SHUTDOWN OPERATIONS
SIGNIFICANCE DETERMINATION PROCESS
PHASE 1 INITIAL SCREENING AND CHARACTERIZATION OF FINDINGS

Effective Date: 05/20/2022

0609G1.01 PURPOSE

The Shutdown Significance Determination Process (SDP) consists of three phases: Phase 1, Initial Screening and Characterization of Findings; Phase 2, Initial Risk Significance Approximation and Basis; and Phase 3, Risk Significance Finalization and Justification. This attachment and its exhibits are designed to provide U.S. Nuclear Regulatory Commission (NRC) inspectors and management with a framework for use in the initial screening and characterization of potentially risk-significant shutdown issues within the Initiating Events, Mitigating Systems, and Barrier Integrity cornerstones for Phase 1 of the SDP. In addition, this process identifies findings of very low risk significance that do not warrant further NRC engagement. This appendix is intended to be used when the plant is shut down with at least one fuel bundle in the reactor and temperature and pressure are within the normal residual heat removal (RHR) / decay heat removal (DHR) operating conditions; otherwise, return to IMC 0609, Attachment 4, "Initial Characterization of Findings."

0609G1.02 ENTRY CONDITIONS

Before entering an issue into the SDP, the inspector will screen the issue to determine its documentation threshold as described in Inspection Manual Chapter (IMC) 0612, Appendix B, "Additional Issue Screening Guidance." If an inspector screens a finding that involves shutdown operations with fuel in the reactor in accordance with IMC 0609, Attachment 4, "Initial Characterization of Findings," and is directed by Attachment 4 to IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," then the inspector will initially screen that finding using the shutdown Phase 1 screening questions found in Exhibits 2–5 of this appendix.

Note: Appendix G is not the appropriate procedure to use for evaluation of events relating to spent fuel pools. Instead, refer to IMC 0609, Appendix A, Exhibit 3, "Barrier Integrity Screening Questions," which contains screening criteria for spent fuel pool events.

0609G1.03 PHASE 1 SDP OVERVIEW

Appendix G of the SDP is a tool which uses a quantitative risk method to characterize the risk of events or conditions during shutdown. All issues, including those at shutdown, that screen more than minor in Appendix B of IMC 0612 are then characterized using IMC 0609, Attachment 4. The inspector would utilize the information from their initial characterization of the finding in IMC 0609, Attachment 4, Tables 1 & 2, but would transfer to this appendix. The purpose of the screening questions in Exhibits 2-5 of this appendix are to determine if the issue can be characterized as Green without entering into a more detailed analysis in Phases 2 or 3. Shutdown SDP Phase 2 guidance for pressurized water reactors (PWRs) is provided in IMC 0609, Appendix G, Attachment 2, “Phase 2 Significance Determination Process Template for PWR during Shutdown.” Shutdown SDP Phase 2 guidance for boiling-water reactors (BWRs) is provided in IMC 0609, Appendix G, Attachment 3, “Phase 2 Significance Determination Process Template for BWR during Shutdown.”

Phase 1 is intended to be accomplished by the inspection staff, with the assistance of a senior reactor analyst (SRA), if needed. Inspectors should collect information needed for determining the significance of the finding, such as the structure, system, or component affected; the nature of the degradation; and the duration of the degraded condition. Inspectors should obtain licensee risk perspectives as early in the SDP process as a licensee is prepared to offer them and use the SDP framework to the extent possible to evaluate the adequacy of the licensee’s input and assumptions.

The Phase 1 screening questions are intended to provide conservative guidance for screening findings of very low safety significance out of further review. Due to the unique nature of shutdown findings and the configuration of the plant during shutdown, the screening questions may not provide sufficient guidance for all findings. If the screening questions are not conservative for the finding being evaluated, the inspector should contact the SRA for additional guidance. The SRA may decide that a Phase 2 or 3 evaluation is more appropriate.

