NRC INSPECTION MANUAL

APOB

INSPECTION MANUAL CHAPTER 0609 APPENDIX A

THE SIGNIFICANCE DETERMINATION PROCESS
FOR FINDINGS AT-POWER

Effective Date: 01/01/2021

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0609A-01 PURPOSE

The Significance Determination Process (SDP) described in this Appendix is designed to provide staff and management with a simplified framework and associated guidance for use in screening at-power findings. This Appendix aids the user in determining if a finding has a very low safety significance (screens to Green) or directs the user to other applicable SDP appendices or to perform a detailed risk evaluation.

This SDP is applicable to at-power findings within the Initiating Events, Mitigating Systems, and Barrier Integrity Cornerstones. The SDP described in this Appendix is implemented by direction from Inspection Manual Chapter (IMC) 0609, Attachment 4, “Initial Characterization of Findings.”

0609A-02 BACKGROUND

Over the years, maintaining the pre-solved tables and risk-informed notebooks from IMC 0609, Appendix A proved to be a challenging task. As plants implemented equipment modifications and associated revisions to the plant risk model, the accuracy of the pre-solved tables and risk-informed notebooks began to degrade. Instead of separately maintaining and updating the plant-specific pre-solved tables and risk-informed notebooks, the agency decided to transition to a software-based system called SAPHIRE (Systems Analysis Programs for Hands-on Integrated Reliability Evaluations). Using SAPHIRE, a user can perform analyses on a regularly maintained site-specific Standardized Plant Assessment Risk (SPAR) model. Updating site-specific SPAR models provides an efficient and effective infrastructure that facilitates risk model fidelity. For legacy, reference, and knowledge transfer purposes, the pre-solved tables, risk-informed notebooks, and associated ROP guidance documents have been archived.

In the transition from the pre-solved tables and risk-informed notebooks to SAPHIRE and the site-specific SPAR models, it is important to note process differences. The pre-solved tables and risk-informed notebooks, by process, provided a second layer of screening and an estimation of the risk impact of the finding. In lieu of the pre-solved tables and risk-informed notebooks, the SDP Workspace, a module within each SPAR model, was developed. The SDP Workspace performs a delta CDF calculation similar in many respects to the risk estimate performed by use of the risk-informed notebooks. However, use of SDP workspace is no longer intended to provide a prescriptive additional layer of screening beyond that which is outlined in Section 0609A-04, “Screening,” of this IMC. Rather, the SDP Workspace is one of many tools the inspection staff and SRAs can utilize to support a detailed risk evaluation (see Section 0609A-05 of this IMC, “Detailed Risk Evaluation,” for more details).

0609A-03 GUIDANCE

This appendix is divided into two functional parts. The first part is a screening tool that uses a series of logic questions to determine whether or not the finding can be characterized as having very low safety significance (i.e., Green) and preclude a more detailed risk evaluation. The second part provides guidance in determining the risk significance of a finding that did not screen to Green in part one.

0609A-04 SCREENING

The screening questions are categorized by cornerstone. As such, there is one set of screening questions for Initiating Events, one for Mitigating Systems, and one for Barrier Integrity (Exhibits 1, 2, and 3 respectively). If more than one cornerstone is affected, the screening questions in all the affected cornerstones apply. In addition, under each cornerstone the screening questions are categorized into sub-sections, so a finding and associated degraded condition might be applicable to more than one sub-section. Typically, the inspection staff completes the screening process with support from the regional SRAs, as needed. The screening questions cover a wide range of instances and scenarios but are not intended to be all inclusive. Therefore, if the inspection staff and/or SRA do not agree with the screening results, other risk tools (e.g., the SDP Workspace) and guidance provided in Section 0609A-05, "Detailed Risk Evaluation," of this IMC can be used to confirm or challenge the screening results. The screening process also directs the user to other applicable SDP appendices as needed (similar to Table 3 of IMC 0609, Attachment 4).

The screening logic questions are designed to systematically determine whether a degraded condition(s) resulting from a finding is of very low safety significance (i.e., Green) or not. If all the logic questions under the applicable cornerstone(s) do not apply, then the finding is screened as Green and the risk evaluation is complete (assuming that there are no unique technical considerations that need to be assessed). Conversely, if any one of the logic questions under a specific cornerstone is applicable to the degraded condition(s), the finding cannot be screened as Green and further risk evaluation is warranted.

In applying the SDP screening questions, inspectors are evaluating the degraded condition in the plant, for which the performance deficiency has been determined to be a proximate cause. In defining the degraded plant condition, inspectors will need to use their judgment, in a reasonable and realistic manner, consistent with previous similar findings. Inspectors are not required to have proof of assumptions used in the SDP but must have a reasonable technical basis. See IMC 0308, Attachment 3 for additional information on the basis of the SDP.

The duration of a plant degraded condition, i.e., the exposure time, is often an important assumption in the SDP and is specifically used to assess the Mitigating Systems screening questions. The exposure time is the duration or time period that the failed or degraded SSC is reasonably known to have existed. The exposure time used in the SDP may not be equivalent to that used for reportability or operability. Inspectors should consult with an SRA if there are questions about determining the exposure time for a finding. The exposure time is often evaluated against the duration of the Technical Specification (TS) allowed outage times, as these periods are generally known to represent configurations of very low risk significance.

