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Nuclear Energy Institute

GUIDELINES FOR PRIORITIZATION AND SCHEDULING IMPLEMENTATION

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NOTICE

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ABBREVIATIONS AND ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient without Scram
CDF	Core Damage Frequency
EDG	Emergency Diesel Generators
EDMG	Extensive Damage Mitigation Guidelines
EOP	Emergency Operating Procedure
EP	Emergency Planning
FLEX	Diverse and Flexible Coping Strategy for Extended Loss of Power
GAET	Generic Assessment Expert Team
HEP	Human Error Probability
I&C	Instrumentation and Control
IDP	Integrated Decision Panel
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination External Events
ISI	In Service Inspection
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
MSPI	Mitigating Systems Performance Index
NEI	Nuclear Energy Institute
NFPA-805	National Fire Protection Association (Standard) 805
NRC	Nuclear Regulatory Commission
NTTF	(NRC Fukushima Lessons Learned) Near Term Task Force
PI	Performance Indicator
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
PSF	Performance Shaping Factor
RA	Recovery Action
RCP	Reactor Coolant Pump
ROP	Reactor Oversight Process
RP	Radiation Protection
SAMA	Severe Accident Mitigation Alternatives
SAMG	Severe Accident Mitigation Guidance
SC	Success Criteria
SDP	Significance Determination Process
SGTR	Steam Generator Tube Rupture
SME	Subject Matter Expert
SRO	Senior Reactor Operator
SSC	Structures, Systems and Components

1.0 INTRODUCTION

1.1 PURPOSE

The purpose of this document is to describe industry's guidance for characterizing and prioritizing regulatory and plant-identified actions and scheduling plant improvements at licensee facilities consistent with safety significance. Generic and plant-specific prioritization and plant-specific scheduling are two elements of the proposed approach for improving the process for managing emerging regulatory issues and addressing industry and regulatory concerns on the cumulative impact of additional regulatory requirements. The results of this prioritization may be used as additional risk-informed input to existing regulatory processes, e.g. requesting exemptions and managing commitments. Revision 0 of this guidance applies to power reactors. Fuel cycle facilities and material licensees will monitor and adjust the process, as necessary, based on lessons learned from the power reactor activities and the unique circumstances applicable to non-power reactor licensees.

Safety impact/importance is the predominant factor in the assignment of scheduling priority. Following safety importance characterization (high, medium, low, very low, none), an overall characterization is performed that takes into account additional factors such as emergency planning, security, equipment reliability, and radiological protection to capture the broader safety significance of any issues in those areas that could not be directly captured under the (nuclear) safety importance. This overall characterization is factored into the plant's existing scheduling process that takes into account other factors, such as availability of personnel and equipment.

The approach is risk-informed, in that generic and plant-specific risk information is an important input to the overall safety impact characterization process. Relevant sources of risk information can be considered, and both qualitative and quantitative approaches may be used. A set of qualitative screening questions is used to support the initial steps of the process. PRA models can be used to inform the process. The ability to factor in the quantitative risk information will rely on the quality of PRA models. For the purposes of scheduling activities, this process provides an appropriate level of technical rigor. The approach is consistent with existing functions such as the reactor oversight process and the 10 CFR 50.59 process. This safety importance characterization is intended only for the purposes of scheduling.

The overall scope of the prioritization process is expected to include:

- Regulatory issues and actions taken to address inspection findings with associated NRC schedule commitment
- Non-regulatory issues or nonsafety-related equipment with safety implications, as identified by risk insights

- Non-regulatory issues and activities, as identified by resource peaks in the business plan

Thus, each plant may have a slightly different scope of actions to be prioritized. For regulatory activities, the first step is a generic safety characterization performed by an industry expert team. This assessment is used to inform a plant-specific assessment of the activity, taking into account the nature of plant-specific risk contributors, such as seismic or flooding. The plant-specific assessment is performed by a multi-disciplinary plant integrated decision-making panel (IDP). Plants may also identify and characterize activities that have no direct regulatory nexus, but rather are identified by the plant to improve performance, reliability, or otherwise affect the design or operation of the facility. Such equipment reliability improvements often have direct and indirect benefits on nuclear safety by reducing initiator frequencies or enhancing the mitigation capability given a plant transient or accident.

Additional guidelines on scope of prioritization process

1. An immediate action necessary for continued safe operation (e.g., to support NRC finding of adequate protection, or to restore compliance with a Technical Specification, or to resolve an environmental compliance issue with an adverse effect on public health and safety, or to remove a threat to personnel safety) should not use the prioritization process.

2. Actions taken to address other non-compliance issues with associated NRC commitments, (e.g., actions taken to address inspection findings), are within the scope of prioritization activities. This is anticipated to be useful when resolution of an issue results in multiple actions of varying importance. Correction of the non-compliance is commensurate with its significance determination and should be scheduled consistent with the safety significance of the action. The results of the prioritization process may be used in a NRC commitment change submittal as justification to change the originally committed to completion date.

3. Immediate repairs necessary for continued power production (e.g., replace damaged main transformer) would not use the prioritization process. Implementation should not adversely impact the scheduling of Priority 1 activities.

4. General O&M, routine facilities maintenance, etc. would not use the prioritization process. This is expected to be budgeted separately from those items subject to prioritization. To the extent that the same skilled personnel

resources may be required, implementation should not adversely impact the scheduling of Priority 1 activities.

5. Some major initiatives that typically receive detailed corporate financial evaluations such as license renewal, extended power uprate, and steam generator replacements may not be appropriate for this prioritization process since they are implemented on their own cost-benefit merits.

1.2 Content of This Guidance Document

Section 2 presents high-level guidance for generic and plant-specific assessment and the prioritization process.

Section 3 presents guidance for generic and plant-specific characterization of safety importance.

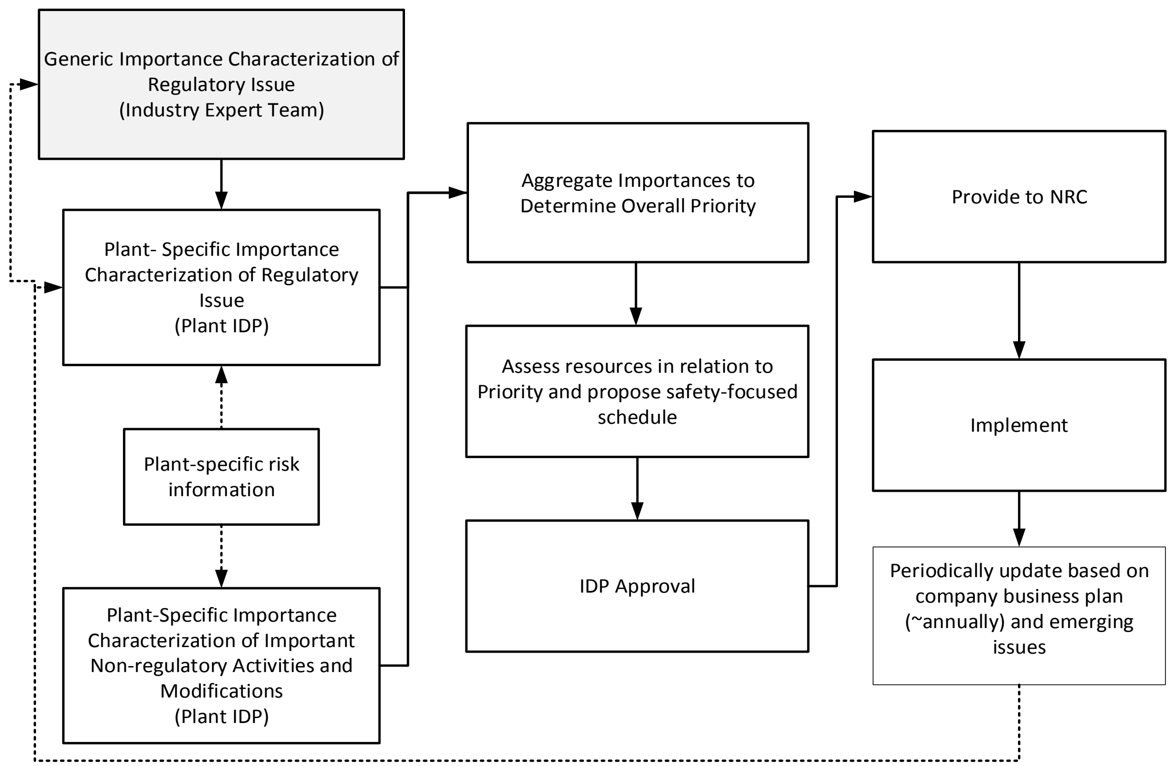
Section 4 presents guidance for generic and plant-specific characterization of security, emergency preparedness, radiological protection, and reliability importance.

Section 5 presents guidance for aggregating the inputs from Sections 3 and 4 and reaching an overall priority of the activity. Guidance for adjusting schedules is also in this section.

The appendix provides examples of the safety importance determination.

Figure 1-1 provides the overall process to be implemented by the plant. As part of the process, feedback between the generic and plant-specific characterizations is an expected outcome.

Figure 1-1
Plant Process for Schedule Prioritization



The overall process addresses the following decision attributes:

- Safety – reactor and spent fuel pool safety; plant personnel safety (other than radiological exposure avoidance) is addressed under "other considerations" on an item-specific basis (Sections 3 and 5.1)
- Security – Both physical security and cyber security (Section 4.1). The considerations are generally consistent with usage of the equivalent Security Cornerstone of the reactor oversight process (ROP).
- Emergency preparedness (EP) (Section 4.2). The considerations are generally consistent with usage of the equivalent Emergency Preparedness Cornerstone of the ROP.
- Radiological protection (RP) – including exposure avoidance for plant personnel (Section 4.3). The considerations are generally consistent with usage of the equivalent Occupational Radiation Safety Cornerstone of the ROP.
- Reliability – Structures, systems and components (both safety-related and non-safety related) (Section 4.4)
- Aggregation of the above to determine priority (Section 5)
- Scheduling (Section 5)

2.0 GENERIC ASSESSMENT EXPERT TEAM (GAET), AND PLANT INTEGRATED DECISION-MAKING PANEL (IDP)

The importance characterization for each category involves a generic component (for regulatory issues), and a plant-specific component (for plant-specific implementation of regulatory issues as well as plant-initiated modifications, etc.).

The generic and plant-specific processes involve the same steps. The generic evaluation is carried out by an industry expert team, known as the Generic Assessment Expert Team (GAET). The GAET evaluation characterizes the importance of the regulatory issue or activity at a generic level and provides an overall assessment and important attributes for consideration in the plant-specific evaluation. The plant-specific process is carried out with the use of a plant integrated decision-making panel (IDP), which reviews the generic characterization provided by the GAET and the plant-specific evaluation provided by a plant subject matter expert (SME), to arrive at plant-specific importance characterization. This importance is determined as one of the following:

- none (no impact)
- very low
- low
- medium
- high

These are intended to be general, approximate characterizations of importance in each category for the purpose of scheduling and sequencing of activities in a safety focused manner. They are not intended for any other use such as cancelling activities. The overall intent is for a practical, efficient and timely process that can be widely implemented.

The GAET provides generic importance characterization information and attributes to the industry. Using this information where applicable, in conjunction with plant-specific evaluation performed by a plant SME, the plant IDP is responsible for making the plant-specific determinations of issue importance. The IDP is separately used to approve the final schedule developed on the basis of the prioritization. The following guidance is provided relative to the makeup of these panels.

The GAET is comprised of industry subject-matter experts with relevant expertise to the issues being evaluated. The GAET composition will vary depending upon the issue. Generally, the GAET is composed of knowledgeable personnel whose expertise represents the important process and functional elements of the Industry, such as operations, engineering, nuclear risk management, industry operating experience, and licensing. The GAET members are expected to have the essential understanding of the issue safety nexus for their assigned issue, and familiarity with the prioritization process guidance and approach. The team can call upon additional personnel, subject matter experts or external consultants, as necessary, to assist in the characterization of issues. Experience, plant knowledge, familiarity with current regulatory issues, and availability to attend the majority, if not all meetings, are important elements in the selection of GAET members. In general, there should be at least five experts designated as members of the GAET with joint expertise in the following fields:

- plant operations (SRO qualified)
- design and systems engineering
- safety analysis
- probabilistic risk assessment and risk-informed decision-making
- licensing.

The plant SMEs are knowledgeable in a particular technical discipline or disciplines (e.g. NFPA 805 implementation or cyber security). They function as the lead presenter of the regulatory issue or activity to the IDP. If a generic assessment is available, this assessment is used by the SME as a key input into the plant-specific assessment along with relevant plant-specific information. The SME should provide his/her evaluation and present the questions and proposed responses to the IDP. The SME should take responsibility to ensure that all relevant generic and plant-specific documents are available to the IDP. The SME should work with the overall coordinator of the prioritization process to ensure that the results of the IDP deliberation are documented and records are maintained.

The IDP is composed of knowledgeable plant personnel whose expertise represents the important process and functional elements of the plant organization, such as operations, engineering (e.g., design, systems, electrical, I&C including information technology, nuclear risk management), industry operating experience, licensing and maintenance.

The IDP can call upon additional plant personnel or external consultants, as necessary, to assist in the evaluation of issues. The precise makeup of the IDP is determined by the licensee. Experience, plant knowledge, and availability to attend the meetings, are important elements in the selection of IDP permanent members. In general, consistent with other licensee expert panels, there should be experts designated as members of the IDP with joint expertise in the following fields:

- plant operations (SRO qualified)
- safety analysis
- design and systems engineering
- probabilistic risk assessment (PRA)
- licensing
- security, emergency planning or other subject matter experts as needed

Members may be experts in more than one field; however, excessive reliance on any one member's judgment should be avoided. The IDP should be aware of the benefits and limitations of the plant-specific PRA and other analyses, and, where necessary, should receive training on the plant-specific PRA, its assumptions, and appropriate implementation. This training is for IDP familiarity and the importance of making well-supported, technical assumptions whether quantitative or qualitative information is used.

The IDP should be familiar with the technical approach and guidance for prioritization. In order to have a full understanding of the issue being characterized, all questions in each applicable step of the guidance should be answered, even if an initial "yes" response has already determined the outcome of that step.

A consensus process should be used for decision-making for both GAET and IDP. Differing opinions should be documented and resolved, if possible. However, a simple majority of the panel is sufficient for final decisions regarding priority of activities. The IDP should apply objective decision criteria and minimize subjectivity.

The IDP should be described in a plant administrative document that includes the designated chairman, panel members, and panel alternates; required training and expectations for the chairman, members, and alternates; requirements for a quorum, attendance records, agendas, and meeting minutes.

2.1 Documentation

GAET: The GAET evaluation results and summary, including basis and description of important considerations/characteristics for plant-specific assessment by the SMEs and IDPs, will be documented and provided to the industry and the NRC for information. Since the prioritization process addresses only scheduling of activities, 10 CFR 50 Appendix B does not apply. Documentation will be maintained to facilitate any subsequent generic update/re-evaluation of the issue, as appropriate. The specific information that should be provided by the GAET includes:

- A description of the specific regulatory issue or proposed activity, including success criteria
- Related and publically available references such as
  - Regulatory documents including Regulatory Analyses; Orders; Commission Papers (SECYs and associated staff requirements memoranda (SRMs)); NUREG and NUREG/CR reports; relevant Commission and Advisory Commission on Reactor Safeguards (ACRS) meeting slides and transcripts; regulatory guides and interim staff guidance; and generic communications such as bulletins and information notices. (Safeguards information shall be treated consistent with current practice).
  - Industry documents including NEI guidance documents and correspondence with the NRC; research reports (e.g., Electric Power Research Institute and Owners Groups); and conference papers
  - International Atomic Energy Agency and Nuclear Energy Agency reports
- Screening question results related to the determination of any impact (Step 1), assessment of more than minimal impact (Step 2), and qualitative/quantitative determination of safety importance level (Step 3A/3B) and associated discussion
- Technical bases for conclusions regarding nuclear safety importance; the generic security significance assessment (if appropriate); and EP and RP issue significance characterization if available. It is expected that the effectiveness determinations for security, EP, and RP will be very plant-specific. Reliability importance assessment is expected to be almost completely plant-specific.
- Considerations and characteristics that may affect the plant-specific importance determination, particularly for safety. For example, the GAET may determine that based on reactor fleet considerations, the existing level of risk of an external initiator is 10-5 to 10-4 / yr CDF on average (Medium). If

information is available, the GAET would convey what attributes could make the plant-specific assessment higher or lower.

IDP: The prioritization process should be documented through plant procedures or other administrative controls. The decisions of the IDP, including a summary of the basis, should be documented and retained as plant records. In particular, the assessment of GAET-identified important issue considerations/characteristics and how they apply to the plant, and a basis for significant plant-specific departures from the general GAET ranking, should be noted. Since the prioritization process affects only scheduling of activities, 10 CFR 50 Appendix B does not apply. Individual licensees will determine an appropriate requirement for documentation to be maintained to facilitate periodic update/re-evaluation of the issue, similar to other plant programs or procedures governing the licensee's expert panels.

Documentation on the prioritization of each issue should be maintained onsite to assist in periodic review/update and to accommodate any NRC audits. The level of documentation should be such that a sufficient basis is provided for a knowledgeable individual to independently review the information and reach the same conclusion. The basis for any engineering judgment and the logic used in the determination should be documented to the extent practicable and to a degree commensurate with the safety significance and complexity of the issue/activity. The items considered by the GAET/SME/IDP must be clearly stated.

For each issue licensees should maintain:

- a copy of the generic package, if applicable
- a copy of the plant-specific package the SME submits to the plant IDP
- a summary of the plant IDP discussion on the issue
- a revised copy of the package, if applicable
- the Priority assigned to the issue and any impact on schedule (e.g., none, accelerate, defer)

For each prioritization period, licensees should maintain:

- a list of issues prioritized during that period and their Priority 1 to 5
- the basis for decision analysis results to differentiate within priority levels, if applicable

- supporting documentation for adjusting licensing/regulatory schedules of issues as applicable

2.2 IMPORTANT ATTRIBUTES OF THE PROCESS

In order to support an aggregation that fully accounts for relevant insights in an integrated manner, for each step in the process, there are important common elements that should be considered in the assessment, as follows:

1. Ensuring the issue and success criteria are well defined

   Although the goal of the overall process is to have clearly defined issues and success criteria prior to evaluation by GAET or IDP, the actual assessment may indicate that additional definition is appropriate. In addition, as the assessment progresses to subsequent steps, the actual conduct of the assessment may identify additional considerations not identified in the initial definition(s). Thus, it is critical that the specific issue and potential options for addressing it are appropriately defined and communicated.

2. Being realistic where appropriate so as to not bias the prioritization

   The level of realism and level of analyses will vary depending on the issue, but in order to avoid biasing, realistic analysis is the objective. A pairwise comparison, generic and plant-specific integrated expert panel, and matrices with wide ranges are included in the process to limit the potential impact of uncertainty. Note that if the risk impact is exceedingly small, or clearly large, a bounding evaluation can suffice.

3. Considering uncertainty

   Although the characterization and importance matrix in Table 3-1 does not require quantitative risk measures, the matrix is based on relative risk and is consistent with the Significance Determination Process (SDP) process of green, white, yellow and red. Thus, each of the entries on current risk differs by about a factor of ten. This should address most concerns on uncertainty for the context of the prioritization process. However, both the GAET and IDP need to be aware of specific issues, such as external events, for which uncertainty considerations may produce risk estimates with multiple orders of magnitude.

4. Considering the need for additional information

   There is the potential that for the assessment of some issues more timely or recent information than originally provided by the GAET will be needed, for example, external flooding at some sites. For such issues, existing NRC-industry practices, including public meetings and interactions between the industry and NRC subject-matter experts, may provide a source of additional information. The decision to pursue additional information should occur sufficiently early in the process such that performing this action does not become the driving factor in delaying a risk-informed prioritization decision and, ultimately, the timely implementation of a regulatory activity.

5. Evaluating the overall nature of the risk impact of a potential action

   Beneficial and adverse effects should be considered (e.g., replacing a small pump with a large pump could reduce the available margin of an emergency diesel generator (EDG); closing and depowering pressurizer power/pilot operated relief valves (PORV) block valves to prevent spurious operation could reduce effectiveness of feed and bleed).

6. Identifying the overall extent of the impact of an individual issue when considering other issues

   The specific intended function of implementation, as well as other correlated or indirect effects, should be considered (e.g., FLEX provides mitigation for more than external hazards even though that is its fundamental intended purpose). In other words, one specific plant modification could impact the specific activity under consideration as well as multiple other separate plant modifications. As discussed above, this could include both positive as well as negative impacts that may not be immediately evident when activities are considered individually. For example, implementation of FLEX impacts the potential benefits of future changes to the station blackout rule. Thus, the sequence of the resolution of several issues could have a beneficial or adverse impact on the priority of an issue. Guidance on pairwise comparison is included to support both a peer check on issue priority as well as for support in identifying any commonalities.

2.3 TYPES OF MODELS AND EVALUATION TOOLS

The models and evaluation tools available or achievable are extensive, and the philosophy for the prioritization process is to use currently available sources of risk information, with understanding of their benefits and limitations. The appropriate model/tool will depend on the issue. For the prioritization process, the best available PRA should be used, i.e., current existing PRA, without resource intensive re-modeling. However, use of PRAs that meet the Quality Standards are beneficial in substantiating any request for exemptions from regulatory due dates. In this context, indicating the level of quality of the tool used for decision-making can provide additional confidence in the characterization of an issue. Choosing a less formal, qualitative approach when more appropriate tools are available should be avoided.

Models/tools include:

1. qualitative checklist or flowchart
2. comparison to a previously ranked issue(s)—which is addressed by using a pairwise comparison
3. review of previous studies (e.g., severe accident mitigation alternatives (SAMA) and issue-specific cost-benefit evaluations)
4. direct use of an existing PRA model
5. adaptation of an existing PRA model
6. development of a focused scope assessment
7. direct, adaptive or new deterministic model, such as to characterize margin in system capability

2.4 EVALUATION

The importance characterization starts with a specific issue and associated issue definition and success criteria. This is a precondition for starting the evaluation. In addition, available information is collected, including NRC and Industry information. Available cost-benefit analyses and SAMA-like analyses are also collected, as available.

In addition the effectiveness of existing or planned programs and processes to address the underlying issue (e.g., ROP, mitigating system performance index (MSPI) program, maintenance rule, fire-protection programs) should be considered. The industry and the NRC may have programs and processes that either could directly, or with changes, address the underlying issue and eliminate the desirability of developing new programs or conducting new analyses. To be effective, such programs and processes would be expected to provide the information and actions needed to address the underlying issue. Further an alternate, smarter action on a plant-specific basis may be identified during the evaluation such that either the cost would be reduced and/or the risk further reduced compared to using the offered success criteria. In such cases, a proposed change in scope to a regulatory issue would require NRC approval, consistent with current regulatory requirements.

When evaluating an issue, it is desirable to the extent practical to evaluate a major issue or program as a whole. For example, NFPA 805 may be evaluated considering the overall changes in risk given full implementation. On further evaluation, it may be possible to separate the issue or program into its piece parts, in order to assign the highest priority to those specific changes that offer the greatest (and soonest) risk reduction.

There are two ways to use the process as follows:

- direct use by the GAET/IDP
- use by a separate team that would follow the process and develop an objective assessment that the GAET/IDP would then use to implement Figure 3-1 or 3-2

The characteristics of the issue will determine the most efficient way. Considerations include complexity of an issue(s) and the potential desire to have refined analyses in advance of the GAET/IDP deliberations.

The success criteria (SC) for a specific issue can range from a potential plant change (e.g., hardware, procedure change, training, staffing) to the conduct of an evaluation.

- For a potential plant change, treat the assessment as if the plant change could impact safety/risk to avoid any presumptive bias on the overall characterization of the issue. (This could include a change aimed at reducing risk [e.g., FLEX] or a change aimed at preventing or minimizing a potential increase in risk due to a future increase in hazard level or frequency [e.g., cyber attacks].)
- For the conduct of an evaluation, treat the assessment as if the evaluation could identify plant changes, which if implemented, could impact safety/risk. (In the cost evaluation, note that both evaluation costs and potential implementation costs will need to be estimated.)

Note: Although the expectation is that an issue and associated definition entering this process is intended to reduce risk/improve safety/security/EP/RP/reliability, there is a potential for the SC to be adverse to risk/safety/security/EP/RP/reliability.

The process addresses this possibility. If an adverse impact is identified, there are alternative paths:

- Continue using the process and address the adverse impact in the overall assessment of benefit and cost.
- Develop and implement a plan for interacting with the NRC using normal processes and procedures (regardless of whether the SC was established by the NRC or the industry). A "plan" here means the approach to communicating with the NRC including, as appropriate, a recommended course of action.

2.5 Insufficient Confidence

This is a sequential screening process. Thus, at any step in the process, except Step 3B, the GAET or plant can continue to the next step if there is insufficient confidence in the assessment result for the previous step. Alternatively, the GAET/plant may develop a plan to gain the information needed to have sufficient confidence. The plan could include interaction with the NRC, conduct of analyses, etc. This applies on a plant-specific basis also. The plant IDP may advise the performance of additional analyses to improve confidence in the outcome of any step. Sufficient confidence exists when the GAET/IDP concludes that the safety importance and/or priority outcome would not change if additional information was obtained or developed.

3.0 SAFETY IMPORTANCE CHARACTERIZATION

Figures 3-1 and 3-2 provide the generic and plant-specific processes for safety importance characterization, respectively.

Figure 3-1
Progressive Screening and Evaluation
Safety Importance (Generic)

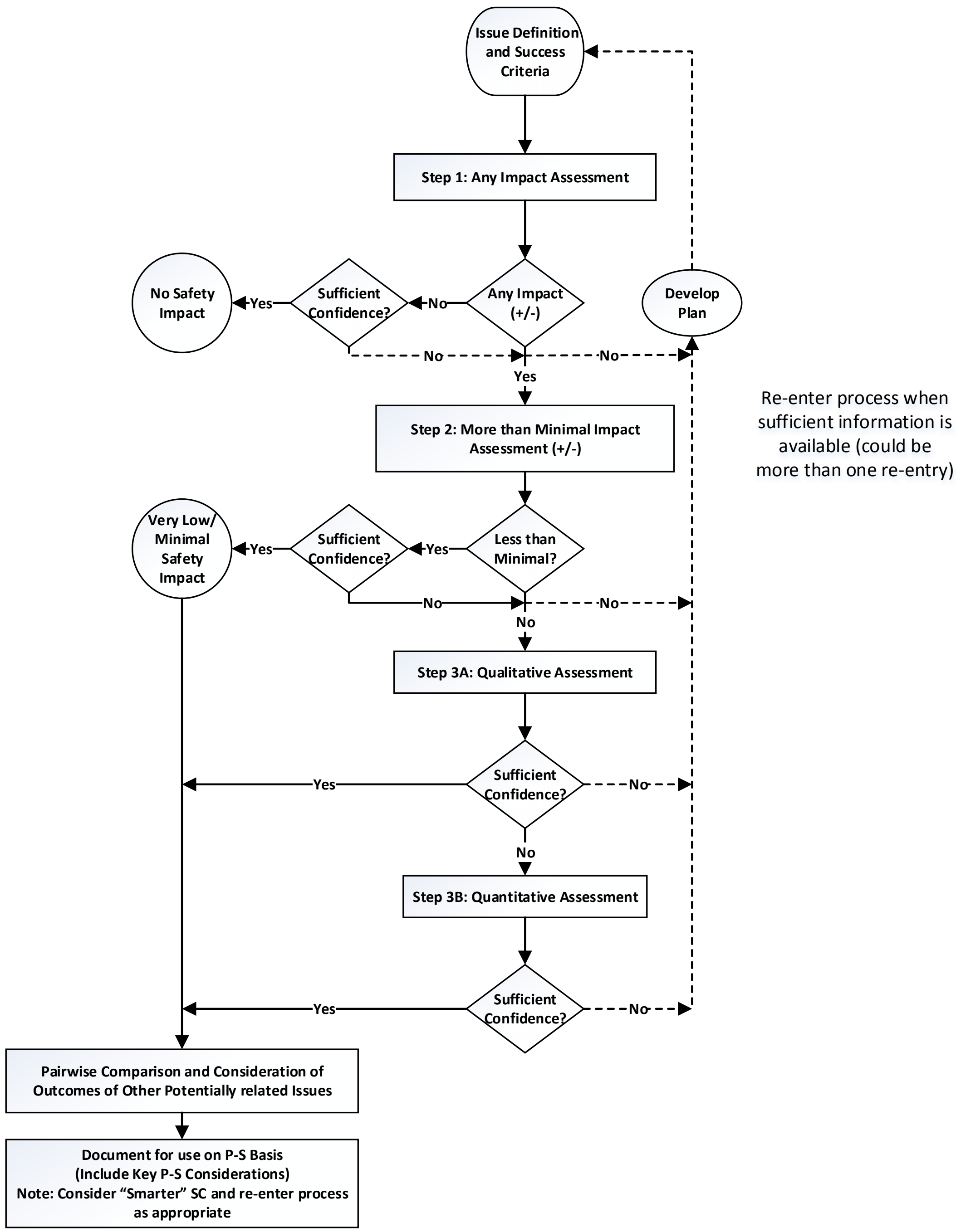
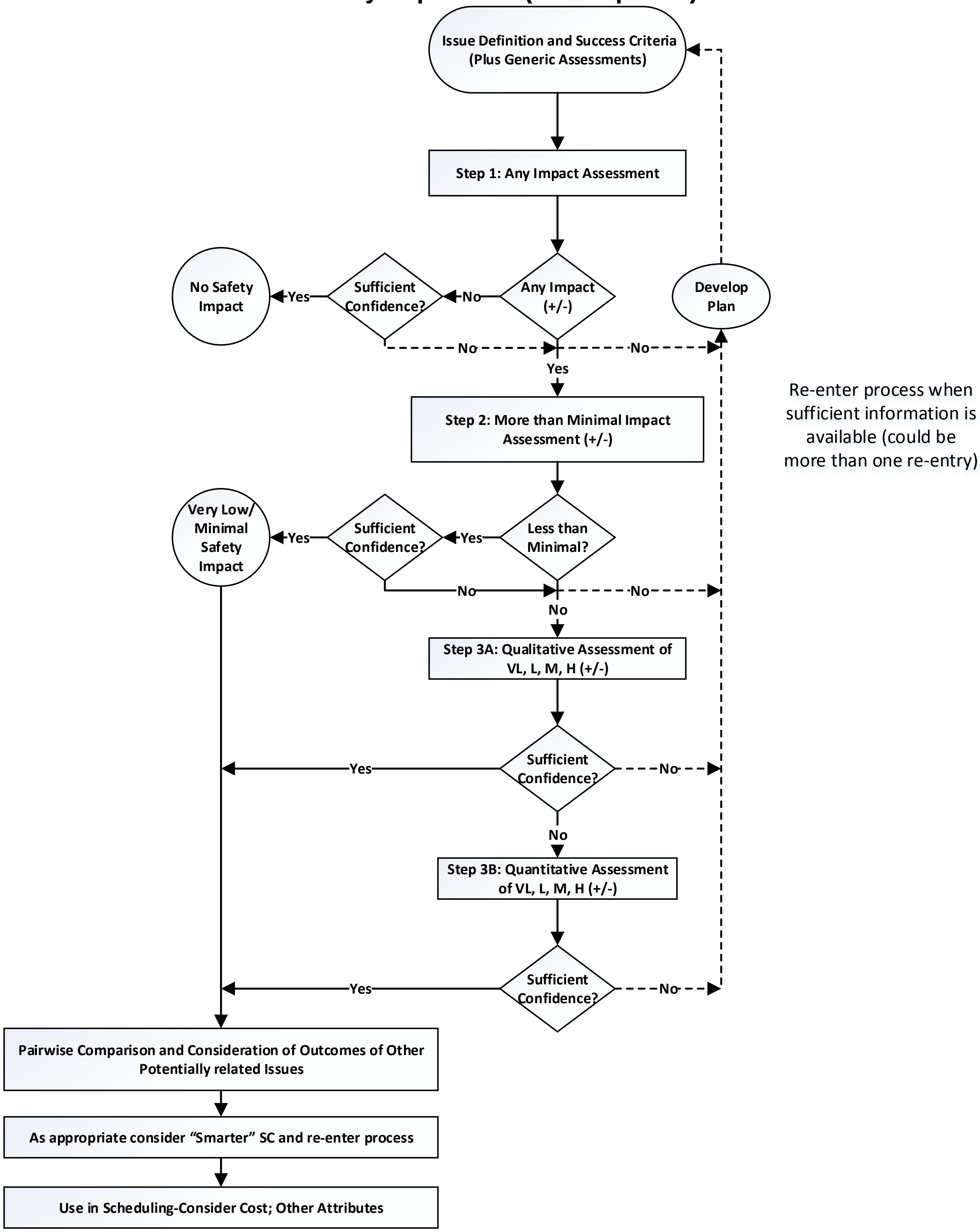


Figure 3-2
Progressive Screening and Evaluation
Safety Importance (Plant-Specific)



The safety importance characterization process is intended to use currently available risk information.

The process is a progressive screening and evaluation, and includes three basic steps: 1) a series of screening questions to address the "no safety impact" step; 2) a series of similar screening questions to address the "more than minimal" impact on safety step; and 3) evaluation using qualitative and quantitative risk estimates to assign high, medium, low, or very low importance to activities that do not screen in Steps 1 and 2. For evaluations, the overall safety importance is determined based on a matrix, provided in Table 3-1.

Examples are provided in the Appendix to further illustrate the process steps.

Step 1 (Screening for any impact)

Step 1 involves screening the issue or activity for "any" impact versus "no" impact on safety. The evaluation should screen for both beneficial and adverse effects.

Thus, a change that decreases/increases the reliability of a function whose failure could initiate an accident would be considered to adversely/beneficially affect risk. Similarly, changes that would introduce a new type of accident or malfunction of structures, systems or components (SSC), or eliminate a type of accident, would screen in.

If a change has both beneficial and adverse effects, the change should be screened in.

The Step 1 screening process is not intended to be excessive or resource intensive and is not concerned with the magnitude of adverse/beneficial effects that are identified. Any change that adversely or beneficially affects risk is screened in. The magnitude of the effect (i.e., is the minimal increase standard met?) is considered in the more detailed evaluation in Step 2.

Screening determinations are made based on the engineering/technical information supporting the potential action. The screening focuses on functions, etc., and ensures the essential distinction between no impact, minimal impact and more than minimal impact addressed in Steps 2 and 3. Technical/engineering information, e.g., design evaluations, that demonstrates changes have no adverse/beneficial effect on functions, methods of performing or controlling functions, or evaluations that demonstrate that intended functions will be accomplished may be used as basis for screening out the potential change.

The guidance and examples here are used to support this screening. The screening on no impact addresses the following set of questions:

Does the proposed activity or issue:

1. **☐ YES ☐ NO Result in an impact on the frequency of occurrence of a risk significant accident initiator?**
2. **☐ YES ☐ NO Result in an impact on the availability, reliability, or capability of SSCs or personnel relied upon to mitigate a risk significant transient, accident, or natural hazard?**
3. **☐ YES ☐ NO Result in an impact on the consequences of a risk significant accident sequence?**
4. **☐ YES ☐ NO Result in an impact on the capability of a fission product barrier?**
5. **☐ YES ☐ NO Result in an impact on defense-in-depth capability or impact in safety margin?**

If ALL the responses are NO, issue or activity screens to NO IMPACT and Nuclear Safety Importance is None.

If ANY response is YES, continue on to Step 2.

In addressing the above questions, there is similarity with the questions in 10 CFR 50.59 and the guidance in NEI 96-07 (Reference 1). Thus, for Question 3 above, consequence is intended to mean radiological dose from risk-significant accident sequences. The impact should be direct, such as an improved containment spray system could reduce radiological releases in a core damage accident. However, reducing the frequency of core damage is addressed elsewhere and is not the intent of this question. In lieu of dose, impact on containment performance (system performance, hydrogen control, isolation, ultimate pressure capacity, etc.) can be used as a surrogate.

Capability addresses the capacity of an SSC or personnel. Consider the following examples:

- The flow capacity of a system could be increased by replacing a pump with a higher capacity pump.
- The tornado resistance of a wall could be increased by adding additional supports.

- The seismic capacity of a relay could be increased by replacing the relay with a higher capacity relay.

Step 2 (Screening for more than minimal impact)

This step involves addressing the following set of questions, which are modified versions of the Step 1 questions:

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?**
2. **☐ YES ☐ NO Result in more than a minimal improvement in the availability, reliability, or capability of SSCs or personnel relied upon to mitigate a risk significant transient, accident, or natural hazard?**
3. **☐ YES ☐ NO Result in more than a minimal decrease in the consequences of a risk significant accident sequence?**
4. **☐ YES ☐ NO Result in more than a minimal improvement in the capability of a fission product barrier?**
5. **☐ YES ☐ NO Result in more than a minimal improvement in defense-in-depth capability or improvement in safety margin?**

If ALL the responses are NO, issue or activity screens to MINIMAL IMPACT and Nuclear Safety Importance is Very Low.

If ANY response is YES, continue on to Step 3.

Guidance on addressing the above questions is provided below. Note that any question answered “NO” in Step 1, will be answered “NO” in Step 2.

Question 1: Does the activity result in more than a minimal decrease in the frequency of a risk-significant accident initiator?

In answering this question, the first step is to identify the risk-significant accident initiators that have been evaluated that could be affected by the proposed activity. For regulatory-initiated actions, this should have been determined on a generic basis by the NRC. Then a determination should be made as to whether the frequency of these accident initiators occurring would be more than minimally decreased. Accident initiators can be divided into categories, whether for at power or low power shutdown conditions, for example:

Accident Initiator Categories (Representative)	Risk Significant?	More than Minimal Decrease or Adverse?
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		
Spent Fuel Pool		
Low power and shutdown conditions		

Risk significance: Risk Significance should be based on matrix benchmarks in Table 3-1, which are based on SDP risk significance. Using readily available information, accident initiators that are not risk-significant, i.e., minimal or less than minimal, <u>generally</u> are those:

- contributing less than 1E-6/year and 1E-7/year for CDF and LERF, respectively (Based on SDP), OR
- contributing less than 1% of total CDF/LERF (consistent with RG 1.174), OR
- contributing to a less than 10% change in frequency (consistent with 50.59 guidance)

If the proposed activity would not meet one of the above criteria, the risk significance of an issue is considered further. If information is not readily available, the risk significance should be determined by comparison to other issues evaluated. While formal guidance on uncertainty treatment is not provided here, the impact on the determination should be considered.

External hazards: Practically, external hazard frequencies cannot be reduced or increased by a plant-initiated or NRC-initiated change. However, the frequency and/or severity might be changed for certain external hazards (such as external flooding) with changes beyond the nuclear power plant site. For example strengthening a dam could reduce the frequency/severity of an external flood that could affect the nuclear power plant site. Such changes can be considered in this process if under the control of the licensee. Otherwise changes related to external hazards will be considered in the second question.

Considerations for changes to accident initiator frequencies: The frequency of accident initiators can be changed in several ways, such as:

Considerations	Potential Action Effect?	More than Minimal or Adverse?
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		
Addition of equipment		
Changes in operating practices		

The industry, the NRC and each plant have programs and practices for managing accident initiator frequency. Existing programs and practices will support determination of changes in frequency (10 CFR 50.59, NFPA 805, aging management programs, piping integrity programs, etc.).

Reasonable engineering practices, engineering judgment and PRA techniques, as appropriate, should be used in determining whether the frequency of occurrence of a risk-significant accident initiator would more than minimally decrease as a result of implementing a proposed activity. A large body of knowledge has been developed in the area of accident frequency and risk-significant sequences through plant-specific and generic studies. This knowledge, where applicable, should be used in determining what constitutes more than a minimal decrease in the frequency of occurrence. The effect of a proposed activity on the frequency of a risk significant accident initiator must be discernible and attributable to the proposed activity in order to exceed the more than minimal decrease standard.

Examples: The following are examples where there is not more than a minimal decrease in the frequency of occurrence of a risk-significant accident initiator.

Example 1

The proposed activity has a negligible effect on the frequency of occurrence of a risk-significant accident initiator. Consistent with the guidance in NEI 96-07, a negligible effect on the frequency of occurrence exists when the change in frequency is so small or the uncertainties in determining whether a change in frequency has occurred are such that it cannot be reasonably concluded that the frequency has actually changed (i.e., there is no clear trend toward decreasing the frequency). An example could be a process change that cannot be demonstrated to have a positive impact, e.g., implementation of a new ASME code on ISI.

Example 2

The change in frequency of occurrence is not more than a minimal decrease if ANY of the following criteria are met:

- The change affects those accident initiators contributing in total less than 1E-6/yr and 1E-7 /yr for CDF and LERF, respectively, OR
- The change affects those accident initiators contributing in total less than 1% of total CDF/LERF (consistent with RG 1.174), OR
- The calculated change in frequency in total is less than 10%.

Question 2: Does the activity result in more than a minimal improvement in the availability, reliability or capability of SSCs or personnel relied upon to mitigate a risk-significant transient, accident or natural hazard?

This includes the reactivity control function, so anticipated transients without scram (ATWS) is addressed here, as ATWS is not an accident initiator, but instead an accident sequence. In answering this question, the first step is to identify the risk significant SSCs and human actions that have been evaluated that could be affected by the proposed activity.

- For regulatory-initiated actions, this may have been determined on a generic basis by the NRC. If not, guidance herein will develop this information.
- Then, a determination should be made as to whether availability, reliability, or capability of SSCs or personnel relied upon to mitigate a risk-significant transient, accident or natural hazard would be more than minimally decreased.

Similar to accident initiators the availability, reliability or capability of SSCs or personnel can be changed in several ways, as described in the table below:

Considerations	**Potential Action Effect?**	**More than Minimal or Adverse?**
Changes in maintenance, testing, 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		
Addition of equipment		
Strengthening of equipment		
Moving equipment (to reduce the impacts of spatial events)		
Eliminating the need for recovery action (RA)		
Improving performance shaping factor related to human performance		
Changes in operating practices		

The industry, the NRC and each plant have programs and practices for managing availability, reliability, capacity and human performance (A/R/C/H). Existing programs and practices will support determination of minor changes in A/R/C/H (10 CFR 50.59, NFPA 805, aging management programs, piping integrity programs, etc.). Potentially major changes (such as changes in redundancy and diversity, additional equipment, strengthening equipment, moving equipment, eliminating RAs and improving performance shaping factors) will require more detailed evaluations.

Risk Significance: Risk significance should be based on matrix benchmarks in Table 3-1, which are based on SDP risk significance. SSCs/human actions that are not risk-significant, i.e., minimal or less than minimal, generally are those associated with potential sequences:

- contributing less than 1E-6/year and 1E-7/year for CDF and LERF, respectively, *unless* the issue being addressed could increase risk above these values, OR
- contributing less than 1% of total CDF/LERF (consistent with RG 1.174), *unless* the issue being addressed could increase risk above these values, OR
- contributing to a less than a 10% change in likelihood of failure (availability, reliability, capability, personnel performance); consistent with 50.59 guidance.

If the proposed activity would not meet one of the above criteria, the risk significance of an issue is considered further. If information is not readily available, the risk significance should be determined by comparison to other issues evaluated.

The term "risk-significant" refers to the structures, systems, and components (SSCs) performing risk-significant functions, including nonsafety-related and safety-related SSCs and human performance. NUMARC 93-01 (Reference 3) provides specific guidance on risk-significant criteria. In determining whether there is more than a minimal decrease, the first step is to determine what SSCs and human actions are affected by the proposed activity. Next, the effects of the proposed activity should be determined. This evaluation should include both direct and indirect effects.

Direct effects are those where the proposed activity affects the issue (e.g., a motor change on a pump or changing the mounting of an electrical cabinet). The activity changes the performance of the SSC by increasing its reliability or increasing its margin to failure under accident conditions. One can directly attribute the overall improvement in how the SSC performs by quantitative analysis, operating experience, or engineering judgment. Indirect effects are those where the proposed activity could affect other risk contributors. For example, installing FLEX equipment to address extended loss of AC power for external initiators could also reduce plant risk by having additional equipment available for internally initiated events such as loss of main feedwater transients.

After determining the effect of the proposed activity on the risk-significant SSCs and human actions, a determination is made of whether the likelihood of failure has decreased more than minimally. Qualitative engineering judgment and/or an industry precedent is typically used in 10 CFR 50.59 evaluations and can be used here to determine if there is more than a minimal decrease in the failure probability.

An appropriate calculation can be used to demonstrate the change in likelihood in a quantitative sense, if available and practical. The effect of a proposed activity on the failure probability must be *discernible and attributable* to the proposed activity in order to exceed the more than minimal decrease standard.

A proposed activity is considered to have a negligible effect on the likelihood of failure when a change in likelihood is so small or the uncertainties in determining whether a change in likelihood has occurred are such that it cannot be reasonably concluded that the likelihood has actually changed (i.e., there is no clear trend toward decreasing the likelihood). A proposed activity that has a negligible effect satisfies the minimal increase standard.

Potential SSC changes, such as increased structural capacity, to address earthquakes, tornadoes and other natural phenomena should also be treated as potentially affecting the likelihood of failure.

Examples: Examples in the Appendix illustrate cases where there would/would not be more than a minimal decrease. [**Note:** The conclusions reached here are not intended to be final as these examples are intended to illustrate the process.]

Question 3: Does the activity result in more than a minimal decrease in the consequences of a risk-significant accident sequence?

In answering this question, the first step is to identify the risk significant sequences that have been evaluated that could be affected by the proposed activity.

- For regulatory-initiated actions, this may be determined on a generic basis by the NRC. If not, this information will need to be developed.

- Then, a determination should be made as to whether the consequences would be more than minimally decreased.

Risk significance: Risk significance should be based on matrix benchmarks in Table 3-1, which are based on SDP risk significance. If available using readily available information, accident sequences that are not risk-significant, i.e., minimal or less than minimal, generally are those:

- contributing less than 1E-6/year and 1E-7/year for CDF and LERF, respectively, OR
- contributing less than 1% of total CDF/LERF (consistent with RG 1.174), OR
- contributing to a less than 10% change in consequences.

If the proposed activity would not meet one of the above criteria, the risk significance of an issue is considered further. For example, a generic regulatory activity is proposed that would address seismic issues. The site characteristics as well as plant-specific PRA are such that the plant is not susceptible to major seismic concerns. The seismic hazard is very low and the plant design sufficiently robust such that the estimated CDF from seismic contribution is well below 1E-6/year and likewise LERF is below 1E-7/year. Therefore, any further decrease in seismic risk would be just a fraction of the existing risk level and would be less than minimal. It is further expected that all U.S. plants have total CDF (including unquantified external hazards) of 1E-4/year or less. If an activity addresses the risks or sequences amounting to only 1% of the total CDF/LERF, then the risk that might be mitigated is less than the 1E-6/year CDF and 1E-7/yr LERF criterion above. For plants with total CDF in the 1E-5/year to 1E-4/year, the incremental benefits of any modifications to address the issue are further diminished. Finally, in addressing the definition of what constitutes a less than minimal decrease in consequences, a 10% decrease in dose for risk-significant sequences is used as the criterion. This threshold has a basis generally consistent with the 10 CFR 50.59 guidance in NEI 96-07 (Reference 1). It is widely acknowledged that there are increasing uncertainties going from the Level 1 portion of a PRA study (core damage frequency estimation) to the Level 2 (containment performance) to the Level 3 (offsite dose consequences). A 10% increase in calculated consequence is such that it could not be reasonably concluded that the consequences have actually changed. Small changes in inputs and assumptions could easily have more of an effect than a calculated change of 10% change in offsite dose from a severe accident sequence.

If information is not readily available, the risk significance should be determined by comparison to other issues evaluated.

In determining if there is more than a minimal decrease in consequences, the first step is to determine which accidents may have their radiological consequences affected as a direct result of the proposed activity. Examples of questions that assist in this determination are:

(1) Will the proposed activity change, i.e., improve, the effectiveness of an action?

(2) Will the proposed activity play a direct role in mitigating the radiological consequences?

In lieu of dose the following should be considered:

- containment bypass
- containment isolation and capacity
- hydrogen
- long-term containment integrity

Question 4: Does the activity result in more than a minimal improvement in the capability of a fission product barrier?

This evaluation focuses on the fission product barriers—fuel cladding, reactor coolant system boundary and containment. Note that the prior question also indirectly addresses containment. Guidance on barrier definitions and impacts on barriers can be found in 10 CFR 50.59 guidance provided in NEI 96-07 (Reference 1). As discussed in NEI 96-07, each barrier has associated with it specific design basis parameters such as fuel cladding temperature, reactor coolant system cool-down rate, and containment pressure. It is expected to be rare that a proposed activity or regulatory issue will result in an impact on the design basis parameters that can be directly calculated. Rather, judgment is required here in ascertaining whether the improvement is more than minimal. For example, an improved fuel design that significantly reduces the potential for pellet-clad interaction probably meets the threshold for more than minimal. A routine change in fuel management strategy that meets all acceptance criteria does not. An improved reactor coolant pump (RCP) seal design that measurably reduces the likelihood of seal failure as well as the leakage rates given loss of seal cooling events is likely to be more than minimal. The addition of an AC-power independent containment spray to mitigate core damage sequences involving station-blackout also is likely to meet the more than minimal improvement threshold. Changing the median failure pressure of containment from 120 psig by 2 psig will not impact the PRA results and is not more than minimal.

Question 5: Does the Activity Result in more than a minimal improvement in the defense in depth capability or safety margin?

Regulatory Guide 1.174 (Reference 2) provides guidance.

Use Step 3A and/or 3B

Note that the user may skip to Step 3B if appropriate quantitative information is readily available or can be developed. It is not necessary to perform both Steps 3A and 3B.

Step 3A (determining high, medium, low, or very low safety importance using qualitative approach)

Step 3A uses Table 3-1, combined with the guidance for Step 2, to place a potential action into a safety importance category as follows. The ranges in the first column are based on the SDP ranges for CDF and LERF.

The table is used as follows:

- First determine the existing risk level (CDF or LERF) associated with the issue using available information. This may be quantitative or based on a comparison to a previously evaluated issue. This establishes the relative risk significance. Note that LERF thresholds are one order of magnitude lower than those of CDF.
- Then determine how much the proposed activity would reduce the relative risk. This establishes the importance of the proposed activity.

There could be some degree of discretion in assigning the existing risk level (row in Table 3-1) and the potential impact of the action in resolving the issue (column). Given these potential uncertainties, if an issue appears to straddle two possible regions of importance, then the higher of the 2 importances should be used.

Note: Grey is used to denote those issues with high relative risk importance for which the proposed activity/action is ineffective. Consideration should be given to identifying an effective activity/action.

The outcomes of Step 3A are:

- high, medium, low, very low importance, OR
- continue to Step 3B, OR
- develop a plan

Table 3-1 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[1]

Current Risk associated with Issue	Potential Impact of Action Resolving Issue (Reduction in Risk)				
	None	Very Small/Minimal	Small	Medium	High
	0%	0 to 25%	25 to 50%	50% to 90%	>90%
	Importance				
Green (VL) LB	Very Low	Very Low	Very Low	Very Low	Very Low
Green (VL) Mid	Very Low	Very Low	Very Low	Very Low	Very Low
Green (VL) UB	Very Low	Very Low	Very Low	Very Low	Very Low
White (L) LB	Very Low	Very Low	Very Low	Very Low	Very Low
White (L) Mid	Very Low	Very Low	Low	Low	Low
White (L) UB	Very Low	Low	Low	Low	Low
Yellow (M) LB	Very Low	Low	Low	Low	Low
Yellow (M) Mid	Very Low	Low	Medium	Medium	Medium
Yellow (M) UB	Very Low	Medium	Medium	Medium	Medium
Red (H) LB		Medium	Medium	Medium	Medium
Red (H) Mid		High	High	High	High
Red (H) UB		High	High	High	High

[1] The thresholds in the left column are consistent with the SDP and are (in units of per yr), for CDF: Green/White = 10^{-6}, White/Yellow = 10^{-5}, Yellow/Red = 10^{-4}; and for LERF: Green/White = 10^{-7}, White/Yellow = 10^{-6}, Yellow/Red = 10^{-5}.

Step 3B (determining high, medium, low, or very low safety importance using quantitative analyses)

In Step 3B, existing information and new information/analyses (e.g. focused scope analyses as needed), is used to estimate the current risk level associated with the issue and the impact of the proposed actions on reducing risk. Based on the outcome of the assessment a safety importance is determined. The types of models possibly available were noted earlier in this document.

Consistent with the SDP process, the safety importance determination using Step 3B is as follows, using the higher of the importances based on changes in CDF and LERF:

HIGH:	ΔCDF > 1E-4 /yr, or ΔLERF > 1E-5 /yr
MEDIUM:	1E-4 /yr ≥ ΔCDF > 1E-5 /yr, or 1E-5 /yr ≥ ΔLERF > 1E-6 /yr
LOW:	1E-5 /yr ≥ ΔCDF > 1E-6 /yr, or 1E-6 /yr ≥ ΔLERF > 1E-7 /yr
VERY LOW:	ΔCDF ≤ 1E-6 /yr, or ΔLERF ≤ 1E-07 /yr

4.0 IMPORTANCE CHARACTERIZATION OF OTHER CATEGORIES

Following safety importance characterization (high, medium, low, very low, none), an overall characterization is performed that takes into account additional factors such as emergency planning, security, and radiological protection. The primary objective of this characterization is to capture the significance of the issue that was not already captured by the factors considered under safety importance.

4.1 SECURITY

Security importance characterization includes two basic steps: 1) a flowchart series of screening questions to address the "no impact" step; and 2) use of qualitative or quantitative effectiveness estimates to assign high, medium, low, or very low importance to activities that do not screen out in Step 1. For Step 1, the flowchart in Figure 4.1-1 is used. For Step 2, the overall security importance is concluded based on a matrix, provided in Table 4-1.

Step 1 (Screening for any impact)

Complete the flowchart in Figure 4.1-1 to determine the current significance associated with the issue.

The IDP should first assess the issue assuming there is no target set impact. Then, a Safeguards qualified IDP should determine if there is an adverse impact on a target set function (noted on Figure 4.1-1 with a dashed line). If no adverse impact, then determinations from the initial IDP assessment are confirmed. If the current significance associated with the issue is anything other than "None," continue to Step 2.

The following considerations should be applied to the Step 1 assessment performed utilizing the flowchart in Figure 4.1-1;

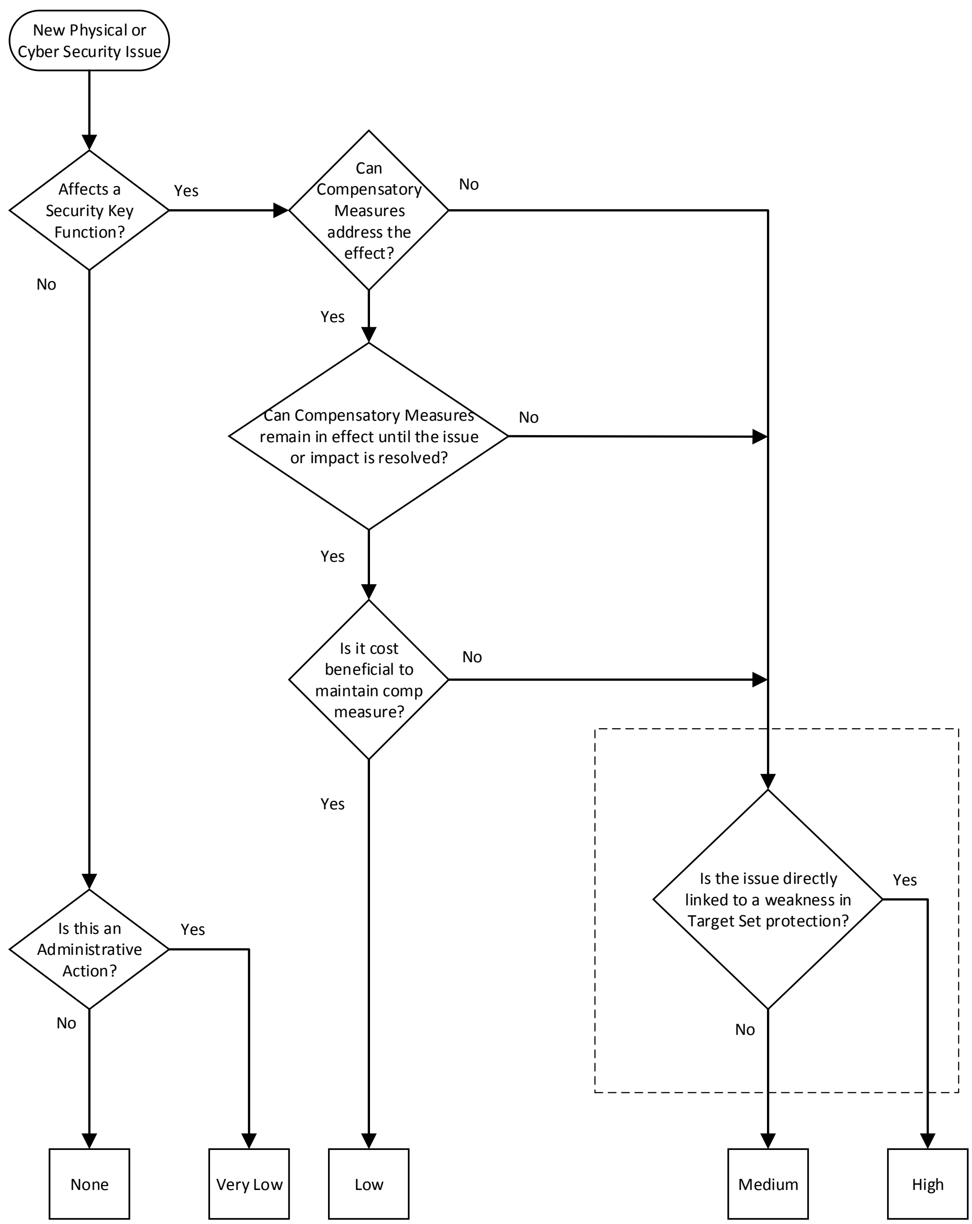
1. Security Key Functions are defined as the ability to Detect, Assess, Delay, and Respond in accordance with the Physical Security Program required by 10 CFR 73.

2. Generally, Compensatory Measures are temporary and not expected to remain long-term. Consideration should be given to the continued effectiveness of Compensatory Measures applied for an extended duration. The NRC has issued several documents on this topic, e.g., NUREG-045 and IN 86-88. These documents should be reviewed during consideration of the use of Compensatory Measures for extended durations.

3. When considering Target Set protection weaknesses, the review should address the impact on the Safety, Security, or Emergency Response function of the Target Set Element impacted. The details of this determination may be considered Safeguards information.

Figure 4.1-1

Security Issue Importance Determination – Step 1



Step 2 (Determine issue's security importance)

See Section 4.4.

4.2 Emergency Preparedness

Emergency preparedness (EP) importance characterization includes two basic steps: 1) a flowchart series of screening questions to address the "no impact" step; and 2) use of qualitative or quantitative effectiveness estimates to assign high, medium, low, or very low importance to activities that do not screen out in Step 1. For Step 1, the flowchart in Figure 4.2-1 is used. For Step 2, the EP importance is concluded based on a matrix, provided in Table 4-1.

Step 1 (Screening for any impact)

If the issue has no nexus to EP, the EP importance is None. If the issue has any nexus to EP, complete the flowchart in Figure 4.2-1 to determine the current significance associated with the issue. If the current significance associated with the issue is anything other than "None," continue to Step 2. Guidance on addressing the flowchart questions is provided below.

1) Activity to maintain or restore compliance with <u>current</u> EP requirements?

Answer "Yes" if the activity is necessary to maintain or restore compliance with current emergency preparedness regulations or the site Emergency Plan (as defined in Regulatory Guide 1.219).

1a) Activity in response to an NRC finding?

1b) Is finding significance greater than Green?

2) Activity to achieve compliance with a <u>new</u> EP requirement?

Answer "Yes" if the activity is necessary to achieve compliance with a new emergency preparedness regulation or related guidance.

2a) New EP requirement supports implementation of a RSPS?

Answer "Yes" if the new EP requirement is associated with implementation of one or more of the four Risk Significant Planning Standards (RSPSs) discussed in NRC Inspection Manual Chapter 609, App B, *Emergency Preparedness Significance Determination Process*.

2b) New EP requirement supports implementation of a PS?

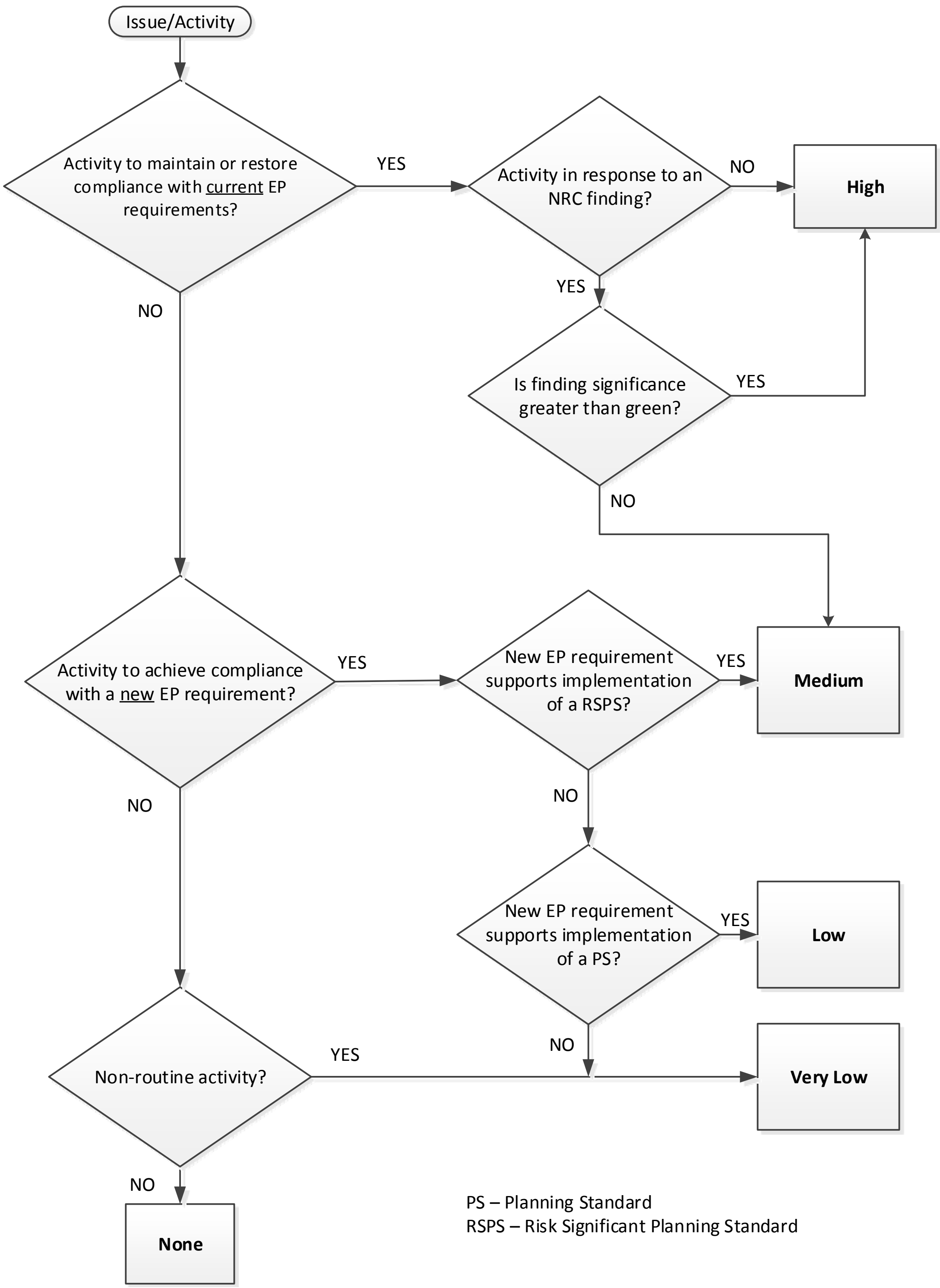
Answer "Yes" if the new EP requirement is associated with implementation of one or more of the non-RSPSs discussed in NRC Inspection Manual Chapter 609, App B; these are referred to simply as Planning Standards (PSs).

3) Non-routine activity?

Answer "Yes" if the activity cannot be adequately addressed or controlled through normal work practices or processes such as a corrective action program or work control. Attributes of such an activity may include the need for a project team and/or budget to address anticipated complexity, cost, duration or needs of multiple stakeholders.

Figure 4.2-1

EP Issue Importance Determination – Step 1



Step 2 (Determine issue's EP importance)

See Section 4.4.

4.3 Radiation Protection

Radiation Protection (RP) importance characterization includes two basic steps: 1) a flowchart series of screening questions to address the "no impact" step; and 2) use of qualitative effectiveness estimates to assign high, medium, low, or very low importance to activities that do not screen out in Step 1. For Step 1, the flowchart in Figure 4.3-1 is used. For Step 2, the overall RP importance is concluded based on a matrix, provided in Table 4-1.

Step 1 (Screening for any impact)

Complete the flowchart in Figure 4.3-1 to determine the current benefit associated with the issue. If the current benefit associated with the issue is anything other than "None" or "Reassess," continue to step 2.

Please note that the decision diamonds entitled "Cost Benefit Achieved" represent the actions taken to assess the projected benefit (e.g., dose savings) achieved by the proposed issue vs. the projected level of effort required, including monetary impact. Site specific monetary values should be used during this assessment.

The first decision diamond addresses the issue of "Public Dose," and could include actions such as:

- System modifications improving effluent treatments
- Improved radiation effluent monitoring capabilities (e.g., detector efficiencies)
- Improved sampling techniques (e.g., C-14 sampling vs. branching calculations)
- The "Site Specific Benefit Rationale Achieved" decision diamond includes an assessment of factors which may be further detailed during the aggregation process (e.g., overall business case for performing the activity, site specific public relations, etc.)

The second decision diamond addresses the issue of "Occupational Exposure" and could include actions, such as:

- Installation of remote monitoring devices in radiological impacted areas (e.g., cameras, dosimetry, other sensors, etc.) that would reduce personnel traffic in the areas

- Modification of High Radiation/Locked High Radiation control systems
- Water chemistry changes impacting source term or personnel exposure
- Site specific ALARA values (dollar per person-rem) should be used for the assessment

The third decision diamond addresses the issue of "Radioactive Waste" and could include actions, such as:

- Use of higher efficiency filters/resin that could result in more "change-outs"
- The need to remove and dispose/store contaminated equipment or material

The fourth decision diamond addresses "Control of Radioactive Material" and could include actions such as:

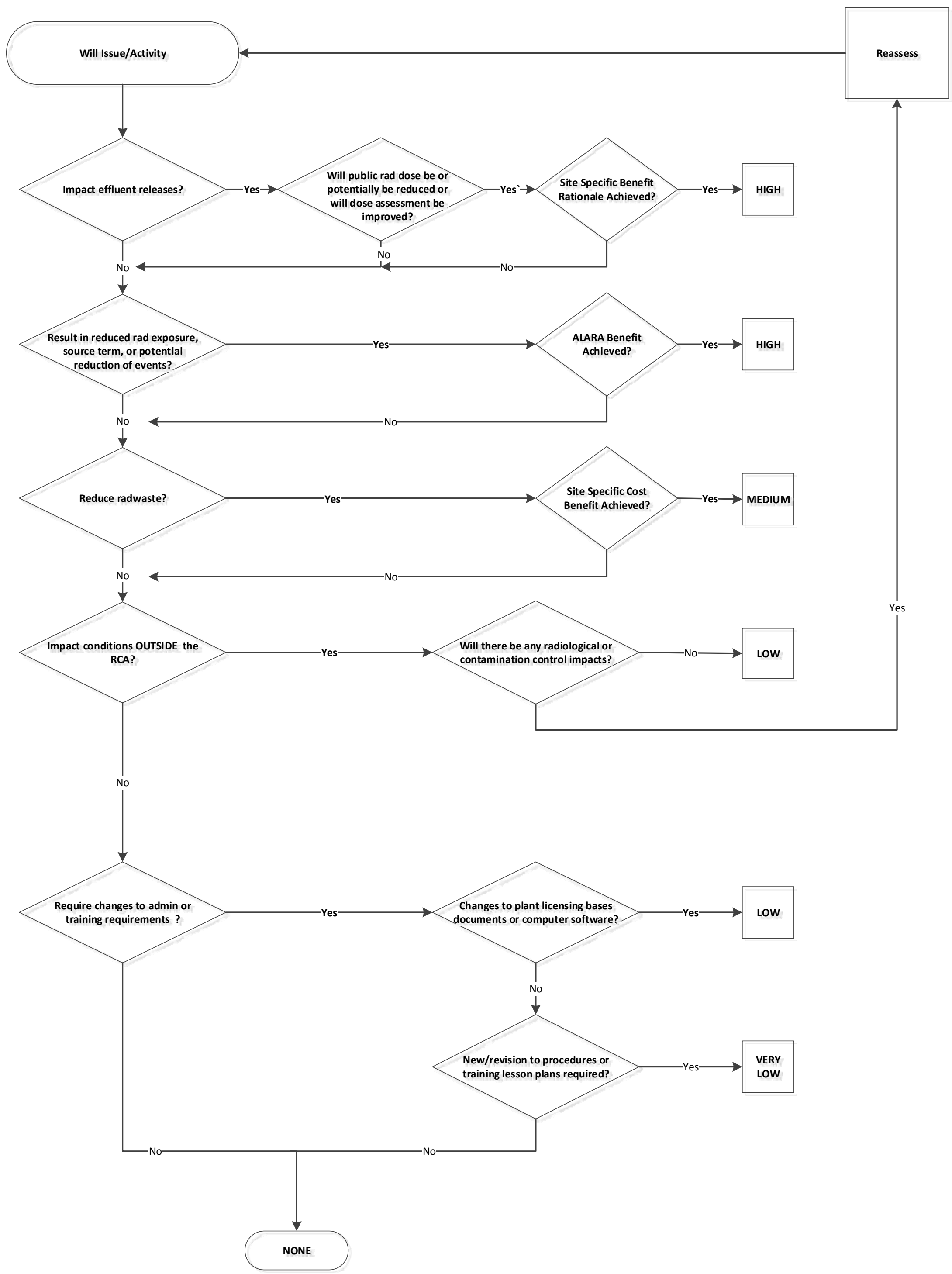
- Potential storage of radioactive material outside of the RCA is needed
- Need for radiography for construction activities outside of the RCA
- Disruption of effluent discharge lines

An outcome of "Reassess" indicates that more information should be gathered to better define the issue/success criteria, come up with a smarter solution (e.g., performance based rule), or otherwise change the proposed action to reduce cost/increase benefit. After reassessing, the process should be re-entered to consider the re-defined issue.

- For example if the storage of radioactive material outside of the RCA could potentially be required by the original plan, "reassess" the plan to determine if the volume of radioactive material generated can be reduced or if there would be an alternate storage location inside the RCA.

Figure 4.3-1

RP Issue Importance Determination – Step 1



Step 2 (Determine issue's RP importance)

See Section 4.4.

4.4 Determining Security, EP, and RP Importance

After completing step 1 in Sections 4.1-4.3, if the current significance associated with the issue is anything other than “None,” continue to Step 2 using Table 4-1.

The table is used as follows:

- First, note the current significance from Step 1 in Sections 4.1-4.3 so as to establish the appropriate row in Table 4-1.
- Then, determine how effective the proposed activity will be in resolving the issue. This establishes the overall importance of the proposed activity. Qualitative and quantitative guidelines regarding effectiveness are provided in Table 4-1 and the discussion below.

Table 4-1 Matrix by Current Significance and Potential Impact

Current significance associated with the issue (from Step 1 Flowcharts)	**Potential Impact of Action Resolving Issue (Effectiveness)**		
	Not Effective	Somewhat Effective	Mostly Effective
	0 to 25%	25 to 80%	>80%
	Importance		
Very Low	Very Low	Very Low	Very Low
Low	Very Low	Very Low	Low
Medium	Very Low	Low	Medium
High	Very Low	Medium	High

Effectiveness relates to the extent to which the full benefit of the proposed change or modification is realized. If there is an available resolution to the issue that would eliminate the concern or significantly mitigate the concern, the “mostly effective” column is appropriate. An action that addresses some but not all aspects of the concern would be considered “somewhat effective.” A proposed resolution that leaves significant aspects of the concern unresolved is deemed “not effective.” If so, it may be appropriate to look for another resolution, if possible.

4.5 RELIABILITY

Reliability is concerned with issues or activities that have some importance and would not otherwise be appropriately captured directly in the safety, security, EP, or RP categories. Reliability should capture the importance of the reliability of SSCs that may be used to generate electricity, or maintain the stewardship of the plant site, that have some nexus with nuclear safety in addition to reliability enhancements of risk-important SSCs. For example, plant aging management, replacement of equipment whose failure could have an adverse impact on overall plant performance in terms of availability, forced outage, power reduction, or potential for a reactor scram may be considered in this category. Note that any quantitative improvement in CDF and/or LERF is directly addressed in the Safety category. However, not including the forward looking aspects of certain projects may underestimate the true impact of the proposed change. Thus, addressing forward looking projects explicitly in the Reliability attribute can be significant to the overall prioritization. The Reliability category may address qualitative aspects of SSC improvements, such as replacing an aging control system on a turbine-driven pump.

A regulatory need for this category is evidenced in the existence of performance indicators (PIs) under NRC's Reactor Oversight Process that include measures of unplanned scrams and unplanned power changes. Exceeding a threshold for a PI might indicate existence of an issue that will become one of some safety importance and could result in the plant being placed in a column of the Action Matrix with heightened regulatory scrutiny, hence the nexus with safety. Reliability importance characterization should facilitate a proactive process to identify and schedule these activities well before approaching a PI threshold rather than a reactive process once an issue has been so far postponed in consideration of other regulatory driven activities that it becomes a regulatory issue itself.

The Reliability importance characterization includes two basic steps: 1) a series of screening questions to address the "no impact" step; and 2) use of qualitative effectiveness estimates to assign high, medium, low, or very low importance to activities that do not screen out in Step 1. For Step 2, the overall Reliability importance is concluded based on a matrix, provided in Table 4-2.

Step 1 (Screening for any impact)

The screening on any impact addresses the following set of questions:

If the Nuclear Safety Importance, from Section 3.0, is anything other than "None" then proceed to the Step 1 questions below. If Nuclear Safety Importance is "None" then the Reliability Importance is "None."

For the proposed activity or issue:

1. ☐ YES ☐ NO Is there a significant risk of SSC failure?

2. ☐ YES ☐ NO Is there a significant replacement lead time?

3. ☐ YES ☐ NO Is there an obsolescence issue?

4. ☐ YES ☐ NO Is there an impact on plant reliability?

5. ☐ YES ☐ NO Is there an impact on SSC or personnel availability due to frequency of preventive maintenance?

If ALL the responses are NO, issue or activity screens to NO IMPACT and Reliability Importance is None.

If ANY response is YES, continue on to Step 2.

Guidance on addressing the above questions is provided below.

Question 1: Is there a significant risk of SSC failure?

In answering this question, the first step is to identify the likelihood of the SSC failing. Is failure imminent, i.e., there have been early warning signs, the SSC has already failed and a temporary repair has been put in place, etc.? (Temporary means more than a compensatory action, but not the same as permanent solution). This is expected to be a qualitative assessment using engineering evaluations; however, a plant-specific calculation may be used to evaluate a potential SSC failure in a quantitative sense.

Next, identify the results of the failure. Will the SSC failure result in a transient, a precursor to a transient, a condition that would make a subsequent transient complicated, a need to operate at reduced power, etc.?

Question 2: Is there a significant replacement lead time?

In answering this question, consider the lead time required for engineering, procurement, fabrication, and installation of a replacement, as applicable. If there is a spare part in the plant warehouse or readily available within a pre-determined distance for the specific site, there is likely not a significant replacement lead time. If significant engineering, procurement or fabrication work must be done, there may be a significant replacement lead time.

Question 3: Is there an obsolescence issue?

In answering this question, consider the impacts of obsolescence that may complicate or compound the time frames cited in response to Question 2, above. If the current SSC cannot be replaced with another SSC that is current technology, form, fit, etc., then the lead times will need to be adjusted accordingly.

Question 4: Is there an impact on plant reliability?

In answering this question, consider both negative and positive impacts of the proposed activity or issue on plant reliability. Will it force a reduction in power or take the plant offline? Could SSC failure result in an unplanned reactor scram or significant plant transient? Is SSC failure more likely during extreme weather events? Will the proposed activity allow the plant to continue to reliably stay online?

Question 5: Is there an impact on SSC or personnel availability due to frequency of preventive maintenance?

In answering this question, for an SSC, consider total out-of-service time that is added due to increased frequency of preventive maintenance or out-of-service time that can be saved with decreased need for preventive maintenance. For personnel, consider whether the proposed activity will decrease the need for preventive maintenance and free personnel resources to address other maintenance needs. Alternately, consider whether NOT implementing the proposed activity will have a significant impact on personnel availability to address other maintenance needs.

Step 2 (Determine issue's reliability importance)

If any response in Step 1 is "yes," determine the timeframe for initial action to prevent unacceptable impacts on reliability, e.g., when personnel must begin the associated engineering process, procurement process, or work scheduling process. The applicable process with the longest lead time for the proposed activity should be used to establish the timeframe for initial action. Thus, the procurement of long lead time equipment might be considered a "short" time frame for action in some cases. Then, using Table 4-2:

- First, note the timeframe for initial action so as to establish the appropriate row in Table 4-2. Table 4-2 uses operating cycles to delineate time frames, thus the actual time frame will vary from plant to plant, e.g., 18 month or 24 month operating cycle. If the plant in question is on an 18 month operating cycle and has an issue for which the initial action must be taken within 20 months to prevent unacceptable impacts on reliability, then the time frame for action is “short,” i.e., less than 36 months or 2 operating cycles. If the same plant has an issue for which the initial action must be taken within 40 months to prevent unacceptable impacts on reliability, then the time frame for action is “long,” i.e., greater than or equal to 36 months or 2 operating cycles.

- Next, determine the characterization of the issue or SSC in question.
  - Safety/Risk Significant SSCs are those so designated by a number of existing risk-informed processes such as Risk Significant SSCs under the Maintenance Rule (10 CFR 50.65) per NUMARC 93-01 Rev 4A (Reference 3), or Safety Significant under 10 CFR 50.69 per NEI 00-04 Rev 0 (Reference 7). Any such SSCs would have either Medium or High Importance depending on the time frame for action.
  - In this characterization, the SSC is not Safety/Risk Significant but its failure has the potential for resulting in a reactor trip, or unplanned plant shutdown or power reduction. Examples might include failures of balance of plant equipment. Any such SSCs would have either Low or Medium Importance depending on the time frame for action.
  - All Other SSCs (or issues) are those that do not fit either of the above categories, but have some nexus with safety by virtue of having answered “yes” in at least one question of Step 1 of Safety. Any such SSC (or issue) would have either Very Low or Low Importance depending on the time frame for action.

Table 4-2 Matrix by Urgency and Characterization

Time frame (in operating cycles) for action associated with the issue	**Characterization of Issue or SSC**		
	SSCs or Issues with Nexus to Safety	Potential for Rx Trip or Unplanned Shutdown/Power Reduction	Safety/Risk Significant SSC
	Importance		
Long (≥ 2)	Very Low	Low	Medium
Short (< 2)	Low	Medium	High

5.0 AGGREGATION TO DETERMINE PRIORITY

After the plant IDP has assigned each issue a level of importance (high, medium, low, very low, or none) in each of the five categories (Safety, Security, EP, RP, and Reliability), the following criteria are used to assign the issue a priority level from 1 to 5. Prioritization and scheduling will be periodically updated based on plant-specific planning, e.g., annually in conjunction with updates to the business plan.

The philosophy behind the approach to prioritization is based on the objective to focus licensees' resources on those issues and activities that have the greatest benefit to public safety. The prioritization process thus assigns higher weight to those issues and activities that are known to directly influence the metrics such as CDF and LERF. However, the prioritization process also recognizes the need to address security, EP, RP, and reliability that typically have some nexus with safety. Consequently, a High in Safety has been equated to the requisite two Highs in the other categories. Likewise, a Medium in Safety is deemed equivalent to a High in Security, EP, RP, or reliability.

Priority 1

- Issue defined by NRC as adequate protection, OR
- High for Safety, OR
- Two or more Highs for any of the four other categories (Security, EP, RP, Reliability)

Priority 2

- Medium for Safety, OR
- One High for any of the four other categories, OR
- Two or more Mediums for any of the four other categories

Priority 3

- Low for Safety, OR
- One Medium for any of the four other categories, OR
- Two or more Lows for any of the four other categories

Priority 4

- Very Low for Safety, OR
- One Low for any of the four other categories

Priority 5

- Does not meet any of the criteria for Priorities 1 through 4

5.1 Regulatory Process for Adjusting Licensing/Regulatory Schedules

As a result of the aggregation described in Section 5.0, each plant may have bins with several Priority 1-5 issues. Priority designation will be an input into the work management and scheduling process for the business plan.

Scheduling

Generally, activities will be implemented as soon as practical considering the next available scheduled outage, if an outage is needed – based on priority. *Parallel implementation of lower priorities is permitted providing it does not result in deferral of implementation of higher priorities.*

- Sufficient resources (financial and skilled personnel) should be dedicated to Priority 1 activities such that the activity will be worked with the maximum feasible effort.
- Priority 2 activities should be worked after maximum feasible resources are assigned to all Priority 1 activities. Work on Priority 2 activities should not impact Priority 1 schedules.
- Priority 3 activities should be worked after maximum feasible resources are assigned to all Priority 1 and 2 activities. Work on Priority 3 activities should not impact Priority 1 and 2 schedules.
- Priority 4 activities should be worked after maximum feasible resources are assigned to all Priority 1, 2 and 3 activities. Work on Priority 4 activities should not impact Priority 1, 2 and 3 schedules.
- Priority 5 activities should be worked after maximum feasible resources are assigned to all Priority 1, 2, 3 and 4 activities. Work on Priority 5 activities should not impact Priority 1, 2, 3 and 4 schedules.

If an activity continues to be subject to deferral, after deferring to the third operating cycle, licensees should decide whether to begin implementation by the end of the next planned refueling outage or submit a request, using the appropriate licensing process, to eliminate the action. Licensees should document this decision with the prioritization document package for the activity.

Tie-Breakers within Priority Level and Other Considerations

Plant-specific processes for decision analysis may be used to determine which activities within a priority level are completed first. For example, if a plant prioritizes 10 activities and has no Priority 1, two Priority 2, four Priority 3, three Priority 4, and one Priority 5, then, the IDP may need to determine which of the Priority 3 and Priority 4 activities get implemented first. The decision analysis may include consideration of:

- Resource allocation (skilled personnel, financial, procurement timing)
- Cost-Benefit ratio
- An approach similar to severe accident mitigation alternatives evaluations under license renewal
- Plant-specific processes or decision analysis tools
- Other considerations, including impact on personnel safety and personnel productivity, such as operator burden or burden on maintenance and security staffing.

Adjusting Licensing/Regulatory Schedules

After assessing an issue using the scheduling and tie-breaking guidance above, if it is determined that the priority of an issue is such that it should be re-scheduled, i.e., deferred, the licensee should enter the appropriate existing process for changing licensing and regulatory schedules. If the schedule to be changed is captured in a regulation, the licensee would process an exemption request per 10 CFR 50.12 or 52.7, as applicable. If the schedule to be changed is captured in a commitment, the licensee would follow the commitment change process as described in NEI 99-04, Rev. 0, *Guidelines for Managing NRC Commitment Changes* (Reference 6).

6.0 REFERENCES

1. NEI 96-07, Guidelines for 10 CFR 50.59 Implementation, Revision 1, November 2000

2. Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Revision 2, May 2011

3. NUMARC 93-01, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Revision 4A, April 2011

4. NUREG/BR-0058, Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission, Revision 4, September 2004

5. NUREG/BR-0184, Regulatory Analysis Technical Evaluation Handbook, January 1997

6. NEI 99-04, Guidelines for Managing NRC Commitment Changes, Revision 0, July 1999

7. NEI 00-04, 10 CFR 50.69 SSC Characterization Guideline, July 2005

APPENDIX A – EXAMPLES OF SAFETY IMPORTANCE DETERMINATION

EXAMPLE 1: INSTALLATION OF IMPROVED REACTOR COOLANT PUMP SEAL DESIGN

Issue: The installation of an improved reactor coolant pump (RCP) seal design (e.g., Byron-Jackson or Flowserve N-9000) could:

- improve system and plant performance, e.g., thermal-hydraulic stability of leakoff/bleedoff flows
- reduce forced shutdowns and reduce transition risk
- potentially reduce likelihood of spontaneous RCP seal LOCAs
- enhance performance during loss of RCP seal cooling and station blackout events

Success criteria: Cost-effective change that improves overall RCP seal performance, improves plant availability, while also improving plant coping capability for loss of all RCP seal cooling events including station blackout scenarios.

References:

1. WCAP-15603-A, Rev. 1 (non-proprietary) (WOG 2000 RCP seal model)
2. WCAP-16175-NP-A, Rev. 0, (RCP seal failure model for CE NSSS)
3. NUREG-1560 (IPE insights) and NUREG-1742 (IPEEE insights)
4. Data NUREGs including NUREG/CR-6928, NUREG/CR-5750, and in particular NUREG/CR-6582 (PWR primary system leaks including RCP seal leakage events)
5. Plant-specific RCP seal design information and PRA insights

Evaluation:

Step 1 (No impact assessment):

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: Catastrophic RCP seal failures in the past have caused reactor coolant system (RCS) leakages beyond normal make-up capability, leading to small LOCAs, so response is **YES**.

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

Justification: A low-leakage RCP seal would enhance mitigation capability for loss of RCP seal cooling events including loss of component cooling water initiators, loss of service water or raw water initiators, and station blackout scenarios, so response is **YES.**

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

Justification: Consistent with 10 CFR 50.59 evaluations, this question asks whether the issue would potentially reduce radiological consequences (dose) given an accident. The improved RCP seal design generally does not *directly* reduce fission product source term (e.g., containment performance) or off-site doses given an accident (e.g., protective actions), so the response is **NO.**

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

Justification: The improved RCP seal design could potentially reduce RCS leakage rates given a loss of RCP seal cooling initiator, therefore, the response is **YES**.

5. ■ **YES** ☐ NO Result in an impact on defense-in-depth capability or impact in safety margin?

Justification: The improved RCP seal design potentially increases the coping time for station blackout sequences, and provides defense against some loss of support system initiators that otherwise are assumed to lead to core damage (e.g., unmitigated loss of component cooling water in some PWRs), so the response is **YES.** (There is no apparent impact in safety margin as typically defined).

Based on the above evaluation, at least one of the questions was answered in the affirmative and the process moves to Step 2.

Step 2 (More than minimal impact 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: Review of several operating experience data reports including NUREG/CR-6928, NUREG/CR-5750, and NUREG/CR-6582 indicates that there have been no RCP seal LOCAs in over 30 years. Therefore, the reduction in small LOCA frequency is judged to be minimal in comparison to all other contributors, so the response is **NO.**

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

Justification: From WCAP-15603-A Rev. 1, the conditional probability of RCS leakage greater than 21 gpm/RCP for Westinghouse seals with qualified O-rings given loss of seal cooling is about 0.21. From BNL-72341-2004 for BJ N-9000 seal designs, the values are:

- 10^{-4} conditional probability of failure for < 4 hr
- 10^{-3} conditional probability of failure for > 4 hr

(dependent on closing bleedoff line and tripping RCPs)

Therefore, there is a more-than-minimal improvement in capability, and the response is **YES.**

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

Justification: As discussed in Step 1, there is no impact on radiological consequences, so the response is **NO.**

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

Justification: Given a loss of RCP seal cooling event, the RCS leakage rates with the enhanced seal design are considerably less than the existing design, so the response is **YES.**

5. ■ **YES** ☐ NO Result in more than a minimal improvement in defense-in-depth capability or improvement in safety margin?

Justification: The substantial reduction in probability of RCP seal LOCA provides significant defense against loss of RCP seal cooling initiators and station blackout, so the response is **YES**. (There is no apparent improvement in safety margin as typically defined).

Based on the above evaluation, at least one of the questions was answered in the affirmative and the process moves to Step 3.

Step 3A (Qualitative assessment)

Table 3-1 is used as a job aid in performing a qualitative (or semi-quantitative) assessment of the issue. If this were a generic characterization and prioritization, relevant industry-wide information on the relative risk associated with spontaneous RCP seal LOCAs, loss of RCP seal cooling initiators, and station blackout from internally and externally initiated events would be useful. If this were a plant-specific prioritization, risk insights from the plant-specific PRA would be used in the process. Generic risk insights could help inform the plant-specific evaluation where the PRA lacks completeness for some external initiators.

Table 3-1 is a two-dimensional matrix that requires as input an order-of-magnitude estimate of the existing level of risk associated with the issue or activity, as well as the potential impact (i.e. effectiveness) resulting from implementation of the change in terms of an approximate measure in the percent reduction in risk associated with resolving the issue.

Existing level of risk: For a plant-specific evaluation, a tabulation of the contribution to CDF and/or LERF from support system initiators and station blackout from internally and externally initiated events would provide and upper bound level of risk. If the PRA model results explicitly provide the risk importance from RCP seal LOCAs this would provide a better estimate. A generic evaluation is given here.

Since the spontaneous RCP seal LOCA frequency has been screened out as relatively low, this aspect will not be evaluated. This is supported by a number of industry studies including the MSPI cross-comparison study in WCAP-16464-NP that indicates that small LOCA contribution to CDF is in the mid-10^{-7} to mid-10^{-6} /yr range, only a small fraction of which is attributable to spontaneous RCP seal

LOCA given no such LOCAs in the past 30 years. Hence, this aspect alone would place the issue in the Very Low importance band (below 10^{-6} /yr CDF for the existing level of risk in the first column of Table 3-1) regardless of the potential impact of the plant modification.

For consequential LOCAs from loss of RCP seal cooling, a number of references including NUREG-1560 and NUREG-1742 (IPE and IPEEE summary reports) or other more recent regulatory impact studies and CDF/risk compilations are useful. Typically for Westinghouse PWRs with total CDFs in the 10^{-5} to mid-10^{-4} /yr range, consequential RCP seal LOCAs are found to contribute some 10s of percent to CDF. Hence, the existing level of risk would probably lie in the lower to mid-level "Yellow" band in the first column of Table 3-1.

Potential risk reduction: Implementation of the low-leakage RCP seal design would reduce the conditional probability of catastrophic seal LOCA by one to two orders of magnitude. In effect, the potential impact in Table 3-1 would be in the "High" column (> 90%).

Importance (generic): The combination of the existing level of risk ("yellow") with the potential impact (high) would place this issue at least in the Low priority band and potentially in the Medium importance band for safety. Given the incompleteness in industry-wide PRAs for all externally-initiated events, a Medium importance for safety (generically) would seem appropriate. Based on plant-specific design and operational considerations, the plant-specific importance could be the same, lower or higher than the generic importance characterization for safety described here.

Step 3B (Quantitative assessment)

The full quantitative assessment, if necessary, would typically be performed on a plant-specific basis. For example, plant risk analysts could make changes directly to the plant-specific PRA models. In this particular example, the RCP seal LOCA models would be reviewed and appropriate changes made to basic event probabilities, coping times, loss of offsite power/station blackout convolution integrals and other supporting PRA models. A direct calculation of the changes in CDF and LERF would be performed. Given uncertainty or incompleteness of the model (e.g., some external initiators not included), some adjustment to the overall results may be necessary.

Example 2: SAMG & EOP Integration

Issue: SAMG & EOP Integration (Fukushima NTTF Recommendation #8)

The regulation would have the following provisions:

- have strategies and guidance for mitigating the consequences of severe accidents
- integrate event and accident mitigating procedures
- identify command and control roles, responsibilities, and authorities during the progression of an event or accident
- conduct related drills, exercises or both
- provide training
- incorporate severe accident situations in written examinations and operating tests for all types of operators.

Success criteria: Cost-effective change that results in improved plant staff performance during beyond-design basis accidents including severe accidents

References:

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 (NTTF) 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. Memorandum to Charles L. Miller (NRC) from Timothy J. Kobetz (NRC), Temporary Instruction 2515/184, "Availability and Readiness Inspection of Severe Accident Management Guidelines" Results," June 6, 2011 (ML11154A109).
6. NEI slides, Industry Perspective on NRC NTTF Recommendation 8 Proposed Rule and Regulatory Basis, November 19, 2013 (ML13330B717).
7. BWROG & PWROG slides, Update on Owners' Groups Activities – NTTF Recommendation 8, November 19, 2013 (ML13330B714).

Evaluation:

Step 1 (No impact assessment):

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: The activity could potentially improve plant staff response to severe accidents, but would not by itself directly impact accident initiator frequency, so the response is **NO.**

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

Justification: The activity could potentially improve plant staff response to severe accidents and thereby potentially reduce the likelihood or consequences of radiological releases, so the response is **YES.**

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

Justification: As discussed in the response to Question 2, improving staff performance during severe accidents could potentially result in reduced radiological releases and thereby impact offsite consequences, so the response is **YES.**

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

Justification: The activity impacts plant staff performance but does NOT *directly* impact the reliability, availability, or performance of equipment used in severe accident management, nor would it directly affect or modify the performance of fuel cladding, RCS integrity, and containment systems, so the response is **NO**.

5. ■ **YES** ☐ NO Result in an impact on defense-in-depth capability or impact in safety margin?

Justification: The activity impacts plant staff performance during severe accidents and therefore may strengthen somewhat the balance of accident prevention and mitigation, so the response is **YES.** (There is no apparent impact in safety margin as typically defined).

Based on the above evaluation, at least one of the questions was answered in the affirmative and the process moves to Step 2.

Step 2 (More than minimal impact 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: As discussed in Step 1, there is no impact on the frequency of occurrence of a risk significant initiator, so the response is **NO.**

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

Justification: As discussed in the Regulatory Basis for NTTF #8, procedures and guidelines already exist for severe accident management based on implementation of Generic Letter 88-20, Supplement 2. Furthermore, on a generic basis, SAMGs have been implemented at all plant sites, plant personnel have been trained, and periodic drills/exercises on the use of the SAMGs are conducted. Of the six provisions identified above under Issue, only the provision for written examinations and operating tests for all types of operators is, in effect, not currently performed. This provision is primarily for inspection purposes rather than necessarily for performance improvement. Furthermore, given the generally fixed amount of operator training time, implementation of this provision may cause dilution of operator focus from more risk-significant transients and accidents. Thus, on a generic basis, it is concluded that the response is **NO.** However, NRC inspections under TI 2515/184 have found plant-by-plant deficiencies. Hence, for a plant-specific evaluation, this question may be answered **YES** if there remains a significant deficiency and more than a minimal improvement in personnel performance results from the implementation of the activity.

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

Justification: As discussed in the response to Question 2, SAMGs have been implemented at all plant sites. Thus, on a generic basis, it is concluded that the response is **NO** regarding more than a minimal improvement in defense-in-depth capability. However, on a plant-specific basis, this question may be answered **YES** as discussed under Question 2.

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

Justification: As discussed in Step 1, there is no impact on the capability of fission product barriers, so the response is **NO.**

5. ☐ YES ■ **NO** Result in more than a minimal improvement in defense-in-depth capability or improvement in safety margin?

Justification: As discussed in the response to Question 2, SAMGs have been implemented at all plant sites. Thus, on a generic basis, it is concluded that the response is **NO** regarding more than a minimal improvement in defense-in-depth capability. However, on a plant-specific basis, this question may be answered **YES** as discussed under Question 2. (There is no apparent impact in safety margin as typically defined).

Based on the above *generic* evaluation, where none of the questions was answered in the affirmative, the activity would screen out, and the generic characterization process would stop. The activity would be characterized generically as Very Low importance for safety. However, as the generic characterization serves as an input to a *plant-specific* assessment, it is possible that a plant-specific evaluation would continue forward. Depending on plant-specific circumstances regarding the fullness of SAMG implementation, training and periodic drills, a different conclusion regarding "minimal improvement" for one or more questions could be reached and the process would then continue to Step 3A and/or 3B.

Step 3A (Qualitative assessment—plant-specific only)

Given that the *generic* characterization is Very Low, this step would be performed on a plant-specific basis only for those plants that met one or more criteria for "more than minimal improvement" under Step 2.

Table 3-1 is used as a job aid in performing a qualitative (or semi-quantitative) assessment of the issue.

Existing level of risk: Insights from the plant-specific PRA could be used to assess the existing level of risk in terms of metrics such as CDF and LERF. The analysts would need to be familiar with the degree to which SAMG/EDMGs actions have been credited in the PRA model. Generally, very few (if any) operator actions associated with SAMGs/EDMGs are credited in the Level 1 PRA for mitigating accidents prior to core damage. Thus, LERF and long term containment integrity may be the metrics mainly impacted by crediting such operator actions. If the PRA model is not complete because not all external events have been considered, adjustments may be necessary.

Potential risk reduction: Some judgment will be necessary regarding the assumption of the potential risk reduction by implementing NTTF #8. Given that all plant sites have implemented SAMGs/EDMGs, then it is a matter of assessing to what degree having a regulatory requirement for procedure maintenance and training could impact operator performance. From Table 3-1, it is reasonable to presume that the regulatory requirement would *not* be 90% or greater effective in improving operator performance, or that at the other extreme there is no improvement whatsoever given that the plant-specific evaluation passed Step 2. Hence, this could help narrow down the potential impact/effectiveness to perhaps the low to medium range columns, for example. At most, the difference between the selection of "low" or "medium" for potential impact would be one level of importance for safety Low versus Very Low, etc.).

Step 3B (Quantitative assessment—plant-specific only)

Alternatively, it may be decided that the PRA models could be used directly in the determination of the risk change. Again, there are generally only a handful of operator actions related to SAMG/EDMG. The human reliability analysis generally would quantify the operator error rate using performance shaping factors (PSF) that adjust the baseline human error probability (HEP). Many of the PSFs such as control room indication or environment are not affected by implementation of the integrated EOPs/SAMGs. Training and the quality of procedures are the most likely PSFs to be affected by the regulatory requirement. One possibility would be to re-quantify the PSFs assuming better (or worse) conditions, revise the HEP and basic event probabilities, and re-quantify the PRA model. It is possible that the PRA model of record assumes ideal conditions so a SAMG program deficiency could mean higher HEPs as the baseline. The difference in LERF thus would reflect the potential improvement resulting from the implementation of the rule on a plant-specific basis.