0609G1.04 INFORMATION ABOUT AP1000 REACTORS

AP1000 reactors are still screened through IMC 0609, Appendix G, Attachment 1 for Phase 1 screening. However, if any issue cannot be screened to Green, analysts will not proceed to Attachment 2 of Appendix G, like they would for existing PWRs. Instead, analysts should use the AP1000 SPAR model to perform a detailed risk evaluation since Attachment 2 does not support AP1000. This detailed risk analysis can be performed by a qualified reliability and risk analyst from APOB branch, or by an SRA. It should be noted that the AP1000 SPAR model is a new model and was developed before the NRC had access to plant procedures, so analysts should carefully review the model for accuracy.

This section identifies some potentially risk-significant aspects of the AP1000 design that are of interest during shutdown conditions. It is not intended to be an all-inclusive list.

04.01 AP1000 Design Features for Shutdown

The AP1000 reactor design features some significant design differences for shutdown over existing PWRs. These design features support improved safety during shutdown and include:

- Self-Venting Suction Line – Most of the residual heat removal pump suction line is sloped upward from the pump to the reactor coolant system hot leg. In the level portions of the piping, there are no local high points. This eliminates potential problems with refilling the pump suction line in a residual heat removal pump if it is stopped due to pump cavitation and or excessive air entrainment. With the self-venting suction line, the line will re-fill and the pumps can be immediately (no additional operator action required) restarted once an adequate level in the hot leg is re-established.

- Step-nozzle Connection – The normal residual heat removal system employs a step-nozzle connection to the reactor coolant system (RCS) hot leg. The step-nozzle connection has two effects on mid-loop operation. One effect is to substantially lower the RCS hot leg level at which a vortex occurs in the residual heat removal suction line due to the lower fluid velocity in the hot leg nozzle. This increases the margin from the nominal mid-loop level to the level where air entrapment into the pump suction begins. Another effect of the step-nozzle is that, if a vortex should occur, the maximum air entrapment into the pump suction has been shown experimentally to be no greater than 5 percent. This level of air ingestion will make air binding of the pump much less likely.

- Normal Residual Heat Removal Throttling During Mid-Loop – The normal residual heat removal pumps are designed to minimize susceptibility to cavitation, and, as a result, the residual heat removal pumps normally operate without the need for throttling flow when the level in the RCS is reduced to a mid-loop level. Only if the RCS is at saturated conditions and the RCS level is at mid-loop will some throttling of a flow control valve be necessary to maintain adequate net positive suction head.

- Hot Leg Level Instrumentation and Automatic Isolation of CVS Letdown – The AP1000 reactor coolant system contains level instrumentation in each hot leg with indication in the main control room. Alarms are provided to alert the operator when the RCS hot leg is approaching a low level. The isolation valves in the line used to drain the reactor coolant system close on a low reactor coolant system level during shutdown operations. These safety-related systems have operability requirements during shutdown in Technical Specifications LCO 3.3.10.

- Wide Range Pressurizer Level – A non-safety-related independent pressurizer level transmitter, calibrated for low temperature conditions, provides water level indication during startup, shutdown, and refueling operations in the main control room and at the remote shutdown workstation. The upper-level tap is connected to an ADS valve inlet header above the top of the pressurizer. The lower-level tap is connected to the bottom of the hot leg. This provides level indication for the entire pressurizer and a continuous reading as the level in the pressurizer decreases to mid loop levels during shutdown operations.

- ADS Valves – The automatic depressurization system first-, second-, and third-stage valves, connected to the top of the pressurizer, are open whenever the core makeup tanks are blocked during shutdown conditions while the reactor vessel upper internals are in place. This arrangement provides a vent path to preclude pressurization of the reactor coolant system during shutdown conditions when decay heat removal is lost. This also allows the in-containment refueling water storage tank to automatically provide injection flow if it is actuated on a loss of decay heat removal. The ADS valves have operability requirements in technical specifications during shutdown conditions for RCS intact conditions and RCS open conditions (LCO 3.4.12 and LCO 3.4.13). Stage 4 of ADS (automatic or manual) is necessary for gravity injection from the In-Containment Refueling Water Storage Tank and for containment recirculation to be successful.

