and Regulatory Efficiency)*
Reference documents	1. NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program," Section 4.3 External Floods 2. NRC Information Notice 2012-02, "Potentially Nonconservative Screening Value for Dam Failure Frequency in Probabilistic Risk Assessments" (ML090510269) 3. NRC Generic Issue 204, "Flooding of Nuclear Power Plant Sites Following Upstream Dam Failure," Screening Analysis (ML113500495) 4. Fukushima Flooding Task Force (FFTF), "Industry Flood Hazard Evaluation Prioritization Scheme," March 15, 2012 (ML12075A048) 5. FFTF, "Completion Schedule for Flooding Reevaluations," May 3, 2012 (ML12125A331) 6. NRC 50.54(f) letters on NTTF 2.1 flooding hazard reevaluation dated 3/12/12 (ML12053A340) and 5/11/12 (ML12097A509) 7. JLD-ISG-2012-05, "Guidance for Performing the Integrated Assessment for External Flooding," Rev. 0
Pre-tabletop activities	1. Participants should review the NRC 50.54(f) letters related to flooding hazard reevaluation, JLD-ISG-2012-05, as well as industry's proposed methodology and completion schedule. 2. Participants should have an overview of the qualitative and quantitative risk insights in NUREG-1742 Section 4.3 to become familiar with aspects of the existing level of risk associated with external flooding. Additional risk insights are available from NRC's screening analysis on GI-204. 3. For the plant-specific prioritization, participants should have information regarding plant-specific external flooding hazards, flood protection features and procedures, and PRA insights (if available).
Tabletop activities	1. Discuss the tabletop exercise objective and desired outcomes 2. Use the guidance documents to characterize and prioritize the particular issue, flooding hazard reevaluations for NTTF Recommendation 2.1. 3. Briefly document the results 4. Identify specific improvements to the guidance documents
Preliminary conclusions to draw from tabletop exercise	Determine a) whether guidance is adequately *structured, robust, transparent, and straightforward,* and b) whether the aids including tables, figures, screening questions, matrices, and worked examples are useful.
Lessons-learned	A list of the major lessons learned from the tabletop exercise should be carried forward to enhance the guidance documents and to improve future tabletop activities

Aspects of NTTF 2.1
Flooding Hazard to Evaluate

- Existing level of risk from external flooding
- Benefits and potential reduction in risk
- Plant categorization
- Defense-in-depth impact

NTTF 2.1 Flooding Hazard

From the NRC 50.54(f) letter of 3/12/12:

- Failure to protect SSCs important to safety from natural phenomena with appropriate safety margins has the potential to result in common-cause failures with significant consequences, as was demonstrated at Fukushima. Additionally, the consequences of an accident from natural phenomena may be aggravated by a "cliff-edge" effect, in that a small increase in hazard (e.g., flooding level) may sharply increase the number of SSCs affected.

- As the state of knowledge of these hazards has evolved significantly since the licensing of many of the plants within the U.S., and given the demonstrated consequences from Fukushima, it is necessary to confirm the appropriateness of the hazards assumed for U.S. plants and their ability to protect against them.

NTTF 2.1 Flooding Hazard (cont.)

New Requirements for Evaluation of Dam Hazards in 10 CFR 100.20:

The staff established a new requirement in 10 CFR 100.20, "Factors to be Considered when Evaluating Sites," in 1996. The requirement in 10 CFR 100.20(b) states that for applications submitted on or after January 10, 1997, the nature and proximity of man-related hazards must be evaluated to establish site parameters for use in determining whether a plant design can accommodate commonly occurring hazards, and whether the risk of other hazards is very low. A parenthetical statement in the new regulation specifically identifies dams as hazards to be evaluated at a plant site.

Tsunami and Regulatory Guide 1.59 Updates:

Following the Sumatra earthquake and its accompanying tsunami in December 2004, the NRC staff initiated a study to examine tsunami hazards at power plant sites. Study results are documented in NUREG/CR-6966, "Tsunami Hazard Assessment at Nuclear Power Plant Sites in the United States of America," which was published in March 2009. As the NTTF report notes, "while tsunami hazards are not expected to be the limiting flood hazard for operating plants sited on the Atlantic Ocean and the Gulf of Mexico, plants in these coastal regions do not currently include an analysis of tsunami hazards in their licensing basis."

NTTF 2.1 Flooding Hazard

From JLD-ISG-2012-05:

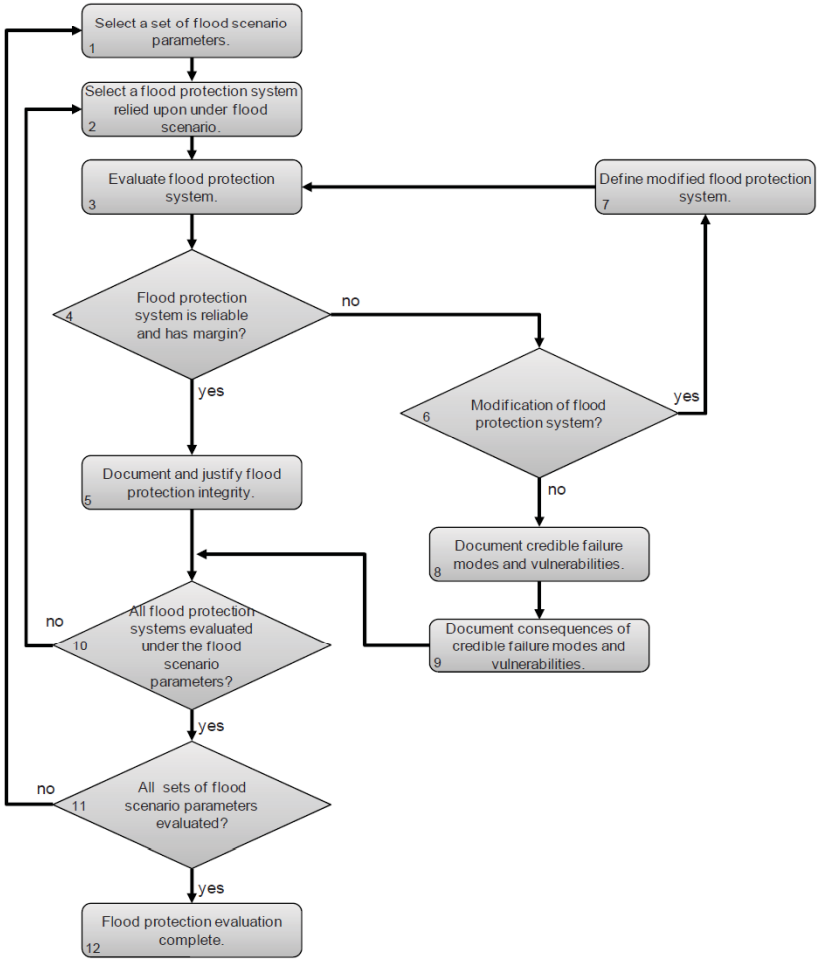


Figure 3: Flood protection evaluation process flowchart

Industry Flood Hazard Evaluation Prioritization Scheme

License Vintage (select one)	**Rating**	**Co-Located Sites** (select one)	**Rating**
1969 - 1975	13	Yes	3
1976 - 1984	8	No	6
1985 - 1996	3		
		Recent Evaluations, after 1995 (select one)	**Rating**
Location/Hazard (select all that apply)	**Rating**	Comprehensive/Detailed	2
River	5	Mixed Detailed/Approx.	6
Lake	3	Nothing Recent	10
Coastal	7		
Upstream Dams	8	**Size of Watershed** (select one)	**Rating**
Onsite Reservoir	3	> 20,000 sq. miles	10
Local Intense Precipitation	2	< 20,000 sq. miles	3
Location/Hazard Summation	2 to 28	Not applicable	0
Licensing Basis (select one)	**Rating**		
Wet	14		
Dry	4		
	Category 1	59 to 81	
	Category 2	37 to 58	
	Category 3	14 to 36	