Also note that as a risk-informed tool, the at-power SDP is focused on initiating events, mitigating system functions, and barrier integrity functions used in probabilistic risk assessments (PRAs), which may differ from design basis transients and accidents as discussed in the Updated Final Safety Analysis Report (UFSAR).

04.01 <u>Initiating Events (Exhibit 1)</u>

The Initiating Events screening questions are categorized into five sub-sections titled Loss of Coolant Accident (LOCA) Initiators, Transient Initiators, Support System Initiators, Steam Generator Tube Rupture (SGTR), and External Event Initiators. Below is additional guidance to support answering the screening questions for each sub-section:

a. LOCA Initiators – Considers small, medium, and large LOCA initiating events. For SDP purposes, a small LOCA is defined as a steam or liquid break in the reactor coolant system (RCS), other than a SGTR, that exceeds the ability to makeup using normal charging (PWR) or control rod drive (BWR) pump flow. Normal makeup flow may include control room actions to start a standby pump or minimize letdown flow, if appropriate for the situation.

b. Transient Initiators – A transient initiator is an event that results in a reactor trip or scram. Some examples of transients are loss of main feedwater, loss of condenser heat sink, and loss of offsite power (LOOP) events.

c. Support System Initiators – Support systems include SSCs needed to start, operate, or control a front-line system, where the front-line system fulfills a critical safety function. Support system initiating events are a subcategory of initiating events where the failure not only causes a loss or challenge to a safety function, but also adversely affects one or more systems needed to respond to shutdown of the reactor. These events not only trigger sequences of events that challenge plant control and safety systems whose failure could potentially lead to core damage or large early release, they also fail all or part of those systems used for mitigation. Examples of support system initiators include loss of service water, loss of vital AC/DC power buses, loss of cooling water and loss of instrument air events. Site-specific support system initiators can be identified in the Plant Risk Information eBook (PRIB). In the rare case that the degraded condition is associated with a support system but does not increase the likelihood of a plant transient or trip, then the finding should still be evaluated by considering its mitigation or other PRA functions under the Mitigating Systems Cornerstone (Exhibit 2).

d. SGTR – Steam generator tube conditions that violate the structural integrity performance criterion (typically three times the differential pressure across a tube during normal full-power steady-state operation, 3ΔPNO) make the tube more susceptible to failure during high pressure, dry steam generator core damage sequences. Steam generator tube conditions that violate accident leakage limits may not be able to meet 10 CFR 100 dose guidelines during design basis accidents.

e. External Event Initiators – In the Initiating Events Cornerstone, the external events of interest are limited to fire and internal flooding. Other external events, in the context of the Initiating Events Cornerstone, are not applicable because the licensee does not have control over these events (e.g., tornado, hurricane). However, the licensee does have control over the systems used to mitigate an external event and that is covered in the Mitigating Systems Cornerstone (Exhibit 2).

04.02 <u>Mitigating Systems (Exhibit 2)</u>

The Mitigating Systems screening questions are categorized into five sub-sections titled Mitigating Systems, Structures, Components (SSCs) and PRA Functionality (except Reactivity Control Systems); External Event Mitigating Systems (Seismic/Flood/Severe Weather Protection Degraded); Reactor Protection System; Fire Brigade; and Flexible Coping Strategies (FLEX). Below is additional guidance to support answering the screening questions for each sub-section:

a. Mitigating SSCs and PRA Functionality (except Reactivity Control Systems) – For the purposes of this sub-section, the SSCs (and their associated functions) of concern are those that provide a risk significant or risk relevant mitigating function in response to an initiating event, i.e., the PRA function. Normally those SSCs that are in the risk model provide a risk significant or risk relevant function; however, that is not always the case (e.g., some SSCs are not modeled explicitly). There are several ways to determine whether an SSC provides a risk significant or risk relevant mitigating function and below are some sources of information to support this determination:

1) Plant Risk Information eBook (PRIB), Table 6 – Table 6 lists systems/functions that are included in the SPAR model. It also provides specific success criteria given a particular initiating event. See PRIB definition in Section 0609A-05 of this IMC, "Detailed Risk Evaluation."
2) PRIB, Table 7 – Table 7 lists the components included in the SPAR model with their associated risk importance measures.
3) SDP Workspace – The SDP workspace contains risk significant and risk relevant SSCs derived from the site-specific SPAR model.
4) UFSAR – Although the systems/functions described in the UFSAR might be different than the systems/functions modeled in the SPAR, the licensed design bases for systems/functions can provide useful information in determining safety significance.
5) Licensee Risk Insights – If provided, risk insights from the licensee risk model (e.g., importance measures, dominant sequences, delta CDF calculations, etc.) and risk/safety significant SSCs from their maintenance rule program can be a good source of risk information.

PRA function refers to the ways in which the SSC can be used in a PRA to prevent an initiating event from resulting in core damage. An SSC may have more or different PRA functions than those functions for which it is credited in the design or licensing basis. For example, the design function of the core spray system may be limited to mitigation of large loss of coolant accidents (LOCAs). As such, the accident analysis may define a certain flowrate required to mitigate that accident. However, the core spray system can be credited in a PRA to provide coolant injection in any scenarios in which coolant injection is needed and pressure can be reduced such that the system can operate. Thus, the PRA function of the core spray system is not limited to the mitigation of large LOCAs and the system may be able to perform some of its other PRA functions without meeting its design flowrate.