04.02 AP1000 Probabilistic Risk Assessment (PRA) Insights

Detailed information can be found in Chapter 19, Probabilistic Risk Assessment of the Vogtle Units 3 and 4 Updated Final Safety Analysis Report (UFSAR) (ADAMS Accession No. ML19171A078).

04.03 AP1000 Mode 4

In the AP1000, Mode 4 has been redefined as Safe Shutdown and corresponds to the range of RCS temperature between 420˚F and 200˚F.

04.04 AP1000 Shutdown Risk Significant Non-Safety-Related Systems

There are shutdown risk-significant, non-safety-related systems which have availability controls specified in the Technical Requirements Manual (TRM). Some of these technical requirements are listed below as reference:

- TRM 3.7.2 RNS (Normal Residual Heat Removal System) - RCS open
- TRM 3.7.3 CCS (Component Cooling Water System) – RCS open
- TRM 3.7.4 SWS (Service Water System) – RCS open
- TRM 3.7.5 Main Control Room Cooling Modes 1-6
- TRM 3.7.6 I&C B&C Room Cooling Modes 1-6
- TRM 3.8.1 AC Power Supplies Mode 5
- TRM 3.8.2 AC Power Supplies RCS open
- TRM 3.8.3 AC Power Supplies Long Term Shutdown Modes 1-6
- TRM 3.8.4 Non-Class 1E DC and UPS System Modes 1-6
- TRM 3.9.1 Containment Penetrations During Movement of Irradiated Fuel Assemblies

04.05 AP1000 Additional Systems Required for Operability

There are additional technical specification systems required for operability during shutdown conditions including:

- TS 3.4.14 RNS Suction Relief Valve LTOP
- TS 3.5.3 Core Makeup Tanks with the RCS Intact
- TS 3.5.5 Passive RHR Hx with the RCS Intact
- TS 3.5.7 IRWST – Mode 5
- TS 3.5.8 IRWST – Mode 6
- TS 3.6.6 Passive Containment Cooling – Mode 5 and 6 > MWt
- TS 3.6.7 Containment Closure During Shutdown
- TS 3.1.9 CVS Demineralized Water Isolation Valves and Two CVS Makeup Line Isolation Valves

04.06 AP1000 RCS Open or Closed Status

Many of the technical specifications for shutdown safety-related systems and availability controls for non-safety-related risk-significant systems are defined by RCS intact and RCS non-intact conditions. For the availability controls, the RCS is considered open when its pressure boundary is not intact. The RCS is also considered open if there is no visible level in the pressurizer. These definitions are different than the RCS open (POS 2) and RCS closed (POS 1).

END

Exhibits:

Exhibit 1 – User Guidance for Appendix G Phase 1: Initial Screening and Characterization of Findings

Exhibit 2 – Initiating Events Screening Questions

Exhibit 3 – Mitigating Systems Screening Questions

Exhibit 4 – Barrier Integrity Screening Questions

Exhibit 5 – External Events Screening Questions

Attachment:

Attachment 1: Revision History for IMC 0609 Appendix G Attachment 1

Exhibit 1 – User Guidance for Appendix G Phase 1: Initial Screening and Characterization of Findings

Step 1: Perform an initial screening of the inspection finding.

CAUTION: Most shutdown finding risk results are driven by the operator failure probabilities. When evaluating shutdown findings, it is important to be aware of any conditions or events that may impact operator response.

1.1 It is important to note that current fleet PWR designs do not have automatic safety actuation systems during shutdown (except for AP1000 plants, which maintain some automatic function). Also, in the current BWR designs there is no requirement to have the automatic low-level injection initiation functional in cold shutdown and refueling. Therefore, the risk significance of many findings will rely on operator's ability to diagnose the problem and perform appropriate actions. Successful operator actions are dependent on plant procedures, available time, complexity of the mitigation response, training, ability to diagnose the problem, etc. Therefore, when evaluating the initial screening of a shutdown finding, it is important to be aware of any conditions or events that may impact the operators' ability to diagnose and respond to a shutdown initiator. If you have any questions or are uncertain about an issue you are evaluating, contact your Regional SRA.

