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RISK-INFORMED REGULATORY FRAMEWORK FOR NEW REACTORS

Public meeting

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Meeting Purpose

Discuss staff's response to the SRM on SECY-10-0121

Agenda

- Background
  - Commission direction per SRM
- Tabletop exercise results
- Gaps and recommendations
- Next steps
- Open discussion

Commission SRM Dated March 2, 2011

- **Commission approved a hybrid of Options 1 and 2**
  - Continue existing risk-informed framework pending a series of tabletop exercises that test existing guidance

- **Commission "reaffirms" existing**
  - safety goals
  - safety performance expectations
  - subsidiary risk goals and associated risk guidance
  - key principles (e.g., RG 1.174)
  - quantitative metrics

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SRM (cont.)

- **Commission expects:**
  - Advanced technologies in new reactors will result in enhanced margins of safety
  - As a minimum, new reactors have the same degree of protection of the public and environment as current generation LWRs
- **New reactors with these enhanced margins and safety features should have greater operational flexibility than current reactors**

Tabletop Exercises

- December 2, 2010: 50.59-like change process for ex-vessel severe accident (EVSA) design features under Section VIII.B.5.c of each design certification rule
- May 4, 2011: Risk-informed inservice inspection of piping
- May 26, 2011 and June 1, 2011: Risk-Informed Technical Specifications (RITS) Initiative 4b on completion times and the Maintenance Rule (a)(4)
- June 29, 2011: RITS Initiative 5b (surveillance frequency control program)
- August 9, 2011: 50.69 and guidance in NEI 96-07 Appendix C on the change processes for Part 52 specific to EVSA design features
- October 5, 2011: RG 1.174; transition options from large release frequency (LRF) as a risk metric to large early release frequency (LERF); and ROP risk-informed case studies including SDP, reactive inspections under Management Directive 8.3, and MSPI
- October 26, 2011: Follow-up discussions with stakeholders on the ROP

Results

- **Risk-informed ISI: No gaps**
  - Risk-neutral effect for a new active plant and a new passive plant, even when sensitivity studies used more restrictive acceptance criteria
  - Numerous regulatory and programmatic controls (e.g., inspection of a minimum set of weld locations is required regardless of risk levels)
  - The 10 year ISI program is dynamic and allows for incorporation of lessons learned and update to risk ranking consistent with Part 52 requirements for PRA maintenance/upgrades

Results (cont.)

- **RITS 4b (completion times): Two key programmatic controls**
  - The risk-informed completion time is limited to a deterministic maximum of 30 days (referred to as the backstop completion time) from the time the TS action was first entered
  - Voluntary use of the risk-managed TS for a configuration which represents a loss of TS specified safety function, or inoperability of all required safety trains, is not permitted

Results (cont.)

- **RITS 4b staff exercises**
  - Staff identified some configurations of equipment outages that would represent 10 years' worth of core damage probability
  - Repeated entry into such condition over time could increase CDF by one or more orders of magnitude, which could approach the baseline CDF of currently operating plants
  - Staff believes these configurations are unlikely or unrealistic, and that there were additional regulatory and programmatic controls that would limit the aggregated risk increase (e.g., performance monitoring, periodic PRA maintenance and upgrade under 50.71(h))

- **Staff concludes no substantive changes to methodology is necessary**

Results (cont.)

- **Maintenance Rule 50.65 (a)(4): No gaps in assessment and management of risk**
  - When PRA approach is combined with other inputs such as the degree of defense in depth and plant transient assessment, factors other than PRA are often more limiting in terms of the risk management action level
  - NUMARC 93-01, Section 11 explicitly acknowledges “there is acknowledged variability in baseline core damage frequency and large early release frequency… determination of the appropriate quantitative risk management action thresholds are plant-unique activities”
  - Some changes to NUMARC 93-01 may be necessary to address changes of scope because of new and different SSCs in the new reactor designs

Results (cont.)