A key concept in assessing whether an SSC can perform its PRA function is mission time. A 24-hour mission time is standard in PRA applications and should be considered in SDP screening as a general rule. The 24-hour mission time used for the purposes of SDP may be different than the time the SSC is required to operate as stated in the accident analysis or design basis for the SSC. Inspectors should consult with an SRA for unique situations or questions about mission time.

When the screening questions refer to a TS allowed outage time (AOT), the AOT is being used to assess the impact of the exposure time during which the SSC could not perform its PRA function. Although TS AOTs were not necessarily derived from risk evaluations, operating experience has shown that an SSC that cannot function for less than its AOT is generally not risk significant. Therefore, a detailed risk evaluation only needs to be performed when the SSC could not function for a period of time greater than that defined in the AOT. For single train systems or single trains within a multi-train system, the period of the AOT is used. For loss of function for two separate TS systems, 24 hours is used to determine if a detailed risk evaluation is warranted. For risk-significant, non-TS SSCs, 3 days is used. For plants that have adopted TSTF-505 and implemented risk-informed completion times (RICTs), the frontstop AOT should be used for screening purposes. RICTs may not be applied in retrospect after a degraded condition occurs.

The screening question that refers to "loss of system and/or function" generally applies to single train systems, or system/function as defined in the PRIB. A system/function is modeled in the PRA but may not have a precise SSC definition. Examples include the recovery of offsite power after a LOOP event, feed and bleed in a PWR plant after AFW system failures, or various plant crosstie capabilities.

b. External Event Mitigating Systems (Seismic/Flood/Severe Weather Protection) – This section is only applicable for findings related to seismic, flooding, and severe weather protection. Findings related to fires or fire protection equipment are not included in this question because those findings should be evaluated using IMC 0609 Appendix F, "Fire Protection Significance Determination Process."

c. Reactor Protection System (RPS) – The main focus of the screening question is to screen findings that result in a minor functional degradation of RPS (e.g., one automatic trip from one instrument) but there are several redundant trips that provide the same function (e.g., three other automatic functional trips). If there is a significant functional degradation to RPS, a detailed risk evaluation is warranted. The determination of what a "significant" or "minor" functional degradation of RPS should be based on reasonable technical judgment of the inspectors, SRA, and management.

d. Fire Brigade – This section screens fire brigade findings to Green that are not expected to result in additional fire growth. Fire brigade findings that are expected to result in additional fire growth are evaluated further using IMC 0609 Appendix M, "Significance Determination Process Using Qualitative Criteria."

e. Flexible Coping Strategies (FLEX) - This screening section is intended for use in assessing inspection findings that are associated with equipment, procedures, training, and other programmatic aspects used specifically for satisfying the requirements of Orders EA-12-049, EA-12-051, and EA-13-109 or for compliance with 10 CFR 50.155. In the event that the equipment serves another function, a different and more limiting SDP tool will be used. For example, if the performance deficiency concerns installed plant equipment that is credited for Phase 1 mitigating strategies, but is also credited for use under normal operating conditions or used to mitigate other transients or accidents (e.g. reactor core isolation cooling pump, turbine-driven auxiliary feedwater pump), the more limiting SDP (e.g., IMC 0609, Appendices A, G, and H) would be used to assess the significance of the issue. This applies to all equipment, procedures, training, and other programmatic aspects that are not credited for the sole purpose of satisfying the requirements of Orders EA-12-049, EA-12-051, EA-13-109, or 10 CFR 50.155. This section is used to screen findings related to all aspects of Phase 1 and Phase 2 mitigating strategies.

For the purpose of this SDP section, a FLEX function should be considered failed if the strategy could not be implemented in accordance with existing plant procedures in the time allotted. This could occur if multiple pieces of equipment fail or if one piece of portable equipment was failed but its failure would not be discovered within enough time for the licensee to install the backup piece of portable equipment. This could also occur if the failure of a piece of equipment would result in additional equipment failures that would require recovery actions or prevent completion of the strategy in accordance with existing procedures within the time allotted.

04.03 Barrier Integrity (Exhibit 3)

The Barrier Integrity screening questions are categorized into five sub-sections titled Fuel Cladding Integrity, Reactor Coolant System (RCS) Boundary, Reactor Containment, Control Room/Auxiliary/Reactor Building or Spent Fuel Pool Building, and Spent Fuel Pool. Below is additional guidance to support answering the screening questions for each sub-section:

a. Fuel Cladding Integrity – The purpose of this section is to screen findings to Green that do not challenge fuel cladding integrity. For the purposes of this SDP, issues that meet any of the following four criteria represent a challenge to fuel cladding integrity and require further evaluation: (1) placed the plant in an unanalyzed condition, (2) adversely impacted any fundamental assumptions regarding fuel failure used in the accident analysis (such as fuel failure temperature or oxidation rate), (3) resulted in reactor coolant activity exceeding TS limits, or (4) resulted in automatic actuation of an SSC necessary to protect against exceeding thermal limits. If degradation of fuel cladding could result in a substantial potential for overexposure, the finding should also be evaluated using IMC 0609 Appendix C, "Occupational Radiation Safety Significance Determination Process."