Table 1 - Categorization Rating System

Category 1 – Most complex flooding evaluation
Category 2 – Average evaluation complexity
Category 3 – Simplest evaluation

Industry's Proposed Completion Schedule

1 Year Completion	2 Year Completion	3 Year Completion
Site	Site	Site
Callaway	Arkansas Nuclear One 1 & 2	Bellefonte *
Calvert Cliffs 1 & 2	Beaver Valley 1&2 *	Brunswick
Comanche Peak 1 & 2	Braidwood 1 & 2	Cooper *
Dresden 2 & 3	Brown's Ferry 1, 2, & 3	Crystal River 3
Fermi 2	Byron 1 & 2	DC Cook 1 & 2
Grand Gulf	Catawba 1 & 2	Diablo Canyon 1 & 2
Hope Creek	Clinton	FitzPatrick
Indian Point 2 & 3	Columbia *	Ft Calhoun *
Nine Mile Point 1 & 2	Davis Besse 1	Ginna
North Anna 1 & 2	Duane Arnold	Millstone 2 & 3
Oconee 1, 2, & 3	Farley 1 & 2	Oyster Creek
Quad Cities 1 & 2	Hatch 1 & 2	Palisades
Salem 1 & 2	Kewaunee	Perry
Shearon Harris 1	LaSalle 1 & 2	Pilgrim
South Texas Project 1 & 2	Limerick 1 & 2	Point Beach 1 & 2
Summer	McGuire 1 & 2	St Lucie 1 & 2
TMI 1	Monticello	San Onofre 2 & 3
Turkey Point 3 & 4	Palo Verde 1, 2, & 3	Seabrook
Vermont Yankee	Peach Bottom 2 & 3 *	Surry 1 & 2
Vogtle 1 & 2	Prairie Island 1 & 2 *	Waterford 3
Watts Bar 1 & 2	River Bend 1	
	Robinson *	
	Sequoyah 1 & 2	
	Susquehanna 1 & 2	
	Wolf Creek	

Flooding Hazard – Step 1

Does the proposed activity or issue:

1. ☐ YES ☐ NO Result in an impact on the frequency of occurrence of a risk significant accident initiator?

Justification:

2. ☐ YES ☐ NO Result in an impact in the availability, reliability, or capability of SSCs and personnel relied upon to mitigate a risk significant transient, accident, or natural hazard?

Justification:

3. ☐ YES ☐ NO Result in an impact in the consequences of a risk significant accident sequence?

Justification:

Flooding Hazard – Step 1 (cont.)

4. ☐ YES ☐ NO Result in an impact in the capability of a fission product barrier?

Justification:

5. ☐ YES ☐ NO Result in an impact in defense-in-depth capability?

Examples include:

a. Strengthen balance of accident prevention and mitigation
b. Reduce reliance on programmatic activities
c. Reduce probability of common-cause failures

Justification:

If ALL the responses are NO and Confidence is sufficient, issue or activity has NO IMPACT and screens out (DROP) STOP.

If any of the responses is YES, then continue to **Step 2** or develop a plan.

Step 2 (More than Minimal Assessment):

Does the proposed activity or issue:

1. ☐ YES ☐ NO Result in more than a minimal decrease in frequency of occurrence of a risk significant accident initiator?

Justification:

2. ☐ YES ☐ NO Result in more than a minimal improvement in the availability, reliability, or capability of SSCs and personnel relied upon to mitigate a risk significant transient, accident, or natural hazard?

Justification:

3. ☐ YES ☐ NO Result in more than a minimal decrease in the consequences of a risk significant accident sequence?

Justification:

Step 2 (cont.):

4. ☐ YES ☐ NO Result in more than a minimal improvement in the capability of a fission product barrier?

Justification:

5. ☐ YES ☐ NO Result in more than a minimal improvement in defense-in-depth capability? Examples include:
   a. Strengthen balance of accident prevention and mitigation
   b. Reduce reliance on programmatic activities
   c. Reduce probability of common-cause failures

Justification:

If ALL the responses are NO and Confidence is sufficient, issue or activity screens to MINIMAL IMPACT.

If any of the responses is YES, then continue to **Step 3** or develop a plan.

Guidance for Step 2, Question 1:

Accident Initiator Categories (Representative)	Risk Significant?	More than Minimal Decrease ?
Transients initiated by frontline systems		
Transients initiated by support systems		
Primary system integrity loss (e.g. SGTR, RCP seal LOCA, LOCA)		
Secondary system integrity loss		
Internal flooding		
Internal fires		
Earthquakes		
External flooding		
Tornados and High Winds		
Other External Hazards		

Guidance for Step 2, Question 2:

Considerations	Potential Action Effect?	More than Minimal Improvement?
Changes in maintenance, training		
Changes in specific SSCs (e.g., installing a more reliable component)		
Changes in materials		
Equipment replacements to address age related degradation		
Changes in redundancy and diversity		
Additional of equipment		
Changes in operating practices		

Step 3A Qualitative Assessment

Matrix by Current Risk and Potential Impact

UB is upper bound of the risk range; Mid is "mid-range" (0.3 times UB); LB is factor of 10 lower than UB

Current Risk associated with Issue *Note: Address the specific issue first; then assess impacts on other risk contributors potentially impacted*	**Potential Impact of Action (Reduction in Risk)**					
	None	Very Small/Minimal	Small	Medium	High	Comments
	0%	0-25%	25-50%	50% to 90%	>90%	Can adjust these initial ranges as appropriate
	Outcome Note: Quantitative values are delta CDF/LERF					
Green (VL) LB	<VL/Green	<VL/Green	<VL/Green	<VL/Green	<VL/Green	No change
Green (VL) Mid	VL/Green	VL/Green	VL/Green	VL/Green	VL/Green	No change
Green (VL) UB	VL/Green	VL/Green	VL/Green	VL/Green	VL/Green	Maximum reduction is 1E-6/1E-7
White (L) LB	VL/Green	VL/Green	VL/Green	VL/Green	VL/Green	
White (L) Mid	VL/Green	VL/Green	L/White	L/White	L/White	White above 25% Category
White (L) UB	VL/Green	L/White	L/White	L/White	L/White	Maximum reduction is 1E-5/1E-6
Yellow (M) LB	VL/Green	L/White	L/White	L/White	L/White	
Yellow (M) Mid	VL/Green	L/White	M/Yellow	M/Yellow	M/Yellow	Only change is 25% Category
Yellow (M) UB	VL/Green	M/Yellow	M/Yellow	M/Yellow	M/Yellow	Maximum reduction is 1E-4/1E-5
Red (H) LB	?	M/Yellow	M/Yellow	M/Yellow	M/Yellow	Addressed by upper bound Yellow
Red (H) Mid	?	H/Red	H/Red	H/Red	H/Red	
Red (H) UB	?	H/Red	H/Red	H/Red	H/Red	

Step 3A Worksheet

Existing Level of Risk

From NUREG-1742 Section 4.3, IPEEE Insights:

Of the licensees' submittals, 12 reported CDF contributions for external flooding ranging from about 2E-8/ry to about 7E-6/ry. Typically, floods induced by dam breaks, hurricanes, or intense precipitation have been treated as leading to a LOOP, which the licensees usually assumed to be irrecoverable, and additional random failures could then lead to core damage. Other submittals listed additional flood-related damage, including the loss of function of the intake structure; failures of diesel fuel oil transfer pumps; and potential failures of safety-related equipment in the diesel generator, auxiliary, and turbine buildings.

Step 3A Worksheet (cont.)