- **RITS 5b (surveillance frequency control program): No gaps**
  - Surveillance frequencies that are controlled by other programs are excluded from the SFCP
  - Equipment covered by inservice testing, for example major pumps and valves, tend to have some of the highest risk importances but are excluded
  - What remains to be implemented under RITS 5b generally are lower risk importance components
  - Unlike RITS 4b, RITS 5b is much more deterministically oriented, with risk impact only a secondary consideration in the criteria for changing surveillance test interval

Results (cont.)

- **50.69: No gaps**
  - Sample application to new reactor design (PWR with active safety systems) shows approximately same categorization distribution (RISC-1,2,3 & 4) as South Texas 1 & 2 pilot based on importance measures
  - Rule has built-in measures to monitor RISC-3 components and take corrective actions (e.g., periodic program review every 2 refuel cycles)

Results (cont.)

- **Part 52 50.59-like change process: Gap identified**
  - Staff generally satisfied with "ex-vessel" portion of NEI 96-07 Appendix C
  - However, changes to severe accident design features that are not specifically intended to address EVSAs (e.g., containment bypass) are not addressed using severe accident criteria as in Section VIII.B.5.c.
- **Recommendation 1**

  Address the potential gap, by a) ensuring that there are sufficient details on all key severe accident features in Tier 1, and b) including a change process in future design certification rulemakings in Section VIII for *non-ex-vessel severe accident features* similar to Section VIII.B.5.c for *ex-vessel severe accident features*

Containment Challenges per § 52.47(a)(23) & § 52.79(a)(38)

