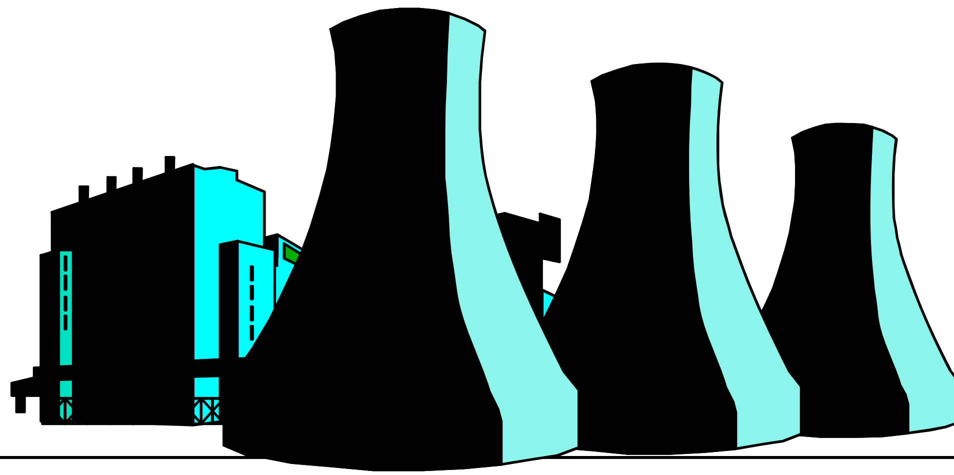
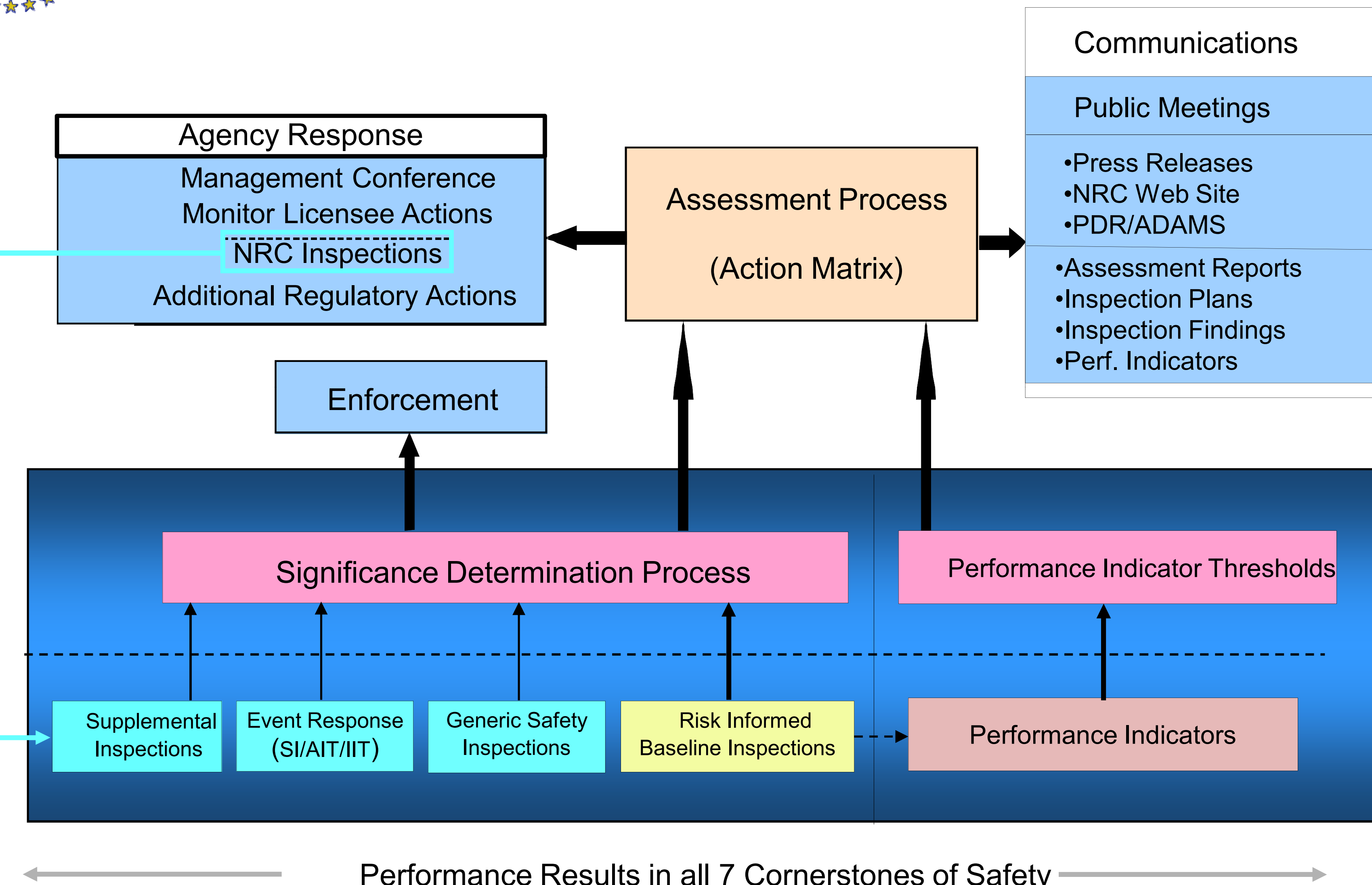
Significance Determination Process (SDP)



ROP Summary



Source: MC 0305 (Operating Reactor Assessment Program) dated 7/6/11
and MC 0308 (Reactor Oversight Program Basis Document) dated 11/8/07

SDP Introduction-
Learning Objectives

- Briefly explain how the SDP is used to risk-inform the inspection process
- Screen an issue and characterize its significance using Phase 1 of the SDP
- Estimate the risk significance of a finding using deterministic SDP
- Estimate the risk significance of a finding using Phase 2 of the SDP
- Estimate the risk significance of a finding using Saphire 8 Risk Software

Objectives of the SDP

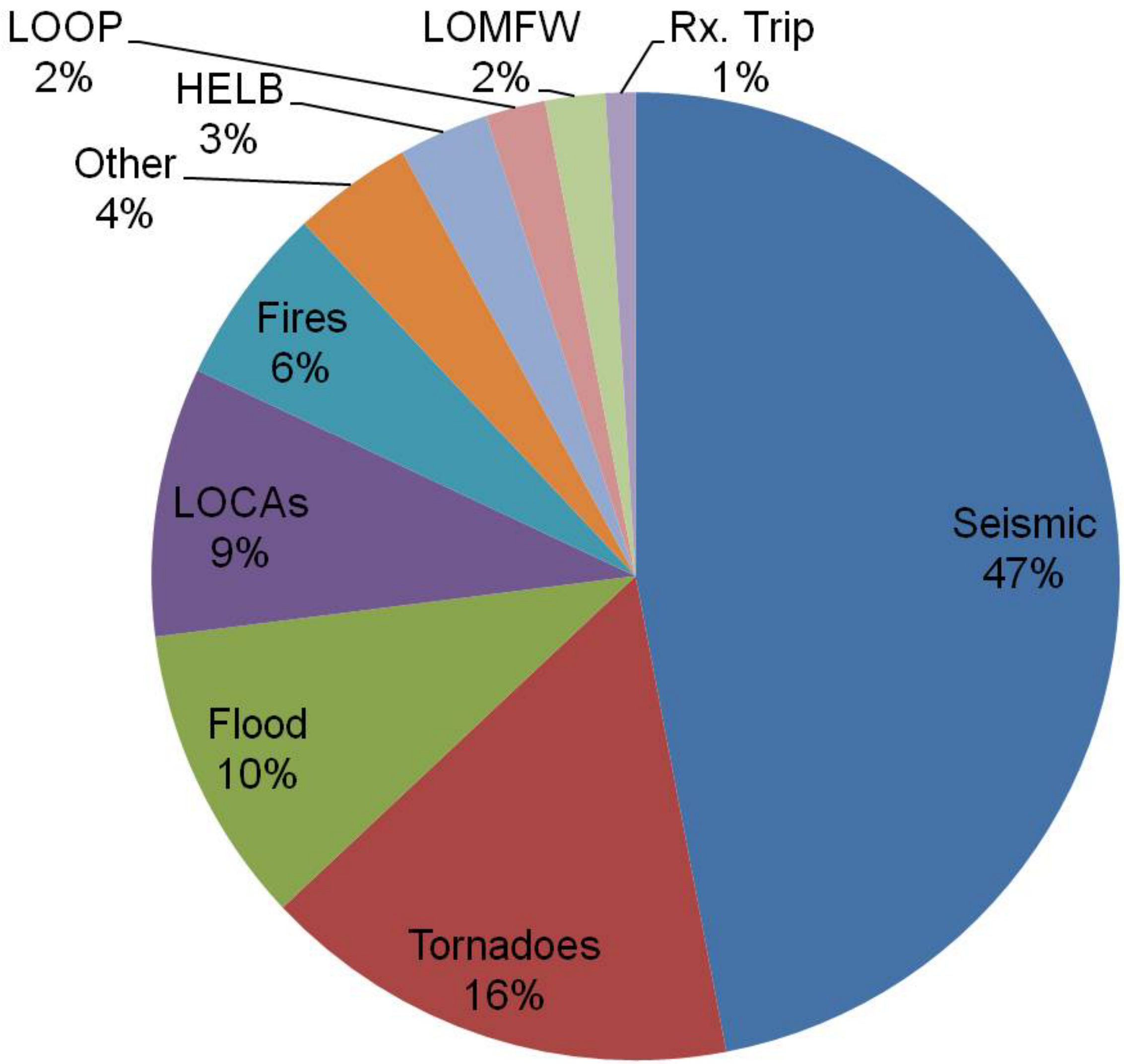
- Characterize the significance of inspection findings in support of the Reactor Oversight Process
- Provide a basis for assessment and enforcement actions associated with inspection findings thereby reducing subjectivity
- Provide stakeholders an objective and common framework for communicating the safety significance of inspection findings
- Provide the staff with plant specific risk information for use in risk-informing the inspection program

Risk-Informed Metrics

- There are two risk-informed metrics that the Commission specified
  - (These metrics are covered in detail in P-105 and P-111)

- Core Damage Frequency (CDF)
  - represents the number of events per unit time which potentially results in a core damage state. Typically, this value is annualized (per year).

CDF



- "At-Power" Risk Only
- Core Damage Frequency = 5.9E-5 /yr (1 in 17,000 yrs.)
- % = Percent of total Core Damage Frequency due to this accident class

Risk-Informed Metrics

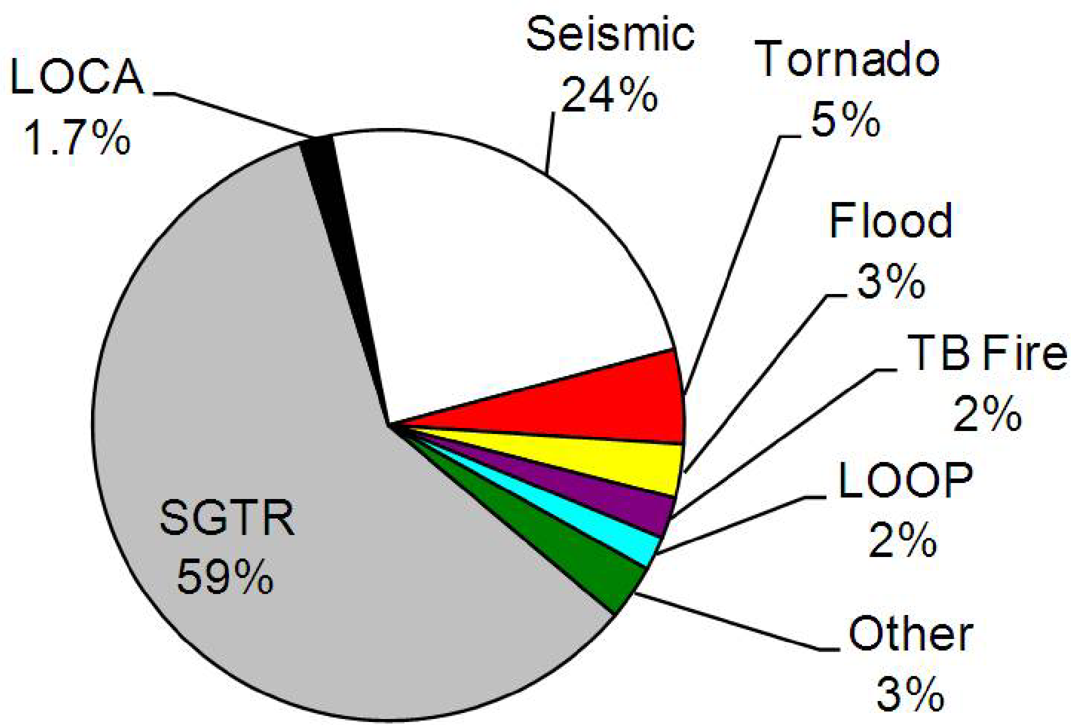
- Large Early Release Frequency (LERF)
  - the frequency of those accidents leading to significant, unmitigated releases from containment in a time-frame prior to effective evacuation of the close-in population such that there is a potential for early health effects.

LERF (cont.)

- *Large* is defined as involving the rapid, unscrubbed release of airborne aerosol fission products to the environment.
- *Early* is defined as occurring before the effective implementation of the off-site emergency response and protective actions.
- A large early release is more significant than a late release because there is little or no time to evacuate or shelter the public. This increases the likelihood of early fatalities.

➢ Remember: risk = frequency $\times$ consequence

LERF (cont.)



LERF Risk Insights

- SGTR and Seismic events dominate LERF and therefore early fatalities.
- SGTR and containment bypass are not dominant contributors to core damage frequency

SDP Risk Thresholds

CONCEPTUAL MODEL FOR EVALUATING LICENSEE PERFORMANCE
GREEN - *Licensee Response Band* Cornerstone objectives fully met. Nominal risk with nominal deviation from expected performance. Very low safety significance. ΔCDF < E-6
WHITE - *Increased Regulatory Response Band* Cornerstone objectives met with *minimal* reduction in safety margin. Low to Moderate Safety Significance. E-6 < ΔCDF < E-5
YELLOW - ***Required Regulatory Response Band*** Cornerstone objectives met with *significant* reduction in safety margin. Substantial Safety Significance. E-5 < ΔCDF < E-4
RED - ***Significant Regulatory Response Band*** Plant performance represents an unacceptable loss of safety margin. It should be noted that should licensee's performance result in a PI reaching the Red Band, margin would still exist before an undue risk to public health and safety would be presented. High Safety Significance. ΔCDF>10-4

Source: MC 0308 Reactor Oversight Process Basis Document) dated 11/8/07

SDP Overview

- Each SDP supports a cornerstone associated with the strategic performance areas

- Reactor Safety SDPs <u>(mostly risk-informed)</u>
  - At-power Findings (Appendix A)
  - Shutdown Findings (Appendix G)
  - Significance Determination Process Using Qualitative Attributes (Appendix M)

- Special SDPs for <u>(deterministic)</u>
  - Emergency Preparedness (Appendix B)
  - Fire Protection (Appendix F)
  - Alternative Mitigating Strategies (B.5.b) (Appendix L)
  - Containment Integrity (Appendix H)
  - Steam Generator Tube Integrity (Appendix J)
  - Maintenance Risk Assessment and Risk Management (Appendix K)
  - Licensed Operator Requalification (Appendix I)

- Radiation Safety <u>(deterministic)</u>
  - Occupational Radiation Safety (Appendix C)
  - Public Radiation Safety (Appendix D)

- Safeguards <u>(risk-informed)</u>
  - Physical Protection (Appendix E)

SDP Phases

- The SDP is a three phase process:
  - Phase 1 screens issues to Green, Phase 2, and/or Phase 3
  - Phase 2 evaluates issues using plant specific risk-informed inspection notebooks that are conservative yet representative of licensee PRA model (i.e., they were benchmarked)
  - Phase 3 is a more detailed review using independent risk tools (e.g., SPAR models)
- Phases 1 and 2 are generally performed by inspection staff, with assistance of a Senior Reactor Analyst (SRA), where necessary.
- Phase 3 is performed by a SRA or other risk analyst.

Phase 1 SDP (MC 0609, Attach 4)

- The Phase 1 Screening Worksheet contains decision logic to determine if the deficiency can be characterized as Green without further analysis.
- Deficiencies generally screen to Green if initiating event frequencies and total function of mitigating and containment systems are not lost.
- Some deficiencies immediately screen to Green based on their low impact to overall plant risk (e.g., radiological barrier systems such as building ventilation).
  - Use guidance in MC 0609, Attach 4, Exhibit 1 to complete a Phase 1 SDP

Phase 1 SDP (MC 0609, Attach 4)

- Table 1 - Record the performance deficiency and factually describe known observations associated with the deficiency
- Table 2 - determine the cornerstone and functions degraded as a result of the performance deficiency. If the finding affects multiple reactor cornerstones (initiating events, mitigating systems, and barrier integrity), the finding should be assigned to the cornerstone that best reflects the dominant risk of the finding.
- Table 3a and Table 3b - identifies the appropriate appendix of IMC 0609 to use based on the cornerstone identified in Table 2.
- Table 4a and Table 4b – Characterization Worksheet - determine if the issue can be characterized as Green.

Deterministic SDPs

- Recall that several SDP Appendices are deterministic in nature, that is, they are not risk-informed.
- Special SDPs:
  - Emergency Preparedness (Appendix B)
  - Fire Protection (Appendix F)
  - Alternative Mitigating Strategies (B.5.b) (Appendix L)
  - Containment Integrity (Appendix H)
  - Steam Generator Tube Integrity (Appendix J)
  - Maint. Risk Assessment and Risk Management (Appendix K)
  - Licensed Operator Requalification (Appendix I)
- Radiation Safety:
  - Occupational Radiation Safety (Appendix C)
  - Public Radiation Safety (Appendix D)

Deterministic SDP Example #1 – The Assignment

Using what we've covered thus far, let's examine a scenario from start to finish.

You've been assigned to a four-person inspection team to evaluate an emergency preparedness drill at a site in your region. The team leader has been assigned and it is not you. Each team member is to prepare for the inspection individually.

- What initial administrative steps should you take to begin preparing for the inspection?

The Assignment (cont.)

The Team Leader assigns you the job of observing licensee actions in the Technical Support Center (TSC) in accordance with IP 71114.01, Exercise Evaluation.

- What additional resources will you need to adequately plan the inspection?
- How do you obtain them?

You arrive on site after the other team members.

- What activities do you expect to perform immediately or shortly after arrival?

The Drill

You take your position in the TSC and, after the drill begins, you observe the activities of the various teams.

While observing the activities of the dose assessment team, you note the following…

The Observation

The dose assessment team failed to recognize a degraded core condition. In this drill, the radiological control manager failed to recognize a degraded core condition (during a release of radioactive material) for 32 minutes after radiation monitor channel indications and field team sample results supported this determination. The radiological control manager's failure to recognize a degraded core condition when indications were available was not corrected by other members of the response team who correctly recognized the condition. The other members of the dose assessment team and the Technical Support Center staff had recognized and communicated that degraded core conditions existed.

The Observation (cont.)

Moreover, the emergency director failed to direct a qualitative evaluation of the core condition, using best available indications, when the radiological control manager reported conflicting indications of core condition with a release in progress. As a result, the protective action recommendation for offsite populations was not upgraded to include evacuation out to 10 miles for the three downwind sectors of the emergency planning zone.

The Issue

- What concern(s) make this observation an issue?
- What is necessary for the issue to be a finding?
- What is necessary for the issue to be a violation?

A Standard

§ 50.47 Emergency plans.

b) The onsite and, except as provided in paragraph (d) of this section, offsite emergency response plans for nuclear power reactors must meet the following standards:

(9) Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.

- Does the issue now represent a finding?
- Does the issue now represent a violation?

HRTD Human Resources Training & Development

The Plot Thickens

When discussing this issue with another team member you learn that a similar problem occurred during the last drill. Specifically, the inspection report stated:

The Plot Thickens (cont.)

“During the biennial emergency preparedness exercise on August 29, 2009, the dose assessment staff failed to recognize that the reactor core was degraded and consequently underestimated offsite doses and issued an incorrect (non-conservative) protective action recommendation for offsite populations. The licensee implemented immediate corrective actions and actions to prevent recurrence of this problem. These actions included revision to the dose assessment procedure, briefing dose assessment teams and decision makers on the identified weakness, promulgation of a white paper containing guidance for decision makers, and administration of dose assessment workshops to emergency response teams to improve command and control and dose assessment technical skills. These actions were completed by March 4, 2010.”

The Plot Thickens…More

- Does the licensee’s failure to adequately correct the last occurrence of this problem represent a finding?

- Does that inadequate corrective action represent a violation?

More Standards

The team leader points out that 10 CFR 50, Appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities,” Paragraph IV.F.2.g, states:

> “All training, including exercises, shall provide for formal critiques in order to identify weak or deficient areas that need correction. Any weaknesses or deficiencies that are identified shall be corrected.”

In addition, 10 CFR 50.47(b)(14) states

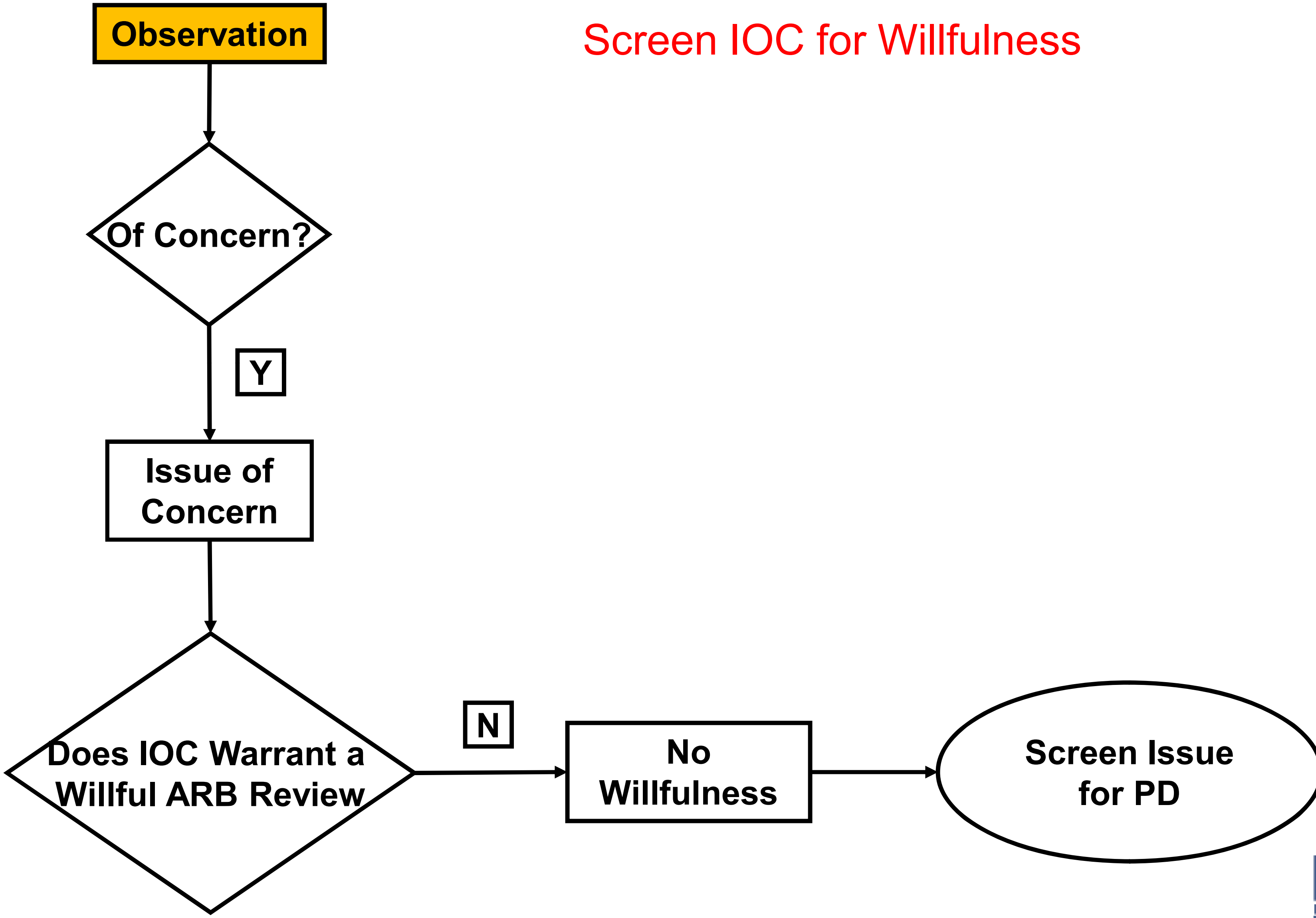
> “Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilities, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected.”

So…

Given these standards, can we begin to characterize the licensee's failure to correct the condition identified in the previous exercise and its recurrence in this most recent exercise?