b. Reactor Coolant System (RCS) Boundary – All issues which address potential violations of regulatory requirements for protection of the reactor pressure vessel against fracture (e.g., pressure-temperature limits, pressurized thermal shock (PTS)) are addressed under the Barrier Integrity Cornerstone and should be reviewed by the applicable technical group in NRR (NRR/DNRL/NVIB). Findings related to RPV fracture toughness requirements must be evaluated in accordance with the ASME Code, Section XI, Appendix E, "Evaluation of Unanticipated Operating Events" which provides deterministic acceptance criteria for evaluating the impact of the out-of-limit condition on the structural integrity of the RPV to determine whether the plant is acceptable for continued operation. All other RCS boundary issues (i.e., leakage) are evaluated under the Initiating Events Cornerstone (Exhibit 1).

c. Reactor Containment – The purpose of this section is to refer findings that primarily impact large early release frequency (LERF) for further evaluation using IMC 0609 Appendix H, "Containment Integrity Significance Determination Process."

d. Control Room/Auxiliary/Reactor Building or Spent Fuel Pool Building – Findings that impact control room habitability require further evaluation. Findings related to the radiological barrier functions of the control room, auxiliary building, reactor building, and spent fuel pool building are not expected to impact CDF or LERF. If degradation of the radiological barrier could result in a substantial potential for overexposure, the finding should also be evaluated using IMC 0609 Appendix C.

e. Spent Fuel Pool – Findings that challenge spent fuel pool design criteria require further evaluation. Further evaluation is performed using IMC 0609 Appendix M because the NRC does not maintain PRA models for spent fuel pools.

0609A-05 DETAILED RISK EVALUATION

The inspection staff and regional SRAs should coordinate efforts, using their specific skills, training, and qualifications, to arrive at an appropriate risk evaluation given the specific circumstances associated with the risk impact of the degraded condition(s) that resulted from the finding. Typically, inspectors develop the finding and the associated functional impact on the equipment and gather plant information to support the detailed risk evaluation. Then the inspectors and SRA collaborate to develop appropriate input assumptions while the SRA normally performs the detailed risk evaluation using the SPAR model, the RASP handbooks, and other risk information as necessary. If the finding does not screen to Green, the regional branch chief responsible for the issue and the SRA shall determine if an Inspection Finding Review Board is warranted using the guidance in IMC 0609 Attachment 5, “Inspection Finding Review Board,” to ensure alignment on the performance deficiency, the inspection finding, any proposed violation(s), and the actions needed to determine the preliminary significance,

All detailed risk evaluations should be peer reviewed by an SRA or Reliability and Risk Analyst. A peer review is recommended but not required for straightforward detailed risk evaluations for Green findings. A peer review is highly recommended for more complicated detailed risk evaluations for Green findings in order to verify reasonable modeling assumptions have been made. Peer reviews are required for any detailed risk evaluations performed for greater than Green findings, as discussed in IMC 0609 Attachment 1, “Significance and Enforcement Review Panel Process.” When the internal events detailed risk evaluation results are greater than or equal to 1.0E-7, the finding should be evaluated for external event risk contribution. Any internal events results that are less than 1.0E-7 can be evaluated for external event risk contribution at the discretion of the regional SRA.[1] If an inspector uses the SDP Workspace to perform a detailed risk evaluation, a regional SRA must review the results to determine if any additional analyses need to be performed.

If more than one cornerstone is affected by the finding and associated degraded condition(s), the risk evaluation of the finding should take into account all of the associated degraded condition(s) from all of the affected cornerstones. However, for the purposes of the power reactor assessment program, the cornerstone which captures the majority fraction of the overall risk evaluation should be identified as the affected cornerstone. The risk tools and guidance available to the staff to perform the detailed risk evaluation are discussed below:

<u>NOTE</u>: The risk tools (e.g., SDP Workspace) and guidance to support the SDP are designed to have users engaged in the process and avoid a “blackbox” approach in determining the risk significance of deficient licensee performance. Users need to be aware of the limitations and specific capabilities of each risk tool and associated guidance to preclude misapplication.

<u>SAPHIRE and SPAR Models</u>:

1) SDP Workspace – The SDP Workspace provides the user with a change in core damage frequency (delta CDF), and change in large early release frequency (delta LERF) calculation with a comprehensive report of results. This tool only accounts for risk associated with internal events (i.e., does not account for external event risk

[1] Until operating experience is gained for AP1000 plants, the finding should be evaluated for external event risk contribution when the internal events detailed risk evaluation results are greater than or equal to 1.0E-8.

contributions) and cannot be adjusted to change the model (e.g., recovery actions, common cause failure).

2) Event Condition Assessment – A workspace that is used by the SRA that allows the analyst more flexibility in adjusting basic events.

3) General Analysis – A workspace that is used by the SRA that allows more flexibility in adjusting both basic events and model logic.

4) Specific SPAR Model Changes – The SRA can alter the SPAR model logic and create a set of changed basic events to reflect the degraded condition(s) and/or event. This approach provides the most flexibility in performing a delta CDF calculation.

5) Plant Risk Information eBook (PRIB) – The PRIB is a summary document associated with the site-specific SPAR model that provides a variety of risk insights.

Changes to SAPHIRE and SPAR Models:

Identified Errors or Discrepancies – Identified errors or discrepancies with SAPHIRE or the site-specific SPAR model should be discussed and vetted by the inspection staff and SRA and then reported to Idaho National Laboratory (INL) via the SAPHIRE webpage at https://saphire.inl.gov/. On the SAPHIRE webpage there is one module to request changes to SAPHIRE (i.e., software) and a separate module to request changes to the SPAR models (which includes changes to the PRIB).