From the GI-204 Screening Analysis:

- In NUREG-1407 -- "For plants designed against current criteria as described in Regulatory Guide 1.59 and applicable Standard Review Plan sections, particularly Section 2.4, floods pose no significant threat of a severe accident because the exceedance frequency of the design basis flood, excluding floods due to failure of upstream dams, is judged to be less than 10^{-5} per year, and the conditional core damage frequency for a design basis flood is judged to be less than 10^{-1} . Thus, CDFs are estimated to be less than 10^{-6} per year for a plant designed against NRC's current criteria."
- The above conclusion regarding contribution of external flooding to core damage frequency is based on a study that excludes floods due to upstream dam failures.

Step 3A Worksheet (cont.)

From IN 2012-02:

- "…a potentially nonconservative screening value for dam failure frequency that originated in 1980's reference documents which may have been referenced by licensees in their probabilistic risk assessment (PRA) for external events. Using a nonconservative screening value for dam failure frequency to evaluate the need for an additional detailed analysis may result in underestimating the risks to the plant associated with external flooding or loss of heat sink from the failure of upstream and downstream dams or levees…Hence, both NSAC-60 and NUREG/CR-5042 provide an insufficient basis for estimating site-specific dam failure frequency."

Step 3A Worksheet (cont.)

Potential Risk Reduction

From the NRC 50.54(f) letter of 3/12/12:

For the sites where the reevaluated flood exceeds the design basis, addressees are requested to submit an interim action plan that documents actions planned or taken to address the reevaluated hazard with the hazard evaluation.

Subsequently, addressees should perform an integrated assessment of the plant to identify vulnerabilities and actions to address them. The scope of the integrated assessment report will include full power operations and other plant configurations that could be susceptible due to the status of the flood protection features. The scope also includes those features of the ultimate heat sinks (UHS) that could be adversely affected by the flood conditions and lead to degradation of the flood protection (the loss of UHS from non-flood associated causes are not included). It is also requested that the integrated assessment address the entire duration of the flood conditions.

Step 3A Worksheet (cont.)

Current risk level:

Potential % reduction in risk:

Priority:

Tabletop Exercise #4
Cyber Security

Ground Rules for Tabletop Exercise #4

Tabletop Exercise #4: Prioritization of Cyber Security	
Dates (tentative)	1. Generic characterization and assessment, December 18-19, 2013 2. Plant-specific prioritization (not anticipated)
Location	NRC Rockville offices, Three White Flint North, Room 6A28
Time	TBD
Objective of tabletop	a) To exercise the draft guidance documents, b) To assess whether the process is *structured, robust, transparent, and straightforward,* c) To identify areas for improvement in the guidance.
Scope of exercise	1. For the generic assessment, assess the issues associated with cyber security. Develop the generic characterization which will serve as input to the plant-specific evaluation(s).
Industry guidance documents	1. Draft NEI *Guidelines for Prioritization and Scheduling Implementation* 2. Draft NEI *Guidance for Generic Assessment Expert Team (Initiative for Improving Nuclear Safety and Regulatory Efficiency)* 3. Draft NEI *Guidance for Integrated Decision Making Panel (Initiative for Improving Nuclear Safety and Regulatory Efficiency)* 4. Security-specific prioritization process TBD
Reference documents	1. SECY-10-0153 [redacted] Cyber Security – Implementation of the Commission's Determination of Systems and Equipment within the Scope of Title 10 of the *Code of Federal Regulations*, Section 73.54 2. NEI 08-09 Rev. 6 Cyber Security Plan for Nuclear Power Reactors 3. NEI Policy Brief on Cyber Security, February 2013
Pre-tabletop activities	1. Participants should review SECY-10-0154 [redacted] 2. Participants should have an overview of the measures taken by industry to address cyber security, including NEI 08-09 and the NEI Policy Brief of February 2013
Tabletop activities	1. Discuss the tabletop exercise objective and desired outcomes 2. Use the guidance documents to characterize and prioritize the particular issue, cyber security. Briefly document the results. 3. Identify specific improvements to the guidance documents
Preliminary conclusions to draw from tabletop exercise	Determine a) whether guidance is adequately *structured, robust, transparent, and straightforward,* and b) whether the aids including tables, figures, screening questions, matrices, and worked examples are useful.
Lessons-learned	A list of the major lessons learned from the tabletop exercise should be carried forward to enhance the guidance documents and to improve future tabletop activities

Regulations and Policy

- 10 CFR 73.54 Protection of digital computer and communication systems and networks, and SECY-10-0153
  - Establish a dedicated cyber security assessment team
  - Identify critical systems and critical digital assets
  - Isolate key control systems
  - Enhance and implement robust controls over the use of portable media
  - Enhance defenses against insider threats
  - Implement cyber security controls to protect equipment deemed most essential
  - Implement measures to maintain effectiveness

Guidance

- Regulatory Guide 5.71 Cyber Security Programs for Nuclear Facilities
- NEI 08-09 Cyber Security Plan for Nuclear Power Reactors

From NEI Policy Brief

Measures to ensure protection against cyber threats:

- Isolated key control systems using either air-gaps, which do not implement any network or internet connectivity, or installed robust hardware-based isolation devices that separate front-office computers from the control system, thus making the front-office computers useless for attacking essential systems. As a result, key safety, security and power generation equipment at the plants are protected from any network-based cyber attacks originating outside the plant.

From NEI Policy Brief (cont.)

- Enhanced and implemented strict controls over the use of portable media and equipment. Where devices like thumb drives, CD, and laptops are used to interface with plant equipment, measures are in place to minimize the cyber threat. These measures include authorizing use of portable assets to the performance of a specific task, minimizing the movement from less secure assets to more secure assets, and virus scanning. As a result, nuclear power plants are well-protected from attacks like Stuxnet, which was propagated through the use of portable media.

From NEI Policy Brief (cont.)

- Heightened defenses against an insider threat. Training and insider mitigation programs have been enhanced to include cyber attributes. Individuals who work with digital plant equipment are subject to increased security screening, cyber security training and behavioral observation.
- Implemented cyber security controls to protect equipment deemed most essential for the protection of public health and safety.

From NEI Policy Brief (cont.)

- Taken measures to maintain effective cyber protection measures. These measures include maintaining equipment listed in the plant configuration management program and ensuring changes to the equipment are performed in a controlled manner. A cyber security impact analysis is performed before making changes to relevant equipment. The effectiveness of cyber security controls is periodically assessed, and enhancements are made where necessary. Vulnerability assessments are performed to ensure that the cyber security posture of the equipment is maintained.