Core-concrete interaction

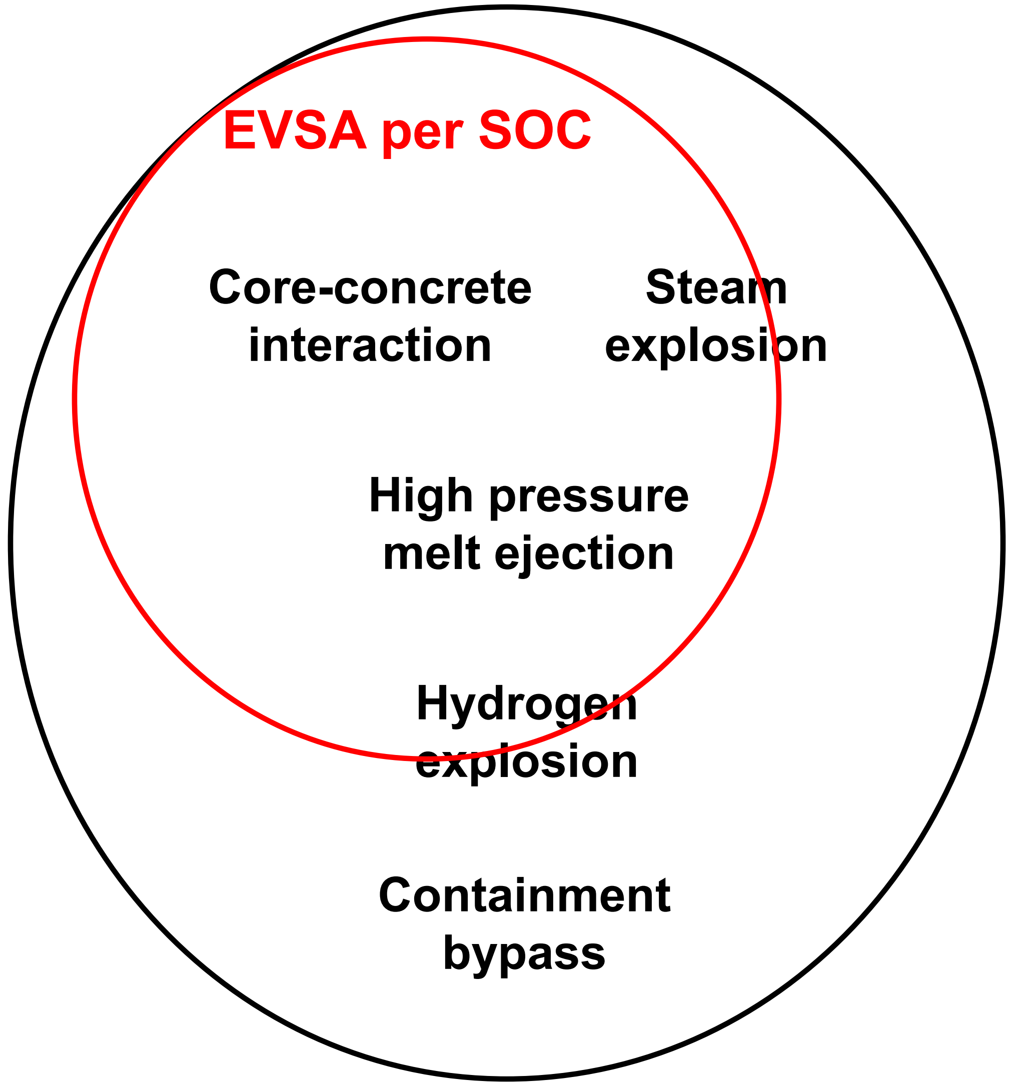
Steam explosion

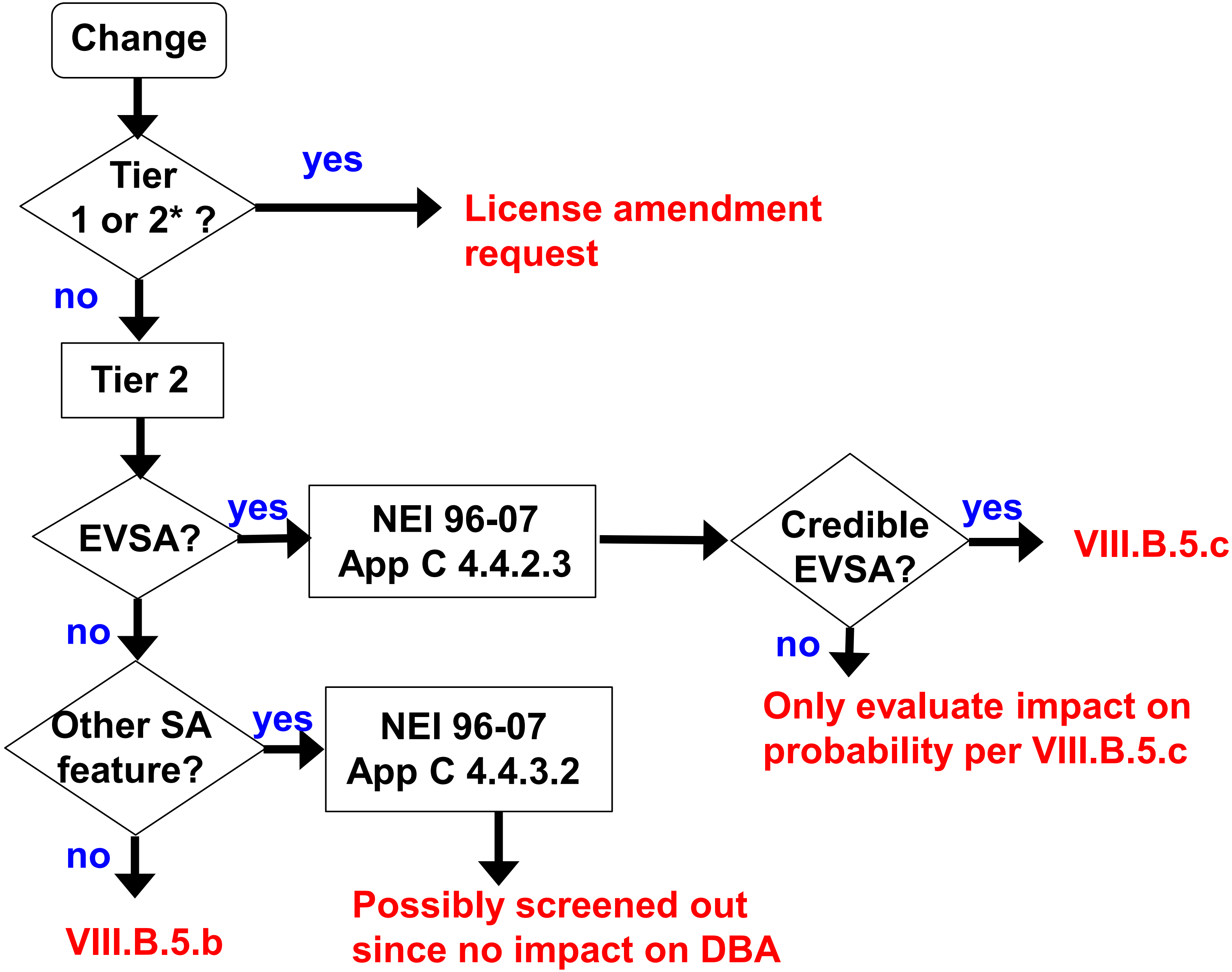
High pressure melt ejection

Hydrogen explosion

Containment bypass

Gap Identified




Change
Tier
1 or 2* ?
yes
License amendment
request
no
Tier 2
EVSA?
yes
NEI 96-07
App C 4.4.2.3
Credible
EVSA?
yes
VIII.B.5.c
no
Only evaluate impact on
probability per VIII.B.5.c
no
Other SA
feature?
yes
NEI 96-07
App C 4.4.3.2
Possibly screened out
since no impact on DBA
no
VIII.B.5.b

Staff's Preliminary Gap Assessment

- Staff reviewed severe accident features for ABWR, AP1000, and ESBWR
- No significant gaps of concern
  - Either it is an EVSA feature and VIII.B.5.c criteria will be used for Tier 2 changes, or
  - If it is a non-ex-vessel severe accident feature, there is generally sufficient detail in Tier 1 as to preclude a significant design change without prior NRC approval
- Staff is verifying preliminary conclusions
- Other standard designs eventually to be addressed
- Await Commission direction per Recommendation 1

Results (cont.)

- **RG 1.174: No gaps**
  - In many of the examples during the exercise, the estimated change in core damage frequency (ΔCDF) was observed to be very low and well below Region II of the acceptance guideline per Figure 3 of RG 1.174
  - Degradation of the level of defense in depth would be an area of close review by the staff
  - Changing a plant feature from highly passive to active thus placing greater reliance on key operator actions would be an area for close review by the staff
  - Proposed changes in or near the boundary of Region II would undergo close scrutiny by the staff, and there should be a compelling reason on the part of the license holder for the proposed change. Alternatives should be assessed by the licensee