Timely SDP Evaluations – To support the SDP timeliness goal, an SRA may make changes to the SPAR model of record, as appropriate, based on information from the inspectors and/or the licensee, to accurately reflect the risk significance of the finding. The SRA should consult with INL on SPAR model changes. These changes must be documented in the associated inspection report and/or SERP package. The SRA should subsequently review the model changes made to determine if those model changes should be incorporated into the plant SPAR model of record.

Guidance Documents:

1) RASP Handbook Volumes 1 (Internal Events), 2 (External Events), and 4 (Shutdown) - These handbooks provide standardized risk guidance and best practices to support determinations across a variety of NRC programs (SDP, Accident Sequence Precursor (ASP), and Management Directive (MD) 8.3, "Event Evaluation").

2) NUREGs – There are many NUREGs that can provide useful information when performing a detailed risk evaluation (e.g., initiating event and failure data, common cause failure modeling techniques).

END

Exhibit 1 - Initiating Events Screening Questions

A. Loss of Coolant Accident (LOCA) Initiators

1. After a reasonable assessment of degradation, could the finding result in exceeding the reactor coolant system (RCS) leak rate for a small LOCA (leakage in excess of normal makeup)?

   □ a. If YES ➛ Stop. Go to Detailed Risk Evaluation section.

   □ b. If NO, continue.

2. After a reasonable assessment of degradation, could the finding have likely affected other systems used to mitigate a LOCA (e.g., Interfacing System LOCA)?

   □ a. If YES ➛ Stop. Go to Detailed Risk Evaluation section.

   □ b. If NO, screen as Green.

B. Transient Initiators

Did the finding cause a reactor trip AND the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition (e.g., loss of condenser, loss of feedwater)? Other events include high-energy line breaks, internal flooding, and fire.

□ a. If YES ➛ Stop. Go to Detailed Risk Evaluation section.

□ b. If NO, screen as Green.

C. Support System Initiators

1. Did the degraded condition result in an actual complete or partial loss of a support system (e.g., component cooling water, service water, instrument air, AC power, DC power)?

   □ a. If YES ➛ Stop. Go to Detailed Risk Evaluation section.

   □ b. If NO, continue.

2. Did the degraded condition increase the likelihood of a complete loss of a support system that would result in a plant trip?

□ a. If YES ➛ Stop. Go to Detailed Risk Evaluation section.

□ b. If NO, screen as Green.

D. <u>Steam Generator Tube Rupture</u>

1. Does the finding involve a degraded steam generator tube condition where one tube cannot sustain three times the differential pressure across a tube during normal full power, steady state operation (3ΔPNO)?

   □ a. If YES ➛ Stop. Go to IMC 0609, Appendix J.

   □ b. If NO, continue.

2. Do one or more SGs violate "accident leakage" performance criterion (i.e., involve degradation that would exceed the accident leakage performance criterion under design basis accident conditions)?

   □ a. If YES ➛ Stop. Go to Detailed Risk Evaluation section and refer to IMC 0609, Appendix J as applicable.

   □ b. If NO, screen as Green.

E. <u>External Event Initiators</u>

Does the finding impact the frequency of a fire or internal flooding initiating event?

□ a. If YES ➛ Stop. Go to Detailed Risk Evaluation section.

□ b. If NO, screen as Green.

Exhibit 2 – Mitigating Systems Screening Questions

A. Mitigating SSCs and PRA Functionality (except Reactivity Control Systems)

1. If the finding is a deficiency affecting the design or qualification of a mitigating SSC, does the SSC maintain its operability or PRA functionality?

   □ a. If YES ➛ Screen as Green.

   □ b. If NO, continue.

2. Does the degraded condition represent a loss of the PRA function of a single train TS system (such as HPCI/HPCS) for greater than its TS allowed outage time?

   □ a. If YES ➛ Stop. Go to Detailed Risk Evaluation section.

   □ b. If NO, continue.

3. Does the degraded condition represent a loss of the PRA function of one train of a multi-train TS system for greater than its TS allowed outage time?

   □ a. If YES ➛ Stop. Go to Detailed Risk Evaluation section.

   □ b. If NO, continue.

4. Does the degraded condition represent a loss of the PRA function of two separate TS systems for greater than 24 hours?

   □ a. If YES ➛ Stop. Go to Detailed Risk Evaluation section.

   □ b. If NO, continue.

5. Does the degraded condition represent a loss of a PRA system and/or function as defined in the PRIB or the licensee's PRA (such as recovery of offsite power or the ability to feed and bleed) for greater than 24 hours?

   □ a. If YES ➛ Stop. Go to Detailed Risk Evaluation section.

   □ b. If NO, continue.

6. Does the degraded condition represent a loss of the PRA function of one or more non-TS trains of equipment designated as risk-significant in accordance with the licensee's maintenance rule program for greater than 3 days?

   □ a. If YES ➛ Stop. Go to Detailed Risk Evaluation section.

   □ b. If NO, screen as Green.