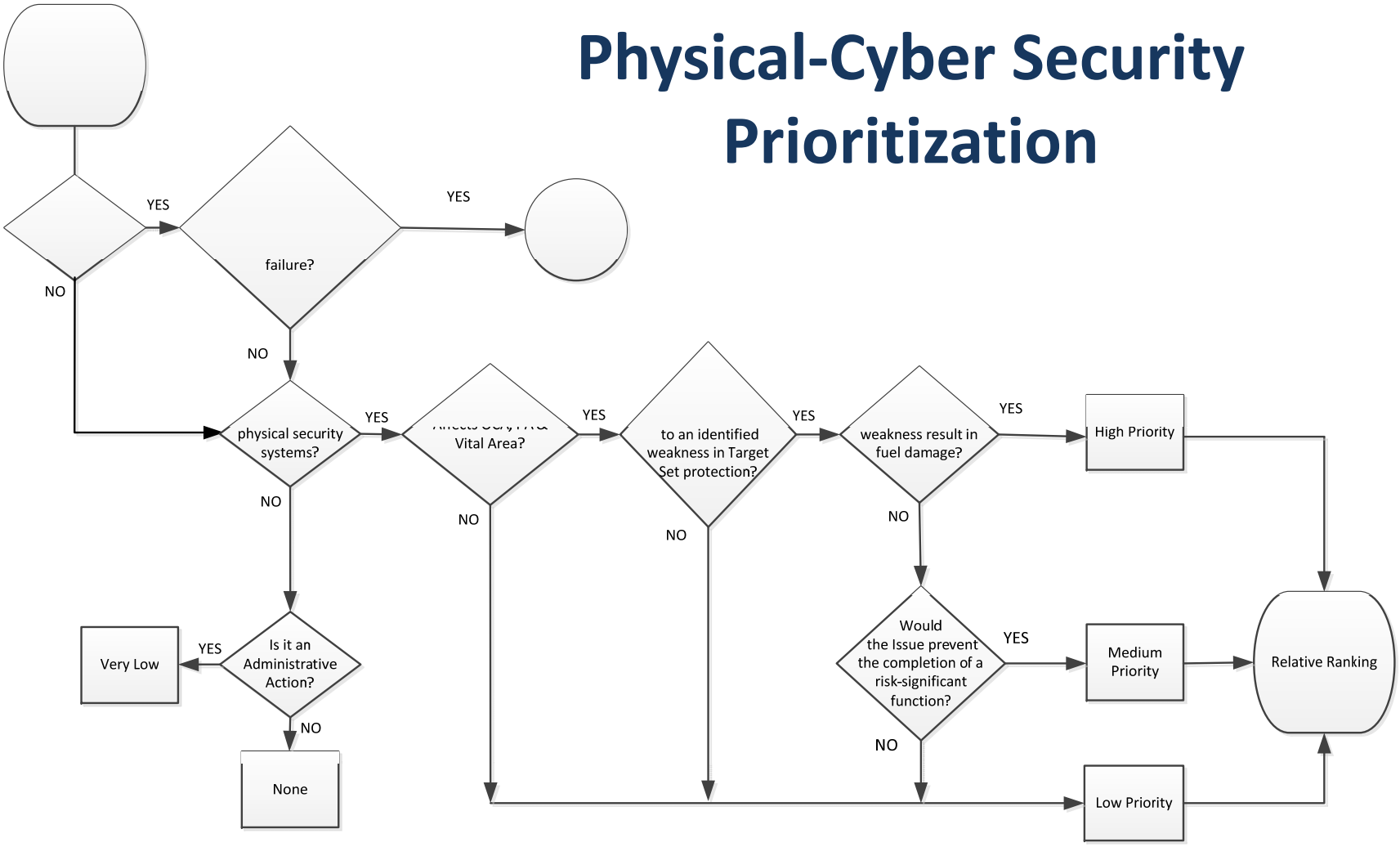
Cyber Security
(High/Medium Significance Criteria)

- Affect cyber (digital or external communications systems)?
- Impact power block (reactor control & safety, generating and transmission systems, if within the owner's control)?
- Adverse impact on target set function?
- Cause fuel damage?
- Cyber issues that impact physical security protection and detection systems assessed through physical security assessment process

Cyber Security
(Medium/Low/Very Low Significance Criteria)

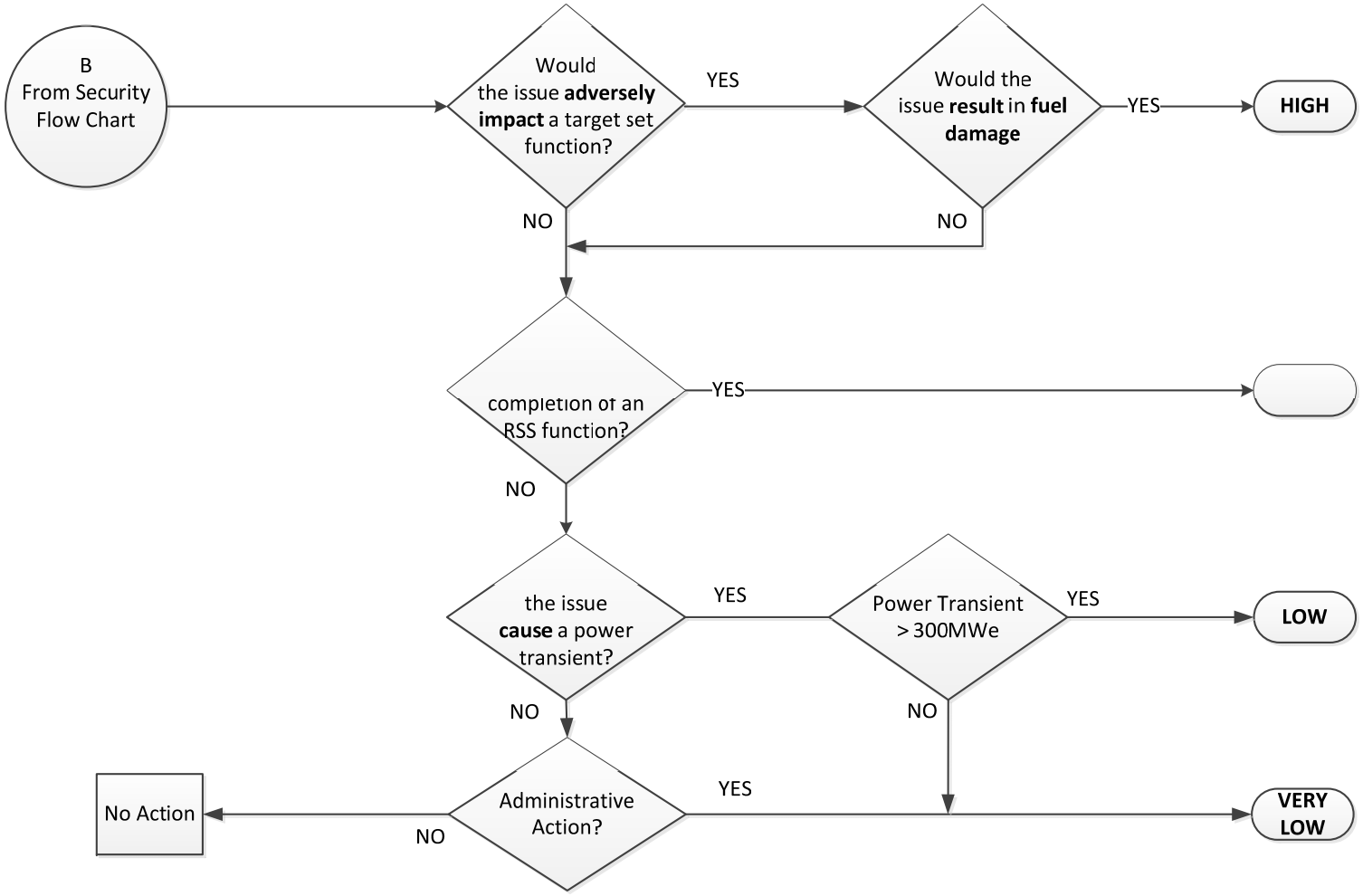
- Prevent completion of a risk-significant function?
  - External communication capability is a risk-significant function
- Cause a power transient greater than 300 MWe?
- Administrative topic?