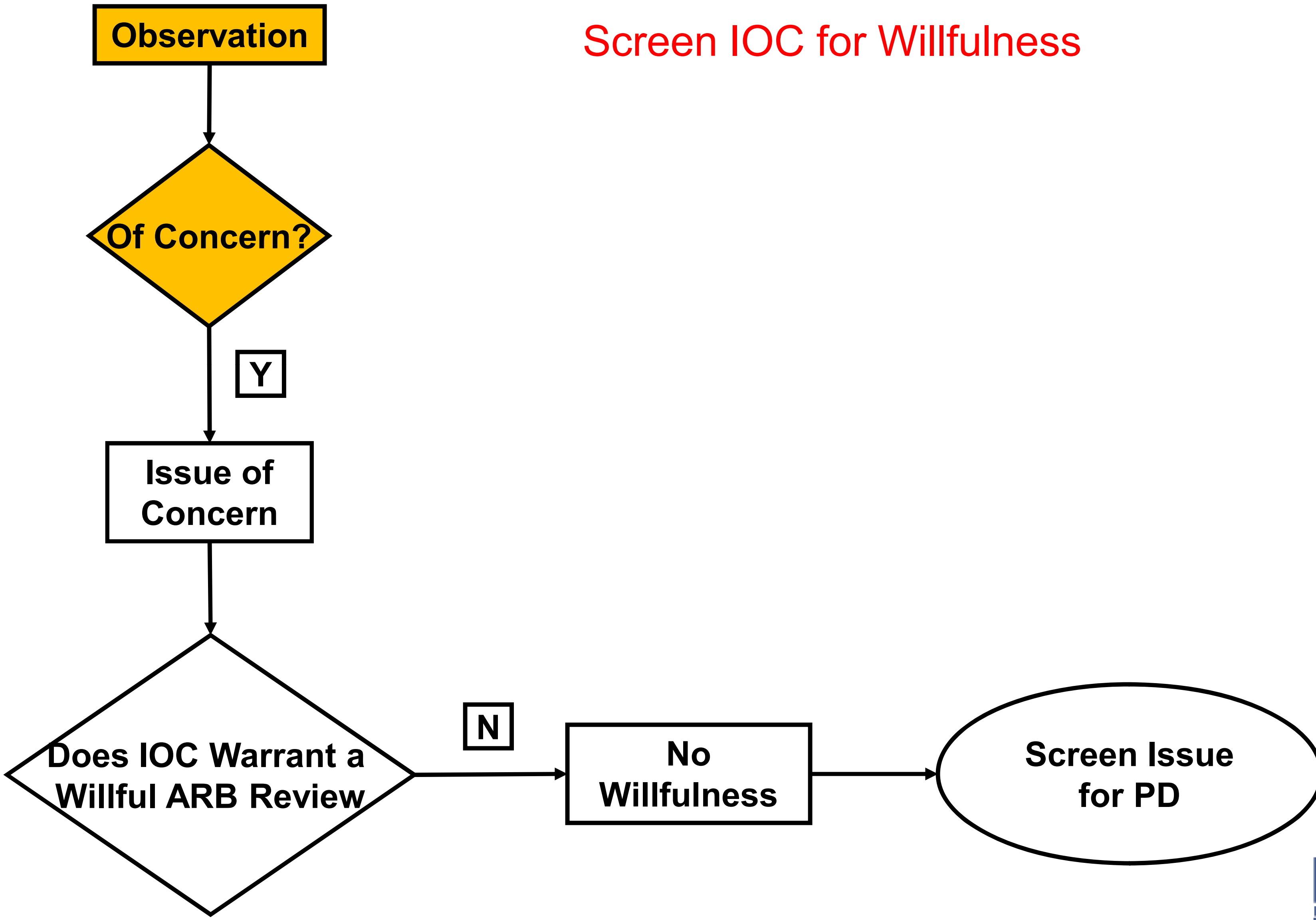
MC 0612, Appendix B, Issue Screening

Screen IOC for Willfulness



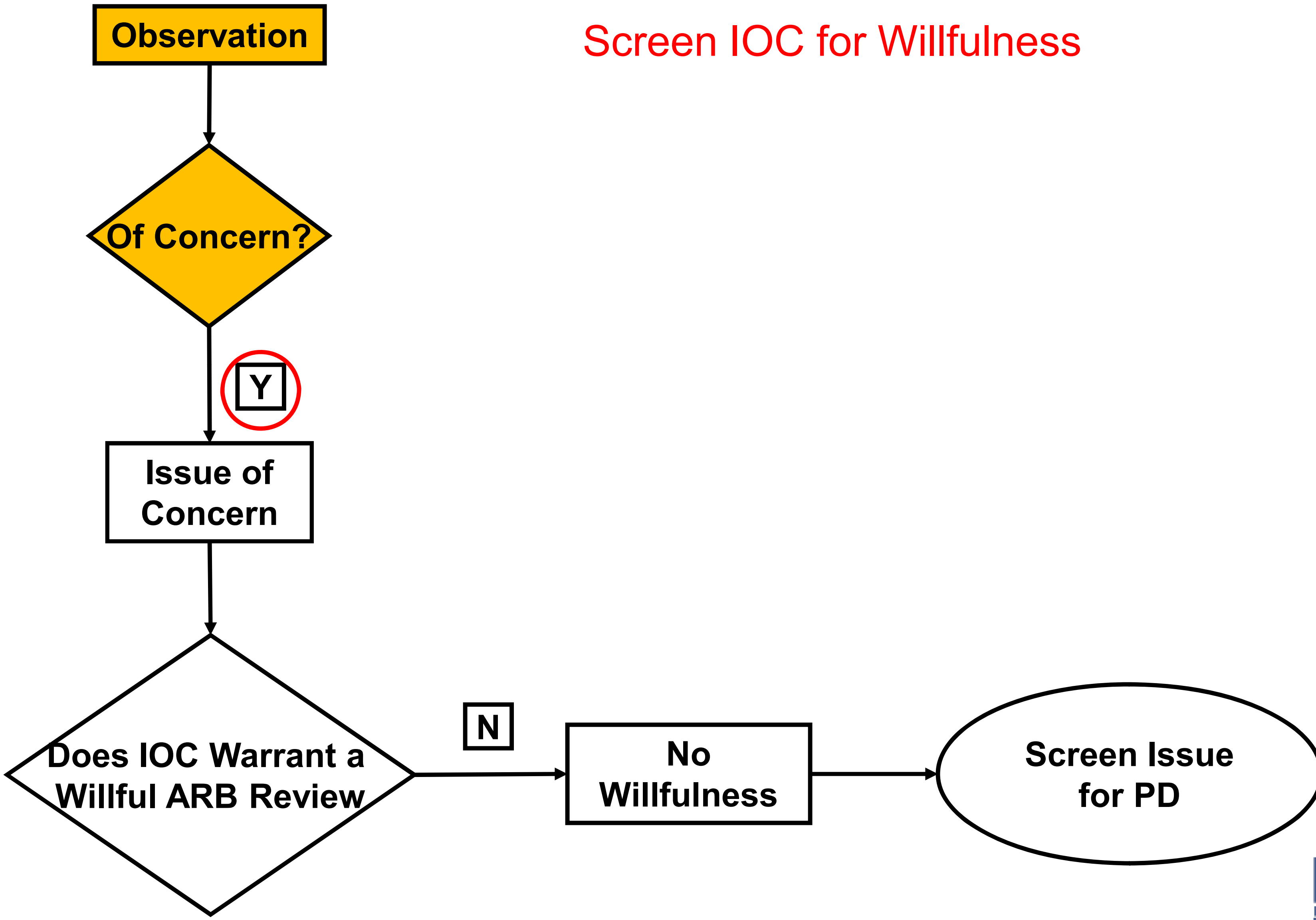
MC 0612, Appendix B, Issue Screening

Screen IOC for Willfulness



MC 0612, Appendix B, Issue Screening

Screen IOC for Willfulness



MC 0612, Appendix B, Issue Screening

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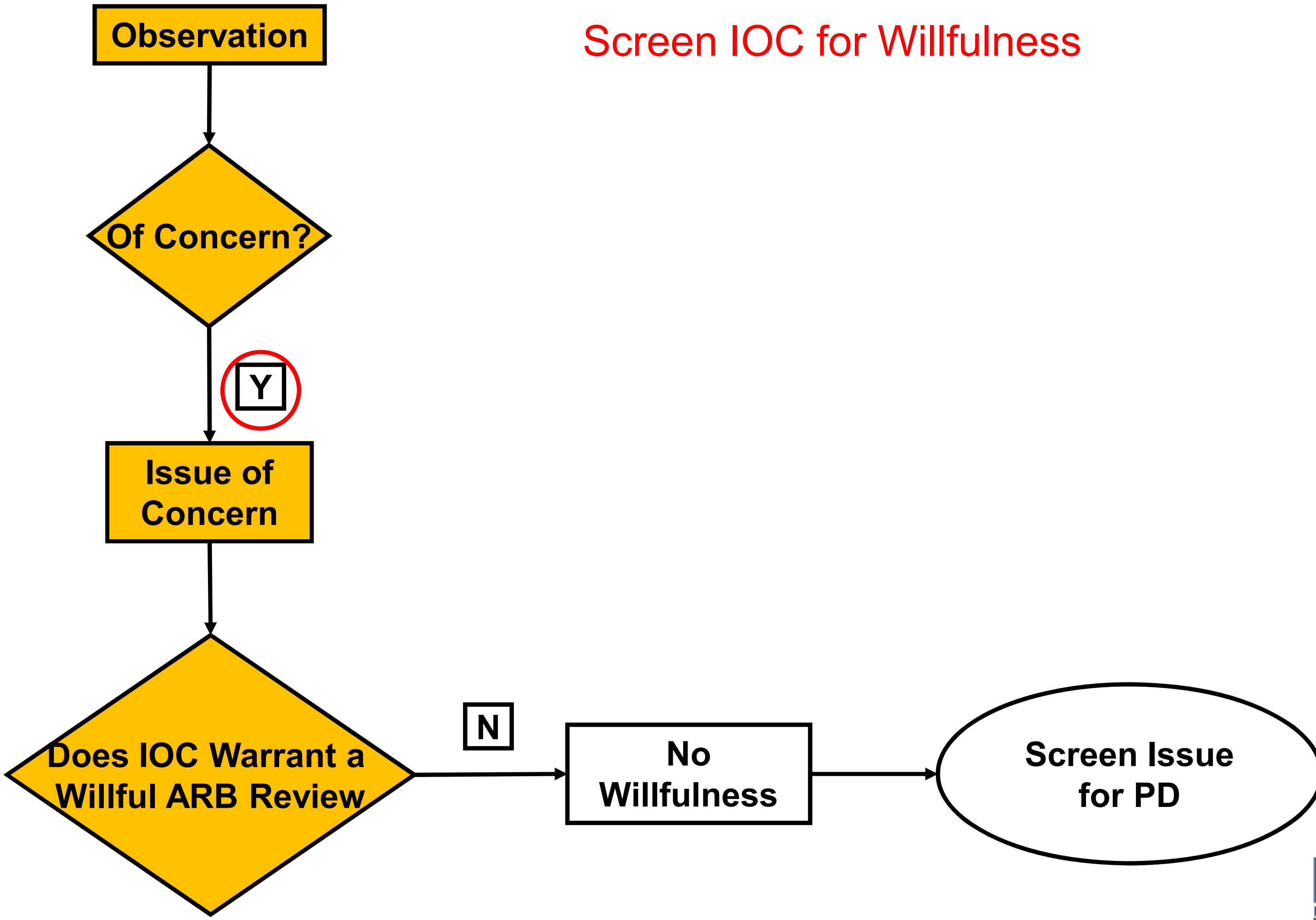

MC 0612, Appendix B, Issue Screening

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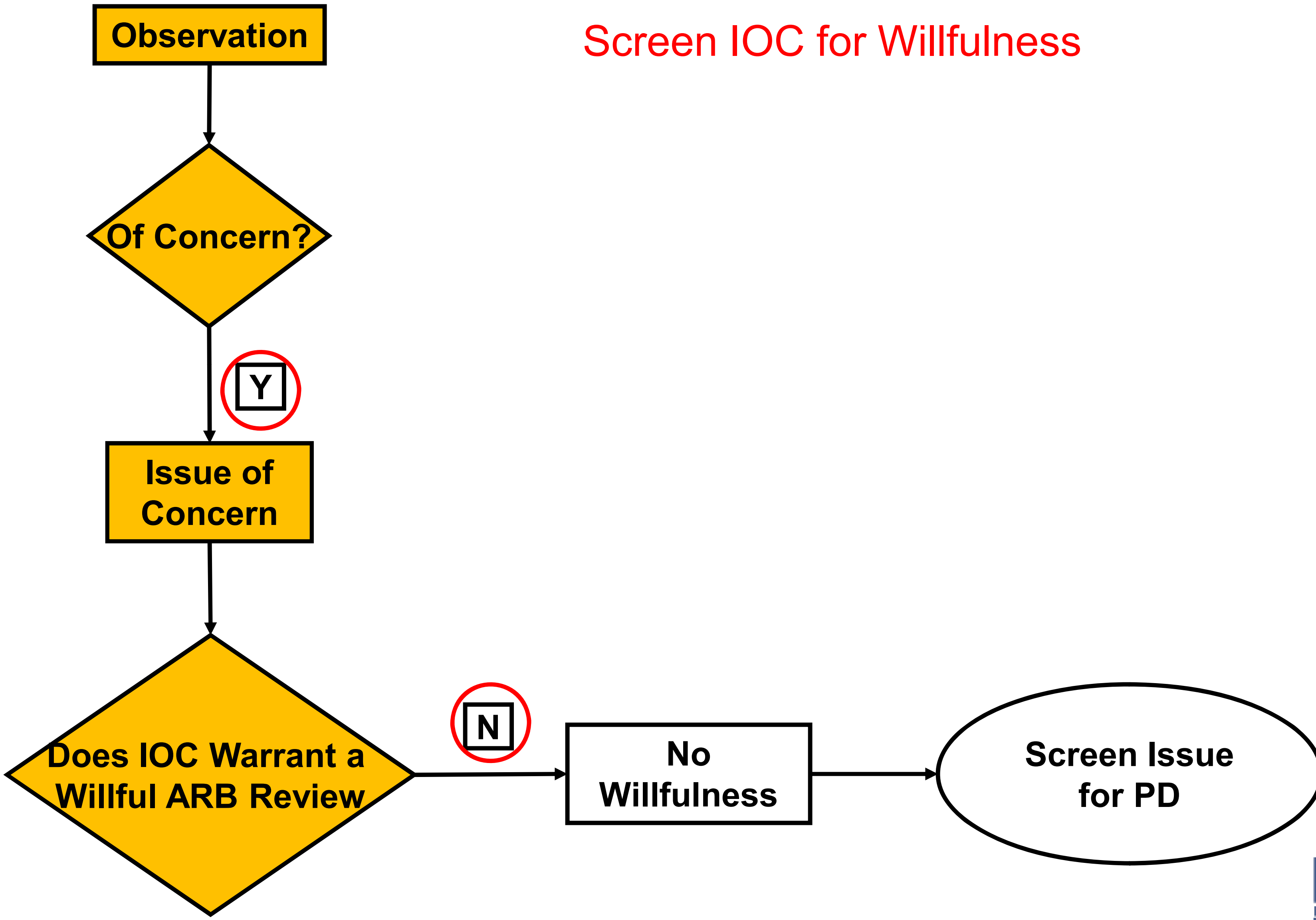
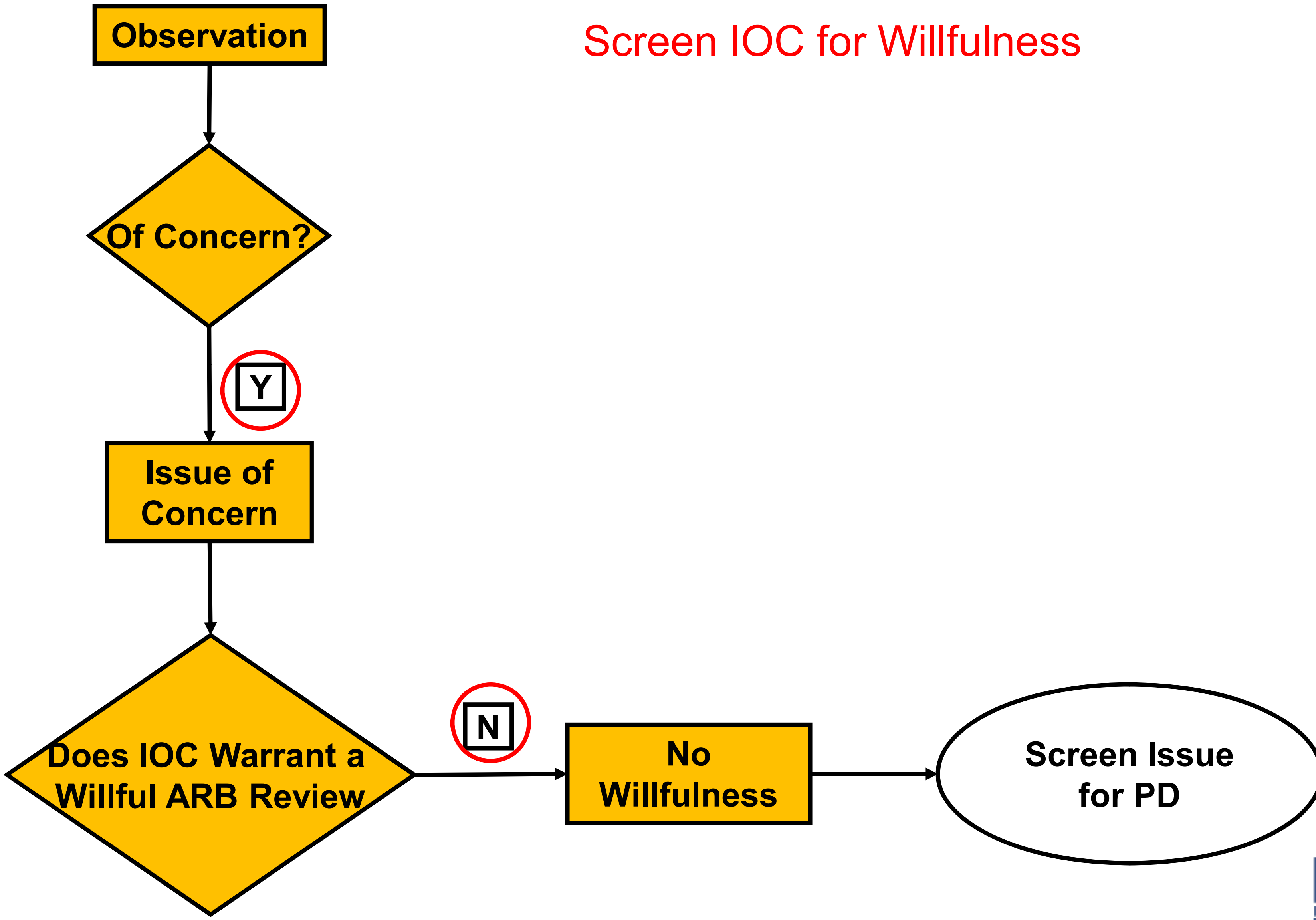


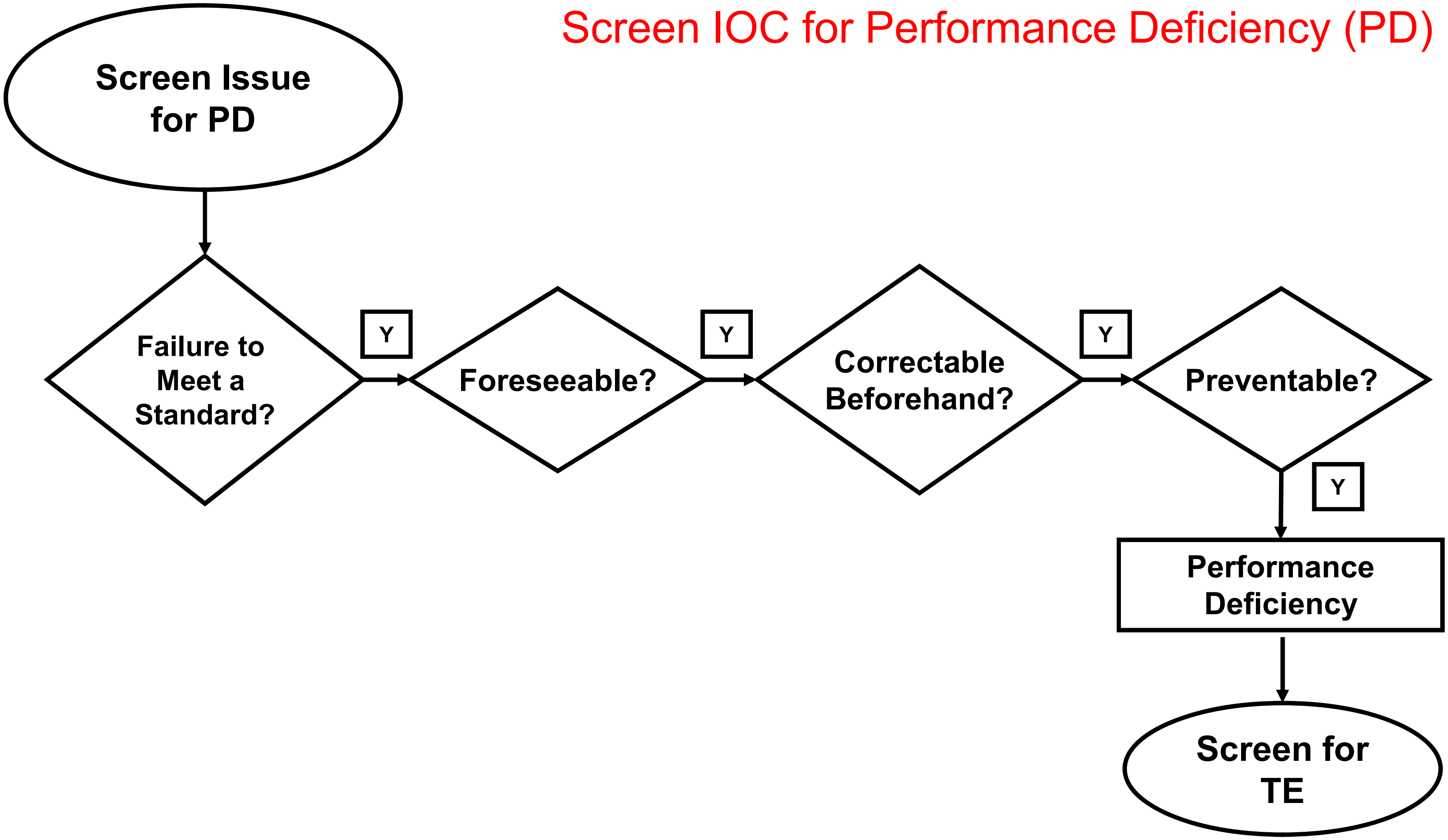
MC 0612, Appendix B, Issue Screening

Screen IOC for Willfulness



MC 0612, Appendix B, Issue Screening

Screen IOC for Performance Deficiency (PD)



MC 0612, Appendix B, Issue Screening

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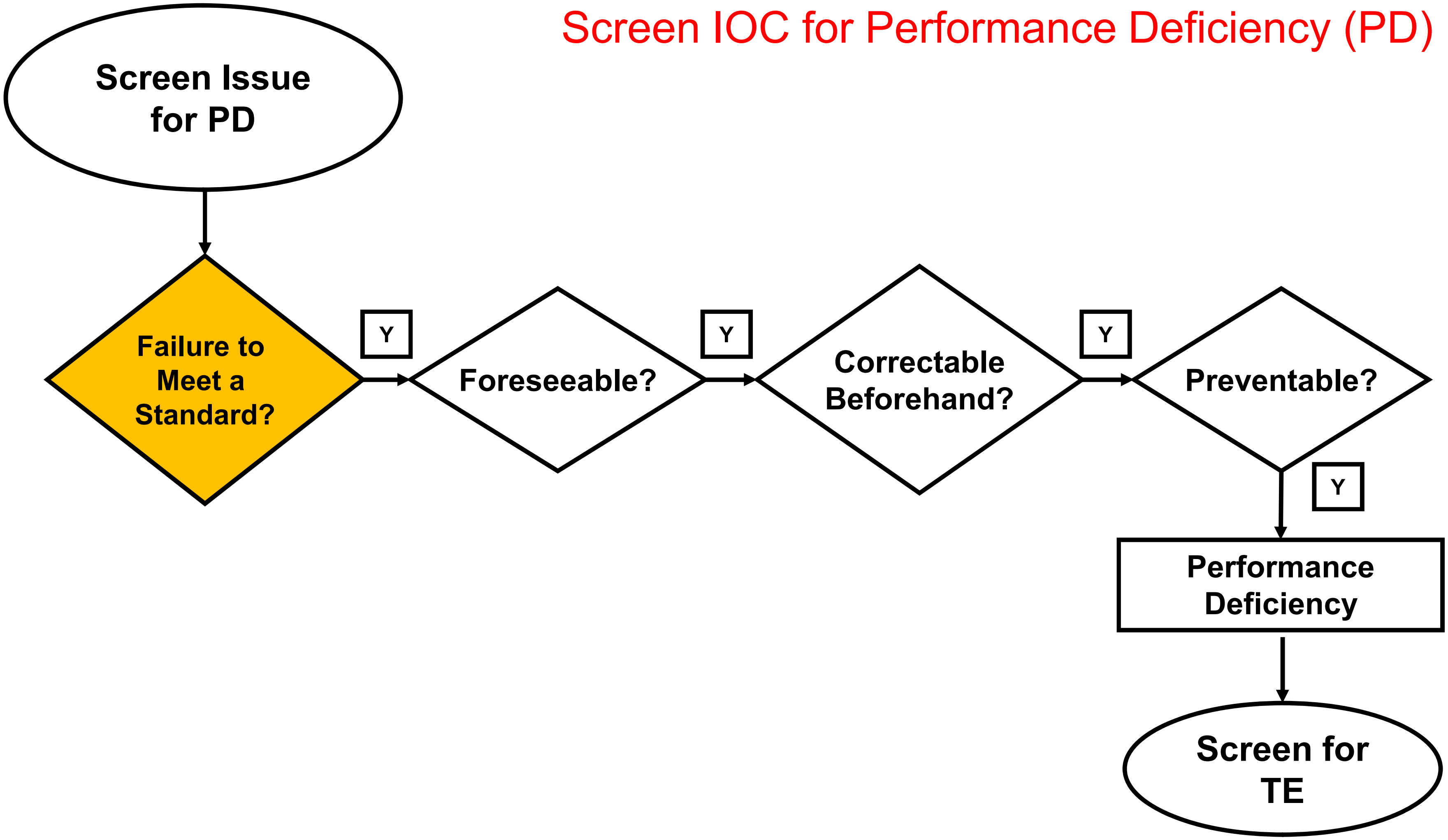

MC 0612, Appendix B, Issue Screening

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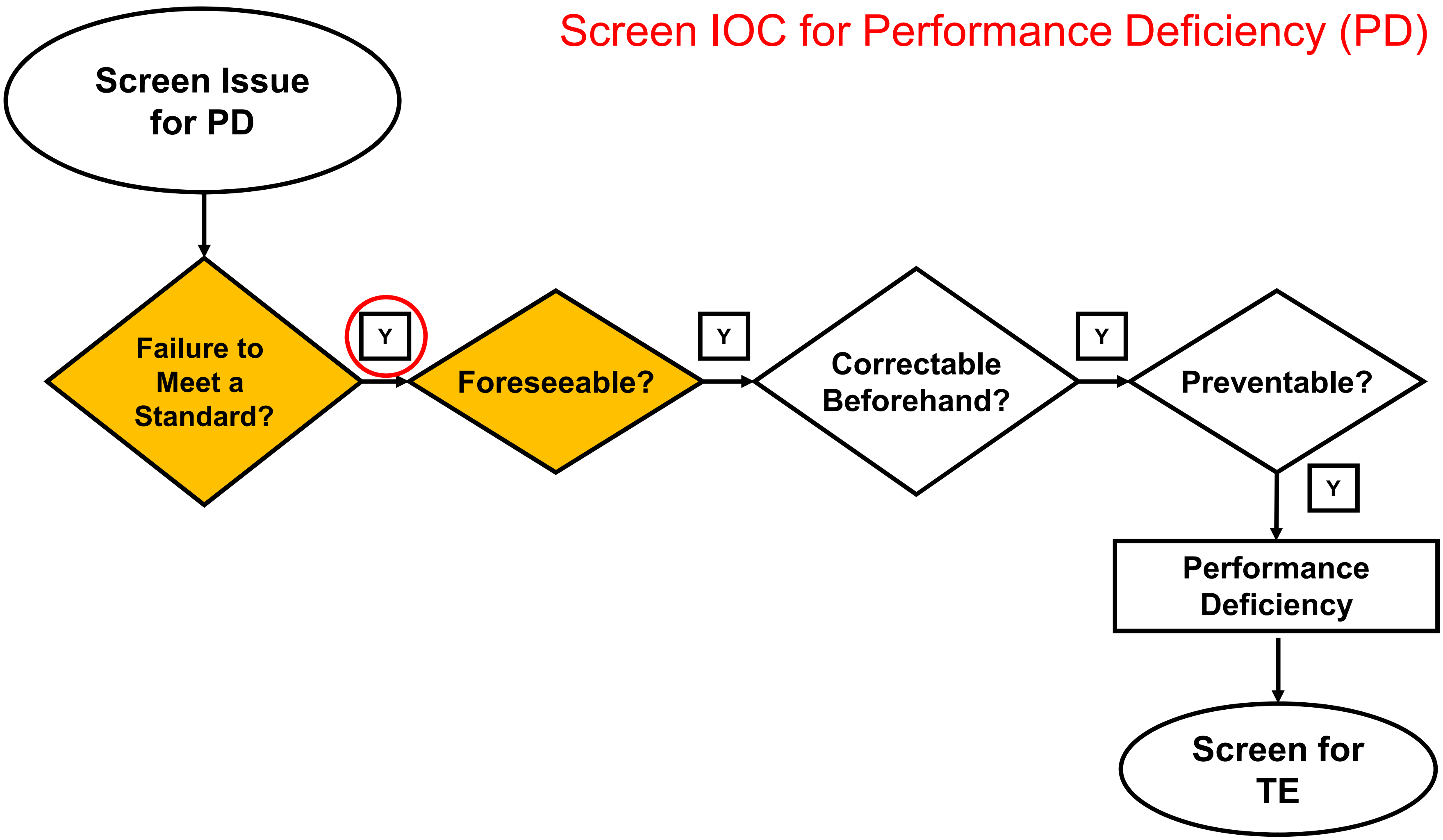