B. <u>External Event Mitigating Systems (Seismic/Flood/Severe Weather Protection)</u>

Does the finding involve the loss or degradation of equipment or function specifically designed to mitigate a seismic, flooding, or severe weather initiating event (e.g., seismic snubbers, flooding barriers, tornado doors) for greater than 14 days?

□ a. If YES ➛ Go to Exhibit 4.

□ b. If NO, screen as Green.

C. <u>Reactor Protection System (RPS)</u>

Did the finding affect a single RPS trip signal to initiate a reactor scram AND the function of other redundant trips or diverse methods of reactor shutdown (e.g., other automatic RPS trips, alternate rod insertion, or manual reactor trip capacity)?

□ a. If YES ➛ Stop. Go to Detailed Risk Evaluation section.

□ b. If NO, screen as Green.

D. <u>Fire Brigade</u>

1. Does the finding involve fire brigade training, qualifications, drill performance, or staffing?

   □ a. If YES ➛ check if the following applies:
   □ The finding would not have significantly affected the ability of the fire brigade to respond to a fire.

   □ b. If the above is checked ➛ screen as Green.

   □ c. If NO, continue.

2. Does the finding involve the response time of the fire brigade to a fire?

   □ a. If YES ➛ check if one or more of the following apply:
   □ The fire brigade's response time was mitigated by other defense-in-depth elements, such as area combustible loading limits were not exceeded, installed fire detection systems were functional, and alternate means of safe shutdown were not impacted.
   □ The finding involved risk-significant fire areas that had automatic suppression systems.
   □ The licensee had adequate fire protection compensatory actions in place.

   □ b. If at least one of the above is checked ➛ screen as Green.

   □ c. If NO, continue.

3. Does the finding involve fire extinguishers, fire hoses, or fire hose stations?

   □ a. If YES ➛ check if one or more of the following apply:
   □ There was no degraded fire barrier and the fire scenario did not require the use of water to extinguish the fire.
   □ The missing fire extinguisher or fire hose was missing for a short time and other extinguishers or hose stations were in the vicinity.

   □ b. If at least one of the above is checked ➛ screen as Green.

   □ c. If none of the boxes under D.1.a, D.2.a, or D.3.a are checked ➛ Stop. Go to IMC 0609, Appendix M.

E. Flexible Coping Strategies (FLEX)

1. Is the inspection finding associated with equipment, training, procedures, and/or other programmatic aspects credited for the sole purpose of satisfying the requirements of Order EA-12-051 or 10 CFR 50.155 for spent fuel pool instrumentation or EA-13-109 for containment venting (i.e., not credited for satisfying EA-12-049 or other portions of 10 CFR 50.155 as well)?

   □ a. If YES ➛ Screen as Green.

   □ b. If NO, continue.

2. Does the inspection finding involve equipment, training, procedures, and/or other programmatic aspects credited in any Phase 1 or 2 FLEX strategy such that any FLEX function (such as extended HPCI/RCIC/AFW operation, providing FLEX DC power, FLEX AC power, or FLEX RCS feed) could not be completed in accordance with existing plant procedures within the time allotted for an exposure period of greater than 21 days?

   □ a. If YES ➛ Stop. Go to Detailed Risk Evaluation section.

   □ b. If NO, screen as Green.

Exhibit 3 – Barrier Integrity Screening Questions

A. Fuel Cladding Integrity

1. Did the finding involve control manipulations that unintentionally added positive reactivity that challenged fuel cladding integrity (e.g., inadvertent boron dilution, cold water injection, two or more inadvertent control rod movements, recirculation pump speed control)?

   □ a. If YES, ➛ Stop. Go to IMC 0609, Appendix M.

   □ b. If NO, continue.

2. Did the finding result in a mismanagement of reactivity by operator(s) that challenged fuel cladding integrity (e.g., reactor power exceeding the licensed power limit, inability to anticipate and control changes in reactivity during crew operations)?

   □ a. If YES, ➛ Stop. Go to IMC 0609, Appendix M.

   □ b. If NO, continue.

3. Did the finding result in the mismanagement of the foreign material exclusion or reactor coolant chemistry control program that challenged fuel cladding integrity (e.g., loose parts, material controls)?

   □ a. If YES, ➛ Stop. Go to IMC 0609, Appendix M.

   □ b. If NO, continue.

4. Did the finding result from fuel handling errors, a dropped fuel assembly, a misplaced fuel bundle, or crane operations over the core or anywhere in the refueling pathway that challenged fuel cladding integrity or resulted in a release of radionuclides?

   □ a. If YES, ➛ Stop. Go to IMC 0609, Appendix M.

   □ b. If NO, screen as Green.

B. Reactor Coolant System (RCS) Boundary

Does the finding involve potential non-compliance with regulatory requirements for protection of the reactor pressure vessel against fracture (e.g., pressure-temperature limits or pressurized thermal shock issues)?

□ a. If YES ➛ Stop. Go to IMC 0609, Appendix M and consult the appropriate technical branch in NRR (NRR/DNRL/NVIB).

□ b. If NO, screen as Green.

C. Reactor Containment:

1. Does the finding represent an actual open pathway in the physical integrity of reactor containment (valves, airlocks, etc.), failure of containment isolation system (logic and instrumentation), failure of containment pressure control equipment (including SSCs credited for compliance with Order EA-13-109), failure of containment heat removal components, or failure of the plant’s severe accident mitigation features (AP1000)?

   □ a. If YES ➛ Stop. Go to IMC 0609, Appendix H.