1.2 Determine the key safety functions, systems, and initiating events affected by the finding using the guidance in Table G1. This table attempts to collect all potential influences on both the human actions and equipment that can affect risk at shutdown. Inspectors should use the information in Table G1 to determine which, if any, categories of Exhibits 2–5 are influenced by specific findings.

PRA function refers to the ways in which the systems, structures, and components (SSCs) 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. IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power" has additional information.

1.3 Identify the applicable exhibit to use:

- ☐ Initiating Events (Exhibit 2)
- ☐ Mitigating Systems (Exhibit 3)
- ☐ Reactor Coolant System (RCS) Barrier (Exhibit 4)
- ☐ Fuel Barrier (Exhibit 4)
- ☐ Containment Barriers (Exhibit 4)
- ☐ External Events (Exhibit 5)

NOTE: When determining the significance of a finding that can be assessed using multiple exhibits, the inspector should use the exhibit that best reflects the dominant risk of the finding.

1.4 Continue to the appropriate exhibit to answer the screening questions. Once entering an exhibit, all questions should be answered. Use the decision logic in the exhibits when answering the screening questions to determine if the issue can be characterized as Green. The examples provided in the exhibits are not all inclusive. If you have any questions or are uncertain about an issue you are evaluating, contact your Regional SRA.

Step 2: If the finding screens as Green, then document in accordance with IMC 0611.

Step 3: If the finding screens as other than Green, perform an Appendix G Phase 2 or Phase 3 analysis as directed by the screening questions in Exhibits 2-5. Any finding that screens other than Green in this attachment is a preliminary assessment and not necessarily the final significance. Program guidance in IMC 0609, Attachment 5, “Inspection Finding Review Board” and IMC 0609, Attachment 1, “Significance and Enforcement Review Panel Process,” should be followed to determine if a planning SERP is required before committing significant resources to a detailed risk evaluation in a Phase 3 analysis.

Table G1 Generic Shutdown Key Safety Functions and System Dependencies[1]			
Safety Function	Major Systems	Supporting Systems	Initiating Event Scenarios
Decay Heat Removal	• Residual Heat Removal • Decay Heat Removal • Shutdown Cooling • Steam Generators (PWR) • Feed and Bleed (Low Pressure Injection, High Pressure Injection, Charging System) (PWR) • Control Rod Drive System (BWR) • Core Spray (BWR) • Passive RHR Heat Exchanger (AP1000) • In-Containment Refueling Water Storage Tank (AP1000) • Automatic Depressurization System • Normal Residual Heat Removal System (RNS in AP1000)	• AC Power • DC Power • RHR/DHR Heat Exchanger • Component Cooling Water (PWR) • Power Operated Relief Valves (PWR) • Instrumentation (i.e., RCS Level, RHR/DHR Heat Exchanger inlet/outlet Temperature, RHR/DHR Flow Indication, Core Exit Thermocouples) (PWRs with reactor head installed only) • Residual Heat Removal Service Water (BWR) • Safety Relief Valves (BWR) • Training • Procedures • Time to Boil and Time to Core Uncovery	• Loss of RHR (LORHR) • Loss of SDC (LOSDC) • Loss of Offsite Power (LOOP) • Loss of Inventory (LOI) • Overdrain (OD) (PWR) • Loss of Level Control[2] (LOLC) (PWR only) • Loss of Component Cooling Water (CCW) (PWR) Loss of Residual Heat Removal Service Water (RHRSW) (BWR)

1 This table is not intended to be all-inclusive. It is intended to give the inspector an overview of important systems and key safety functions to consider when characterizing the shutdown finding.

2 Loss of level control requires a Phase 2 or Phase 3 evaluation if:

(1) inadvertent loss of 2 feet of RCS inventory when not in mid-loop OR
(2) inadvertent loss of 2 inches of RCS inventory when in mid-loop conditions, OR
(3) inadvertent entry into reduced inventory or mid-loop conditions.