MC 0612, Appendix B, Issue Screening

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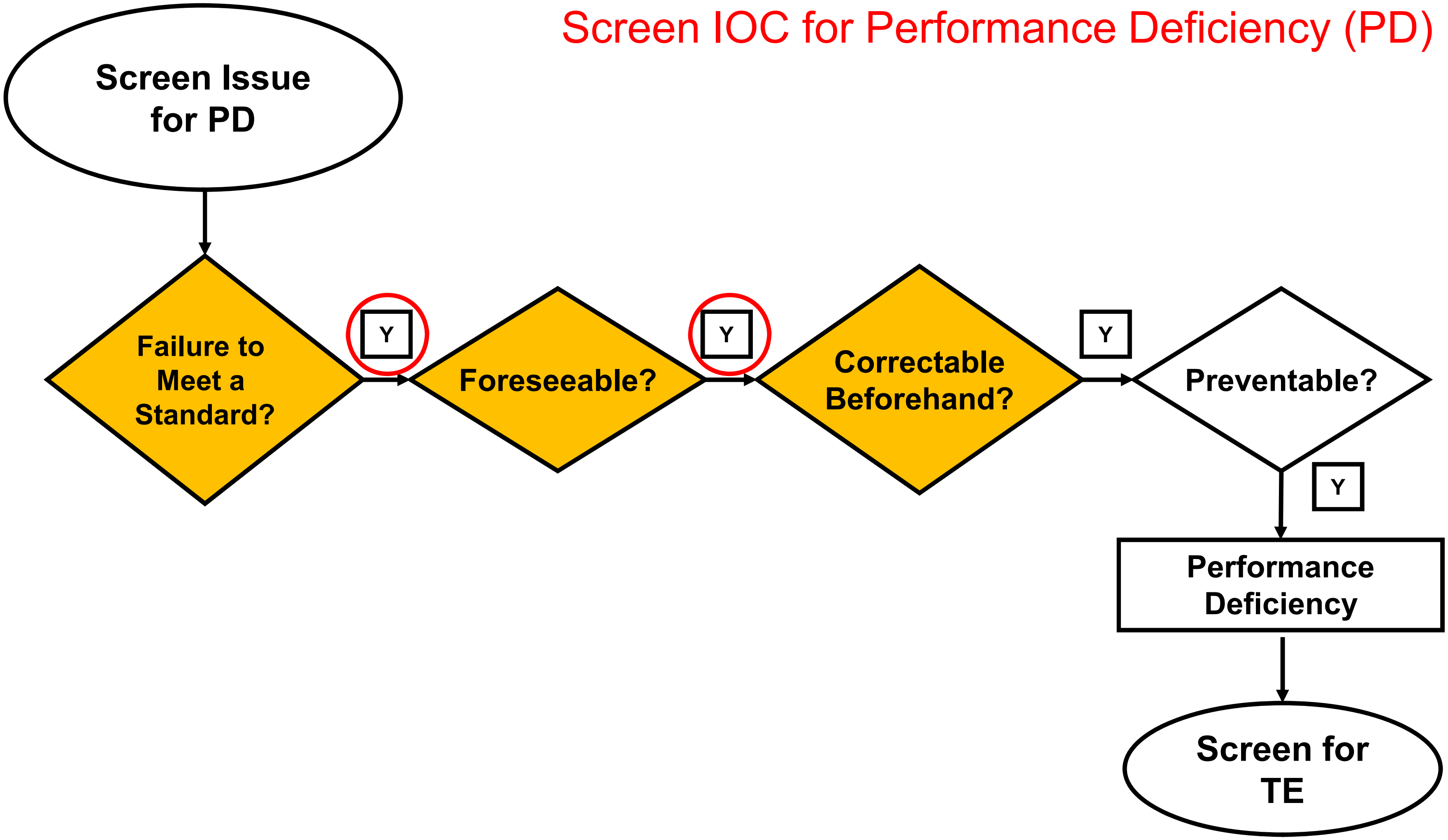

MC 0612, Appendix B, Issue Screening

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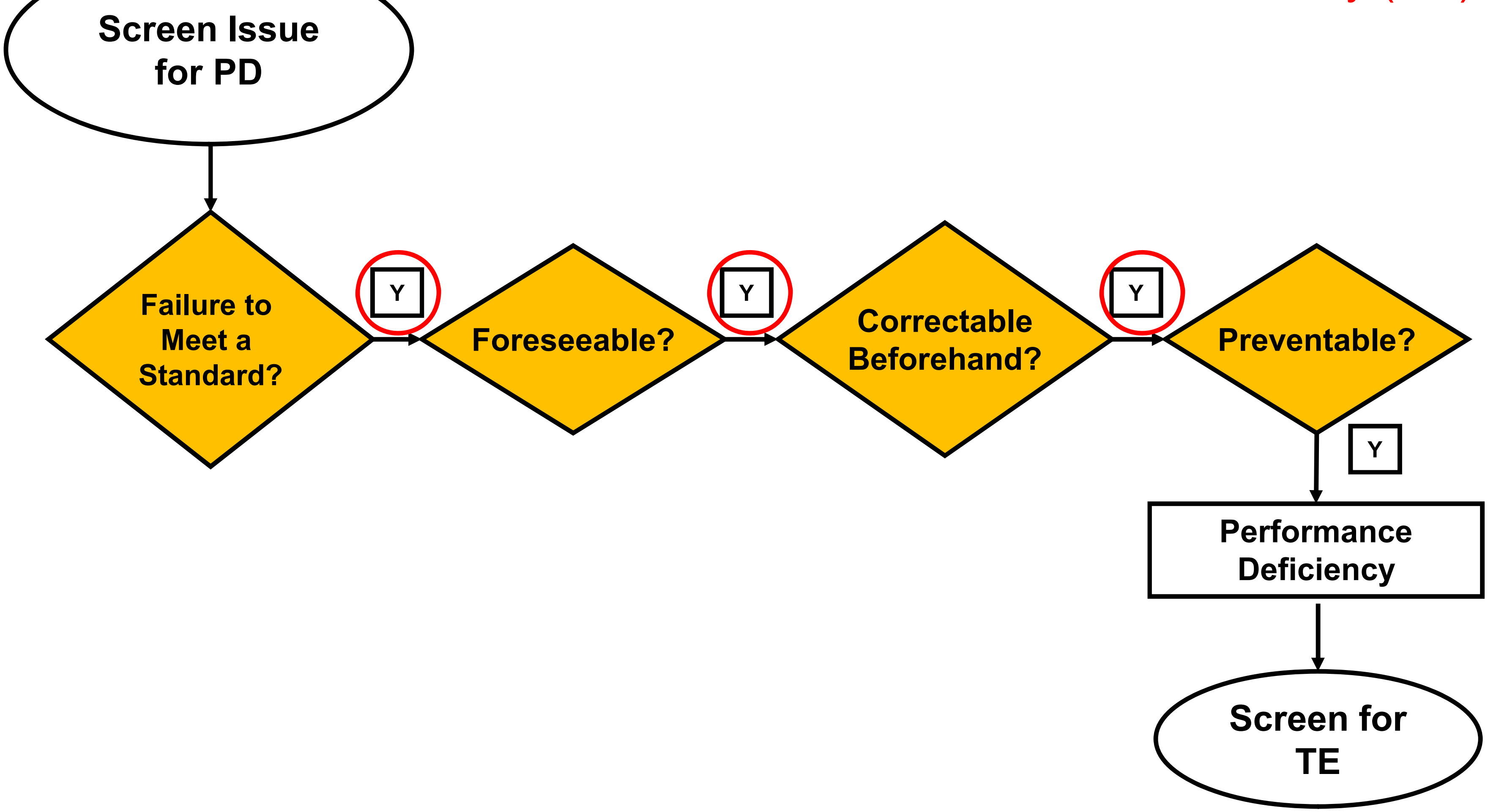

MC 0612, Appendix B, Issue Screening

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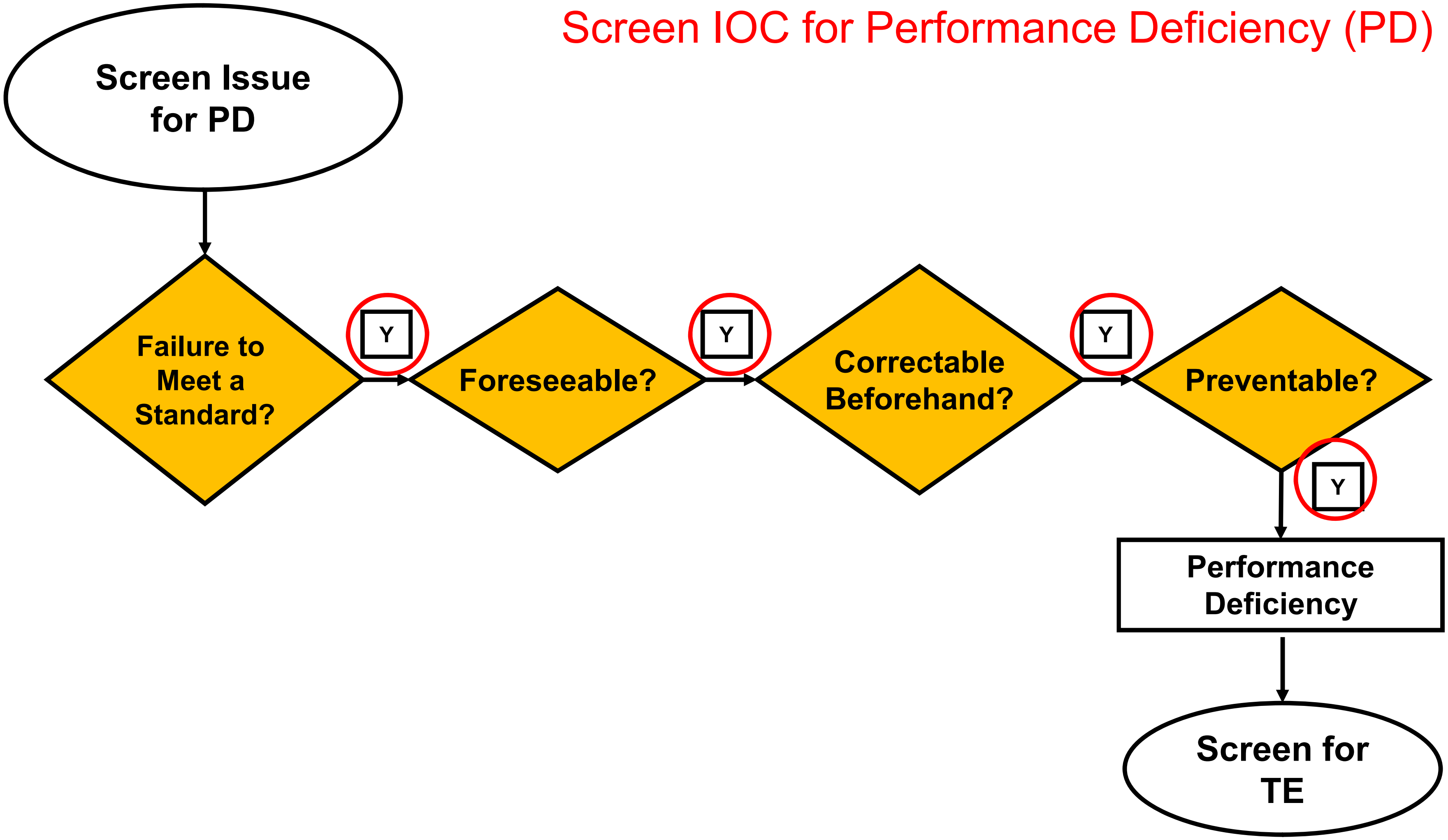
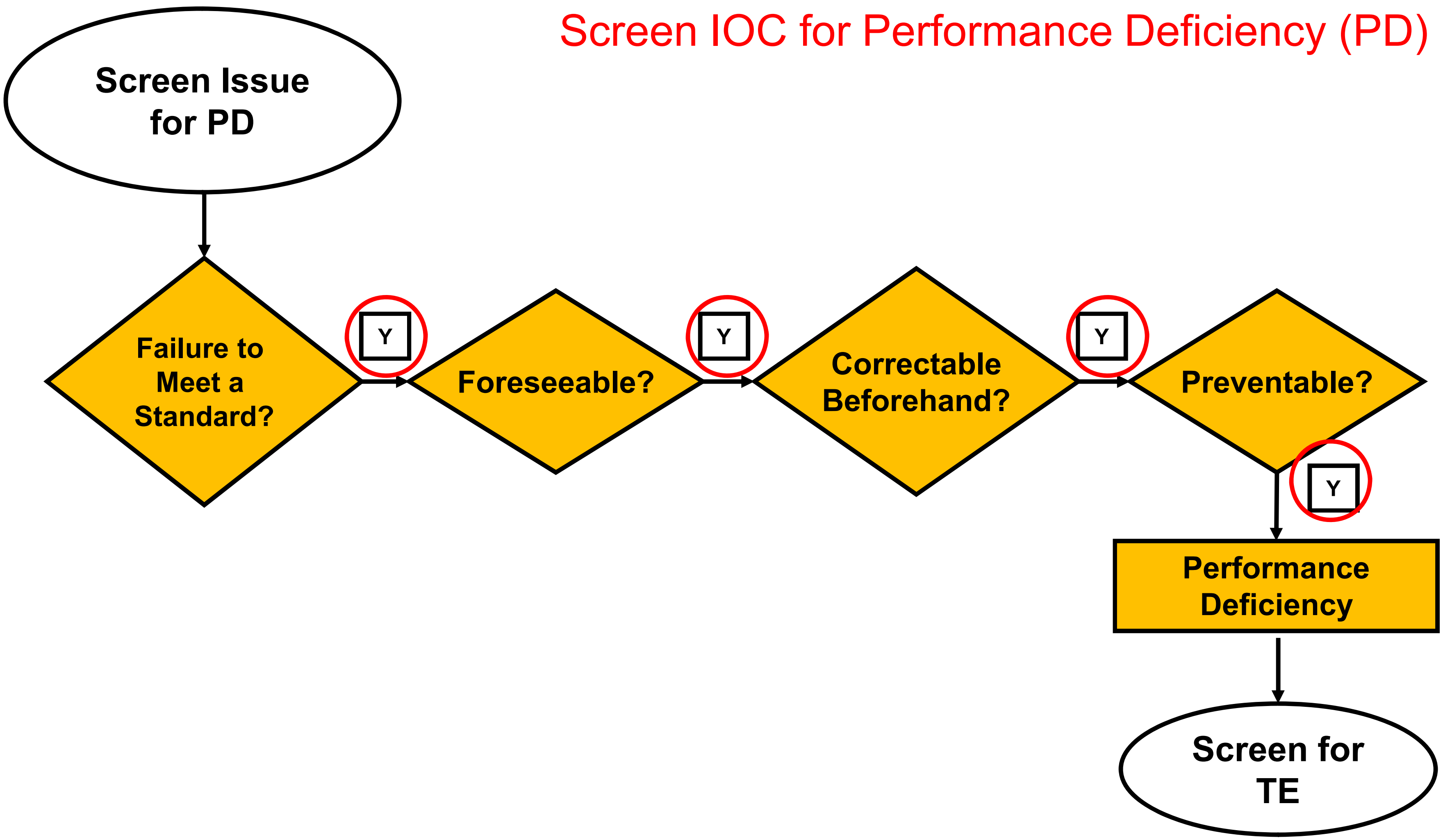


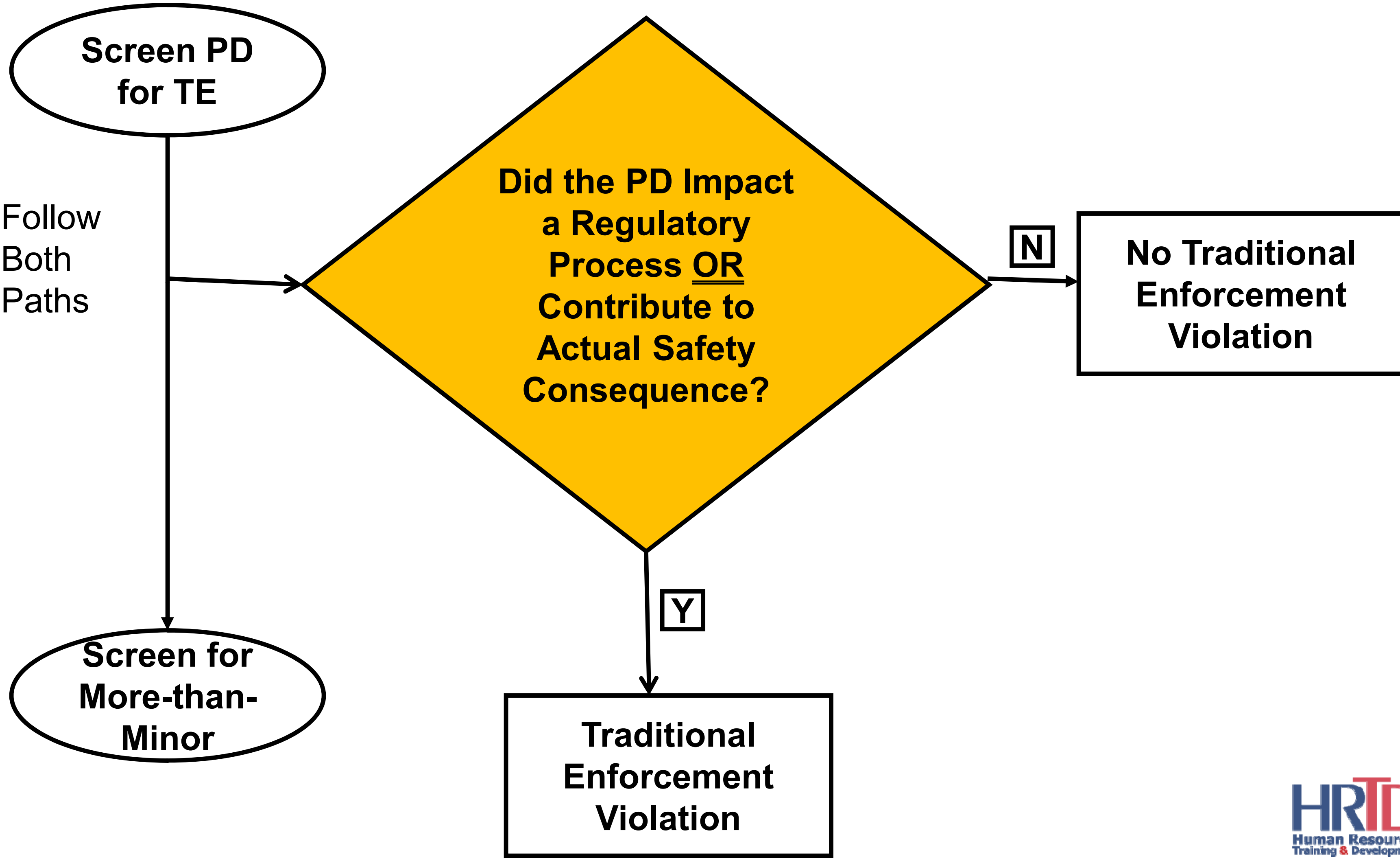
MC 0612, Appendix B, Issue Screening

Screen IOC for Performance Deficiency (PD)



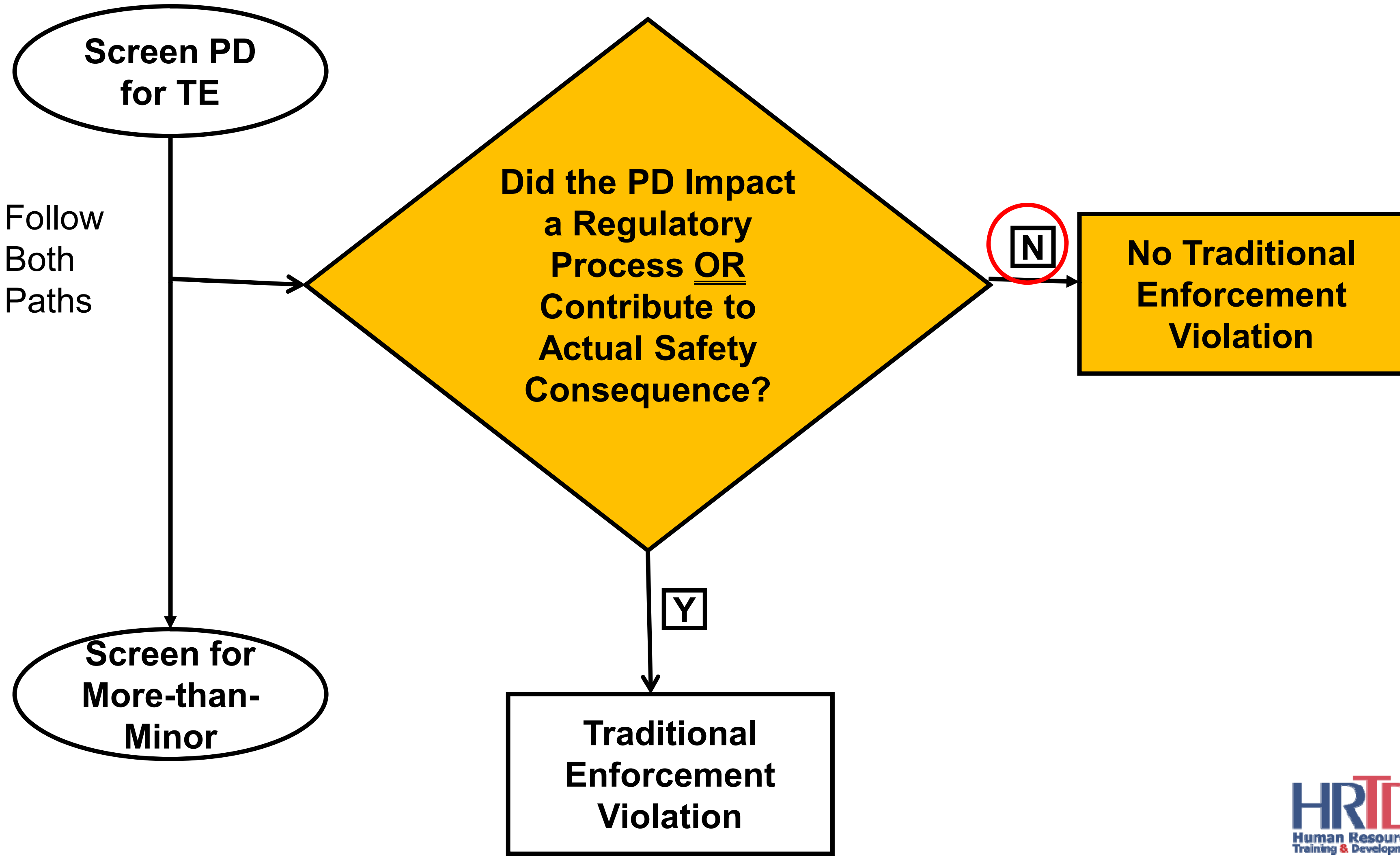
MC 0612, Appendix B, Issue Screening

Screen PD for Traditional Enforcement (TE)



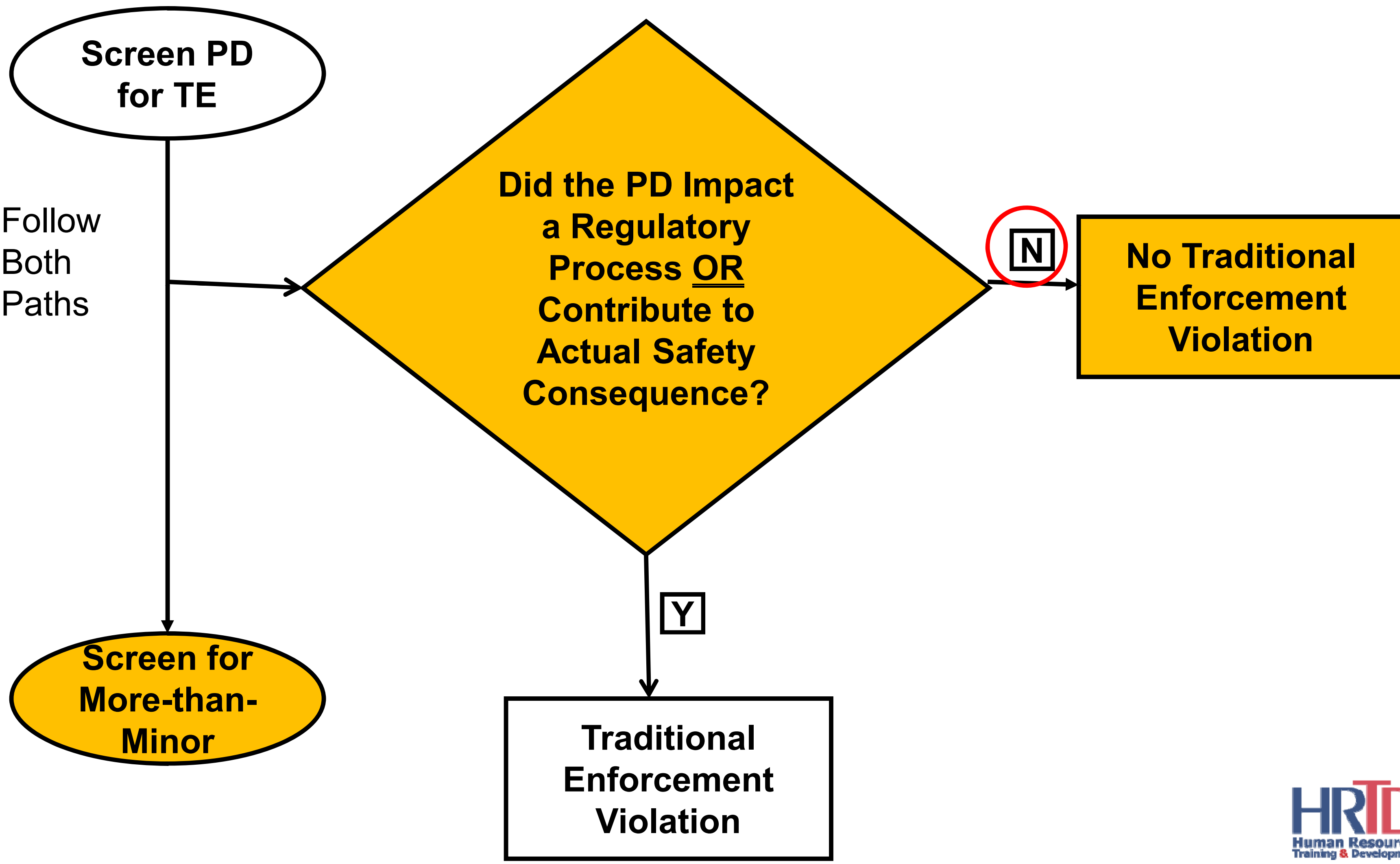
MC 0612, Appendix B, Issue Screening

Screen PD for Traditional Enforcement (TE)