Physical-Cyber Security Prioritization



NOTE: As used in this document the term issue may be a cyber-security intrusion, a potential cyber-security intrusion, or a security action or potential action

Cyber Security Prioritization



Tabletop Exercise #5
Reliable SFP Instrumentation

Ground Rules for Tabletop Exercise #5

Tabletop Exercise #5: Prioritization of Reliable Spent Fuel Pool Instrumentation	
Dates (tentative)	1. Generic characterization and assessment, December 18-19, 2013 2. Plant-specific prioritization, TBD
Location	NRC Rockville offices, Three White Flint North, Room 6A28
Time	TBD
Objective of tabletop	a) To exercise the draft guidance documents, b) To assess whether the process is *structured, robust, transparent, and straightforward,* c) To identify areas for improvement in the guidance.
Scope of exercise	1. For the generic assessment, assess the issues associated with the NRC Order EA-12-051 regarding reliable spent fuel pool (SFP) instrumentation, and develop the generic characterization which will serve as input to the plant-specific evaluation(s)
Industry guidance documents	1. Draft NEI *Guidelines for Prioritization and Scheduling Implementation* 2. Draft NEI *Guidance for Generic Assessment Expert Team (Initiative for Improving Nuclear Safety and Regulatory Efficiency)* 3. Draft NEI *Guidance for Integrated Decision Making Panel (Initiative for Improving Nuclear Safety and Regulatory Efficiency)*
Reference documents	1. SECY-13-0112, Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor 2. EPRI TR 3002000498, Spent Fuel Pool Risk Assessment Integration Framework (Mark I and II BWRs) and Pilot Plant Application 3. NRC Order EA-12-051 Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (ML12056A044) 4. NEI 12-02 Rev. 1, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation" 5. JLD-ISG-2012-03, "Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation, Interim Staff Guidance," Revision 0 6. SFP design information and any relevant SFP risk insights (plant-specific only)
Pre-tabletop activities	1. Participants should have an overview of the qualitative and quantitative risk insights in SECY-13-0112 and EPRI TR 3002000498 to become familiar with aspects of the existing level of risk associated with beyond-design-basis SFP accidents. 2. Participants should review the NRC Order EA-12-051, NEI 12-02 Rev. 1 guidance, and NRC Interim Staff Guidance JLD-ISG-2012-03 to understand the scope, requirements, and guidance for reliable SFP instrumentation. 3. For the plant-specific prioritization, participants should have information regarding plant-specific SFP design and existing instrumentation, including implementation of relevant aspects of NEI 12-06 on FLEX.
Tabletop activities	1. Discuss the tabletop exercise objective and desired outcomes 2. Use the guidance documents to characterize and prioritize the particular issue, Reliable SFP Instrumentation, using Steps 1, 2 and 3A 3. Briefly document the results 4. Identify specific improvements to the guidance documents
Preliminary conclusions to draw from tabletop exercise	Determine a) whether guidance is adequately *structured, robust, transparent, and straightforward,* and b) whether the aids including tables, figures, screening questions, matrices, and worked examples are useful.
Lessons-learned	A list of the major lessons learned from the tabletop exercise should be carried forward to enhance the guidance documents and to improve future tabletop activities

Aspects of Reliable SFP Instrumentation to Evaluate

- Existing level of risk
  - LERF from SFP accidents
  - Effect of SFP accidents on reactor CDF and LERF
- Benefits and potential reduction in risk (ΔLERF)
- Defense-in-depth impact

SFP Instrumentation Order

From the NRC Order EA-12-051 :

- The lack of information on the condition of the spent fuel pools (at Fukushima) contributed to a poor understanding of possible radiation releases and adversely impacted effective prioritization of emergency response actions by decision makers.
- Adequate inventory of water under accident conditions is necessary to provide containment, as well as the cooling and shielding safety functions.

SFP Instrumentation Order (cont.)

- Fukushima demonstrated the confusion and misapplication of resources that can result from beyond-design-basis external events when adequate instrumentation is not available.
- The spent fuel pool level instrumentation at U.S. nuclear power plants is typically narrow range and, therefore, only capable of monitoring normal and slightly off-normal conditions.

SFP Instrumentation Order (cont.)

- Although the likelihood of a catastrophic event affecting nuclear power plants and the associated spent fuel pools in the United States remains very low, beyond-design-basis external events could challenge the ability of existing instrumentation to provide emergency responders with reliable information on the condition of spent fuel pools.
- Benefits can be derived from the availability of more diverse instrumentation.
- These new requirements provide a greater capability, consistent with the overall defense-in-depth philosophy.

From NEI 12-02 Rev 1

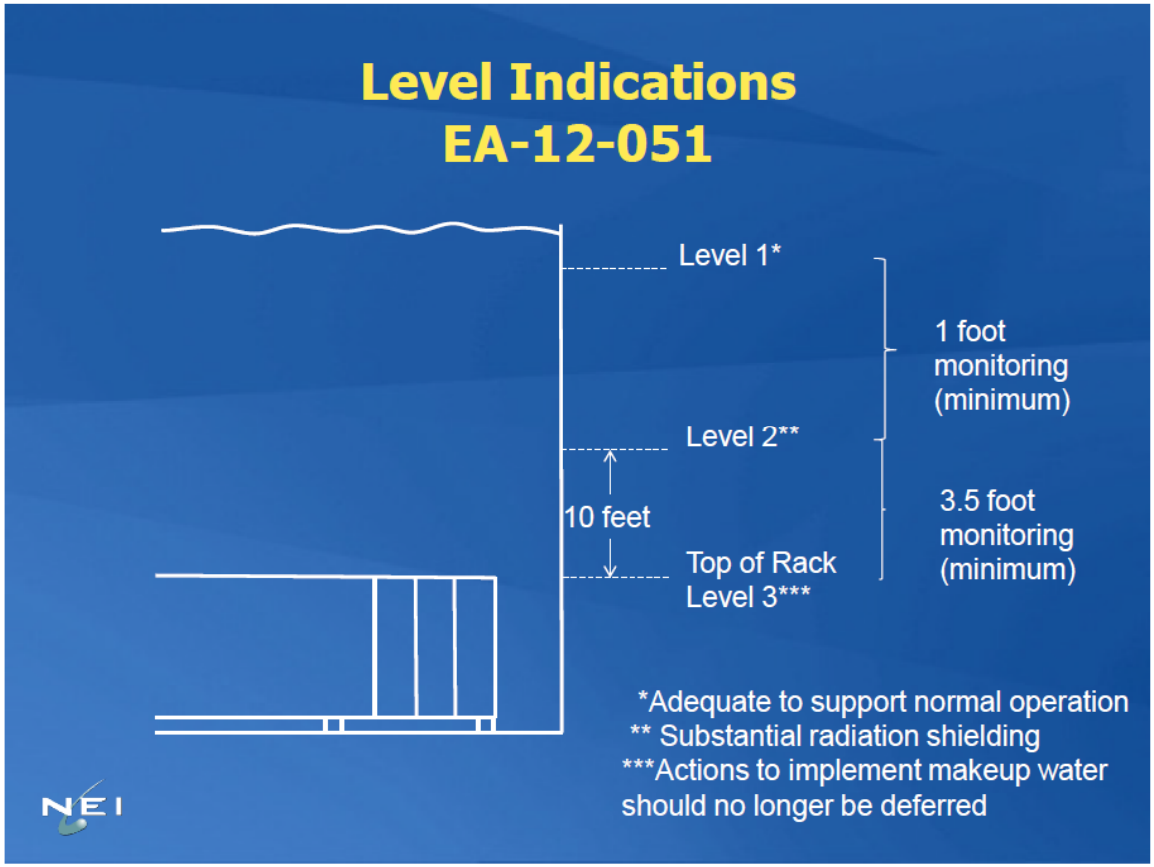


Figure 1

2.3.1. Level 1 – level that is adequate to support operation of the normal fuel pool cooling system

NEI 12-02 (Cont.)

- If portable components are used as part of a backup instrument channel then, to limit personnel resources required for deployment, it shall be designed such that it can easily be deployed by a maximum of two trained personnel within 30 minutes at the spent fuel pool (i.e., no more than 1 person-hour).
- All channels of SFP level instrumentation shall provide the capability of connecting the channel to a source of power (e.g., portable generators or replaceable batteries) independent of the normal plant AC and DC power systems.

Reliable SFP Instrumentation – Step 1

Does the proposed activity or issue:

1. ☐ YES ☐ NO Result in an impact on the frequency of occurrence of a risk significant accident initiator?

Justification:

2. ☐ YES ☐ NO Result in an impact in the availability, reliability, or capability of SSCs and personnel relied upon to mitigate a risk significant transient, accident, or natural hazard?

Justification:

3. ☐ YES ☐ NO Result in an impact in the consequences of a risk significant accident sequence?

Justification:

Reliable SFP Instrumentation – Step 1 (cont.)

4. ☐ YES ☐ NO Result in an impact in the capability of a fission product barrier?