   □ b. If NO, continue.

2. Does the finding involve an actual reduction in function of hydrogen igniters in the reactor containment?

   □ a. If YES ➛ Stop. Go to IMC 0609, Appendix H.

   □ b. If NO, screen as Green.

D. Control Room, Auxiliary, Reactor, or Spent Fuel Pool Building:

1. Does the finding only represent a degradation of the radiological barrier function provided for the control room, auxiliary building, spent fuel pool, SBGT system (BWR), or EGTS system (PWR ice condenser)?

   □ a. If YES ➛ Stop. Screen as Green.

   □ b. If NO, continue.

2. Does the finding represent a degradation of the barrier function of the control room against smoke or a toxic atmosphere?

   □ a. If YES ➛ Stop. Go to Detailed Risk Evaluation section.

   □ b. If NO, screen as Green.

E. Spent Fuel Pool (SFP)

1. Does the finding adversely affect decay heat removal capabilities from the spent fuel pool causing the pool temperature to exceed the maximum analyzed temperature limit specified in the site-specific licensing basis?

   □ a. If YES ➛ Stop. Go to IMC 0609, Appendix M.

   □ b. If NO, continue.

2. Does the finding result from fuel handling errors, dropped fuel assembly, dropped storage cask, or crane operations over the SFP that caused mechanical damage to fuel clad AND a detectible release of radionuclides?

   □ a. If YES ➛ Stop. Go to IMC 0609, Appendix M (refer to IMC 0609, Appendix C as applicable).

   □ b. If NO, continue.

3. Does the finding result in a loss of spent fuel pool water inventory decreasing below the minimum analyzed level limit specified in the site-specific licensing basis?

   □ a. If YES ➛ Stop. Go to IMC 0609, Appendix M.

   □ b. If NO, continue.

4. Does the finding affect the SFP neutron absorber, fuel bundle misplacement (i.e., fuel loading pattern error) or soluble Boron concentration (PWRs only)?

   □ a. If YES ➛ Stop. Go to IMC 0609, Appendix M.

   □ b. If NO, screen as Green.

Exhibit 4 – External Events Screening Questions

1. If the equipment or safety function is failed or unavailable, are ANY of the following three statements TRUE? The loss of this equipment or function by itself during the external initiating event it was intended to mitigate:

    - would cause a plant trip or an initiating event;
    - would degrade two or more trains of a multi-train system or function;
    - would degrade one or more trains of a system that supports a risk significant system or function.

    □ a. If YES ➛ Stop. Go to Detailed Risk Evaluation section.

    □ b. If NO, Continue.

2. Does the finding involve the total loss of any PRA function, identified by the licensee through a PRA, IPEEE, or similar analysis, that contributes to external event initiated core damage accident sequences (i.e., initiated by a seismic, flooding, or severe weather event)?

    □ a. If YES ➛ Stop. Go to Detailed Risk Evaluation section.

    □ b. If NO, screen as Green.

ATTACHMENT 1
Revision History for IMC 0609 Appendix A

Commitment Tracking Number	Accession Number Issue Date Change Notice	Description of Change	Description of Training Required and Completion Date	Comment Resolution and Closed Feedback Form Accession Number (Pre-Decisional, Non-Public Information)
	04/21/00 CN 00-007	Initial issue		
	12/28/00 CN 00-029	Revised to incorporate changes based on inspector feedback. Enhancements generated by IIPB and SPSB risk analysts based on initial implementation experience to date have also been added. A significant change is the credit given for operator actions in step 2.3 of the document. Clarification changes have also been made to the phase 1 screening worksheets. Phase 2 worksheets are in the process of being updated to include plant and site-specific information. This document is an integral part of the Significant Determination Process for reactor inspection findings for At-Power operations and will be used by resident and region-based inspectors as well as by SRAs.		
	02/05/01 CN 01-003	Revised to correct formatting problems with charts and tables, and to make minor editorial changes.		
	03/18/02 CN 02-009	Revised: 1) to correct identified problems with the appendix, 2) to incorporate the rules for using the site specific risk-informed inspection notebook, 3) to simplify the process of accounting for external initiators in characterizing the risk significant inspection findings, and 4) to provide guidance on evaluating concurrent inspection findings.		

Commitment Tracking Number	Accession Number Issue Date Change Notice	Description of Change	Description of Training Required and Completion Date	Comment Resolution and Closed Feedback Form Accession Number (Pre-Decisional, Non-Public Information)
	ML042600558 09/10/04 CN 04-023	Multiple editorial changes to enhance user friendliness of the document. For example, re-format action steps, provided additional examples, added the reference to Appendix J for steam generator issues.	N/A	
	ML043560116 12/01/04 CN 04-027	Corrected two errors on page 4 of the worksheet, under MS cornerstone for screening issues and under BI cornerstone guidance for question 3 for screening to Green.	N/A	
	ML052790196 11/22/05 CN 05-030	Enhanced guidance to help meet timeliness requirements for finalizing the SDP for inspection findings.	N/A	
	ML063470288 03/23/07 CN 07-011	Incorporate references to the site-specific inspection notebooks and associated Pre-Solved Tables; In Attachment 2, update the site-specific risk-informed inspection notebooks usage rules; Attachment 3, provide user guidance for screening of external events risk contributions.	1. Training has been provided to the SRAs at last two SRA counterpart meetings, and the SRAs have provided training to the region based and resident inspectors (10/2006) 2. Formalized training will be introduced through the P-111 course (FY 2008)	ML070720624

Commitment Tracking Number	Accession Number Issue Date Change Notice	Description of Change	Description of Training Required and Completion Date	Comment Resolution and Closed Feedback Form Accession Number (Pre-Decisional, Non-Public Information)
	ML063060377 01/10/08 CN 08-002	Removed the Phase 1 Initial Screening and Characterization of Findings process to create the new IMC 0609, Attachment 4. Added clarification statement to Step 2.1.2 and Usage Rule 1.1 that the maximum exposure time used in SDP is limited to one year.	N/A	ML073460588
	ML101400574 06/19/12 CN 12-010	Updated the guidance to reflect the transition from the pre-solved tables and risk-informed notebooks to SAPHIRE and the site-specific SPAR models. Moved the Initiating Events, Mitigating Systems, and Barrier Integrity screening questions from IMC 0609, Attachment 4 to this appendix. Incorporated feedback from ROP FBFs 0609.04-1458 and 0609A-1575. This is a complete reissue.	Senior Reactor Analysts and headquarters staff provided detailed instructor-led training to resident inspectors, region based inspectors, and other regional staff. June 2012	ML12142A091 Closed FBF: 0609.04-1458 ML12171A225 0609A-1575 ML12171A231
N/A	ML19198A183 7/17/19 CN	Made draft publicly available to discuss at the July 31, 2019 ROP monthly public meeting	N/A	N/A
	ML19011A338 12/13/19 CN 19-040	Updated guidance to direct users to contact NRR for issues with pressure-temperature limits (ROP FBF 0609A-2070), moved some of the reactivity control questions to the Barrier Integrity Cornerstone exhibit to align with IMC 0612 (ROP FBF 0609A-2134), revised the fire brigade and support system initiator questions for clarity (ROP FBFs 0609A-2167 and	No required training on specific changes to this revision.	ML19014A063 Closed ROP FBFs 0609A-2070 ML19014A104 0609A-2134 ML19014A205

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		2311), added a question regarding fuel cladding integrity, separated and revised the mitigating systems questions to account for single train systems and PRA functions (ROP FBFs 0609A-2260 and 2318), and incorporated FLEX questions from IMC 0609 Appendix O. Questions were modified to screen FLEX findings that are solely related to EA-13-109 and containment pressure control systems to Appendix H (ROP FBF 0609A-2355). IMC 0609, Attachment 4 was reviewed to align with this appendix regarding support system initiators and spent fuel pool applicability (ROP FBF 0609A-2290 and 2085). Inspector training related to use of risk-informed thinking and tools is planned (ROP FBF 0609A-1924). Document was reviewed and minor changes were made to allow for use with new reactor designs (AP1000). In accordance with Management Directive 8.13 and COMSECY-16-0022, the Commission was notified of the described changes via SECY-19-0037, "Reactor Oversight Process Self-Assessment for Calendar Year 2018," (ADAMS Accession No. ML19042A100). The Commission was also notified of the revisions in a Commissioner Assistants' Note (ADAMS Accession No. ML19302F254).	General training is planned as part of the Regional Risk Informed Decision-Making action plan.	0609A-2167 ML19014A106 0609A-2260 ML19014A107 0609A-2311 ML19014A108 0609A-2318 ML19014A109 0609A-1924 ML19253A002 0609A-2085 ML19253A003 0609A-2290 ML19253A004 0609A-2355 ML19253A005

Commitment Tracking Number	Accession Number Issue Date Change Notice	Description of Change	Description of Training Required and Completion Date	Comment Resolution and Closed Feedback Form Accession Number (Pre-Decisional, Non-Public Information)
	ML20308A592 11/03/2020 CN DRAFT	Made draft publicly available to discuss at the November 18, 2020, ROP monthly public meeting.	N/A	N/A
	ML20226A093 11/30/20 CN 20-066	Combined and revised the FLEX screening questions to clarify when a detailed risk evaluation should be performed (ROP FBF 0609A-2407). Revised the FLEX screening questions and background information to incorporate issuance of 10 CFR 50.155 and to move information to the basis document, IMC 0308 Att 3 App A. Updated correct branch to contact for findings related to protection again reactor pressure vessel fracture (Exhibit 2, Question B) after NRR/NRO merger (ROP FBF 0609A-2408). Changed the method of further evaluation for these findings from a detailed risk evaluation to IMC 0609 Appendix M based on SRA feedback. Added guidance to Section 0609A-05 to recommend peer reviews for all detailed risk evaluations (ROP FBFs 0308.03A-2178, 0609-2179) and to refer to IMC 0609 Att 5. Added a new question to Exhibit 3 for fuel handling errors to align with a change made to IMC 0609 Att 4 that routes those types of findings to App A. Revised the LOCA initiator screening question (Exhibit 1, Question A.2) based on SRA feedback. Added screening guidance consistent with revisions to the basis document for those areas that had no additional guidance in Section 0609A-04.	No training required on specific changes in this revision.	ML20226A181 Closed ROP FBFs 0609A-2407 ML20226A221 0609A-2408 Editorial - rejected from the feedback process upon receipt 0609-2179 ML20226A217 0308.03A-2178 ML20226A207