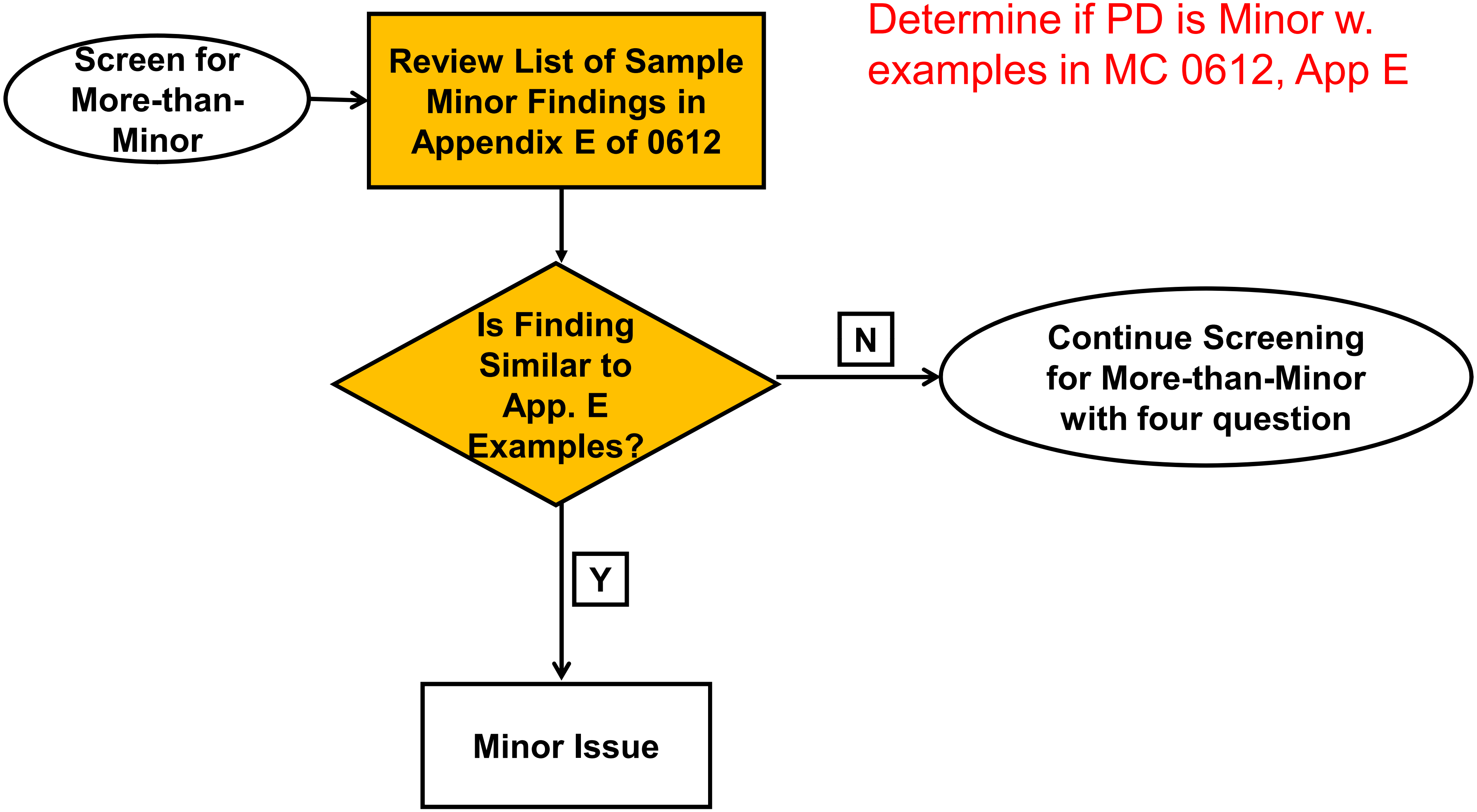
MC 0612, Appendix B, Issue Screening

Screen PD for Traditional Enforcement (TE)



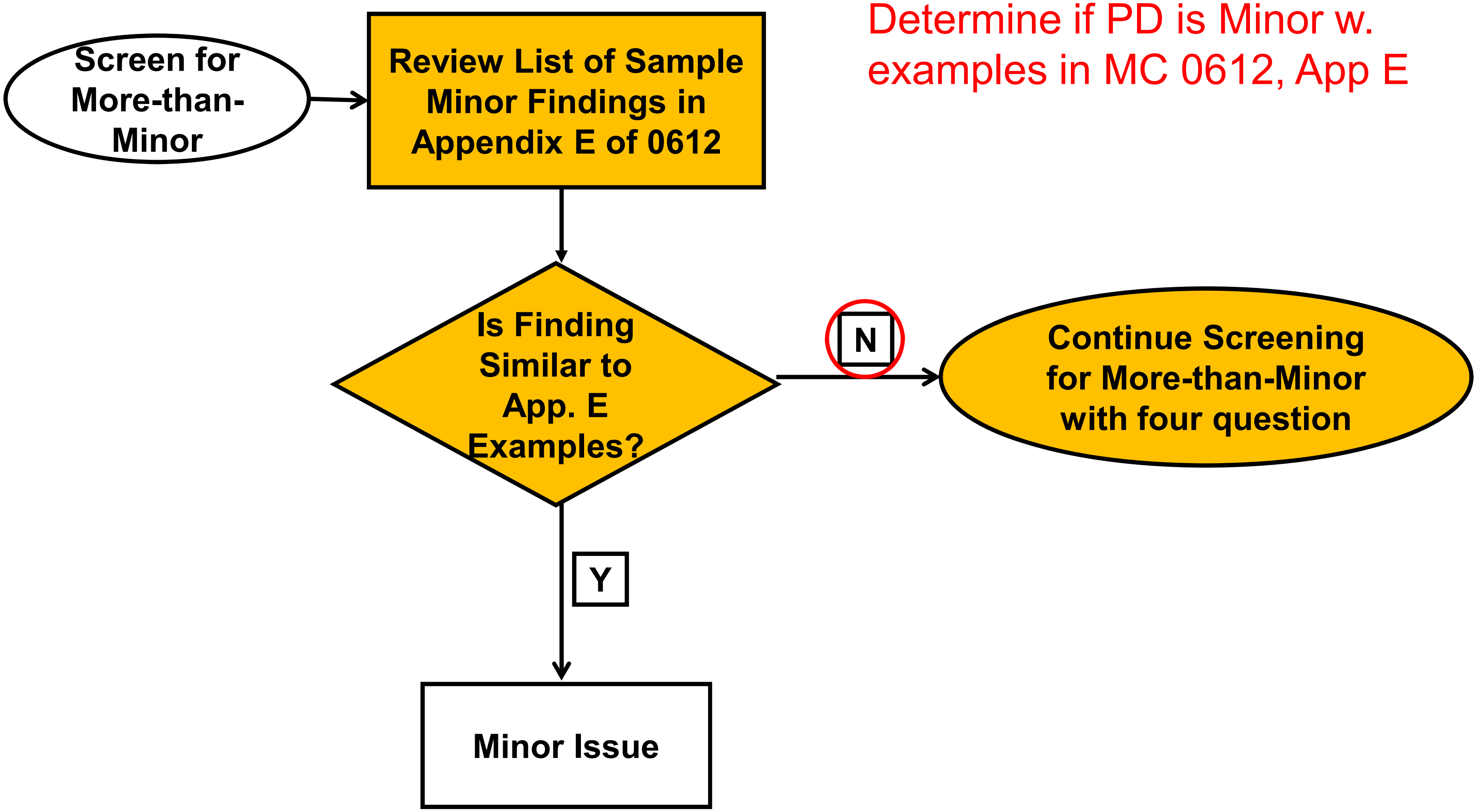
MC 0612, Appendix B, Issue Screening

Determine if PD is Minor w. examples in MC 0612, App E



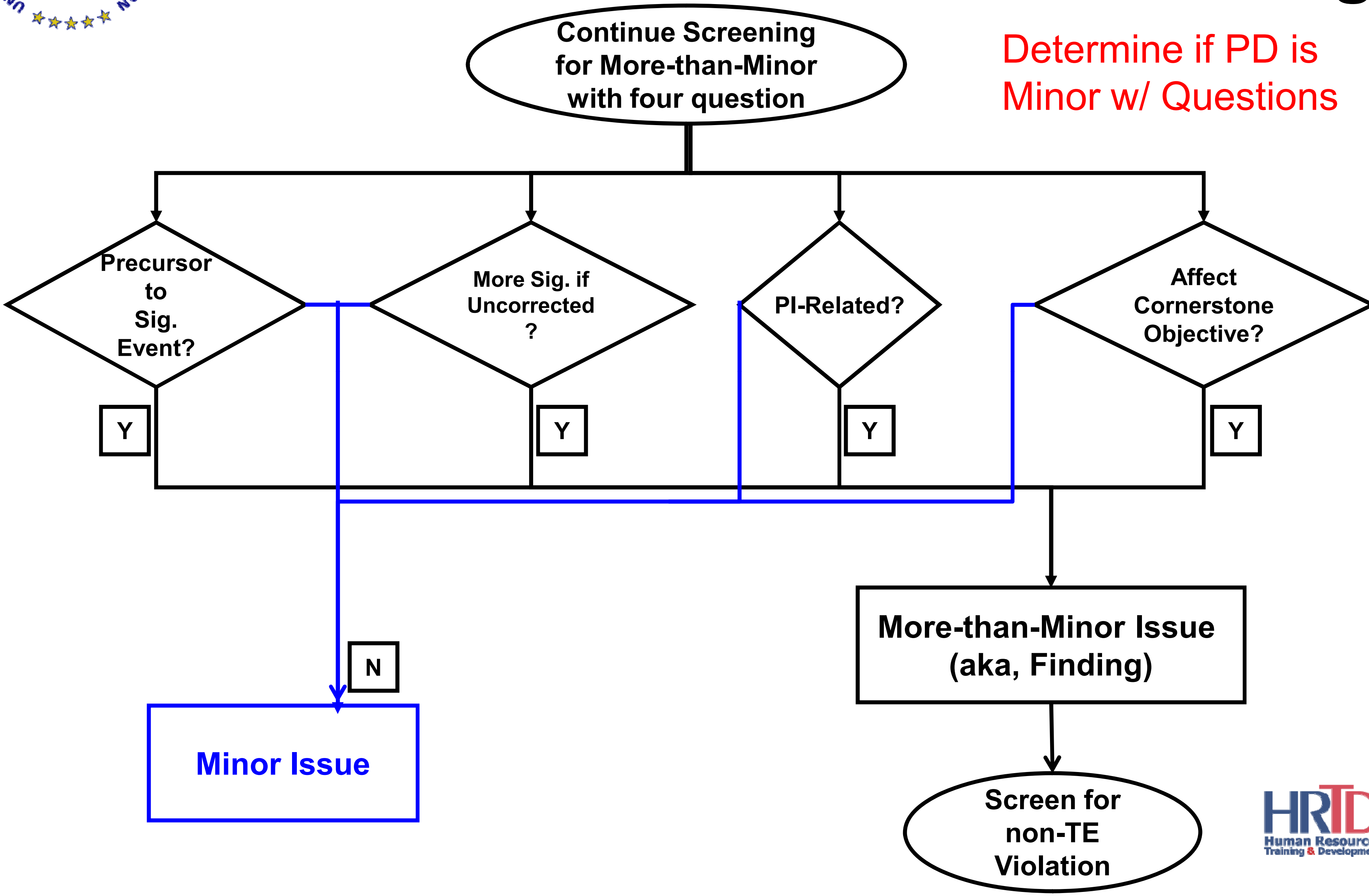
MC 0612, Appendix B, Issue Screening

Determine if PD is Minor w. examples in MC 0612, App E



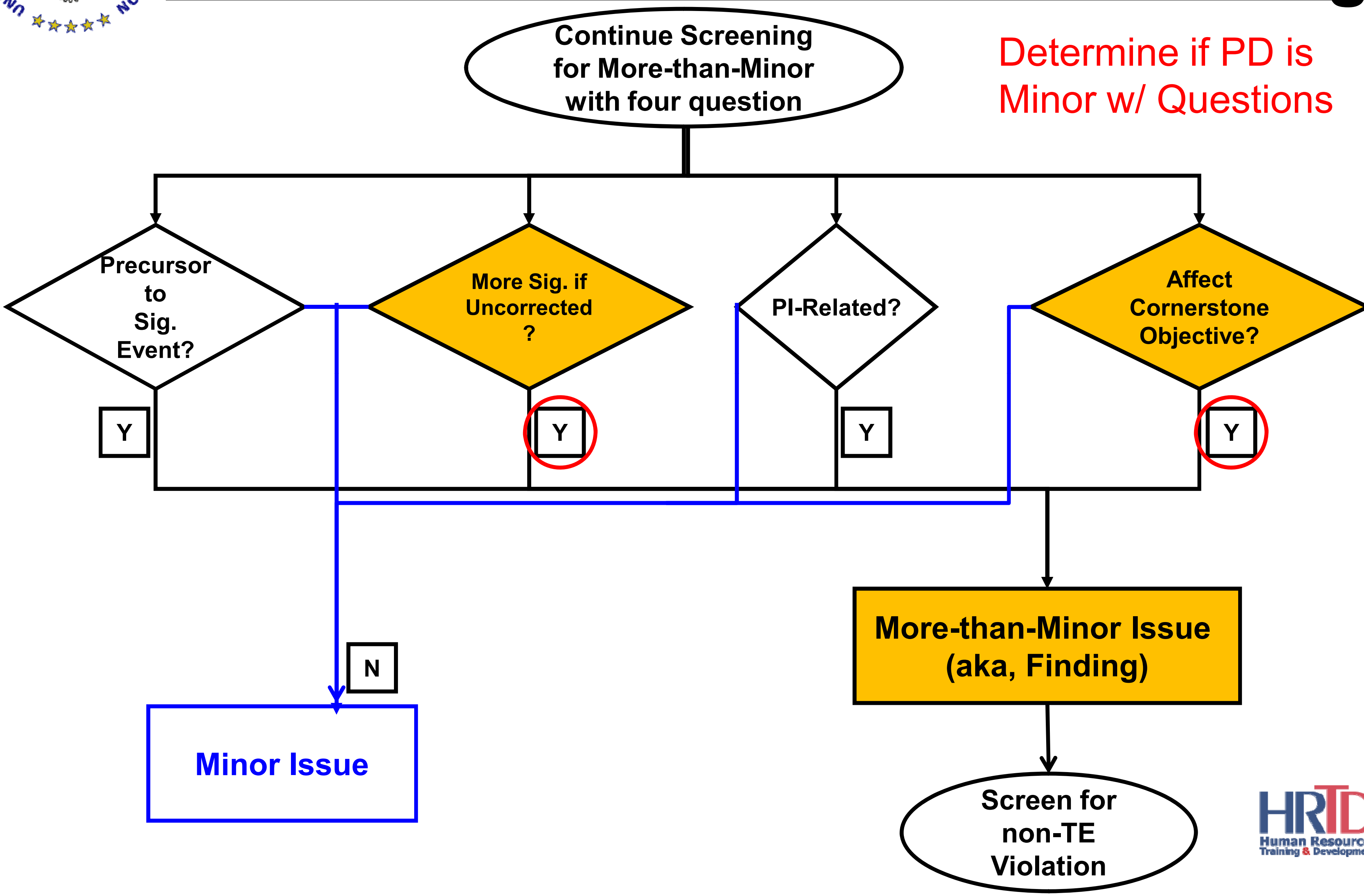
MC 0612, Appendix B, Issue Screening

Determine if PD is Minor w/ Questions



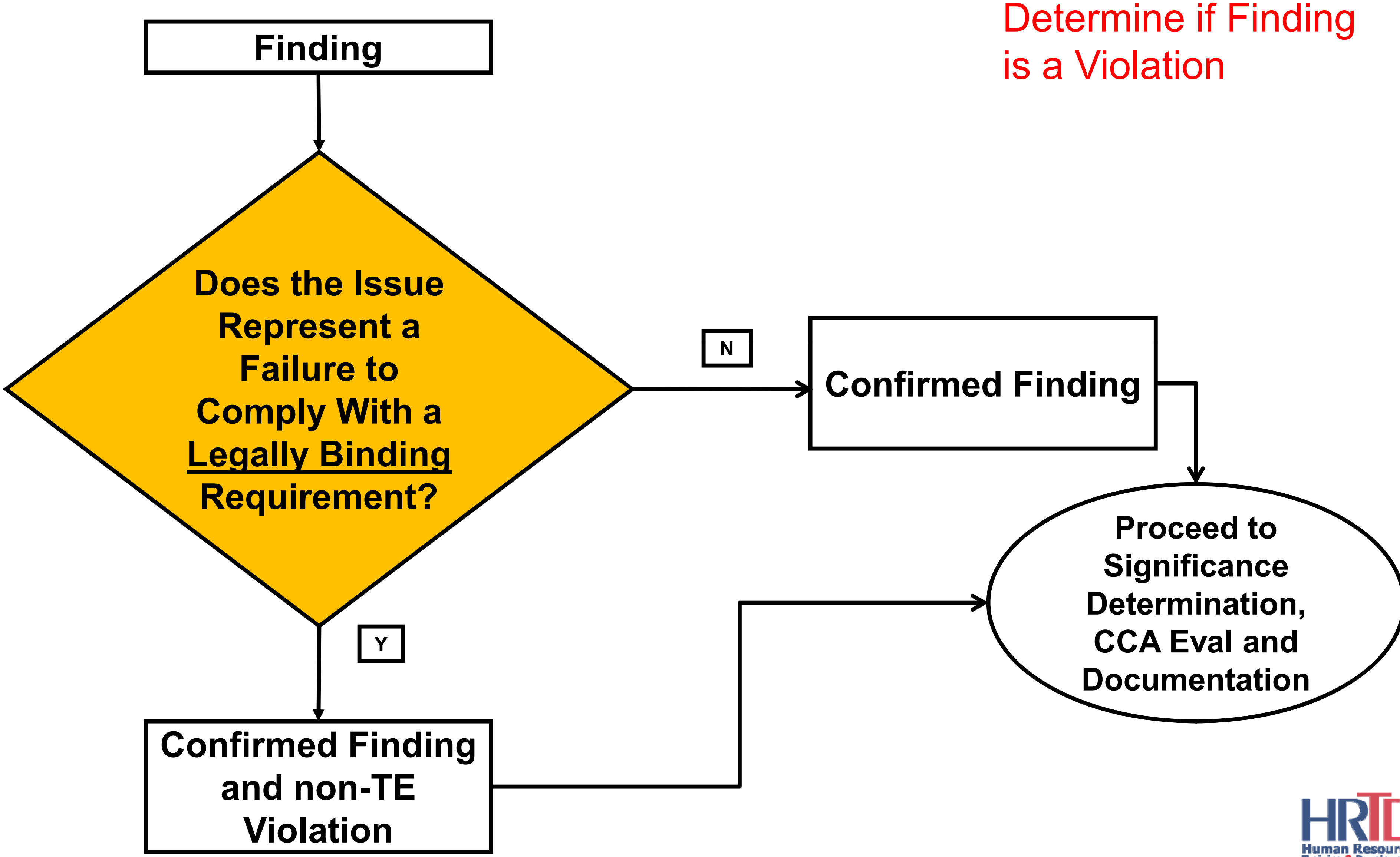
MC 0612, Appendix B, Issue Screening

Determine if PD is Minor w/ Questions



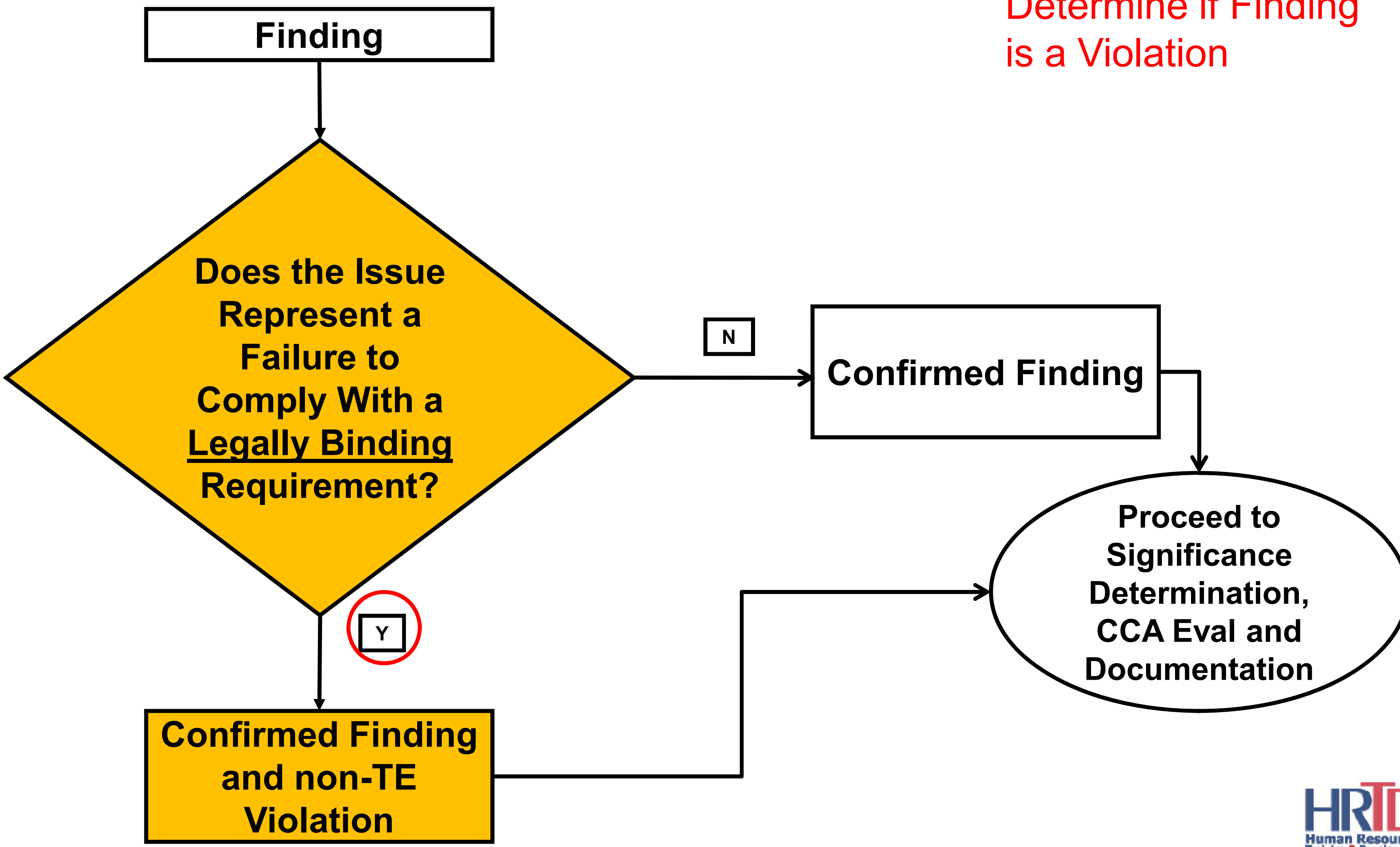
MC 0612, Appendix B, Issue Screening

Determine if Finding is a Violation



MC 0612, Appendix B, Issue Screening

Determine if Finding is a Violation



Summarizing…

- The Observation is an IOC
- The IOC is a PD
- The PD is more-than-minor
- The PD is a Finding
- Traditional Enforcement does <u>not</u> apply
- The IOC is a Violation
- Risk Significance is yet to be determined

Determine Risk Significance of the Violation

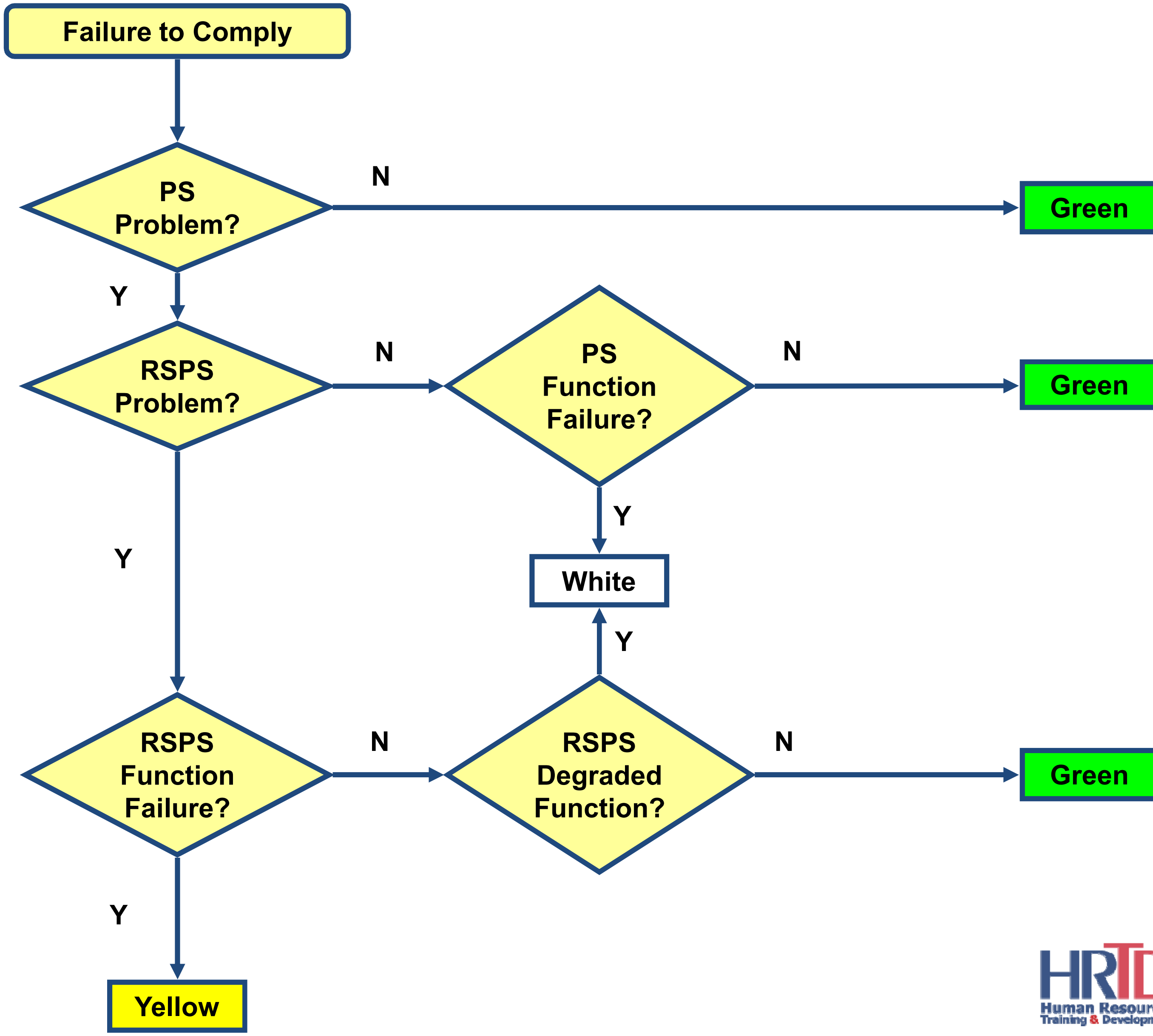
Table 3a - SDP PHASE 1 SCREENING WORKSHEET FOR EMERGENCY PREPAREDNESS, OCCUPATIONAL & PUBLIC RADIATION, AND SECURITY CORNERSTONES
IF the finding is in the licensee's: 1.emergency preparedness area, **THEN STOP. Go to** IMC 0609, Appendix B. 2.occupational radiation safety area, **THEN STOP. Go to** IMC 0609, Appendix C. 3.public radiation safety area, **THEN STOP. Go to** IMC 0609, Appendix D. 4.security area, **THEN STOP. Go to** IMC 0609, Appendix E.

- So Phase 1 of the SDP sends us to MC 0609, Appendix B (Emergency Preparedness Significance Determination Process)

Determine Risk Significance of the Violation

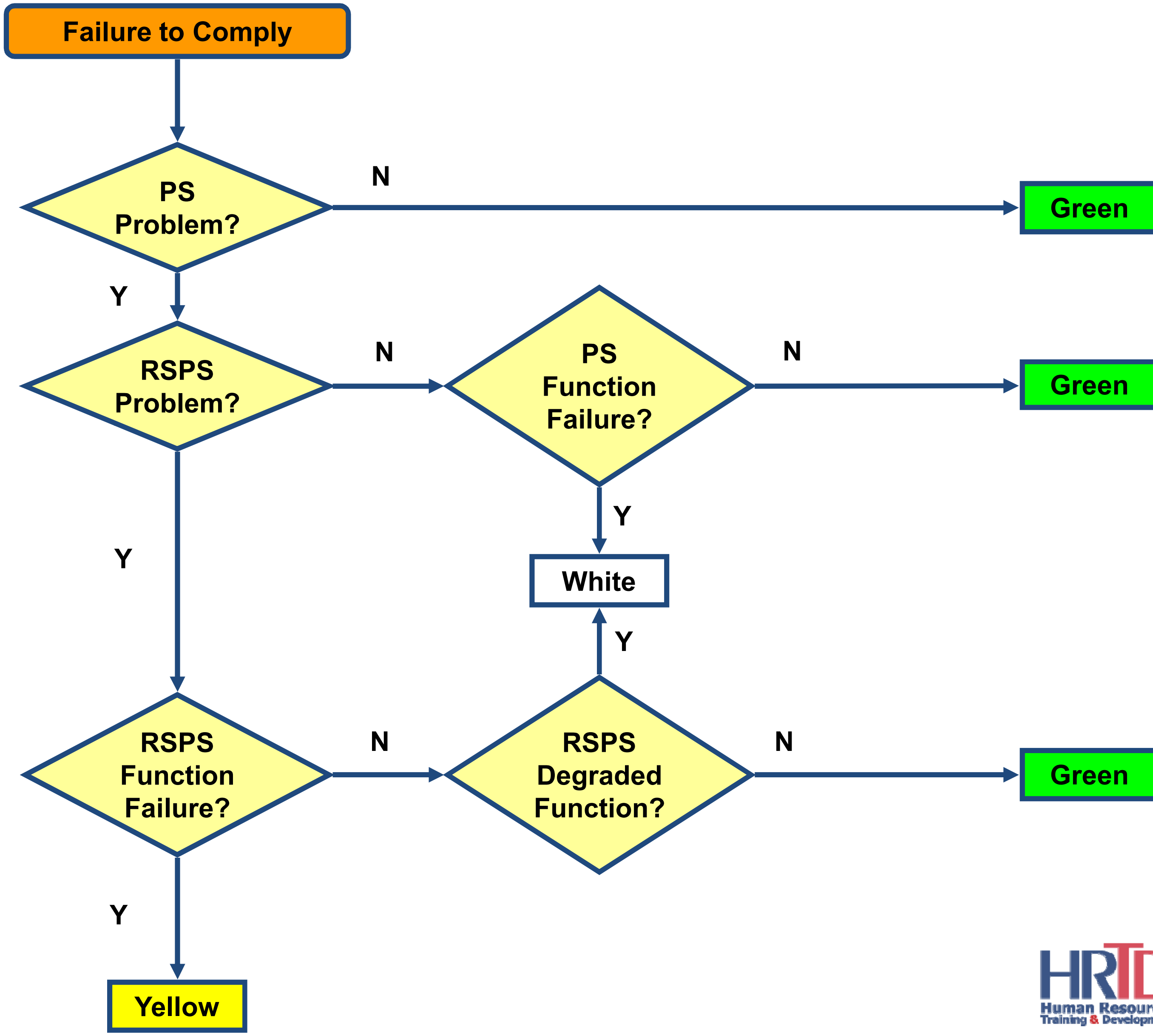
- MC 0609 Appendix B, “Emergency Preparedness Significance Determination Process” applies.
- Choice of flowcharts:
  - Failure to Comply
  - Actual Event Implementation Problem

Determining Significance with MC 0609 Appendix B Flowchart



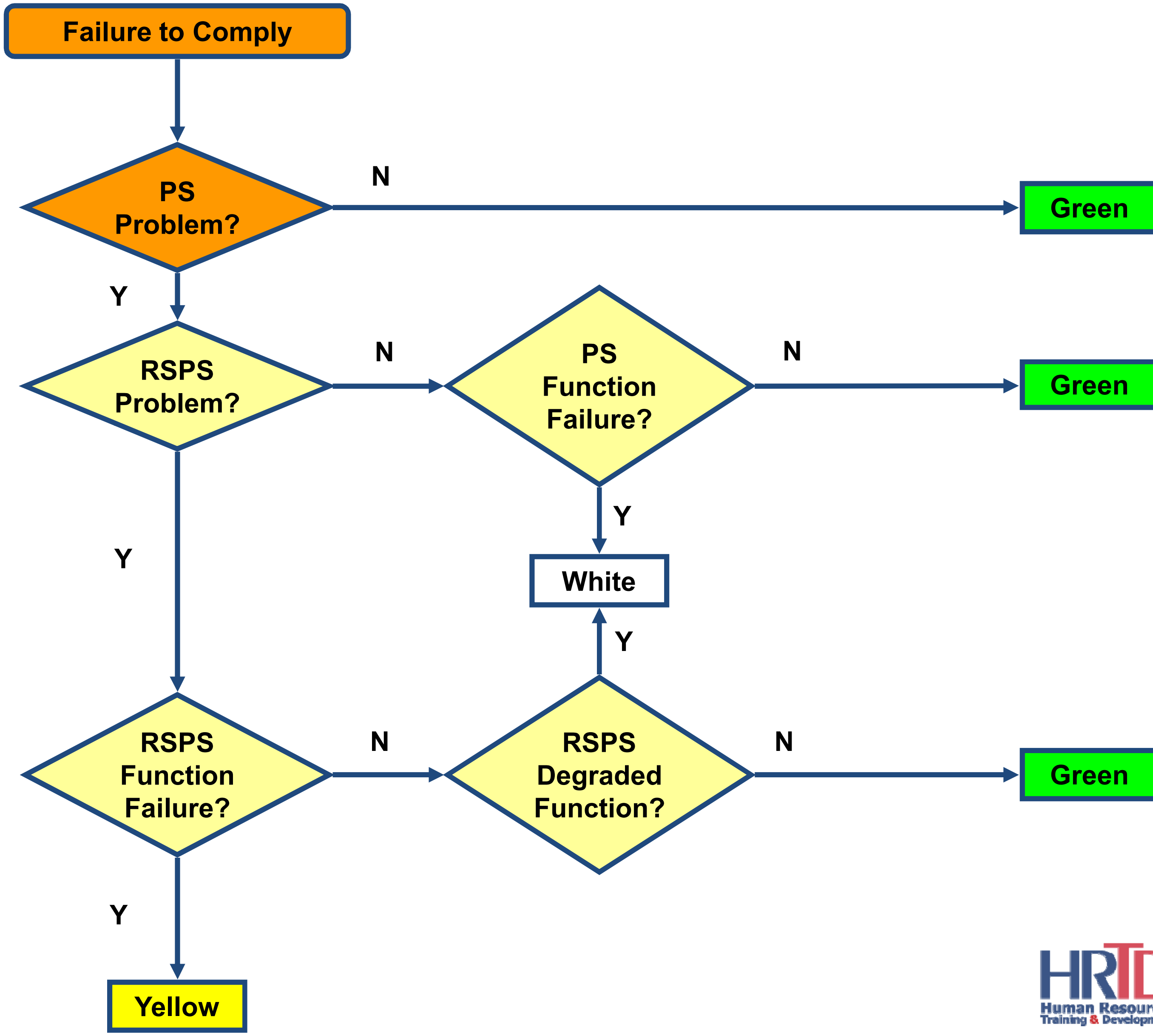
Determining Significance with MC 0609 Appendix B Flowchart

- Failure to Comply Starts the Process



Determining Significance with MC 0609 Appendix B Flowchart

- Failure to Comply Starts the Process
- What's a "PS Problem?"



Determining Significance with MC 0609 Appendix B Flowchart

- What's a "PS Problem?"
- From MC 0609, Appendix B

PLANNING STANDARD (PS): Any of the sixteen Emergency Preparedness Planning Standards defined in 10 CFR 50.47(b), including the RISK-SIGNIFICANT PLANNING STANDARDS and related sections of Appendix E to 10 CFR Part 50.

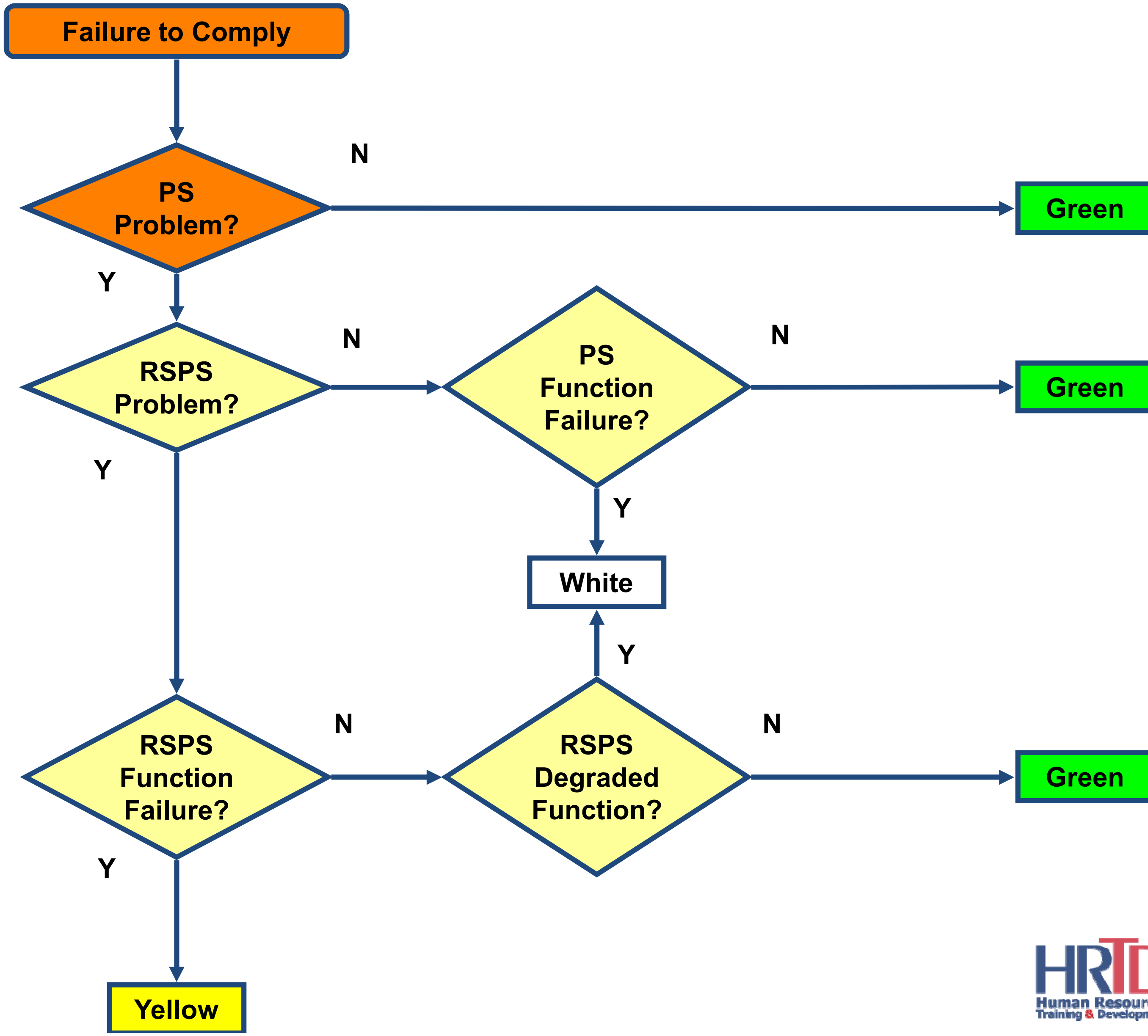
Recall that the corrective action failure represented a failure to meet 10 CFR 50.47, Emergency Plans.

"b) The onsite and, except as provided in paragraph (d) of this section, offsite emergency response plans for nuclear power reactors must meet the following standards:

(14) Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilities, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected."

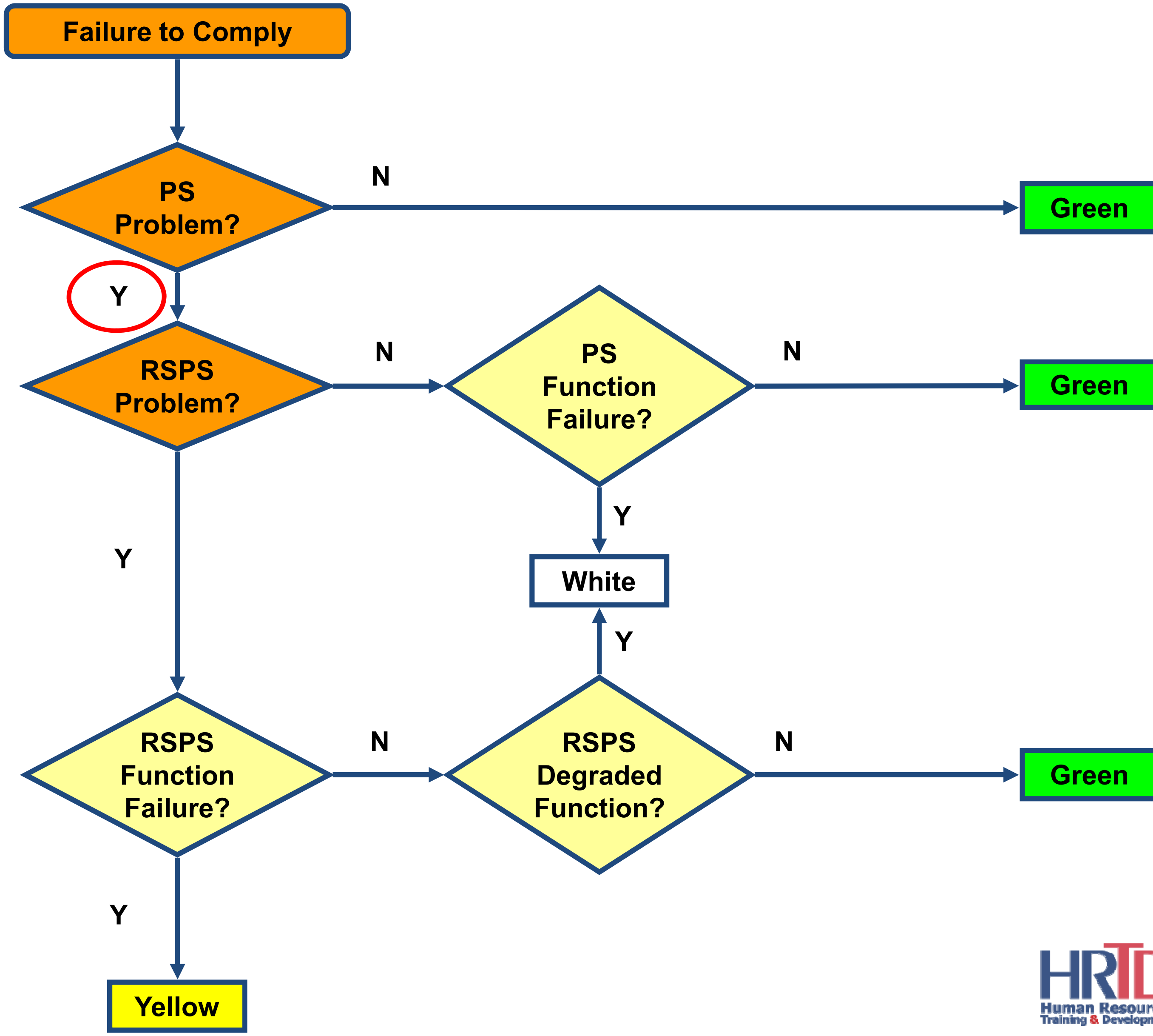
Determining Significance with MC 0609 Appendix B Flowchart

- MC 0609 Appendix B Flowchart
- Failure to Comply Starts the Process
- So, is there a "PS Problem?"- If so, what is it?



Determining Significance with MC 0609 Appendix B Flowchart

- MC 0609 Appendix B Flowchart
- Failure to Comply Starts the Process
- There is a PS problem – the failure to correct previously identified deficiencies.
- What's an "RSPS problem?"



Determining Significance with MC 0609 Appendix B Flowchart

- What's a "RSPS Problem?"
- From MC 0609, Appendix B

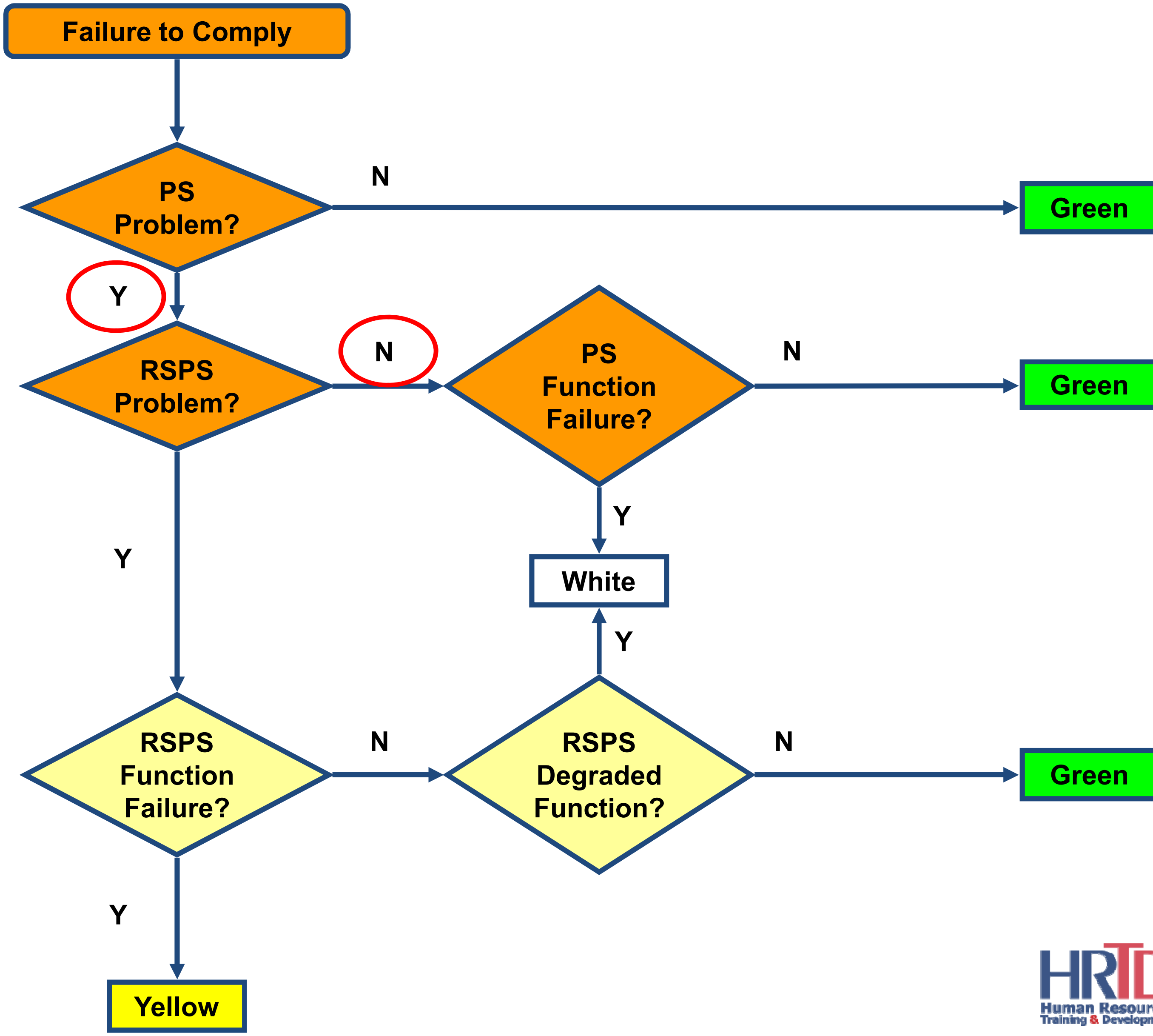
  RISK-SIGNIFICANT PLANNING STANDARD (RSPS): Any of the following four Planning Standards defined in 10 CFR 50.47(b):

  Specifically-10 CFR 50.47(b)(4), (5), (9), or (10), including the related sections of Appendix E to 10 CFR Part 50.

- Is the issue a Risk Significant Standard Problem?

Determining Significance with MC 0609 Appendix B Flowchart

- MC 0609 Appendix B Flowchart
- Failure to Comply Starts the Process
- There is a PS problem – the failure to correct previously identified deficiencies.
- There is NOT a RSPS problem
- Now, what's a "PS Function Failure?"



Determining Significance with MC 0609 Appendix B Flowchart

- What's a "PS Function Failure?"
- From MC 0609, Appendix B

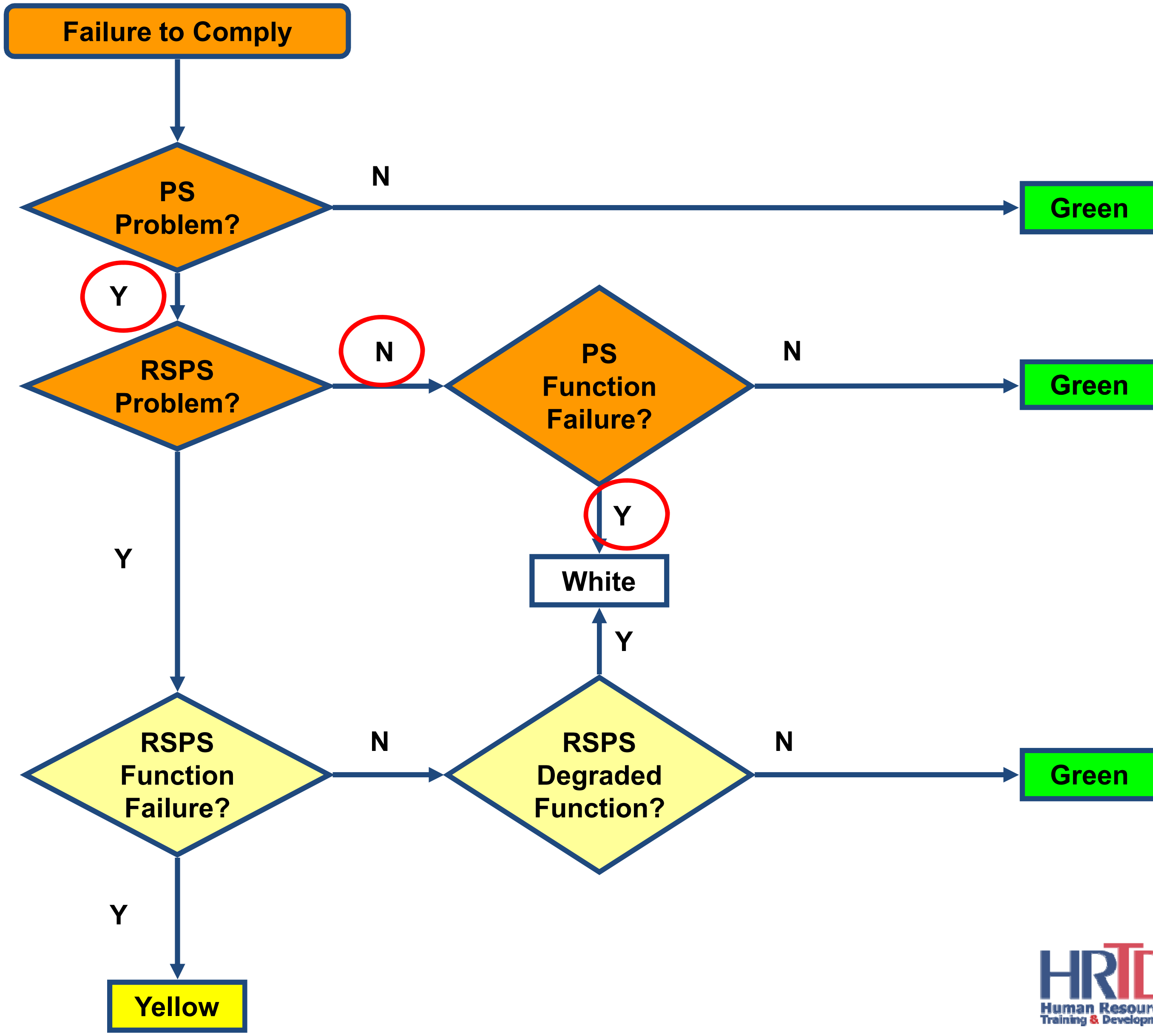
FAILURE TO COMPLY - As defined in Section 2.1 of this appendix, failure to comply means that a program is noncompliant with a regulatory requirement. Loss of PS function means that program elements are not adequate, not compliant with the planning standards of 10 CFR 50.47(b), or otherwise not functional to such an extent that the function of the planning standard is not available for emergency response.

WEAKNESS - As applied to emergency preparedness, a weakness is a level of performance demonstrated during a drill or exercise that could have precluded effective implementation of the Emergency Plan in the event of an actual emergency. Weaknesses are not confined to performance problems that result in a loss of PS function. For example, an inaccurate or untimely classification, notification, or Protective Action Recommendation (PAR) development is a weakness associated with an RSPS (i.e., a Drill and Exercise Performance (DEP) PI opportunity failure). However, a WEAKNESS also exists if a performance problem occurs associated with an accurate and/or timely classification, notification or PAR development that was anticipated by the scenario (i.e., a DEP PI successful opportunity)...

•Does the finding represent a planning standard function failure?

Determining Significance with MC 0609 Appendix B Flowchart

- MC 0609 Appendix B Flowchart
- Failure to Comply Starts the Process
- There is a PS problem – the failure to correct previously identified deficiencies.
- There is NOT a RSPS problem
- There was a PS function failure.

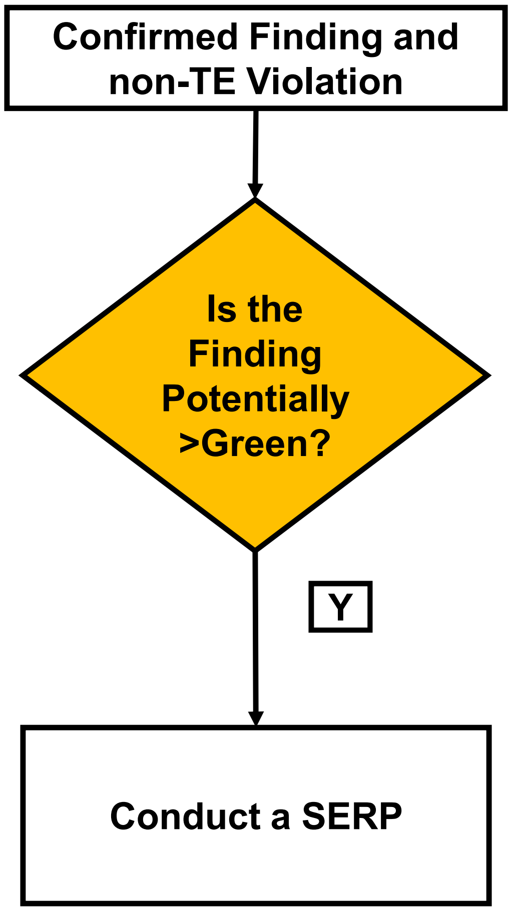


Safety Culture Consideration

Review the details involved with the finding and determine

- How the CCA requirements from MC 0612 are met for the finding?
- Which if any cross-cutting aspect(s) from MC 0310 apply?