Justification:

5. ☐ YES ☐ NO Result in an impact in defense-in-depth capability?

Examples include:

a. Strengthen balance of accident prevention and mitigation
b. Reduce reliance on programmatic activities
c. Reduce probability of common-cause failures

Justification:

If ALL the responses are NO and Confidence is sufficient, issue or activity has NO IMPACT and screens out (DROP) STOP.

If any of the responses is YES, then continue to **Step 2** or develop a plan.

Step 2 (More than Minimal Assessment):

Does the proposed activity or issue:

1. ☐ YES ☐ NO Result in more than a minimal decrease in frequency of occurrence of a risk significant accident initiator?

Justification:

2. ☐ YES ☐ NO Result in more than a minimal improvement in the availability, reliability, or capability of SSCs and personnel relied upon to mitigate a risk significant transient, accident, or natural hazard?

Justification:

3. ☐ YES ☐ NO Result in more than a minimal decrease in the consequences of a risk significant accident sequence?

Justification:

Step 2 (cont.):

4. ☐ YES ☐ NO Result in more than a minimal improvement in the capability of a fission product barrier?

Justification:

5. ☐ YES ☐ NO Result in more than a minimal improvement in defense-in-depth capability? Examples include:
   a. Strengthen balance of accident prevention and mitigation
   b. Reduce reliance on programmatic activities
   c. Reduce probability of common-cause failures

Justification:

If ALL the responses are NO and Confidence is sufficient, issue or activity screens to MINIMAL IMPACT.

If any of the responses is YES, then continue to **Step 3** or develop a plan.

Guidance for Step 2, Question 1:

Accident Initiator Categories (Representative)	Risk Significant?	More than Minimal Decrease ?
Transients initiated by frontline systems		
Transients initiated by support systems		
Primary system integrity loss (e.g. SGTR, RCP seal LOCA, LOCA)		
Secondary system integrity loss		
Internal flooding		
Internal fires		
Earthquakes		
External flooding		
Tornados and High Winds		
Other External Hazards		

Add line item for Spent Fuel Pools →

Guidance for Step 2, Question 2:

Considerations	Potential Action Effect?	More than Minimal Improvement?
Changes in maintenance, training		
Changes in specific SSCs (e.g., installing a more reliable component)		
Changes in materials		
Equipment replacements to address age related degradation		
Changes in redundancy and diversity		
Additional of equipment		
Changes in operating practices		

Step 3A Qualitative Assessment

Matrix by Current Risk and Potential Impact

UB is upper bound of the risk range; Mid is "mid-range" (0.3 times UB); LB is factor of 10 lower than UB

Current Risk associated with Issue *Note: Address the specific issue first; then assess impacts on other risk contributors potentially impacted*	**Potential Impact of Action (Reduction in Risk)**					
	None	Very Small/Minimal	Small	Medium	High	Comments
	0%	0-25%	25-50%	50% to 90%	>90%	Can adjust these initial ranges as appropriate
	Outcome Note: Quantitative values are delta CDF/LERF					
Green (VL) LB	<VL/Green	<VL/Green	<VL/Green	<VL/Green	<VL/Green	No change
Green (VL) Mid	VL/Green	VL/Green	VL/Green	VL/Green	VL/Green	No change
Green (VL) UB	VL/Green	VL/Green	VL/Green	VL/Green	VL/Green	Maximum reduction is 1E-6/1E-7
White (L) LB	VL/Green	VL/Green	VL/Green	VL/Green	VL/Green	
White (L) Mid	VL/Green	VL/Green	L/White	L/White	L/White	White above 25% Category
White (L) UB	VL/Green	L/White	L/White	L/White	L/White	Maximum reduction is 1E-5/1E-6
Yellow (M) LB	VL/Green	L/White	L/White	L/White	L/White	
Yellow (M) Mid	VL/Green	L/White	M/Yellow	M/Yellow	M/Yellow	Only change is 25% Category
Yellow (M) UB	VL/Green	M/Yellow	M/Yellow	M/Yellow	M/Yellow	Maximum reduction is 1E-4/1E-5
Red (H) LB	?	M/Yellow	M/Yellow	M/Yellow	M/Yellow	Addressed by upper bound Yellow
Red (H) Mid	?	H/Red	H/Red	H/Red	H/Red	
Red (H) UB	?	H/Red	H/Red	H/Red	H/Red	

Step 3A Worksheet

Existing Level of Risk

From SECY-13-0112 (Beyond DBE SFP accident at BWR Mark I)

Table 1 Frequency of SFP Fuel Uncovery (/yr)

Initiating Event Class	**NUREG-1353** (1989) (BWR, best-estimate[1])	**NUREG-1738** (2001)
Seismic events	7×10^{-6}	2×10^{-6} (LLNL) 2×10^{-7} (EPRI)[2]
Cask / heavy load drop	3×10^{-8}	2×10^{-7}
LOOP – severe weather	-	1×10^{-7}
LOOP – other	-	3×10^{-8}
Internal fire	-	2×10^{-8}
Loss of pool cooling	6×10^{-8}	1×10^{-8}
Loss of coolant inventory	1×10^{-8}	3×10^{-9}
Inadvertent aircraft impacts	6×10^{-9}	3×10^{-9}
Missiles – general	1×10^{-8}	-
Missiles - tornado Pneumatic seal failures	- 3×10^{-8}	$< 1 \times 10^{-9}$ -

[1] These numbers have not been multiplied by the stated conditional probability of having a Zirconium fire of 0.25.
[2] NUREG-1738 presented results for the two different seismic hazard models in wide use at the time (the Electric Power Research Institute (EPRI) and Lawrence Livermore National Labs (LLNL) models).

Step 3A Worksheet (cont.)

SECY-13-0112 (cont.)

Table 33 Overall Consequence Results

SFP Fuel Loading	High Density (1x4)		Low Density	
Seismic Hazard Frequency[1] (/yr) (PGA of 0.5 to 1.0g)	1.7E-05		1.7E-05	
50.54(hh)(2) Mitigation Credited	Yes	No	Yes	No
Conditional[2] Probability of Release	0.036%	0.69%	0.036%	0.69%
Hydrogen Combustion Event	"Not Predicted"	"Possible"	"Not Predicted"	"Not Predicted"
Conditional[3] Consequences (Release Frequency-Averaged[4])				
Cumulative Cs-137 Release at 72 hours (MCi)	0.26	8.8[(8)]	0.19[(7)]	0.11
	Measures Related to Health and Safety of Individuals			
Individual Early Fatality Risk	0	0	0	0
Individual Latent Cancer Fatality Risk[5] Within 10 Miles	3.4E-04	4.4E-04	3.4E-04	2.0E-04
	Measures Related to Cost Benefit Analysis			
Collective Dose (Person-Sv)	47k	350k	47k	27k
Land Interdiction[6] (mi^2)	230	9,400	230	170
Long-term Displaced Individuals[6]	120k	4,100k	120k	81k
Consequences per year (Release Frequency-Weighted[4])				
Release Frequency (/yr)	6.1E-09	1.2E-07	6.1E-09	1.2E-07
	Measures Related to Health and Safety of Individuals			
Individual Early Fatality Risk (/yr)	0	0	0	0
Individual Latent Cancer Fatality Risk[5] Within 10 Miles (/yr)	2.1E-12	5.2E-11	2.1E-12	2.4E-11
	Measures Related to Cost Benefit Analysis			
Collective Dose (Person-Sv/yr)	2.9E-04	4.1E-02	2.9E-04	3.2E-03
Land Interdiction[6] (mi^2/yr)	1.4E-06	1.1E-03	1.4E-06	2.0E-05
Long-term Displaced Individuals[6] (Persons/yr)	7.1E-04	4.9E-01	7.1E-04	9.5E-03