Table G1 Generic Shutdown Key Safety Functions and System Dependencies[1]			
Safety Function	Major Systems	Supporting Systems	Initiating Event Scenarios
Inventory Control	• Low Pressure Injection • High Pressure Injection • Charging System (PWR) • Control Rod Drive System (BWR) • Core Spray (BWR) • Automatic Isolation Capability of CVS Letdown Valves (AP1000) • Core Makeup Tanks (CMT) with RCS intact; Tech Spec 3.5.3 (AP1000) • Automatic Depressurization System (ADS) LCO 3.4.12 – RCS intact and 3.4.13 - RCS open (AP1000) • Inside Containment Refueling Water Storage Tank (IRWST) in Modes 5 and 6 (AP1000)	• Drain Down Isolation Valve(s) • AC Power • DC Power • RHR/DHR Heat Exchanger • RHR/DHR Relief Valves • Power Operated Relief Valves (PWR) • Instrumentation (i.e., RCS Level, RHR/DHR Heat Exchanger inlet/outlet Temperature and RHR/DHR Flow Indication, Core Exit Thermocouples (PWRs with reactor head installed only)) • Safety Relief Valves (BWR) • Training • Procedures • Time to Boil and Time to Core Uncovery	• Loss of Inventory (LOI) • Overdrain (OD) (PWR) • Loss of Level Control (LOLC) (PWR only) • Loss of Off-site Power (LOOP)
Electric Power Availability	• Emergency Diesel Generators • Onsite Standby Diesel Generators (AP1000) • Offsite Power Feeds • Offsite Transformers	• AC and DC Buses • Batteries and Battery Charges • Motor Generators • Inverters • Training • Procedures • Time to Boil and Time to Core Uncovery	• All Initiators

Table G1 Generic Shutdown Key Safety Functions and System Dependencies[1]			
Safety Function	Major Systems	Supporting Systems	Initiating Event Scenarios
Reactivity Control	• RPS • Control rod and associated drive mechanisms • Chemical and Volume Control System (PWR) • Standby Liquid Control (BWR)	• AC Power • DC Power • Nuclear Instrumentation • Training • Procedures • Time to Boil and Time to Core Uncovery	• Reactivity (inadvertent criticality)
Containment	• Hydrogen Control • Containment Closure Capability for both existing PWR and AP1000 • Penetrations • Passive Core Cooling System (PXS) only in modes 5 and 6 with greater than 6MWt (AP1000)	• AC Power • DC Power • Motive Power to close Hatches (assuming loss of AC power) • Temporary closures/penetrations • Training • Procedures • Time to Boil and Time to Core Uncovery	• All Initiators

Exhibit 2 - Initiating Events Screening Questions

A. Shutdown Initiators

1. Does the finding increase the likelihood of a shutdown initiating event?

   ☐ If YES ➛ Stop. Go to Appendix G Phase 2.

   ☐ If NO, continue.

B. Loss of Coolant Accident – Loss of Inventory (LOI) Initiators

2. Did a LOI event result in a leakage such that if the leakage were undetected and/or unmitigated it would cause the currently operating decay heat removal method to fail in 24 hours or less (e.g., level would drop to below the hot leg suction of the operating decay heat removal pump (PWR), or to the shutdown cooling isolation low level setpoint (BWR))?

   ☐ If YES ➛ Stop. Go to Appendix G Phase 2.

   ☐ If NO, continue.

3. Is the LOI event self-limiting such that leakage will stop before impacting the operating method of decay heat removal?

   ☐ If YES, continue.

   ☐ If NO ➛ Stop. Go to Appendix G Phase 2.

C. Transient Initiators

4. LOOP - Did the initiator occur when refuel canal/cavity was flooded?

   ☐ If YES, continue.

   ☐ If NO ➛ Stop. Go to Appendix G Phase 2.

5. LOOP – (additional PWR question) Did the initiator occur when refuel canal/cavity was flooded and the upper internals are still installed?[3]

   ☐ If YES ➛ Stop. Go to Phase 3.

   ☐ If NO ➛ continue.

3 This question addresses a concern regarding insights from NUREG/CR-5820, “Consequences of the Loss of the Residual Heat Removal Systems in Pressurized Water Reactors.” That report found that significant voiding can occur in the upper plenum and for some plants with limited flow through the upper support plate, there is a potential for core uncovery.

6. LOOP - Did the initiator occur when the time to boil off RCS inventory to the top of active fuel (TAF) was shorter than the time to recover offsite power?

   ☐ If YES ➛Stop. Go to Appendix G Phase 2.

   ☐ If NO, continue.

7. LORHR - Did the initiator occur when refuel canal/cavity was flooded?

   ☐ If YES, continue.

   ☐ If NO ➛ Stop. Go to Appendix G Phase 2.

8. LORHR (additional PWR question) - Did the initiator occur when refuel canal/cavity was flooded and the upper internals are still installed?[4]

   ☐ If YES ➛Stop. Go to Phase 3.

   ☐ If NO, continue.

9. Loss of Level Control (LOLC) or Over Drain (OD) - For PWRs, did the initiator occur when reactor level was in reduced inventory?

   ☐ If YES ➛Stop. Go to Appendix G Phase 2.

   ☐ If NO, continue.

D. External Event Initiators

10. Does the finding increase the likelihood of a fire or internal/external flood that could cause a shutdown initiating event?

   ☐ If YES ➛ Stop. Go to Phase 3.

   ☐ If NO, screen as Green.

4 This question addresses a concern regarding insights from NUREG/CR-5820, "Consequences of the Loss of the Residual Heat Removal Systems in Pressurized Water Reactors." That report found that significant voiding can occur in the upper plenum and for some plants with limited flow through the upper support plate, there is a potential for core uncovery.

Exhibit 3 – Mitigating Systems Screening Questions

A. Mitigating System Structure Component (SSC) and PRA Functionality

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?

   ☐ If YES, screen as Green.

   ☐ If NO, continue.

2. Does the finding represent a loss of system safety function? Examples of system safety function are listed in IMC 0609, Appendix G and in Table G1 of this attachment; however, they include decay heat removal, inventory control, electric power availability, reactivity control, and containment.

   ☐ If YES ➛Stop. Go to Appendix G Phase 2.

   ☐ If NO, continue.

3. Does the finding represent an actual loss of safety function of at least a single train for greater than its technical specification (TS) Allowed Outage Time, OR two separate safety systems out-of-service for greater than their TS Allowed Outage Time?

   ☐ If YES ➛ Stop. Go to Appendix G Phase 2.

   ☐ If NO, continue.

4.a. If the cavity is flooded, does the finding represent an actual loss of safety function of one or more non-TS trains of equipment during shutdown designated as risk-significant (e.g., 10CFR50.65), for greater than 24 hours?

   ☐ If YES ➛ Stop. Go to Appendix G Phase 2.

   ☐ If NO, continue.

4.b. If the cavity is not flooded, does the finding represent an actual loss of safety function of one or more non-TS Trains of equipment during shutdown designated as risk-significant (e.g., 10CFR50.65), for greater than 4 hours?

   ☐ If YES ➛ Stop. Go to Appendix G Phase 2.

   ☐ If NO, continue.

5.a. For PWRs, does the finding degrade RCS level indication and/or core exit thermal couples (CETs) when the cavity is not flooded?

   ☐ If YES ➛ Stop. Go to Appendix G Phase 2.

   ☐ If NO, continue.

5.b. For BWRs, does the finding degrade a functional auto-isolation, regardless of whether it is required to be operable or not, of RHR on low reactor vessel level?

☐ If YES ➛ Stop. Go to Appendix G Phase 2.

☐ If NO, continue.

6. Does the finding involve an open, cold leg penetration without an adequate, large hot leg vent path (such as a steam generator plenum manway)? These types of finding are a concern due to the potential of creating a hot leg to cold leg differential pressure that could force water out of the core. Vent paths in the pressurizer or reactor vessel head are often not adequate to prevent pressurization of the reactor coolant system after the boiling point is reached. Information Notice 88-36 provides more information.

☐ If YES ➛ Stop, go to Phase 3.

☐ If NO, continue.

B. External Event Mitigating Systems (Seismic/Flood/Severe Weather Protection Degraded)

7. Does the finding involve a seismic, flooding, or severe weather initiating event?

☐ If YES ➛ Go to Exhibit 5.

☐ If NO, continue.

8. Does the finding involve issues related to fire protection, fire brigade, fire hoses, fire extinguishers, or hose stations (question 9-11)?

☐ If YES ➛ Continue with part C below.

☐ If NO, then screen as Green.

C. Fire Protection

9. Does the finding involve fire brigade training, qualification, 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.

10. 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.

11. 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 C.9.a, C.10.a, or C.11.a are checked ➛ Stop. Go to IMC 0609, Appendix M.

Exhibit 4 – Barrier Integrity Screening Questions

A. RCS or Fuel Barrier

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

   ☐ If YES ➛ Perform a phase 3 and contact the appropriate technical branch in NRR/DMLR/Vessels and Internals Branch (MVIB).[5]

   ☐ If NO, continue.

2. Does the finding only involve fuel bundle misplacement or misorientation in the reactor core?

   ☐ If YES, screen as Green.

   ☐ If NO, continue.

3. Low Temperature Over Pressurization (LTOP) – For PWRs, does the finding involve either an (1) inadvertent safety injection actuation, or (2) the unavailability of a PORV or LTOP relief valve or their associated setpoints during LTOP operations when LTOP is required?

   ☐ If YES ➛ Stop. Go to Phase 3 and contact the appropriate technical branch in NRR/DMLR/Vessels and Internals Branch (MVIB).

   ☐ If NO, continue.

4. Freeze Seal – Does the finding increase the potential for failure of the freeze seal or if unmitigated have the potential to cause a disruption in RHR/DHR or a LOI event?

   ☐ If YES ➛ Stop. Go to Phase 3.

   ☐ If NO, continue.

5. Steam Generator (SG) Nozzles Dams – For PWRs, does the finding involve improper SG nozzle dam installation (e.g., hot leg manway must be opened first, hot leg SG nozzle dam installed last), inadequate SG nozzle dam RCS vent path, deficiencies of the SG nozzle dams (Ref GL 88-17 and IN 88-36), or SG nozzle dam functionality?

   ☐ If YES ➛ Stop. Go to Phase 3.

   ☐ If NO, continue.

5 Violations of 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.

6.a. Criticality – For PWRs, does the finding involve the potential for, or an actual, RCS boron dilution event?

- ☐ <u>If YES ➛ Stop. Go to IMC 0609, Appendix M.</u>
- ☐ If NO, continue.

6.b. Criticality – For BWRs, does the finding involve two or more adjacent control rods with the potential for, or an actual, addition of positive reactivity?

- ☐ <u>If YES ➛ Stop. Go to IMC 0609, Appendix M.</u>
- ☐ If NO, continue.

7. Drain Down Path or Leakage Path - Does the finding degrade the ability to isolate a drain down or leakage path?

- ☐ <u>If YES ➛ Stop. Go to Phase 3.</u>
- ☐ If NO, continue.

B. <u>Containment Barrier</u>

8. Does the finding degrade the ability to close or isolate the containment (this includes but is not limited to equipment and personnel hatches and permanent and temporary penetrations)?

- ☐ <u>If YES ➛ Stop. Go to IMC 0609, Appendix H.</u>
- ☐ If NO, continue.

9. Does the finding degrade the physical integrity of reactor containment (valves, penetrations, containment isolation components)?

- ☐ <u>If YES ➛ Stop. Go to IMC 0609, Appendix H.</u>
- ☐ If NO, continue.

10. Does the finding involve an actual reduction in function of hydrogen control for a BWR Mark III containment, a PWR ice condenser containment or an AP1000 containment?

- ☐ <u>If YES ➛ Stop. Go to IMC 0609, Appendix H.</u>
- ☐ If NO, screen as Green.

Exhibit 5 – External Events Screening Questions

1. If the equipment or safety function (examples of safety function are decay heat removal, inventory control, reactivity control, and containment) is assumed to be completely 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 any of the initiating events used by Table G1 for the plant in question;
    - would degrade two or more trains of a multi-train safety system or function, or would degrade the only available train, which would defeat the entire safety function;
    - would degrade one or more trains of a system that supports a safety system or function.

    ☐ If YES ➛ STOP. Go to Phase 3.

    ☐ If NO, continue.

2. Does the finding involve the total loss of any safety function, identified through a probabilistic risk assessment, Individual Plant Examination External Events, or similar analysis, that contributes to external event-initiated core damage accident sequences (i.e., initiated by a seismic, flooding, or severe weather event)?

    ☐ If YES ➛ STOP. Go to Phase 3.

    ☐ If NO, screen as Green.

Attachment 1: Revision History for IMC 0609 Appendix G, Attachment 1

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)
N/A	ML041470333 05/25/04 CN 04-015	Initial issuance	N/A	N/A
	ML13050A934 05/09/14 CN 14-011	IMC 0609 App G, Att. 1 is revised to enhance the usability of this appendix, based on feedback received from the SRA. The formatting was updated to be consistent with IMC 0609 Appendix A. The checklists from the previous revision, for PWRs and BWRs, were combined into one list in the various Exhibits in the attachment using screening questions and decision logic. The content was updated and reworked to be more user-friendly for inspectors to screen findings to determine if they are Green or a more detailed analysis is needed. Incorporated feedback from ROPFF 0609G1-1911 and 0609G-1323. This is a complete reissue no red line.	N/A	ML13162A640 0609G-1323 ML14120A177 0609G1-1911 ML1412A166

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)
N/A	ML19094B822 01/08/20 CN 20-004	Exhibit 4, steps A.1 and 3 were modified to include instructions to contact the EVIB branch staff for evaluating violations of the P-T limits curve. And an informational note with supplemental information was added. Exhibit 2 steps 5 and 8 were changed to require analysts to perform a Phase 3 analysis on a PWR if the cavity is flooded but the upper internals have not been removed. This reflects insights from NUREG/CR-5820 which found the potential for core damage to occur even with the cavity flooded in certain situations. Exhibit 3 step 6 is a new step added intending to direct users to a Phase 3 analysis for findings involving an open cold leg penetration without an adequate vent. This concern was discussed in Information Notice 88-36. Added a note to section 2 “Entry Conditions” notifying users that Appendix G is not appropriate for issues related to a spent fuel pool and to refer to IMC 0609 Appendix A, Exhibit 3 “Barrier Integrity Screening Questions”. Revised section 1.4 to state that once entering an exhibit, all questions should be answered. (continued next page)	N/A	ML19156A184 0609G1-2071 ML19112A186 0609G1-2097 ML19112A187 0609G-2086 ML19112A191 0609G1-2171 ML19112A188 0609G2-2098 ML19112A189

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		(continued from previous page) Added a new section 4 “Information about AP1000 Reactors” to highlight some design difference that affect shutdown risk. Added information to Table G1 to include certain systems applicable to the AP1000 design. Added a statement in Exhibit 1, step 1.1 letting users know that AP1000 have some automatic actuation functions during shutdown. Also added information that AP1000 problems that cannot be screened to Green must be assessed via the AP1000 SPAR model and not through Attachment 2. Screening questioned revised for consistency with recent changes to IMC 0609, Appendix A. Clarified on table G1 that a loss of level control only applies to a PWR and removed incorrect criteria for a loss of level control in a BWR. Step 3 of exhibit 1 was modified to remind users that anything that can’t be screened to Green in Attachment 1 is only a preliminary risk result. It may be appropriate to refer to IMC 0609 Significance Determination Process to determine if a planning SERP is required before performing a detailed risk analysis.		

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N/A	ML22096A212 05/20/22 CN 22-010	Exhibit 3, part C title was changed from Fire Brigade to Fire Protection since part C applies to more than just the fire brigade. A change was made to Exhibit 3 B. and question 8 was added that will screen to Green unless the issue is related to fire brigade, fire hoses, extinguishers, or hose stations. This was to avoid sending the reader to Appendix M inappropriately. Another change in Exhibit 4, question 3, removed the number “3” since it caused confusion since the elements contained in “3” for PORV/LTOP were contained within part “2”.	N/A	N/A