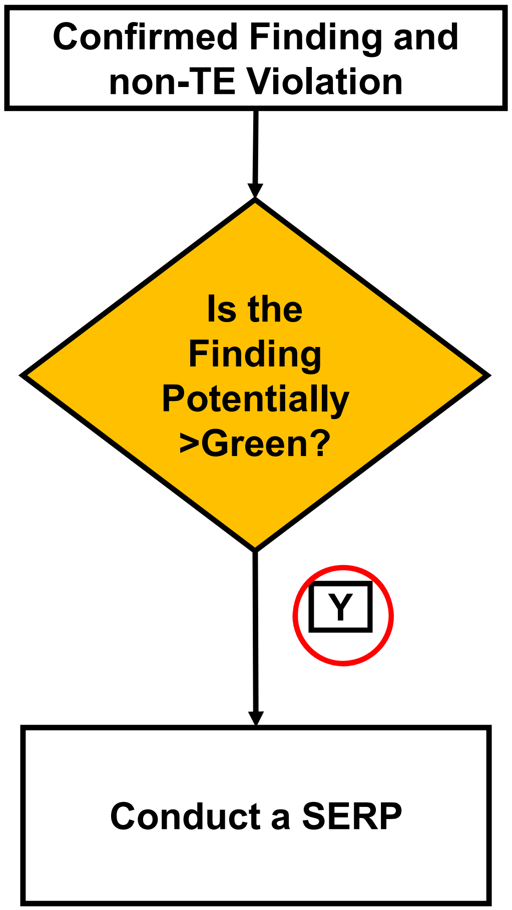
MC 0612, Appendix B, Issue Screening



Screened the violation through MC 0609, Attachment 4 (SDP Phase 1) and MC 0609, Appendix B (SDP Phase 2).

Violation processed through Phase 2 with White Significance.

MC 0612, Appendix B, Issue Screening



A Significance Enforcement Review Panel must convene to provide a management review of the preliminary significance characterization and basis of any findings that are potentially White, Yellow, Red, or Greater than Green.

Fully Characterizing the Issue

At this stage of the inspection, the issue…

- Is a Finding
- Is a Violation of 10 CFR 50.47(b)(14) and/or 10 CFR 50 Appendix E IV.F.2.g
- Is of a level of significance that is characterized as WHITE, which a SERP will review
- Has a cross-cutting aspect which contributes to the cause and will be discussed in the exit meeting and documented in the report details.

Deterministic SDP Example #2

RP Occupational (Access Control)

- A non-cited violation, with three examples, was identified as a result of the licensee's failure to barricade, conspicuously post, and lock or guard a restricted high radiation areas (dose rates greater than 1000 millirems per hour) to prevent unauthorized entry.

- Example One: A radiation protection technician walked through a door to Steam Generator Bay B on the 994-foot elevation of the containment building and left the door unguarded and open with the posting not conspicuous. General area dose rates were as high as 1500 millirem per hour in the bay.

  Example Two: The ladder leading to the Steam Generator bay was locked with a sheet metal gate, but the gate was flanked on the side by rails which were approximately 3 feet high. This would have allowed an individual to bypass the gate by simply stepping over the railing. General area dose rates were as high as 4 rem per hour in the bay.

- Example Three: A permanent ladder leading into the reactor cavity was controlled by locking the ladder climbing rails at the top of the ladder. An individual could either step around the ladder barrier or go underneath it and enter the reactor cavity. Additionally, scaffolding was erected to house a set of temporary stairs into the cavity. An individual could bypass the locked door by climbing on the outside of the scaffolding and down into the reactor cavity using a ladder-like structure which was part of the scaffolding. General area dose rates were as high as 5 rem per hour in the reactor cavity.

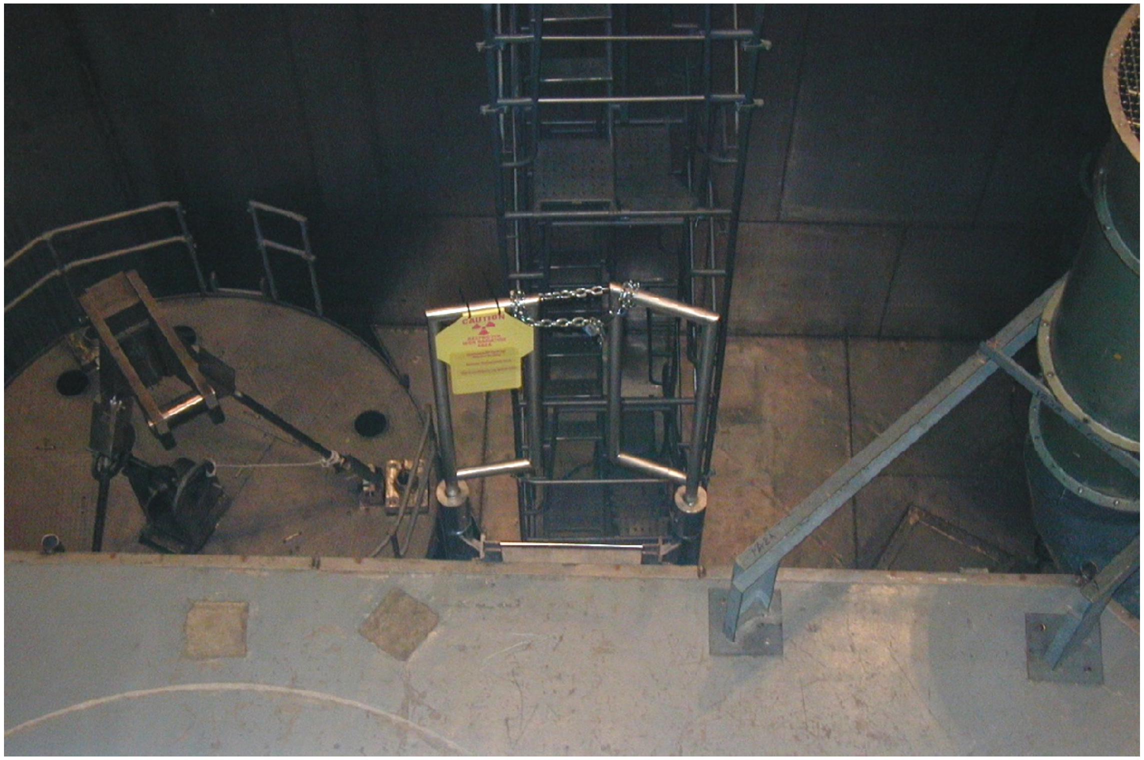
Deterministic Example #2

- The ladder provides access to the steam generator platform. There is a short sheet metal barrier that a worker could crawl over (since it is too short to be considered an adequate barrier) or around (the side shrouds are too shallow), enabling unauthorized access to this locked High Radiation Area. This is a violation of the licensee's high radiation area technical specifications and is a PI "hit."

HRTD
Human Resources Training & Development

Deterministic Example #2

- The "locked" swing gate guarding the ladder does not prevent unauthorized entry since a worker could go under the partially opened gate or swing around the outside of the gate. Additionally, the ladder rungs were in place and no ladder shroud was used.

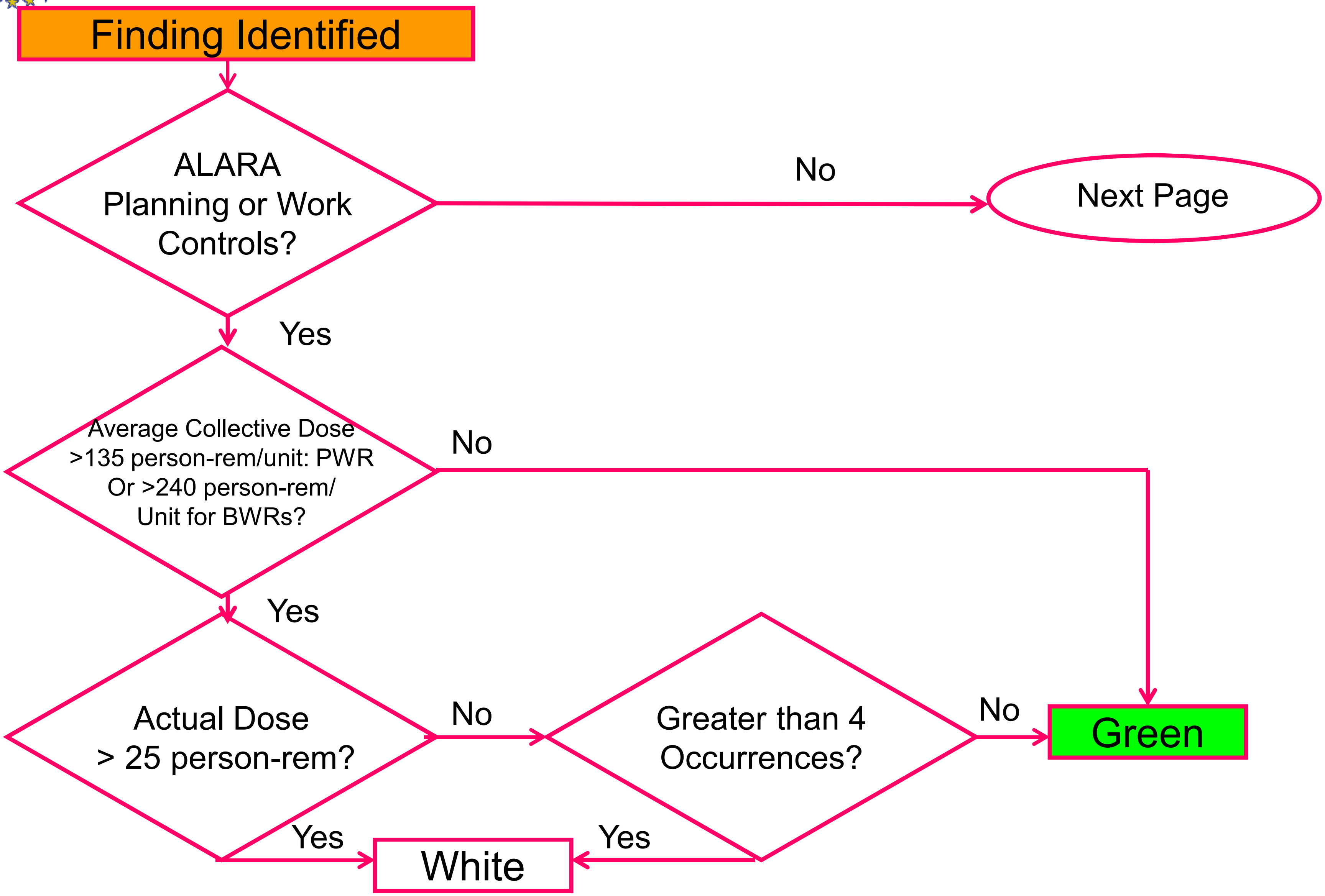


Determine Risk Significance of the Violation

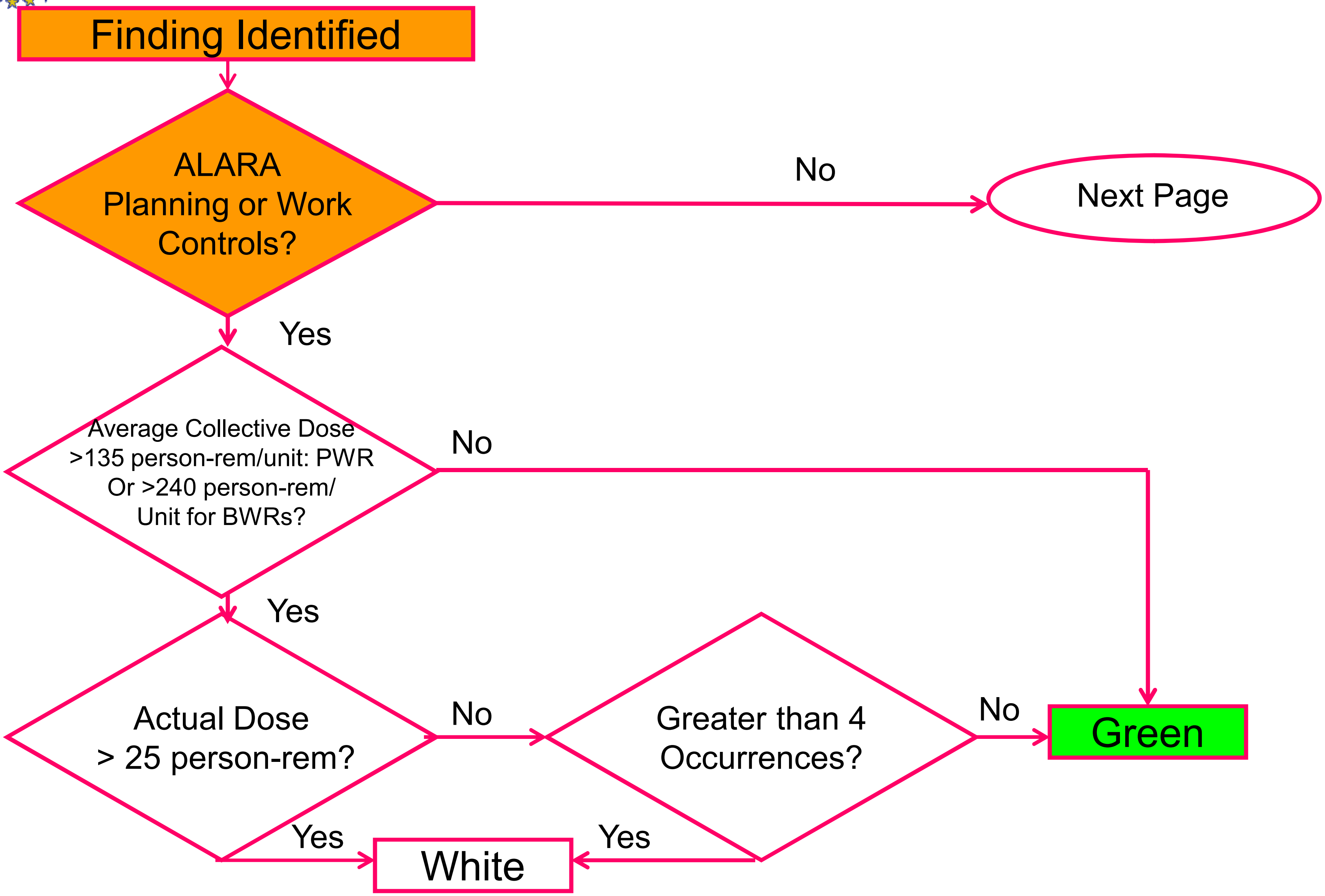
Table 3a - SDP PHASE 1 SCREENING WORKSHEET FOR EMERGENCY PREPAREDNESS, OCCUPATIONAL & PUBLIC RADIATION, AND SECURITY CORNERSTONES
IF the finding is in the licensee's: 1.emergency preparedness area, **THEN STOP. Go to** IMC 0609, Appendix B. 2.occupational radiation safety area, **THEN STOP. Go to** IMC 0609, Appendix C. 3.public radiation safety area, **THEN STOP. Go to** IMC 0609, Appendix D. 4.security area, **THEN STOP. Go to** IMC 0609, Appendix E.

- So Phase 1 of the SDP sends us to MC 0609, Appendix C (Occupational Radiation Safety Significance Determination Process)

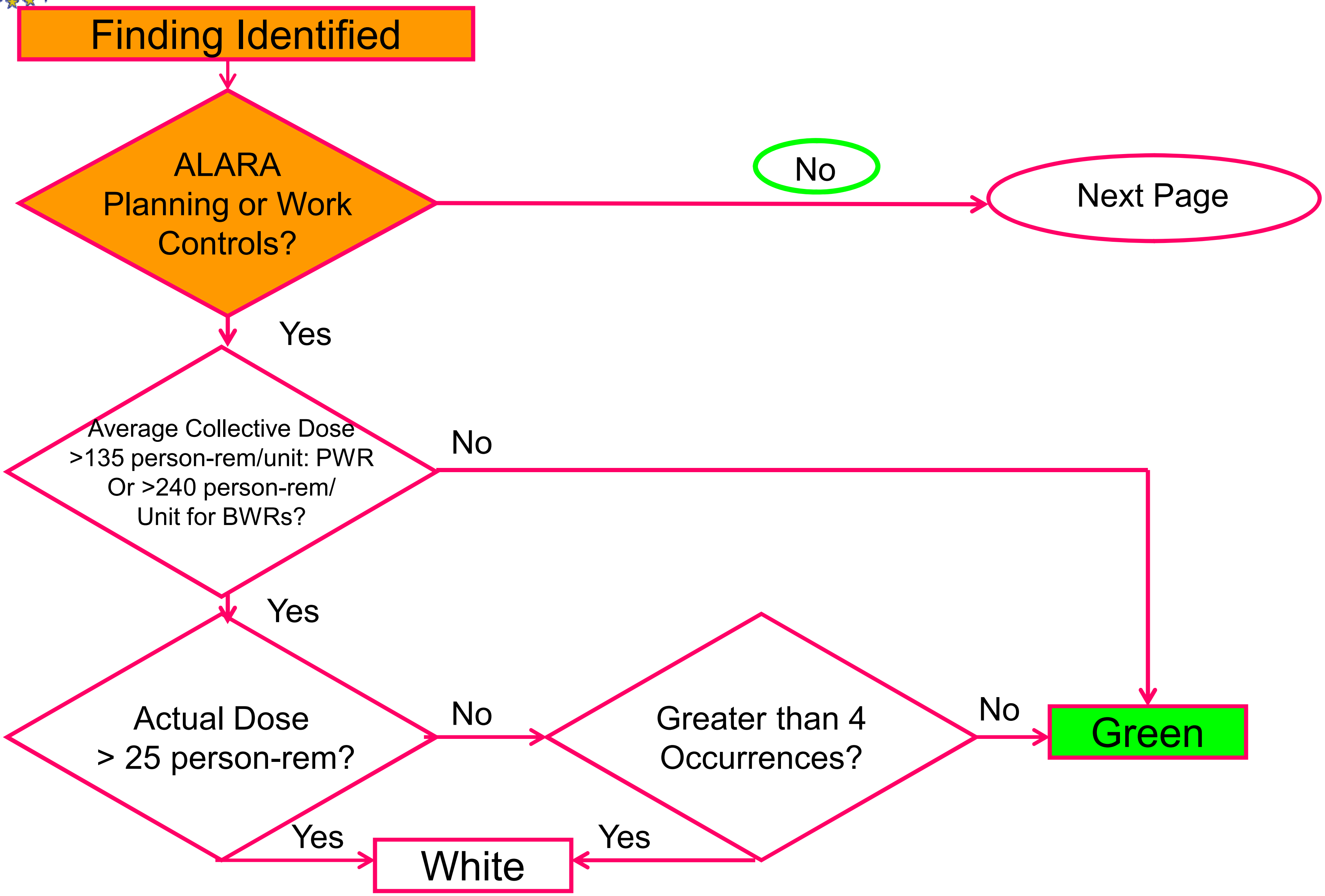
Determining Significance with MC 0609 Appendix C Flowchart



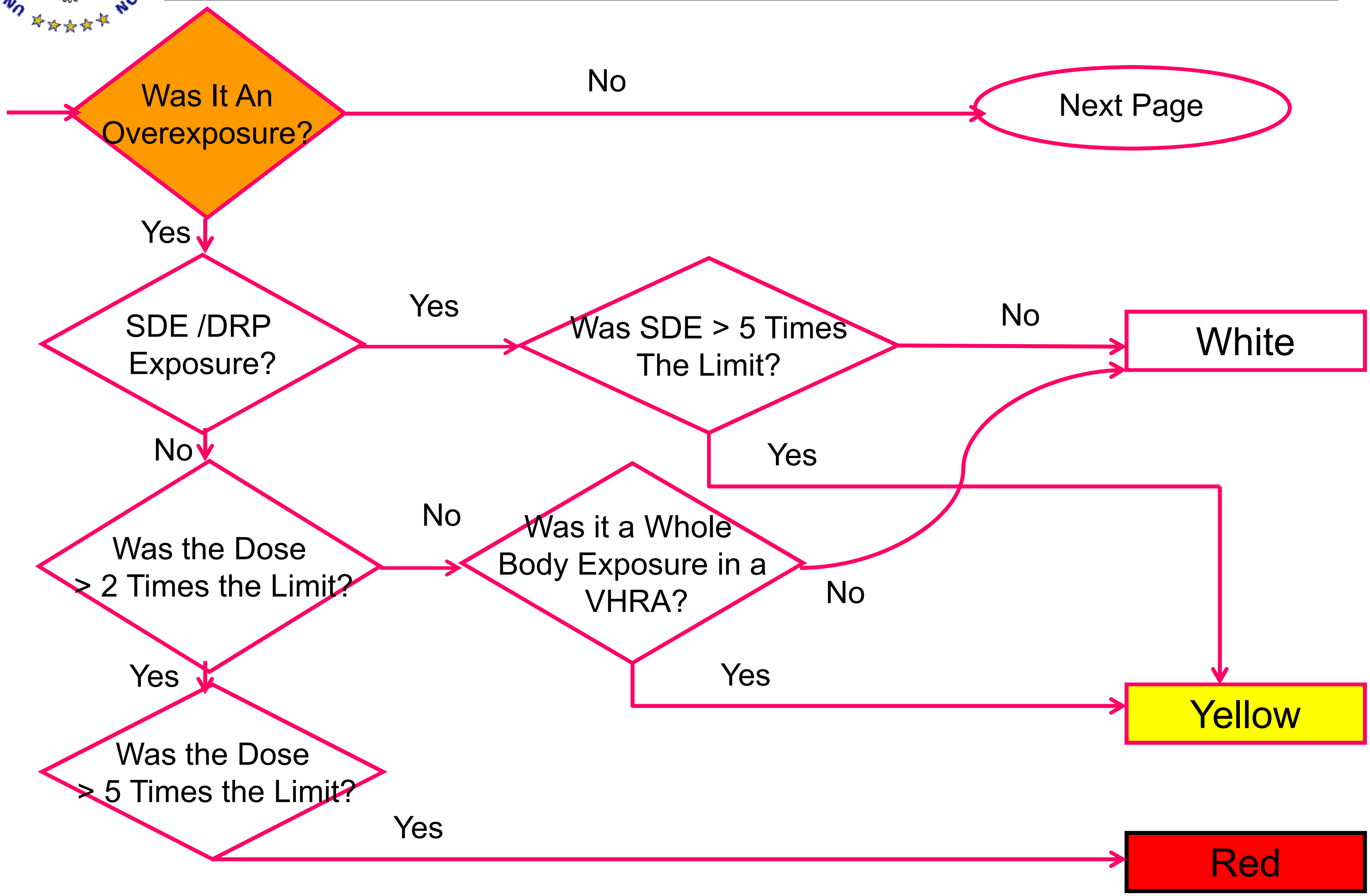
Determining Significance with MC 0609 Appendix C Flowchart



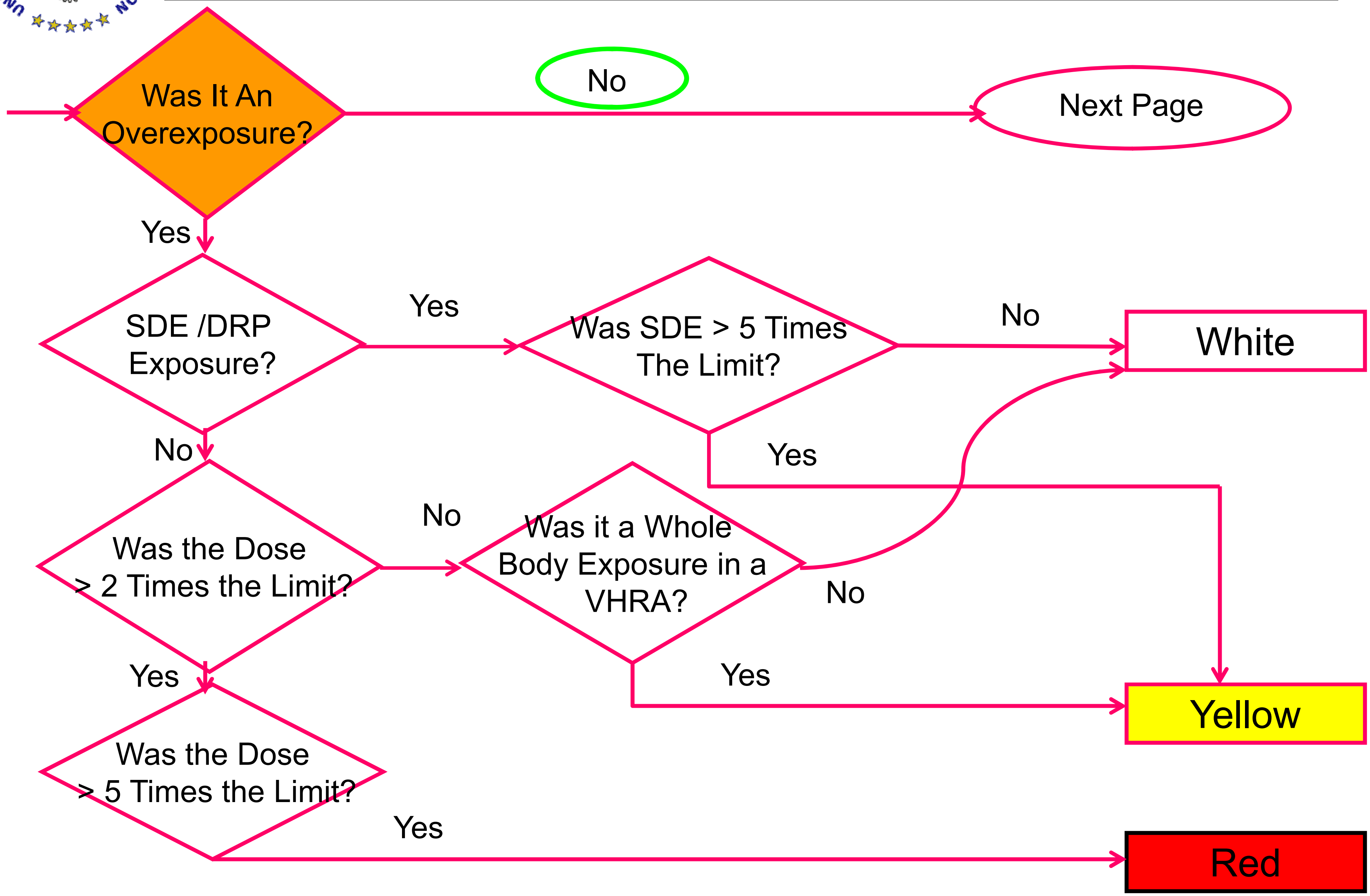
Determining Significance with MC 0609 Appendix C Flowchart