Step 3A Worksheet (cont.)
From SECY-13-0112

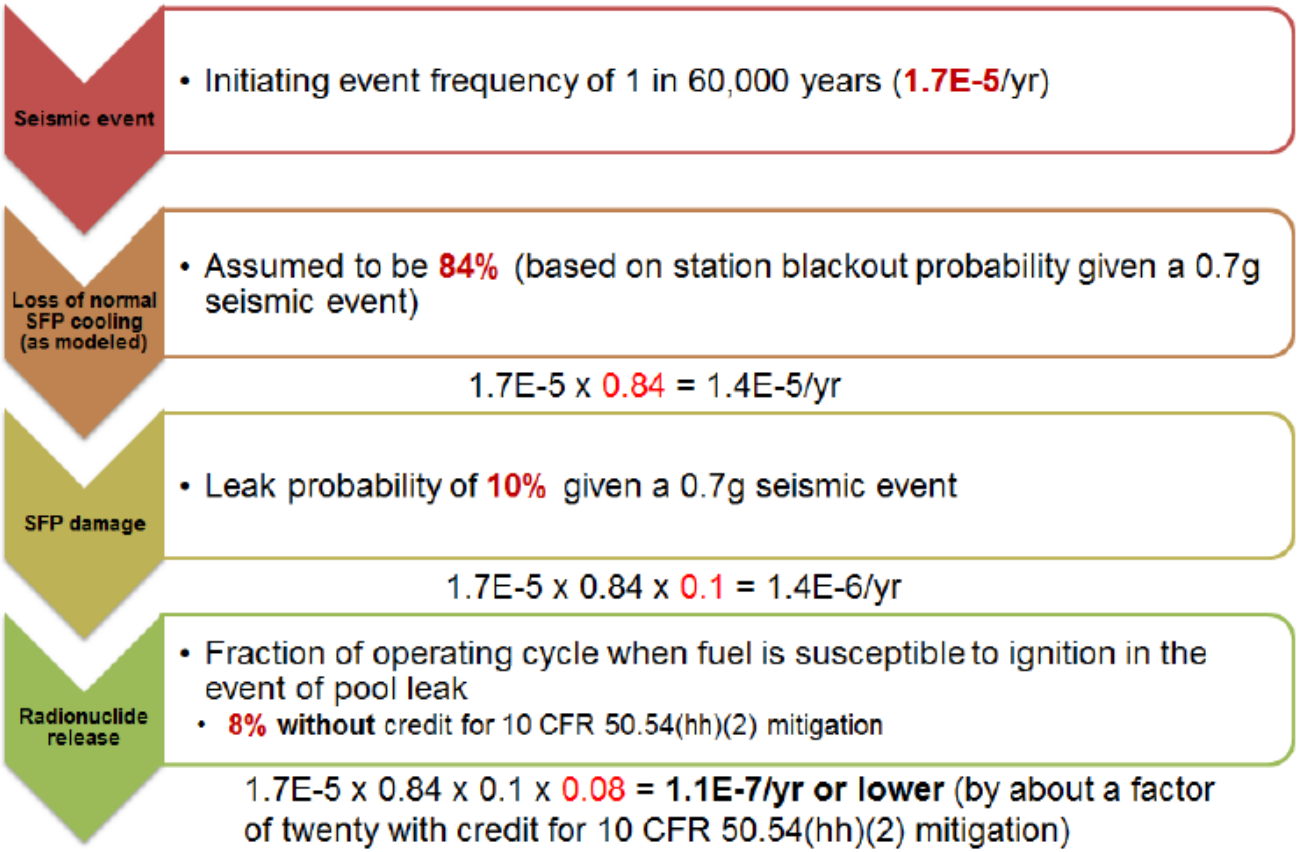


Figure 138: Affecting Likelihood of SFP Release from a Severe Seismic Event Event

Step 3A Worksheet (cont.)

From EPRI TR 3002000498, "Spent Fuel Pool Risk Assessment Integration Framework (Mark I and II BWRs) and Pilot Plant Application"

Table 6-2
Heavy Load Drop and Incremental Manual Shutdown Initiator Contributions to CDF

Initiator	Incremental Contribution to CDF (/yr)	% Increase in CDF Relative to Model of Record (MOR)
%DROP-CASK	4.33E-09	0.12%
%CASK-SHAFT	4.51E-10	0.01%
All other SFP initiators leading to core damage	1.91E-09	0.05%
Total	6.69E-09	0.18%

Table 6-3
Heavy Load Drop and Incremental Manual Shutdown Initiator Contribution to Reactor LERF

Initiator	Incremental Contribution to LERF (/yr)	% Increase in LERF Relative to Model of Record (mor)
%DROP-CASK	1.38E-10	0.03%
%CASK-SHAFT	ε	ε
All other SFP initiators leading to core damage	8.94E-10	0.20%
Total	1.03E-09	0.23%

Step 3A Worksheet (cont.)
From EPRI TR 3002000498

Table 6-5
SFP Fuel Damage Frequency (FDF) from All Evaluated Causes[1]

Contributor [1]	CDF (Total)[6] [2]	FDF (Total)[1] [3]	FDF (SFP Events Only)[2] [4]	Simultaneous CDF and FDF (Combined Reactor and SFP Events)[3] [5]	CDF (Without SFP Fuel Damage)[4] [6]	Incremental FDF Resulting from Severe Accident Progression in Reactor[5] [7]
FPIE	3.65E-06	2.71E-07	1.46E-07	1.25E-07	3.52E-06	1.05E-07
Seismic PRA	8.30E-06	3.27E-06	3.93E-07	2.88E-06	5.42E-06	1.81E-07
Shutdown	1.78E-07	3.94E-09	2.87E-09	1.07E-09	1.78E-07	1.07E-09
Shutdown Seismic PRA	2.48E-07	6.32E-08	ε	6.32E-08	1.85E-07	1.62E-09
TOTAL	1.24E-05	3.61E-06	5.42E-07	3.07E-06	9.30E-06	2.89E-07

Table 6-7
Summary of Combined Severe Accident LERF Sources (/yr)

Contributor [1]	CDF (Total) [2]	Total Reactor LERF[1] [3]	SFP LERF (SFP Events Only)[2] [4]	Simultaneous SFP and Reactor LERF (SFP and Reactor Events)[3] [5]	Reactor LERF (Reactor Events Only)[4] [6]	Total SFP LERF[5] [7]
FPIE	3.65E-06	4.41E-07	1.54E-07	1.26E-08	4.27E-07	1.66E-07
Seismic PRA	8.30E-06	6.78E-06	3.76E-07	2.78E-06	4.00E-06	3.16E-06
Shutdown	1.78E-07	1.69E-07	2.86E-09	1.02E-09	1.68E-07	3.88E-09
Shutdown Seismic PRA	2.48E-07	2.23E-07	ε	6.30E-08	1.60E-07	6.30E-08
TOTAL	1.24E-05	7.61E-06	5.33E-07	2.86E-06	4.76E-06	3.39E-06

Step 3A Worksheet (cont.)
From EPRI TR 3002000498

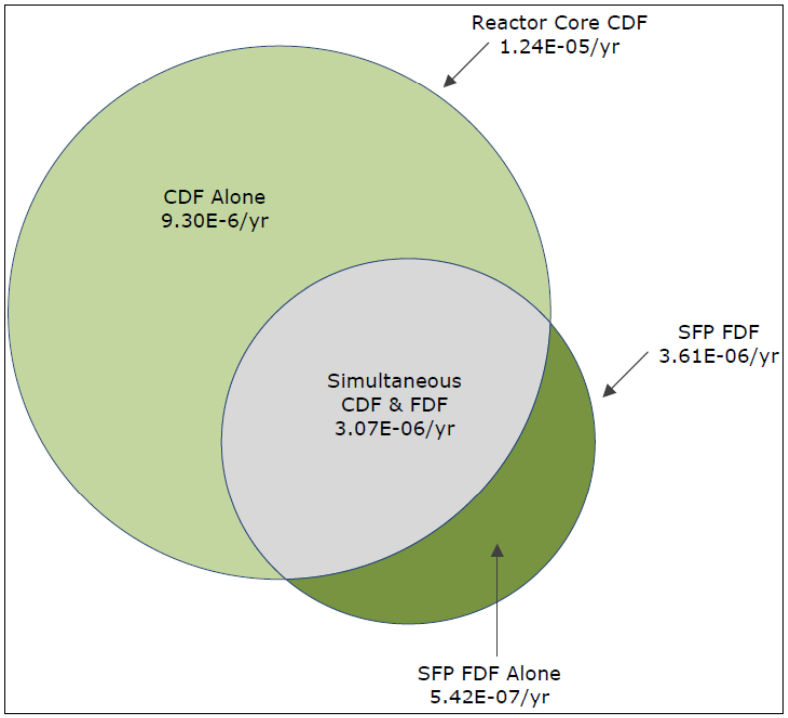


Figure 6-5
Venn Diagram of Contributors to CDF and FDF
(Not drawn to scale)