LRF-to-LERF Transition

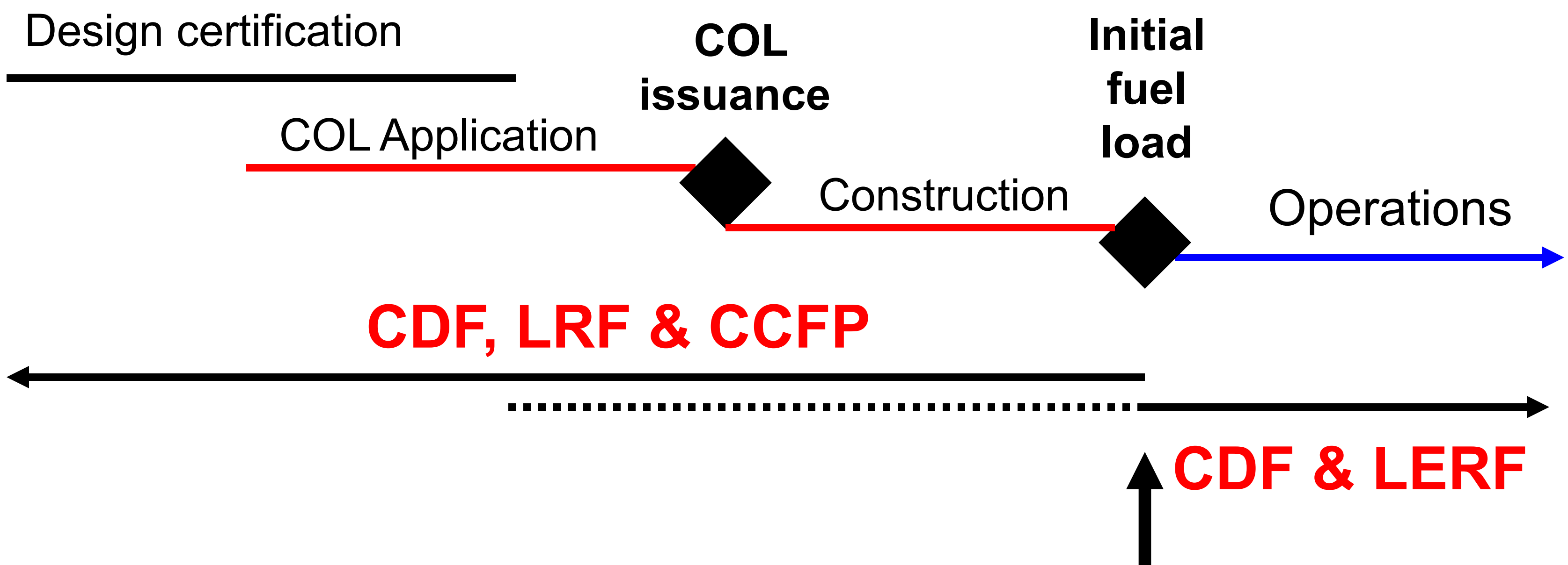
- **LRF vs. LERF**
  – Commission goals for new reactors are based on a conditional containment failure probability (CCFP) of less than 0.1, and a LRF of less than 10^{-6}/yr, as well as 10^{-4}/yr for core damage frequency (CDF)
  – Operating reactors use CDF and LERF as risk metrics

- **LRF issues**
  – LRF (and CCFP) have not been defined by the staff
  – Each design center has chosen different definitions
  – LERF is used in the ASME/ANS level 1 PRA standard, in risk-informed staff guidance (e.g., RG 1.174), and ROP
  – No existing or proposed level 2 PRA standard provides a universal definition of LRF

LRF-to-LERF Options

- Option 2A: continue use of LRF (& CCFP) indefinitely
- Option 2B: continue use of LRF (& CCFP) indefinitely and add LERF at initial fuel load
- Option 2C: transition from LRF to LERF at or prior to initial fuel load; discontinue regulatory use of LRF (& CCFP) thereafter

Option 2C



- LERF calculated at or prior to initial fuel load. CDF & LERF used for RG 1.174 acceptance guidelines going forward.
- Last regulatory use of LRF & CCFP

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Option 2C

- **Advantages**
  - **Consistent with SRM direction**
  - **Harmonizes metrics for all operating reactors, both current and new, going forward**
- **Disadvantages**
  - **LRF & CCFP, part of original design objective in design certification, no longer tracked**
  - **LRF not available to assist in determining impact on late containment failure in RG 1.174**
    - **Augment discussion on long-term containment performance in Section 2.2 of RG 1.174 by referring to the containment performance objectives in SECY-90-016 and SECY-93-087**

Option 2C (cont.)

- **Containment performance objectives per SECY-90-016 and SECY-93-087:**

  The containment should maintain its role as a reliable, leak-tight barrier (for example, by ensuring that containment stresses do not exceed ASME Service Level C limits for metal containments, or Factored Load Category for concrete containments) for approximately 24 hours following the onset of core damage under the more likely severe accident challenges and, following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products.

Results (cont.)

- **Recommendation 2**

  Staff recommends Option 2C to harmonize risk-informed applications for the new reactors consistent with the risk metrics used by the currently operating fleet

Results (cont.)

- **Other programs not assessed in tabletops**
  - Risk-informed inservice testing of pumps and valves (RG 1.175)
  - Integrated leak rate testing interval extension (NEI 94-01)
  - 50.46a
  - NFPA 806
- **Little short-term interest by COL applicants**
- **Alternative source term (RG 1.183) implemented at all new designs with COLAs except ABWR**

ROP Tabletop Approach

- Tested various realistic scenarios to confirm the adequacy of the current ROP risk-informed processes for regulatory decision-making or identify areas for improvement
- Used a broad cross-section of well-vetted cases, developed from actual greater-than-green examples from the current fleet of reactors:
  - Significance Determination Process (SDP) findings
  - Mitigating Systems Performance Index (MSPI) data
  - Management Directive (MD) 8.3 applications
- Applied similar situations to the new reactor designs, filling in gaps with realistic hypothetical situations and reasonable assumptions, and then compared the risk values and resultant regulatory response

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SDP Tabletops

RESULTS

- Existing risk thresholds for determining significance of inspection findings are generally acceptable
- Greater-than-green inspection findings would likely involve common cause failures and/or long exposures of risk-significant components
- Existing process does not always ensure an appropriate regulatory response for degradation of passive components and barriers commensurate with safety

CONCLUSION

- SDP analyses could be augmented with additional qualitative considerations (deterministic backstop) to appropriately address performance issues

MD 8.3 Tabletops

RESULTS

- Existing risk thresholds for invoking reactive inspections are adequate for new reactors
- Deterministic criteria used initially for event screening and then within a range of response determined by risk values
- Risk values heavily influence whether or not a reactive inspection is warranted and, if so, at what level
- Variations in or minor revisions to risk models used can potentially result in an inadequate response

CONCLUSION

- Contribution of existing deterministic criteria could be modified or new deterministic criteria developed for initiating reactive inspections for new reactors

MSPI Tabletops

RESULTS

- Existing MSPI is not adequate and would be largely ineffective in determining an appropriate regulatory response for active new reactor designs
- Meaningful MSPI may not even be possible for passive systems using the current formulation of the indicator
- Existing performance limit (backstop) could be further leveraged for active new reactor designs

CONCLUSION

- Alternate PIs in the mitigating systems cornerstone could be developed and/or additional inspection could be used to supplement insights currently gained through MSPI

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ROP Options

GROUND RULES ACROSS 3 ROP OPTIONS

- Maintain current risk thresholds for new reactor designs
- Consistent with integrated risk-informed decision-making concepts in RG 1.174
- Afford greater operational flexibility based on enhanced safety margins

A. USE AS IS

- Use the existing risk-informed ROP tools for new reactor applications without making any changes
- No additional action or resources needed, but existing tools may not always provide for an appropriate regulatory response

ROP Options (cont.)

B. AUGMENT EXISTING PROCESSES

- SDP: Use existing risk-informed SDP, but augment with deterministic backstops to ensure an appropriate regulatory response to address performance issues
- MD 8.3: Modify the contribution of existing deterministic criteria or develop new criteria for determining the appropriate regulatory response to plant events
- MSPI: Develop alternative to MSPI or augment existing guidance to emphasize performance limit for active new reactor designs, and increase inspection of passive mitigating systems for passive new reactor designs
- Proposed enhancements could be developed using existing resources and working with stakeholders

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ROP Options (cont.)

C. DEVELOP DETERMINISTIC TOOLS

- Do not use the existing risk-informed ROP tools
- Capture risk insights to a lesser extent than the current fleet using deterministic guidance consistent with new reactor design certification and licensing basis
- Additional resources may be necessary to research and develop the new guidance documents

Staff Recommendation: Option B

- Staff would obtain Commission approval for proposed changes to ROP at least one year prior to implementation
- Process enhancements could be further refined based on experience and lessons learned

Next steps

- ACRS Reliability & PRA Subcommittee March 7
- Full ACRS April 12
- SECY due to be issued early June, 2012