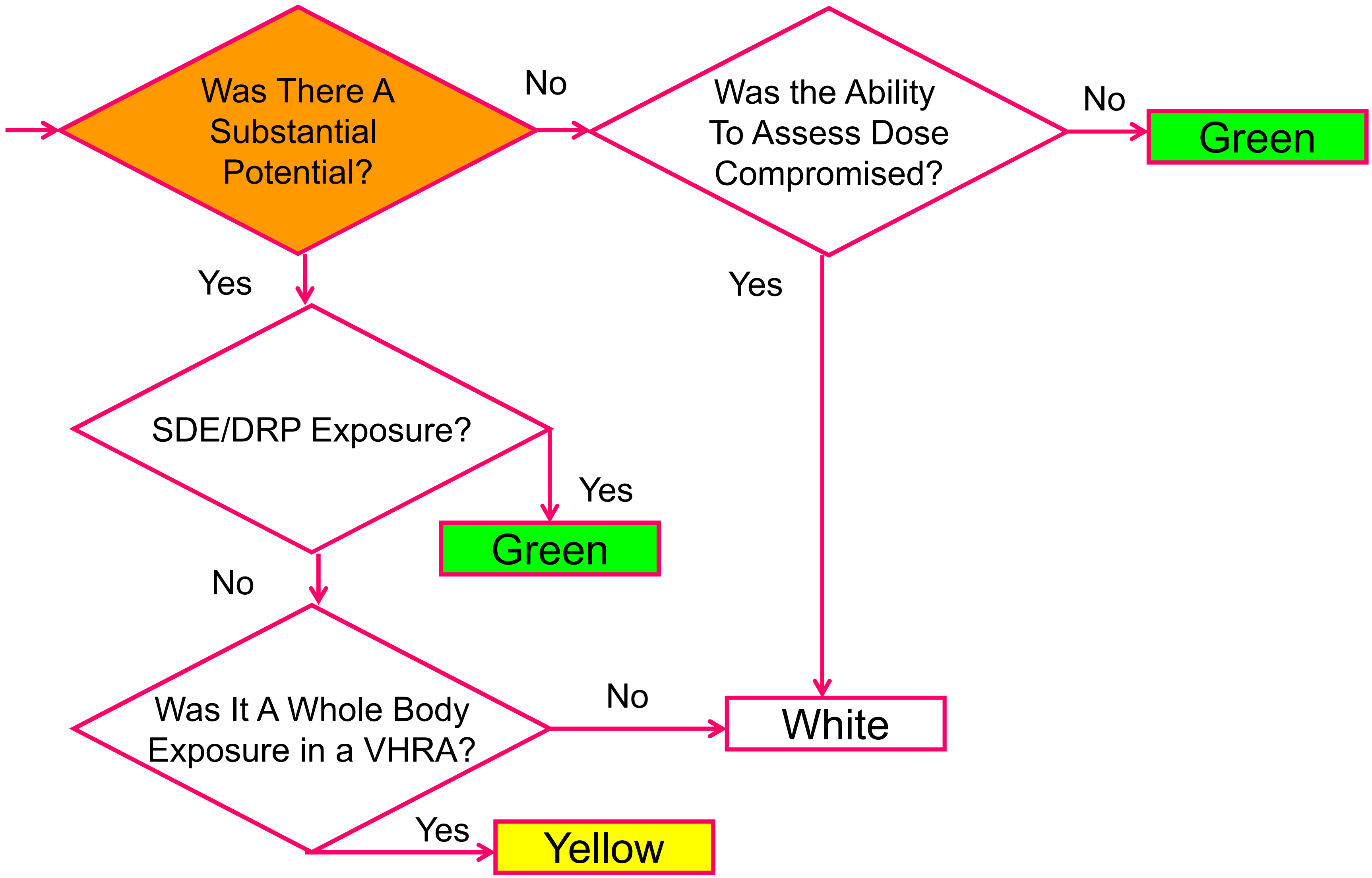
Determining Significance with MC 0609 Appendix C Flowchart



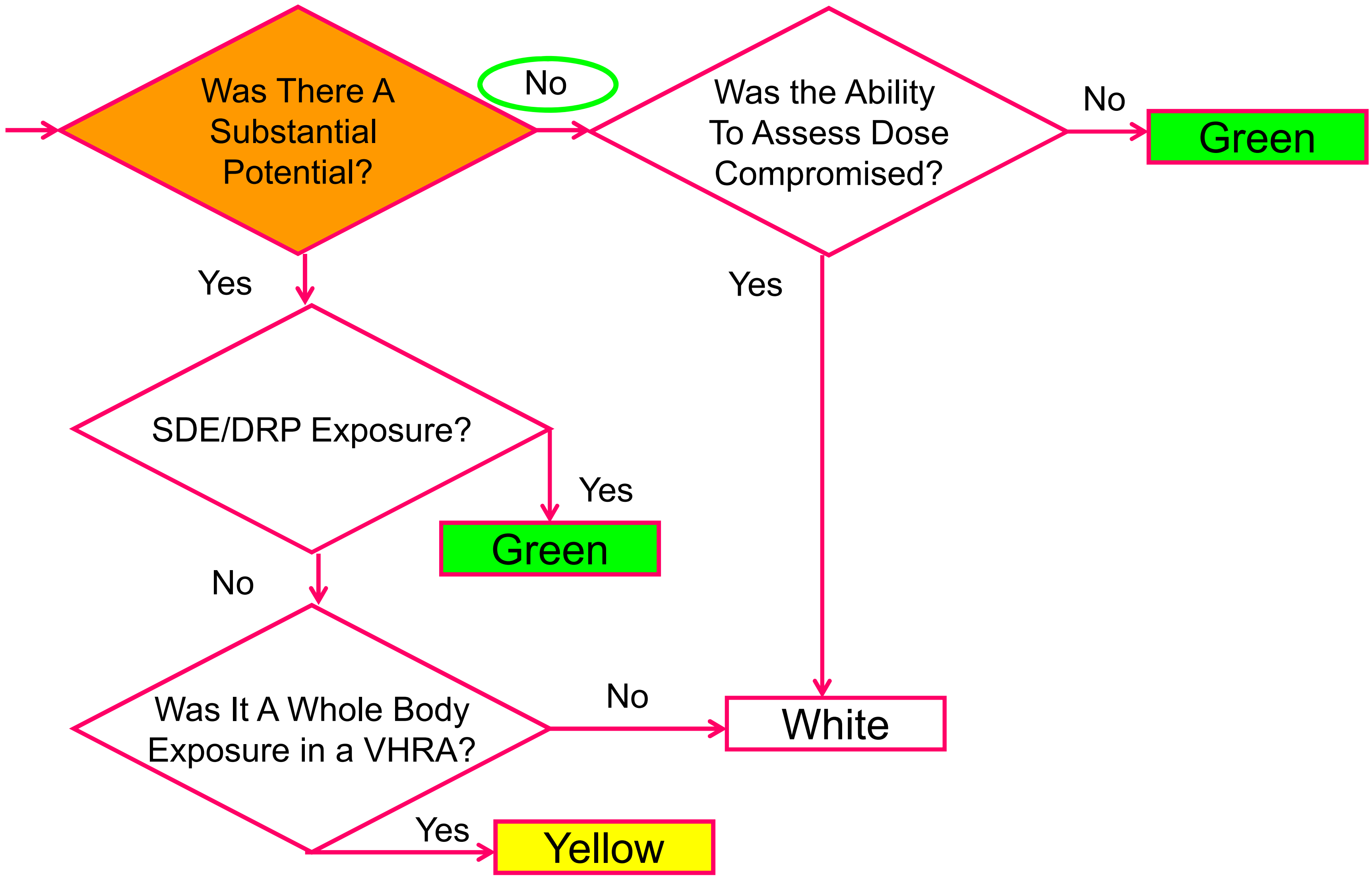
Determining Significance with MC 0609 Appendix C Flowchart



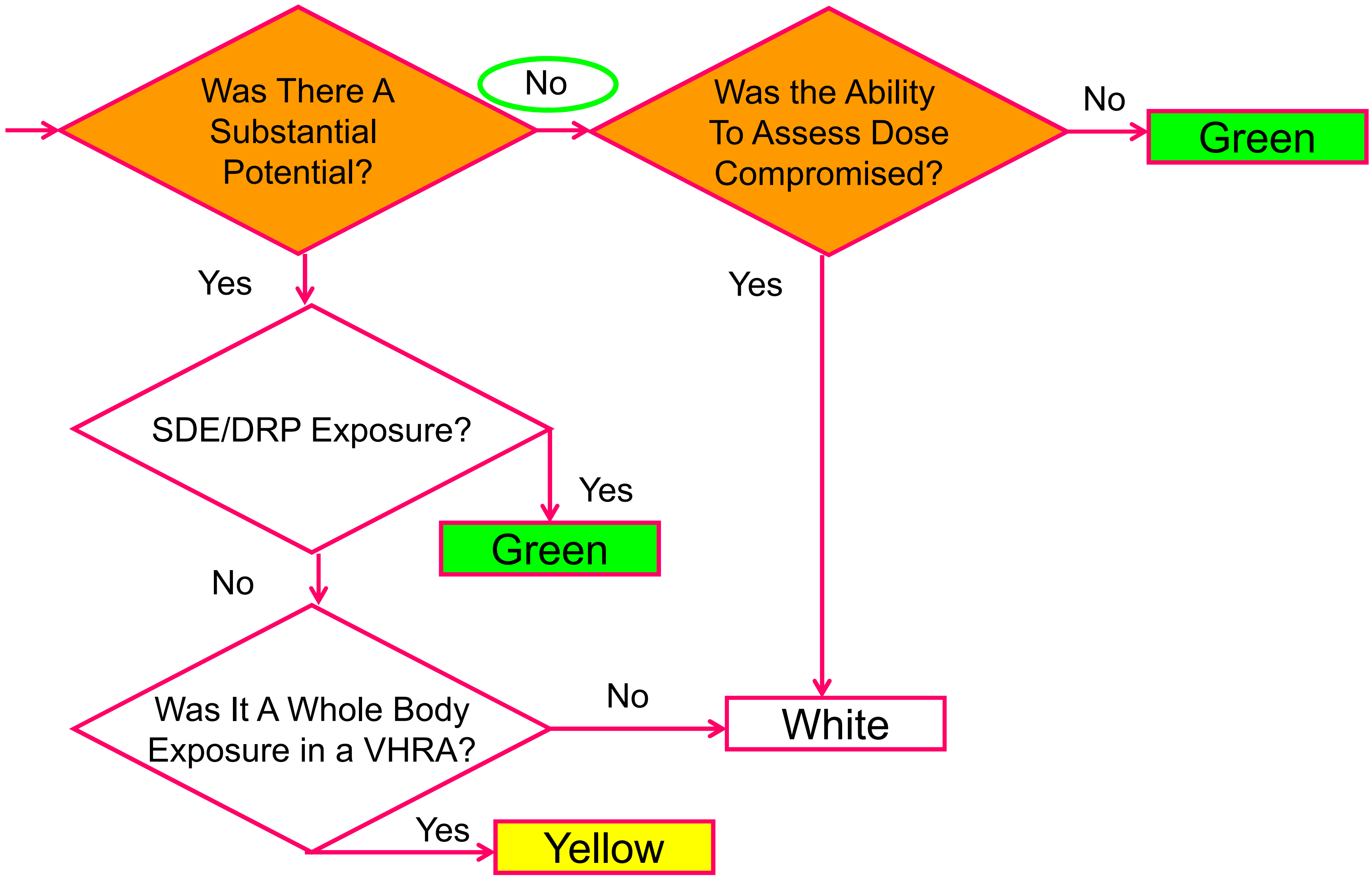
Determining Significance with MC 0609 Appendix C Flowchart



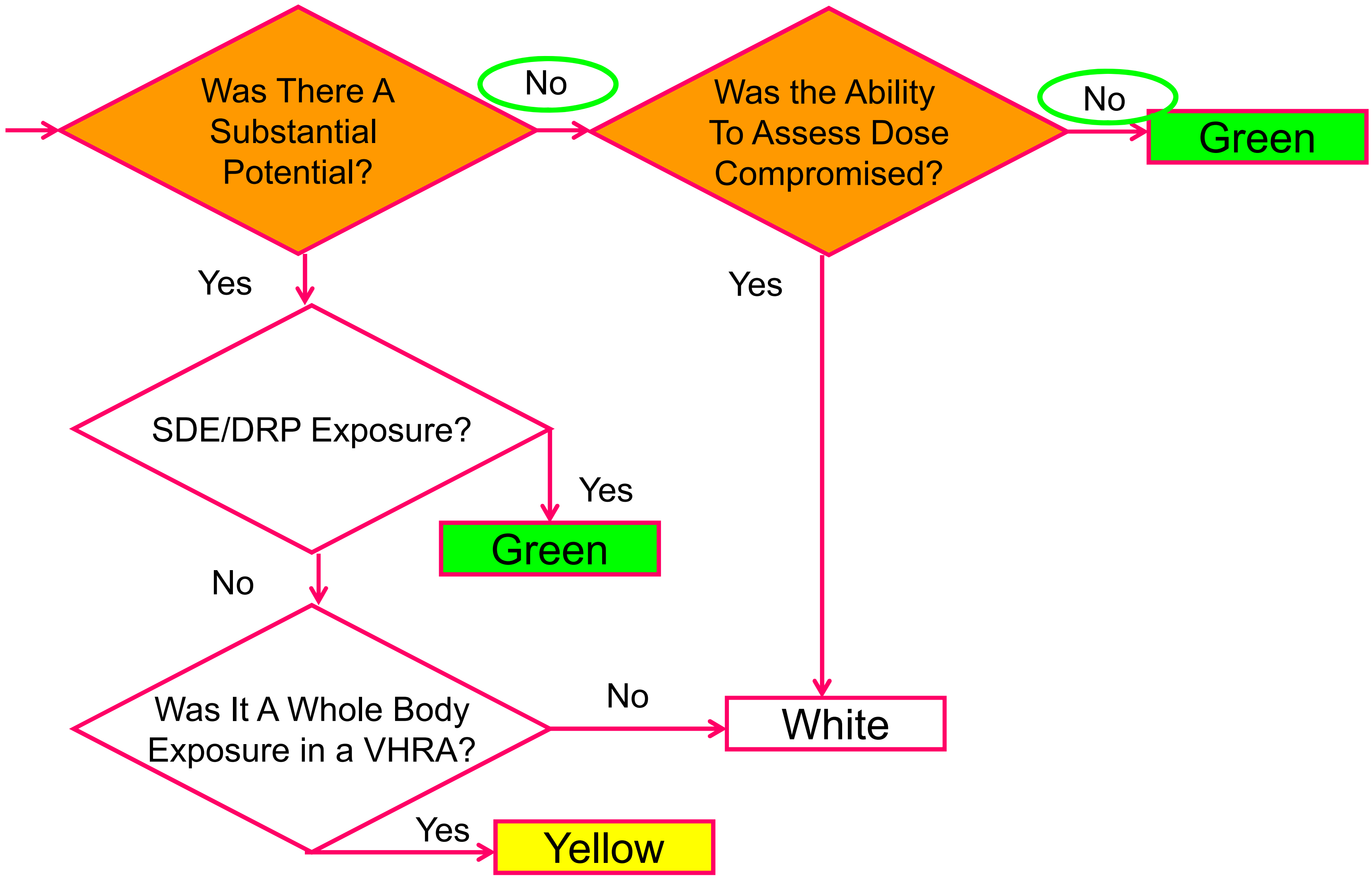
Determining Significance with MC 0609 Appendix C Flowchart



Determining Significance with MC 0609 Appendix C Flowchart



Determining Significance with MC 0609 Appendix C Flowchart



Risk-Informed SDPs

Recall - Before performing a Phase 2 SDP, you must have a finding and a completed Phase 1 SDP

Reactor Safety SDPs

- At-power Findings (MC 0609, Appendix A)
- Shutdown Findings (MC 0609, Appendix G)
- Significance Determination Process Using Qualitative Attributes (MC 0609, Appendix M)

Phase 2 (Risk-Informed Notebook)

- The Phase 2 SDP is based on a simplified PRA model.
- For all plants in the US, notebooks have been developed that are used to:
  - Identify the initiating event(s) impacted by the inspection finding
  - Identify the functional level accident sequence(s) affected
  - Identify the systems available to perform the critical safety functions
  - Determine the increase in core damage frequency of the finding
- The notebooks use order of magnitude values for unavailabilities of mitigating systems and initiating event frequencies
  - ➢ Let's take a look at a Phase 2 Notebook in detail

Phase 2 (Risk-Informed Notebook)

- Step 1 - Select Initiating Event Scenarios
- Step 2 - Estimate the Initiating Event Likelihood
- Step 3 - Determine the Remaining Mitigation Capability
- Step 4 - Estimate Risk Significance of Finding
- Step 5 - Screen for External Event Contribution
- Step 6 - Screen for LERF significance
    - ➢Let’s examine each step in detail

Source: MC0609 (Significance Determination Process), Appendix A, Attachment 1

Phase 2 (Risk-Informed Notebook)

- **Step 1 - Select Initiating Event Scenarios**
- Step 2 - Estimate the Initiating Event Likelihood
- Step 3 - Determine the Remaining Mitigation Capability
- Step 4 - Estimate Risk Significance of Finding
- Step 5 - Screen for External Event Contribution
- Step 6 - Screen for LERF significance

- Enter Table 2, with the equipment or safety function that was assumed to be impacted by the inspection finding.

  In this case, the finding is associated with the loss of one high head safety injection (HHSI) pump.
- Determine the initiating event worksheet(s) that must be evaluated.

Table 2 Initiators and System Dependency for Generic PWR Nuclear Power Plant

Affected Systems	Major Components	Support Systems	Initiating Event Scenarios
Engineered Safeguards Features Actuation System (ESFAS)	Three actuation trains, each with a load sequencer	120V vital AC, DC	All
Essential Cooling Water System (ECWS)	Three trains, each with one pump	4.16-kV, 480V (for MOVs), DC, ESFAS	All
High Head Safety Injection (HHSI) System	Two pumps (800 gpm @1275 psi, shutoff head = 1650 psid)	4.16-kV, 480V, DC, ESFAS, SI pump room cooling(8)	**All except LLOCA, ATWS, LODC**
Instrument Air (IA)	Two IA compressors (per unit). Back up is two station air compressors	Offsite power, BOP diesel(5)	LOIA
Low Head Safety Injection (LHSI) System	Three pumps	4.16-kV, 480V, DC, ESFAS, SI pump room cooling(8)	All except ATWS, LCCW, LODC
Main Steam Isolation System	For each steam generator: one MSIV [FW isolation and Control Valves (10)]	Offsite power and IA, DC, ESFAS	SGTR, MSLB
	For each steam generator: one PORV	480V, DC, 120V vital AC	All except LLOCA, and MLOCA
	For each steam generator: five safety relief valves	None	TPCS, LOOP, ATWS, LEAC

Phase 2 (Risk-Informed Notebook)

- Step 1 - Select Initiating Event Scenarios
- **Step 2 - Estimate the Initiating Event Likelihood**
- Step 3 - Determine the Remaining Mitigation Capability
- Step 4 - Estimate Risk Significance of Finding
- Step 5 - Screen for External Event Contribution
- Step 6 - Screen for LERF significance

- Enter Table 1 with exposure time associated with the finding. Assume > 30 days.
- Determine the initiating event likelihood (IEL) for each initiating event identified in Step 1.
- If the finding increases the likelihood of an initiating event, increase the IEL value in accordance with the SDP usage rules.

Table 1 - Categories of Initiating Events for Generic PWR Nuclear Power Plant

Row	Approximate Frequency	Example Event Type	Initiating Event Likelihood (IEL)		
I	> 1 per 1-10 yr	Loss of Power Conversion System (TPCS)	1	2	3
II	1 per 10-10^2 yr	Loss of offsite power (LOOP), Loss of Class 1E 125V DC Bus A or B (LODC)	2	3	4
III	1 per 10^2 - 10^3 yr	Steam Generator Tube Rupture (SGTR), Stuck open PORV/SRV (SORV), Small LOCA including RCP seal failures (SLOCA), Main Steam Line Break Outside Containment (MSLB)	3	4	5
IV	1 per 10^3 - 10^4 yr	Medium LOCA (MLOCA), LOOP with Loss of One Class 1E 4.16-kV Bus (LEAC)	4	5	6
V	1 per 10^4 - 10^5 yr	Large LOCA (LLOCA), Loss of Component Cooling Water (LCCW)	5	6	7
VI	less than 1 per 10^5 yr	ATWS[1)]	6	7	8
			>30 days	3-30 days	<3 days
			Exposure Time for Degraded Condition		

Phase 2 (Risk-Informed Notebook)

- Step 1 - Select Initiating Event Scenarios
- Step 2 - Estimate the Initiating Event Likelihood
- **Step 3 - Determine the Remaining Mitigation Capability**
- Step 4 - Estimate Risk Significance of Finding
- Step 5 - Screen for External Event Contribution
- Step 6 - Screen for LERF significance

- For each inspection notebook worksheet identified in Step 1, determine which safety functions were impacted by inspection finding.
- Circle affected functions within associated sequences on each worksheet that contain one or more of affected safety functions
- If the inspection finding increases the likelihood of an initiating event, circle all sequences on the worksheet for that particular event.

Phase 2 (Risk-Informed Notebook)

- Step 1 - Select Initiating Event Scenarios
- Step 2 - Estimate the Initiating Event Likelihood
- **Step 3 - Determine the Remaining Mitigation Capability (cont.)**
- Step 4 - Estimate Risk Significance of Finding
- Step 5 - Screen for External Event Contribution
- Step 6 - Screen for LERF significance

- Enter Table 5, "Remaining Mitigation Capability Credit," and determine the remaining mitigation capability credit for each of the functions affected.

- Determine if an operator could recover the affected function in time to mitigate the assumed initiating event. If the criteria for recovery credit are met, enter a recovery credit of 1.

Type of Remaining Mitigation Capability	Remaining Mitigation Capability Credit $X = -\log_{10}(\text{failure prob})$
Recovery of Failed Train Operator action to recover failed equipment that is capable of being recovered after an initiating event occurs. Action may take place either in the control room or outside the control room and is assumed to have a failure probability of approximately 0.1 when credited as "Remaining Mitigation Capability." Credit should be given only if the following criteria are satisfied: (1) sufficient time is available; (2) environmental conditions allow access, where needed; (3) procedures describing the appropriate operator actions exist; (4) training is conducted on the existing procedures under similar conditions; and (5) any equipment needed to perform these actions is available and ready for use.	1
1 Automatic Steam-Driven (ASD) Train A collection of associated equipment that includes a single turbine-driven component to provide 100% of a specified safety function. The probability of such a train being unavailable due to failure, test, or maintenance is assumed to be approximately 0.1 when credited as "Remaining Mitigation Capability."	1
1 Train A collection of associated equipment (e.g., pumps, valves, breakers, etc.) that together can provide 100% of a specified safety function. The probability of this equipment being unavailable due to failure, test, or maintenance is approximately 1E-2 when credited as "Remaining Mitigation Capability."	2
1 Multi-Train System A system comprised of two or more trains (as defined above) that are considered susceptible to common cause failure modes. The probability of this equipment being unavailable due to failure, test, or maintenance is approximately 1E-3 when credited as "Remaining Mitigation Capability," regardless of how many trains comprise the system.	3
2 Diverse Trains A system comprised of two trains (as defined above) that are not considered to be susceptible to common cause failure modes. The probability of this equipment being unavailable due to failure, test, or maintenance is approximately 1E-4 when credited as "Remaining Mitigation Capability."	4 (=2+2)
Operator Action Credit Major actions performed by operators during accident scenarios (e.g., primary heat removal using bleed and feed, etc.). These actions are credited using three categories of human error which represents a failure probability between 5E-3 and 5E-2, and Operator Action = 3 which represents a failure probability between 5E-4 and 5E-3. probabilities (HEPs). These categories are Operator Action = 1 which represents a failure probability between 5E-2 and 0.5, Operator Action = 2	1, 2, or 3

Source: MC 0609, App. A (Determining the Significance of Reactor Inspection Findings for At-Power Situations), dated 1/10/08

Table 3.1 SDP Worksheet for Generic PWR Nuclear Power Plant — Transients with Loss of PCS (TPCS) (1)

Safety Functions Needed:		Full Creditable Mitigation Capability for Each Safety Function:			
Secondary Heat Removal (AFW)		1/3 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train) with (1/1 SG PORV or 1/5 safety relief valves) per SG that is fed by AFW			
High Pressure Injection for FB (EIHP)		**1/2 HHSI pumps (1 multi-train system)**			
Primary Heat Removal, Feed/Bleed (FB)		2/2 pressurizer PORVs open for Feed/Bleed (operator action = 2) (2)			
High Pressure Recirculation (LPR)		1/3 LHSI trains and with associated 1/3 RHR heat exchangers or 2/6 RCFCs with cooling flow aligned to CCW (1 multi-train system)			
Circle Affected Functions		**IEL**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Recovery of Failed Train**	**Results**
1 TPCS - AFW - LPR (3) 1 + 4 + 3	8				
2 TPCS - AFW - FB (4) 1 + 4 + 2	7				
3 TPCS - AFW - EIHP (5) 1 + 4 + 3	8	1	4 + 2(indicates single train credit)	1	8

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

Operator open manual valve.

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Phase 2 (Risk-Informed Notebook)

- Step 1 - Select Initiating Event Scenarios
- Step 2 - Estimate the Initiating Event Likelihood
- Step 3 - Determine the Remaining Mitigation Capability
- **Step 4 - Estimate Risk Significance of Finding**
- Step 5 - Screen for External Event Contribution
- Step 6 - Screen for LERF significance

- Determine the sequence risk significance for each of the sequences circled in Step 3.
- Complete “Counting Rule Worksheet.” The result is the risk significance of the inspection finding based on the internal initiating events that lead to core damage.

Counting Rule Worksheet

- Add the affected sequences from the affected initiating events.
- Assume that you have:
- Five 9's,
- Seven 8's,
- One 7,
- One 6, and
- One 5.
- Total risk is 1E-5 (Yellow).

Counting Rule Worksheet

Step	Instructions		
(1)	Enter the number of sequences with a risk significance equal to 9.	(1)	5
(2)	Divide the result of Step (1) by 3 and round down.	(2)	1
(3)	Enter the number of sequences with a risk significance equal to 8.	(3)	7
(4)	Add the result of Step (3) to the result of Step (2).	(4)	8
(5)	Divide the result of Step (4) by 3 and round down.	(5)	2
(6)	Enter the number of sequences with a risk significance equal to 7.	(6)	1
(7)	Add the result of Step (6) to the result of Step (5).	(7)	3
(8)	Divide the result of Step (7) by 3 and round down.	(8)	1
(9)	Enter the number of sequences with a risk significance equal to 6.	(9)	1
(10)	Add the result of Step (9) to the result of Step (8).	(10)	2
(11)	Divide the result of Step (10) by 3 and round down.	(11)	0
(12)	Enter the number of sequences with a risk significance equal to 5.	(12)	1
(13)	Add the result of Step (12) to the result of Step (11).	(13)	1
(14)	Divide the result of Step (13) by 3 and round down.	(14)	0
(15)	Enter the number of sequences with a risk significance equal to 4.	(15)	0
(16)	Add the result of Step (15) to the result of Step (14).	(16)	0

- If the result of Step 16 is greater than zero, then the risk significance of the inspection finding is of high safety significance (RED).
- If the result of Step 13 is greater than zero, then the risk significance of the inspection finding is at least of substantial safety significance (YELLOW).
- If the result of Step 10 is greater than zero, then the risk significance of the inspection finding is at least of low to moderate safety significance (WHITE).
- If the result of Steps 10, 13, and 16 are zero, then the risk significance of the inspection finding is of very low safety significance (GREEN).

Phase 2 Result: ☐ **GREEN** ☐ **WHITE** ■ **YELLOW** ☐ **RED**

Phase 2 (Risk-Informed Notebook)

- Step 1 - Select Initiating Event Scenarios
- Step 2 - Estimate the Initiating Event Likelihood
- Step 3 - Determine the Remaining Mitigation Capability
- Step 4 - Estimate Risk Significance of Finding
- **Step 5 - Screen for External Event Contribution**
- Step 6 - Screen for LERF significance

- The plant-specific SDP Phase 2 worksheets do not currently include external initiating events contribution (e.g., fire, seismic).

- If the phase 2 SDP result for an inspection finding represents an increase in risk of greater or equal to 1E-7 per year, then an SRA or other NRC risk analyst performs an analysis to estimate the increase in risk due to external initiators.

Phase 2 (Risk-Informed Notebook)

- Step 1 - Select Initiating Event Scenarios
- Step 2 - Estimate the Initiating Event Likelihood
- Step 3 - Determine the Remaining Mitigation Capability
- Step 4 - Estimate Risk Significance of Finding
- Step 5 - Screen for External Event Contribution
- **Step 6 - Screen for LERF significance**

- If any of the sequence results are greater than or equal to 1E-7 per year and involve any of the sequence types listed below, then the finding is screened for LERF contribution using IMC 0609, Appendix H (this is an SRA responsibility).

- ISLOCA, transients (includes SBO scenarios), or small LOCAs for all reactor containment types
- ATWS for BWR Mark I and II reactor containment types
- SGTRs for all PWR reactor containment types

HRTD Human Resources Training & Development

Significance Determination Process for At-Power Inspection Findings

- Two Example Exercises

Phase 2 SDP-Exercise #1

- While performing a complete system walkdown of the high head safety injection (HHSI) system, an inspector identified that a normally locked open manual valve in the discharge flow path of one train was closed.

- The valve position for this valve was not indicated in the control room. This valve was also not in the flow path during quarterly surveillance testing of the system.

Phase 2 SDP-Exercise #1 (cont.)

- It was subsequently determined that the valve had been out of position since maintenance was last performed on the system ten months prior.
- The inspectors determined that the criteria for crediting operator recovery of the HHSI train were satisfied and that credit for recovery of the train was appropriate.
- The generic PWR risk-informed inspection notebook will be used for this exercise.

Table 2 Initiators and System Dependency for Generic PWR Nuclear Power Plant

Affected Systems	Major Components	Support Systems	Initiating Event Scenarios
Engineered Safeguards Features Actuation System (ESFAS)	Three actuation trains, each with a load sequencer	120V vital AC, DC	All
Essential Cooling Water System (ECWS)	Three trains, each with one pump	4.16-kV, 480V (for MOVs), DC, ESFAS	All
High Head Safety Injection (HHSI) System	Two pumps (800 gpm @1275 psi, shutoff head = 1650 psid)	4.16-kV, 480V, DC, ESFAS, SI pump room cooling(8)	**All except LLOCA, ATWS, LODC**
Instrument Air (IA)	Two IA compressors (per unit). Back up is two station air compressors	Offsite power, BOP diesel(5)	LOIA
Low Head Safety Injection (LHSI) System	Three pumps	4.16-kV, 480V, DC, ESFAS, SI pump room cooling(8)	All except ATWS, LCCW, LODC
Main Steam Isolation System	For each steam generator: one MSIV [FW isolation and Control Valves (10)]	Offsite power and IA, DC, ESFAS	SGTR, MSLB
	For each steam generator: one PORV	480V, DC, 120V vital AC	All except LLOCA, and MLOCA
	For each steam generator: five safety relief valves	None	TPCS, LOOP, ATWS, LEAC

Table 1 - Categories of Initiating Events for Generic PWR Nuclear Power Plant

Row	Approximate Frequency	Example Event Type	Initiating Event Likelihood (IEL)		
I	> 1 per 1-10 yr	Loss of Power Conversion System (TPCS)	1	2	3
II	1 per 10-10^2 yr	Loss of offsite power (LOOP), Loss of Class 1E 125V DC Bus A or B (LODC)	2	3	4
III	1 per 10^2 - 10^3 yr	Steam Generator Tube Rupture (SGTR), Stuck open PORV/SRV (SORV), Small LOCA including RCP seal failures (SLOCA), Main Steam Line Break Outside Containment (MSLB)	3	4	5
IV	1 per 10^3 - 10^4 yr	Medium LOCA (MLOCA), LOOP with Loss of One Class 1E 4.16-kV Bus (LEAC)	4	5	6
V	1 per 10^4 - 10^5 yr	Large LOCA (LLOCA), Loss of Component Cooling Water (LCCW)	5	6	7
VI	less than 1 per 10^5 yr	ATWS[1]	6	7	8
			>30 days	3-30 days	<3 days
			Exposure Time for Degraded Condition		

Table 3.1 SDP Worksheet for Generic PWR Nuclear Power Plant — Transients with Loss of PCS (TPCS) [1]

Safety Functions Needed:	Full Creditable Mitigation Capability for Each Safety Function:
Secondary Heat Removal (AFW)	1/3 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train) with (1/1 SG PORV or 1/5 safety relief valves) per SG that is fed by AFW
High Pressure Injection for FB (EIHP)	**1/2 HHSI pumps (1 multi-train system)**
Primary Heat Removal, Feed/Bleed (FB)	2/2 pressurizer PORVs open for Feed/Bleed (operator action = 2) [2]
High Pressure Recirculation (LPR)	1/3 LHSI trains and with associated 1/3 RHR heat exchangers or 2/6 RCFCs with cooling flow aligned to CCW (1 multi-train system)

Circle Affected Functions		IEL	Remaining Mitigation Capability Rating for Each Affected Sequence	Recovery of Failed Train	Results
1 TPCS - AFW - LPR (3) 1 + 4 + 3	8				
2 TPCS - AFW - FB (4) 1 + 4 + 2	7				
3 TPCS - AFW - **EIHP** (5) 1 + 4 + 3	8	**1**	**4 + 2(indicates single train credit)**	**1**	**8**

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

Operator open manual valve.

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Table 3.2 SDP Worksheet for Generic PWR Nuclear Power Plant — Small LOCA (SLOCA)

Safety Functions Needed:	Full Creditable Mitigation Capability for Each Safety Function:
Early Inventory, HP Injection (EIHP) Secondary Heat Removal (AFW) Primary Heat Removal, Feed/Bleed (FB) Low Pressure Injection (LPI) Low Pressure Recirculation (LPR)	**1/2 HHSI pumps (1 multi-train system)** 1/3 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train) 2/2 PORVs open for Feed/Bleed (operator action = 2) 1/3 LHSI pumps (1 multi-train system) 1/3 LHSI pumps with associated 1/3 RHR heat exchangers or 2/6 RCFCs with cooling flow from CCW (1 multi-train system)

Circle Affected Functions		IEL	Remaining Mitigation Capability Rating for Each Affected Sequence	Recovery of Failed Train	Results
1 SLOCA - LPR (2,4,7) 3 + 3	6				
2 SLOCA - AFW - FB (5) 3 + 4 + 2	9				
3 SLOCA - **EIHP** (8) 3 + 3	6	**3**	**2**	**1**	**6**

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

Operator open manual valve

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Table 3.3 SDP Worksheet for Generic PWR Nuclear Power Plant — Stuck Open PORV (SORV)(1)

Safety Functions Needed:	Full Creditable Mitigation Capability for Each Safety Function:
Isolation of Small LOCA (BLK)	The closure of the block valve associated with stuck open PORV (operator action = 2) (2)
Early Inventory, HP Injection (EIHP)	**1/2 HHSI pumps (1 multi-train system)**
Secondary Heat Removal (AFW)	1/3 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train)
Primary Heat Removal, Feed/Bleed (FB)	1/1 remaining PORVs open for Feed/Bleed (operator action = 2)
Low Pressure Injection (LPI)	1/3 LHSI pumps (1 multi-train system)
Low Pressure Recirculation (LPR)	1/3 LHSI pumps with associated 1/3 RHR heat exchangers or 2/6 RCFCs with cooling flow from CCW (1 multi-train system)

Circle Affected Functions		IEL	Remaining Mitigation Capability Rating for Each Affected Sequence	Recovery of Failed Train	Results
1 SORV - BLK - LPR (2, 4, 7) 3 + 2 + 3	8				
2 SORV - BLK - AFW - FB (5) 3 + 2 + 4 + 2	11				
3 SORV - BLK - **EIHP** (8) 3 + 2 + 3	8	3	2 + 2	1	8

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

Operator open manual valve

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Table 3.4 SDP Worksheet for Generic PWR Nuclear Power Plant — Medium LOCA (MLOCA)

Safety Functions Needed:	Full Creditable Mitigation Capability for Each Safety Function:
Early Inventory, HP Injection (EIHP) Low Pressure Injection (LPI) Low Pressure Recirculation (LPR)	**1 remaining HHSI train (1 single train system)** ½ remaining LHSI trains (1 multi-train system) ½ remaining LHSI trains with associated 1/3 RHR heat exchangers or 2/6 RCFCs with cooling flow from CCW (1 multi-train system)

Circle Affected Functions		IEL	Remaining Mitigation Capability Rating for Each Affected Sequence	Recovery of Failed Train	Results
1 MLOCA - LPR (2) 4 + 3	7				
2 MLOCA - LPI (3) 4 + 3	7				
3 MLOCA - EIHP (4) 4 + 2	6	4	0	1	5

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

Operator open manual valve

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Table 3.6 SDP Worksheet for Generic PWR Nuclear Power Plant — Loss of Offsite Power (LOOP)

Safety Functions Needed:	Full Creditable Mitigation Capability for Each Safety Function:
Emergency AC Power (EAC)	1/3 Standby Diesel Generators (1 multi-train system)
Secondary Heat Removal (TDAFW)	1/1 TDAFW pump (1 ASD train) with 1/5 safety relief valves per SG that is fed by AFW
Secondary Heat Removal (AFW)	1/3 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train)
Recovery of AC Power in < 2 hrs (REC2)	Recovery of AC power (operator action = 1) [1]
Recovery of AC power in < 5 hrs (REC5)	Recovery of AC power (operator action = 2) [3,4]
Early Inventory, HP Injection (EIHP)	**1/2 HHSI pumps (1 multi-train system)**
Primary Heat Removal, Feed/Bleed (FB)	2/2 pressurizer PORVs open for Feed/Bleed (operator action = 2)
Low Pressure Recirculation (LPR)	1/3 LHSI trains and with the associated 1/3 RHR heat exchangers or 2/6 RCFCs with cooling flow aligned to CCW (1 multi-train system)

Circle Affected Functions		IEL	Remaining Mitigation Capability Rating for Each Affected Sequence	Recovery of Failed Train	Results
1 LOOP - AFW - LPR (3) 2 + 4 + 3	9				
2 LOOP - AFW - FB (4) 2 + 4 + 2	8				
3 LOOP - AFW - **EIHP** (5) 2 + 4 + 3	9	2	4 + 2	1	9
4 LOOP - EAC - LPR (7, 11) 2 + 3 + 3 (AC Recovered)	8				
5 LOOP - EAC - **EIHP** (8, 13) 2 + 3 + 3	8	2	3 + 2	1	8
6 LOOP - EAC - REC5 (9) 2 + 3 + 2	7				
7 LOOP - EAC - TDAFW - FB (12) 2 + 3 + 1 + 2 (AC Recovered)	8				
8 LOOP - EAC - TDAFW - REC2 (14) 2 + 3 + 1 + 1	7				

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

Operator open manual valve

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Table 3.7 SDP Worksheet for Generic PWR Nuclear Power Plant -Steam Generator Tube Rupture (SGTR) (1)

Safety Functions Needed:	Full Creditable Mitigation Capability for Each Safety Function:
Secondary Heat Removal (AFW)	1/3 MDAFW trains (1 multi-train system) (2)
Early Inventory, HP Injection (EIHP)	**1/2 HHSI pumps (1 multi-train system)**
Primary Heat Removal, Feed/Bleed (FB)	2/2 pressurizer PORVs open for Feed/Bleed (operator action = 2)
Pressure Equalization (EQ)	Operator depressurizes RCS to less than setpoint of relief valve of SG using 1/3 pressurizer spray valves or 2/2 pressurizer PORVs (operator action = 2)
Isolation of Faulted SG (ISOL)	Operator isolates the faulted SG by closing 1/1 MSIV and associated Feedwater Isolation Valve (operator action = 2)
Cooldown and depressurization (DEPR)	Operator cools down and depressurizes the RCS using 1/4 SG PORVs or ½ pressurizer PORVs (operator action = 2)
Low Pressure Recirculation (LPR)	1/3 LHSI trains and with the associated 1/3 RHR heat exchangers or 2/6 RCFCs with cooling flow aligned to CCW (1 multi-train system)
Low Pressure Injection (SDC)	1/3 RHR trains (pumps & HXs) and ½ charging pumps (operator action = 3) (3)

Circle Affected Functions		IEL	Remaining Mitigation Capability Rating for Each Affected Sequence	Recovery of Failed Train	Results
1 SGTR - EQ - ISOL (3) 3 + 2 + 2	7				
2 SGTR - **EIHP** - SDC (5) 3 + 3 + 3	9	3	2 + 3	1	9
3 SGTR - **EIHP** - DEPR (6) 3 + 3 + 2	8	3	2 + 2	1	8
4 SGTR - **EIHP** - EQ (7) 3 + 3 + 2	8	3	2 + 2	1	8
5 SGTR - AFW - LPR (9) 3 + 3 + 3	9				
6 SGTR - AFW - ISOL (10) 3 + 3 + 2	8				
7 SGTR - AFW - FB (11) 3 + 3 + 2	8				
8 SGTR - AFW - **EIHP** (12) 3 + 3 + 3	9	3	3 + 2	1	9

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

Operator open manual valve

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Table 3.9 SDP Worksheet for Generic PWR Nuclear Power Plant — Main Steam Line Break Outside Containment (MSLB)

Safety Functions Needed:	Full Creditable Mitigation Capability for Each Safety Function:
MSLB Isolated (MSIV)[1]	3/4 MSIVs close [failure means at least 2 MSIVs failed] (1 multi-train)
High Pressure Injection (EIHP)	**1/2 HHSI pumps (1 multi-train system)**
Secondary Heat Removal (AFW)	1/3 MDAFW trains (1 multi-train system)
Feedwater valves close (FWVC)	Isolation of the feed to the SG whose MSIV did not close by auto trip of MFW pumps or isolation of MFW line, and operators close the valves feeding the SG from AFW, or trip of the AFW pump (operator action =2) [2]
Stop Injection (STIN)	Operators stop high pressure injection (operator action = 1) [3]
Primary Heat Removal, Feed/Bleed (FB)	2/2 pressurizer PORVs open for Feed/Bleed (operator action = 2)
High Pressure Recirculation (LPR)	1/3 LHSI pumps and with the associated 1/3 RHR heat exchangers or 2/6 RCFCs with cooling flow aligned to CCW (1 multi-train system)

Circle Affected Functions		IEL	Remaining Mitigation Capability Rating for Each Affected Sequence	Recovery of Failed Train	Results
1 MSLB - FWVC - STIN (3) 3 + 2 + 1	6				
2 MSLB - AFW - LPR (5) 3 + 3 + 3	9				
3 MSLB - AFW - FB (6) 3 + 3 + 2	8				
4 MSLB - **EIHP** - FWVC (8) 3 + 3 + 2	8	3	2 + 2	1	8
5 MSLB - **EIHP** - AFW (9) 3 + 3 + 3	9	3	2 + 3	1	9
6 MSLB - MSIV (10) 3 + 3	6				

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

Operator open manual valve

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Table 3.10 SDP Worksheet for Generic PWR Nuclear Power Plant — Loss of Component Cooling Water (LCCW) (1)

Safety Functions Needed:	Full Creditable Mitigation Capability for Each Safety Function:
RCP Trip (RCP)	Operator trips the RCPs to prevent a seal LOCA (operator action = 2) (2)
Seal Injection using PDP (PDP)	Operator starts PDP for seal injection (operator action = 2) (2)
High Pressure Injection (EIHP)	**1/2 HHSI trains (1 multi-train system)**
Secondary Heat Removal (AFW)	1/3 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train)

Circle Affected Functions		IEL	Remaining Mitigation Capability Rating for Each Affected Sequence	Recovery of Failed Train	Results
1 LCCW - AFW (2) **5 + 4**	**9**				
2 LCCW – EIHP (3) **5 + 3**	**8**	**5**	**2**	**1**	**8**
3 LCCW - RCP (4) **5 + 2**	**7**				

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

Operator open manual valve

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Table 3.12 SDP Worksheet for Generic PWR Nuclear Power Plant — LOOP and Loss of One Class 1E 4.16-kV Bus (LEAC)(1)

Safety Functions Needed:	Full Creditable Mitigation Capability for Each Safety Function:
PORV Recloses (PORV)	2/2 Pressurizer PORVs reclose after opening during transient (1 train)
Secondary Heat Removal (AFW)	½ MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train) with 1/5 safety relief valve per SG that is fed by AFW
High Pressure Injection for FB (EIHP)	**1 HHSI pump (1 train)**
Primary Heat Removal, Feed/Bleed (FB)	2/2 pressurizer PORVs open for Feed/Bleed (operator action = 2)
Low Pressure Recirculation (LPR)	½ LHSI pumps with (associated ½ RHR heat exchangers or 2/4 RCFCs with cooling flow aligned to CCW) (1 multi-train system)

Circle Affected Functions		IEL	Remaining Mitigation Capability Rating for Each Affected Sequence	Recovery of Failed Train	Results
1 LEAC - AFW - LPR (3) 4 + 4 + 3	11				
2 LEAC - AFW - FB (4) 4 + 4 + 2	10				
3 LEAC - AFW - EIHP (5) 4 + 4 + 2	10	4	4 + 0	1	9
4 LEAC - PORV - LPR (7) 4 + 2 + 3	9				
5 LEAC - PORV - EIHP (8) 4 + 2 + 2	8	4	2 +0	1	7
6 LEAC - PORV - AFW (9) 4 + 2 + 4	10				

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

Operator open manual valve

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Counting Rule Worksheet			
Step	**Instructions**		
(1)	Enter the number of sequences with a risk significance equal to 9.	(1)	5
(2)	Divide the result of Step (1) by 3 and round down.	(2)	1
(3)	Enter the number of sequences with a risk significance equal to 8.	(3)	7
(4)	Add the result of Step (3) to the result of Step (2).	(4)	8
(5)	Divide the result of Step (4) by 3 and round down.	(5)	2
(6)	Enter the number of sequences with a risk significance equal to 7.	(6)	1
(7)	Add the result of Step (6) to the result of Step (5).	(7)	3
(8)	Divide the result of Step (7) by 3 and round down.	(8)	1
(9)	Enter the number of sequences with a risk significance equal to 6.	(9)	1
(10)	Add the result of Step (9) to the result of Step (8).	(10)	2
(11)	Divide the result of Step (10) by 3 and round down.	(11)	0
(12)	Enter the number of sequences with a risk significance equal to 5.	(12)	1
(13)	Add the result of Step (12) to the result of Step (11).	(13)	1
(14)	Divide the result of Step (13) by 3 and round down.	(14)	0
(15)	Enter the number of sequences with a risk significance equal to 4.	(15)	0
(16)	Add the result of Step (15) to the result of Step (14).	(16)	0

- If the result of Step 16 is greater than zero, then the risk significance of the inspection finding is of high safety significance (RED).
- If the result of Step 13 is greater than zero, then the risk significance of the inspection finding is at least of substantial safety significance (YELLOW).
- If the result of Step 10 is greater than zero, then the risk significance of the inspection finding is at least of low to moderate safety significance (WHITE).
- If the result of Steps 10, 13, and 16 are zero, then the risk significance of the inspection finding is of very low safety significance (GREEN).

Phase 2 Result: ☐ **GREEN** ☐ **WHITE** ■ **YELLOW** ☐ **RED**

Phase 2 SDP- Exercise #2

- Consider a hypothetical inspection finding that involves the failure of the licensee to identify a 180 degree circumferential crack on a weld on a 2-inch line connected to the reactor coolant system.
- Evidence of the crack remained unidentified for four months.
- The inspectors determined that a small loss of coolant accident would result if this weld failed.

Phase 2 SDP-Exercise #2 (cont.)

- Assume that recovery credit is not appropriate for the circumstances surrounding this hypothetical finding.

- The generic PWR risk-informed inspection notebook will be used for this exercise.

Notebook Usage Rules

Rule 1.2 - Finding (Not Involving a Support System) that Increases the Likelihood of an Initiating Event

If the amount of increase in the frequency of the initiating event due to the inspection finding is not known, increase the IEL for the applicable initiating event by one order of magnitude. If specific information exists that indicates the IEL should be increased by more than one order of magnitude, consult with the regional SRA to determine the appropriate IEL.

Source: MC0609, Appendix A, Attach 2 (Site Specific Risk-Informed Inspection Notebook Usage Rules), dated 1/10/08

Table 1 - Categories of Initiating Events for Generic BWR Nuclear Power Plant

Row	Approximate Frequency	Example Event Type	Initiating Event Likelihood (IEL)		
I	> 1 per 1-10 yr	Loss of Power Conversion System (TPCS)	1	2	3
II	1 per 10-10^2 yr	Loss of offsite power (LOOP) Loss of Class 1E 125V DC Bus A or B (LODC)	2	3	4
III	1 per 10^2 - 10^3 yr	Steam Generator Tube Rupture (SGTR), Stuck open PORV/SRV (SORV), Small LOCA including RCP seal failures **(SLOCA)**, Main Steam Line Break Outside Containment (MSLB)	3	4	5
IV	1 per 10^3 - 10^4 yr	Medium LOCA (MLOCA), LOOP with Loss of One Class 1E 4.16-kV Bus (LEAC)	4	5	6
V	1 per 10^4 - 10^5 yr	Large LOCA (LLOCA), Loss of Component Cooling Water (LCCW)	5	6	7
VI	less than 1 per 10^5 yr	ATWS (1)	6	7	8
			> 30 days	3-30 days	< 3 days
			Exposure Time for Degraded Condition		

Notes:

Table 3.2 SDP Worksheet for Generic PWR Nuclear Power Plant — Small LOCA (SLOCA)

Safety Functions Needed:	**Full Creditable Mitigation Capability for Each Safety Function:**
Early Inventory, HP Injection (EIHP)	1/3 HHSI pumps (1 multi-train system)
Secondary Heat Removal (AFW)	1/3 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train)
Primary Heat Removal, Feed/Bleed (FB)	2/2 PORVs open for Feed/Bleed (operator action = 2)
Low Pressure Injection (LPI)	1/3 LHSI pumps (1 multi-train system)
Low Pressure Recirculation (LPR)	1/3 LHSI pumps with associated 1/3 RHR heat exchangers or 2/6 RCFCs with cooling flow from CCW (1 multi-train system)

Circle Affected Functions		**IEL**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Recovery Credit**	**Results**
1 **SLOCA - LPR** (2,4,7) 3 + 3	6	2	3	0	5
2 **SLOCA - AFW - FB** (5) 3 + 4 + 2	9	2	4 + 2	0	8
3 **SLOCA - EIHP** (8) 3 + 3	6	2	3	0	5

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Counting Rule Worksheet

Step	Instructions		
(1)	Enter the number of sequences with a risk significance equal to 9.	(1)	0
(2)	Divide the result of Step (1) by 3 and round down.	(2)	**0**
(3)	Enter the number of sequences with a risk significance equal to 8.	(3)	**1**
(4)	Add the result of Step (3) to the result of Step (2).	(4)	**1**
(5)	Divide the result of Step (4) by 3 and round down.	(5)	**0**
(6)	Enter the number of sequences with a risk significance equal to 7.	(6)	**0**
(7)	Add the result of Step (6) to the result of Step (5).	(7)	**0**
(8)	Divide the result of Step (7) by 3 and round down.	(8)	**0**
(9)	Enter the number of sequences with a risk significance equal to 6.	(9)	0
(10)	Add the result of Step (9) to the result of Step (8).	(10)	**0**
(11)	Divide the result of Step (10) by 3 and round down.	(11)	**0**
(12)	Enter the number of sequences with a risk significance equal to 5.	(12)	**2**
(13)	Add the result of Step (12) to the result of Step (11).	(13)	**2**
(14)	Divide the result of Step (13) by 3 and round down.	(14)	**0**
(15)	Enter the number of sequences with a risk significance equal to 4.	(15)	**0**
(16)	Add the result of Step (15) to the result of Step (14).	(16)	**0**

- If the result of Step 16 is greater than zero, then the risk significance of the inspection finding is of high safety significance (RED).
- If the result of Step 13 is greater than zero, then the risk significance of the inspection finding is at least of substantial safety significance (YELLOW).
- If the result of Step 10 is greater than zero, then the risk significance of the inspection finding is at least of low to moderate safety significance (WHITE).
- If the result of Steps 10, 13, and 16 are zero, then the risk significance of the inspection finding is of very low safety significance (GREEN).

Phase 2 Result: ☐ GREEN ☐ WHITE ■ **YELLOW** ☐ **RED**

Risk-Informed SDP Exercise, Phase 1 and 2

Deficient Reactor Building Coatings and their impact on the Emergency Sump using SDP Phase 1 and 2

http://www.nrc.gov/reactors/operating/ops-experience/pwr-sump-performance/function-containment-sump.html

Workshop #6
SDP Phase 1 and 2

Phase 3 (Senior Reactor Analyst)

- Risk Significance Estimation Using Risk Basis That Departs from the Phase 1 or 2 Process
  - If necessary, Phase 3 will refine or modify, with sufficient justification, the earlier screening results from Phases 1 and 2.
  - In addition, Phase 3 will address findings that cannot be evaluated using the Phase 2 process (e.g., external event contributors).
  - Phase 3 analysis will use appropriate PRA techniques and rely on the expertise of NRC risk analysts.

Phase 3 (Senior Reactor Analyst)

A Phase 3 analysis includes:

- Phase 1 and 2 results
- PRA tools used for the Phase 3 assessment
- Affected accident sequences
- Influential assumptions
- Sensitivity of results to each assumption
- Contributions of greatest uncertainty factors

Risk effects of Large Early Release Frequency, internal flooding and external events are also evaluated.

Phase 3 analysis is documented in a Significance and Enforcement Review Panel (SERP) package and presented to SERP members for a preliminary decision.

SERP, MC 0609 Attach 1

- SERP decision presented to licensee in a Preliminary Determination Letter (a.k.a, Choice Letter)

- Licensee has choice to:
  - respond by letter
  - attend a Regulatory Conference
  - accept preliminary result

- If preliminary result is changed due to new information or insights, SERP reconvenes and determines final significance of finding

- Final Significance Determination letter (and NOV) sent to licensee describing finding and regulatory significance

Real World
SDP and SERP Packages

SDP Recap

- The SDP process provides a structured way to evaluate significance
- Areas requiring personal judgment and subjectivity are minimized
- Two different people looking at the same issue should reach similar conclusions
  - Where they differ will be readily apparent and will serve to focus discussions
- Each cornerstone has SDPs
  - Important to read the material – flowcharts are helpful, but not comprehensive
- Cross-cutting aspects are important considerations to enable comprehensive assessment of licensee performance.

SDP Risk Thresholds

CONCEPTUAL MODEL FOR EVALUATING LICENSEE PERFORMANCE
GREEN - *Licensee Response Band* Cornerstone objectives fully met. Nominal risk with nominal deviation from expected performance. Very low safety significance. ΔCDF < E-6
WHITE - *Increased Regulatory Response Band* Cornerstone objectives met with *minimal* reduction in safety margin. Low to Moderate Safety Significance. E-6 < ΔCDF < E-5
YELLOW - ***Required Regulatory Response Band*** Cornerstone objectives met with *significant* reduction in safety margin. Substantial Safety Significance. E-5 < ΔCDF < E-4
RED - ***Significant Regulatory Response Band*** Plant performance represents an unacceptable loss of safety margin. It should be noted that should licensee's performance result in a PI reaching the Red Band, margin would still exist before an undue risk to public health and safety would be presented. High Safety Significance. ΔCDF>10-4

Source: MC 0308 Reactor Oversight Process Basis Document) dated 11/8/07

Saphire 8

SAPHIRE is a probabilistic risk and reliability assessment software tool. SAPHIRE stands for *Systems Analysis Programs for Hands-on Integrated Reliability Evaluations*. The system was developed for the U.S. Nuclear Regulatory Commission (NRC) by the Idaho National Laboratory.

Although MC 0609 has not been updated, Saphire 8 will replace the Phase 2 notebooks and field inspectors will be expected to use Saphire 8 with the guidance of an SRA.

➢Let's explore the use of Saphire 8 using the deficient RB Coatings example

Risk-Informed SDP Exercise, Saphire 8

Deficient Reactor Building Coatings and their impact on the Emergency Sump using Saphire 8

Workshop #7


saphire.inl.gov
8
SAPHIRE
Systems Analysis Programs for Hands-on Integrated Reliability Evaluations

Review - Learning Objectives

- Explain how the SDP is used to risk- inform the inspection process
- Perform an issue screening and characterize its significance using Phase 1 of the SDP
- Estimate the risk significance of a finding using deterministic SDP
- Estimate the risk significance of a finding using Phase 2 of the SDP
- Estimate the risk significance of a finding using Saphire 8 Risk Software

That's all folks!

It's been a pleasure!

Before you leave:

- Please straighten up your area
- Return any building access cards

Please, complete the on-line evaluation of this course when prompted by iLearn (via email)