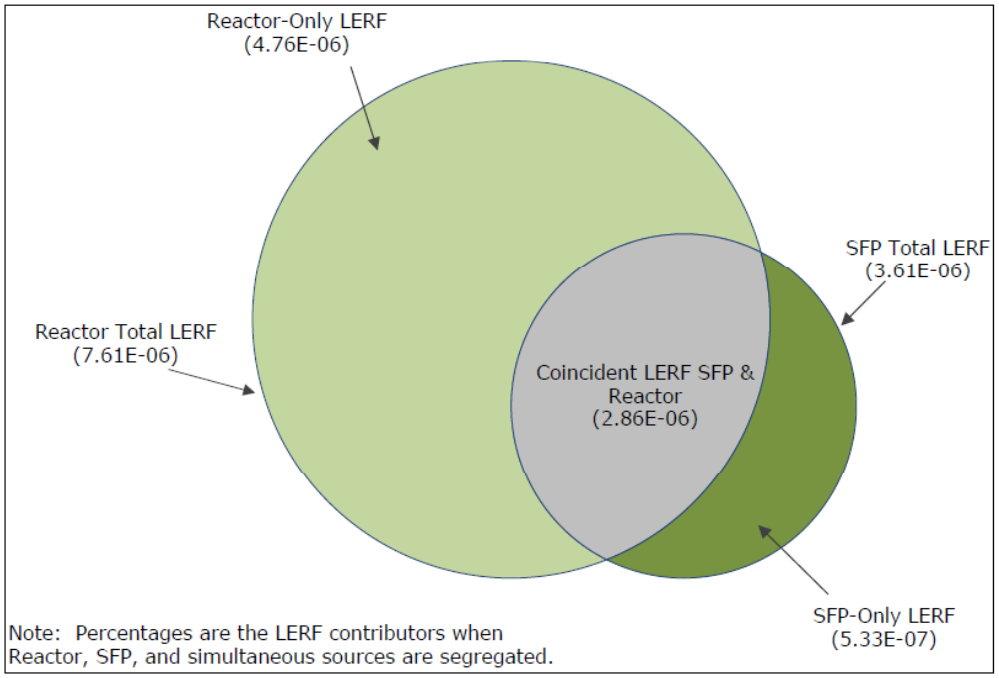


Figure 6-8
Venn Diagram of Contributors to LERF (Total LERF = 8.14E-06/yr)
(Not drawn to scale)

Step 3A Worksheet (cont.)

Potential Risk Reduction

From SECY-13-0112

In the cases studied, which in general did not account for multiple or concurrent reactor and SFP accidents, the precise time to diagnose the need for SFP mitigation did not have an effect on the course of most scenarios.

> Nevertheless, the improved reliable and available SFP indication required by the NRC Order of March 12, 2012 (EA-12-051) is important to ensure that plant personnel can effectively prioritize emergency actions. The availability of such instrumentation may have changed the mitigation mode (makeup versus sprays) deployed to mitigate events that resulted in a release.

Step 3A Worksheet (cont.)

From SECY-13-0112
(SFP water level is at the top of the fuel rack)

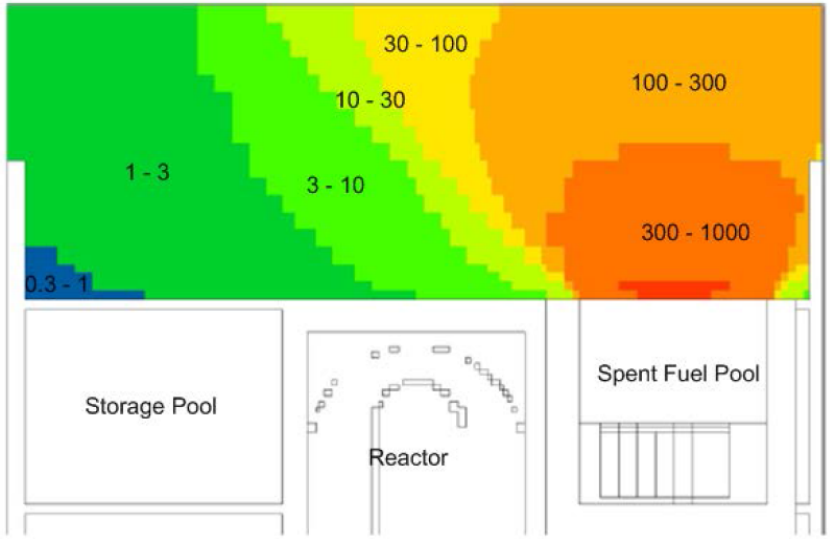
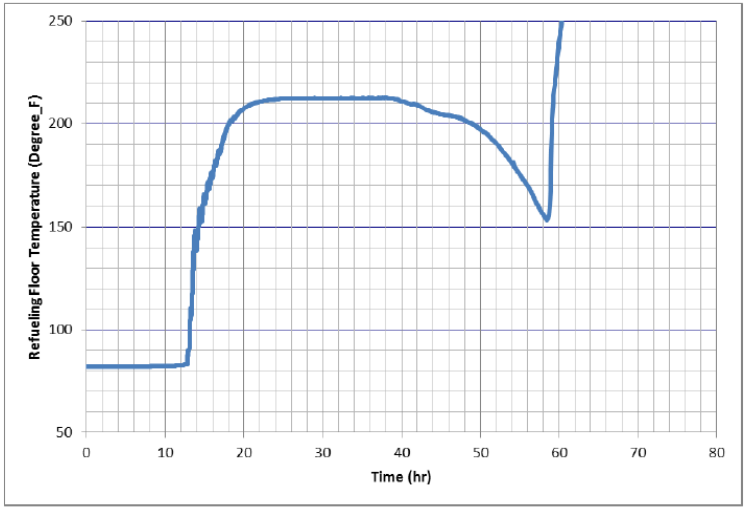


Figure 98 Approximate dose rate of elevation contours, water at the top of fuel hardware, around the time of defueling (rem per hour).

Small leak scenario



Step 3A Worksheet (cont.)

Potential Risk Reduction

From EPRI TR 3002000498

Contributor	Initiator Frequency	Time to Fuel Damage[1]	Comment
Loss of SFP Cooling	Low	2 to 5 days	Slow boil-off of inventory results in an extensive amount of time for response
Seismic Structural Failure of SFP	Extremely Low	<1 hour	Postulated LOCA in SFP leads to rapid challenge to fuel integrity
Heavy Load Drop Structural Failures	Extremely Low	<1 hour	Postulated LOCA in SFP leads to rapid challenge to fuel integrity
Reactor-Related Phenomena Causing Structural Failure	Extremely Low	>4 hours	Reactor severe accident progression required, and then phenomena to fail SFP leading to subsequent used fuel melting

(1) Earliest postulated time for full core offload.

Step 3A Worksheet (cont.)

Current risk level:

Potential % reduction in risk:

Priority: