White Paper for Proposed Revision to NEI 99-02
Criteria for AP1000® Unplanned Scrams with Complications

Introduction:

This white paper highlights the need to update the Unplanned Scrams with Complications (USwC) performance indicator criteria in NEI 99-02 (Reference 1) for the **AP1000®** Nuclear Power Plant due to the operational and design differences associated with this new Generation III+ reactor type. Revisions to the criteria are proposed based on the unique design features of the AP1000, which relies on safety-related passive safety systems for decay heat removal and transient event mitigation. The proposed criteria better reflect the capabilities and characteristics of the AP1000 nuclear power plant and will establish appropriate criteria for increasing regulatory oversight when needed.

Background:

NEI 99-02 describes the purpose of the USwC performance indicator as, "This indicator monitors that subset of unplanned automatic and manual scrams that either require additional operator actions beyond that of the 'normal' scram or involve the unavailability of or inability to recover main feedwater. Such events or conditions have the potential to present additional challenges to the plant operations staff and therefore, may be more risk-significant than uncomplicated scrams."

Prior to operation of the first AP1000 in the United States, the NRC developed SECY-18-0091 (Reference 2) in recognition that the performance indicator processes described in NEI 99-02 would need to be updated due to this new type of Generation III+ reactor. Consideration of these operational and design differences resulted in the development of substantial changes to the USwC performance indicator as described in a previous white paper (Reference 3). These changes were subsequently incorporated into NEI 99-02, Revision 8, via Frequently Asked Question (FAQ) 21-01 (Reference 4). At that time, there was no available operating experience for the AP1000 in the United States. Recent operational events at Vogtle Unit 3, which included two countable USwC events, where the passive residual heat removal (PRHR) system and safeguards were actuated, have resulted in reconsideration of the criteria for the USwC performance indicator for the AP1000 (References 5 and 6).

The Generation III+ safety-related design of the AP1000 relies on passive safety features to remove decay heat and to mitigate transient events. The AP1000 also has non-safety-related active features that are credited for risk mitigation, such as startup feedwater (similar to Auxiliary or Emergency Feedwater in Generation II plants) and a normal residual heat removal system. This combination of passive and active capabilities gives the AP1000 a much lower risk profile as compared to existing Generation II power plants. The USwC performance indicator described in NEI 99-02 was originally developed for Generation II power plants, based on the known design and operation of these facilities. At the time of development of the performance indicator guidance in 1998 and 1999 there was substantial operational experience for the existing Generation II plants in the United States.

FAQ 21-01 provided the initial attempt to account for the differences between the AP1000 and the USwC performance indicator at that time. Now with additional operational experience for AP1000, this white paper proposes adjustment of the USwC performance indicator.

Proposed Changes for the USwC Performance Indicator Guidance for AP1000:

The current USwC performance indicator guidance for the AP1000, as described in NEI 99-02, is based on the following screening questions:

1. Did two or more control rods fail to fully insert?

2. Did the turbine fail to trip?
3. Was power lost to any battery backed Class 1E DC and UPS System (IDS) bus (For AP1000 only)?
4. Was a Safeguards Actuation signal received? (For AP1000 only)
5. Was Main Feedwater unavailable or not recoverable using approved plant procedures during the scram response?
6. Was the scram response procedure unable to be completed without entering another EOP?

These questions are evaluated for possible revision for AP1000, as follows:

Question 1: Did two or more control rods fail to fully insert?

The guidance in NEI 99-02 was modified to account for the key design difference of the AP1000, which is the use of Gray Rod Cluster Assemblies (GRCAs). The guidance appropriately states that the GRCAs are not considered when answering this question, because the GRCAs are not considered when determining the need to borate and the GRCAs are not considered in the Shutdown Margin (SDM) calculation while in MODE 1. Therefore, no improvement of the guidance for Question 1 is currently needed.

Question 2: Did the turbine fail to trip?

The guidance for this question in NEI 99-02 was not modified for AP1000. The design of the AP1000 includes the reactor trip input to turbine trip actuation similar to Generation II plants. Therefore, no improvement of the guidance for Question 2 is currently needed.

Question 3: Was power lost to any battery backed Class 1E DC and UPS System (IDS) bus (For AP1000 only)?

The guidance in NEI 99-02 was modified to account for the key design difference of the AP1000, which is the battery backed Class 1E DC and UPS System. Therefore, no improvement of the guidance for Question 3 is currently needed.

Question 4: Was a Safeguards Actuation signal received? (For AP1000 only)

Under FAQ 21-01, the guidance in NEI 99-02 was modified to account for the key design difference, which is the use of "Safeguards Actuation," which was judged to be the AP1000 equivalent for the "Safety Injection" signal criterion used for Generation II plants.

The Vogtle Electric Generating Plant (VEGP) Units 3&4 UFSAR Section 7.3.1.1 states that the safeguards actuation is initiated based on any of the following conditions:

1. Low-3 pressurizer pressure
2. Low-2 lead-lag compensated steam line pressure
3. Low-2 cold leg temperature
4. High-2 containment pressure
5. Manual initiation

During recent operating events where safeguards actuation signals were generated, the progression of the events was consistent with AP1000 passive safety system response. Specifically, both licensee event reports state that there were no safety consequences, and that decay heat was removed by the passive residual heat removal system. Description of AP1000 passive safety system response, based on the design of the AP1000, as described in the VEGP Units 3&4 UFSAR, is provided in the appendix to this white paper.

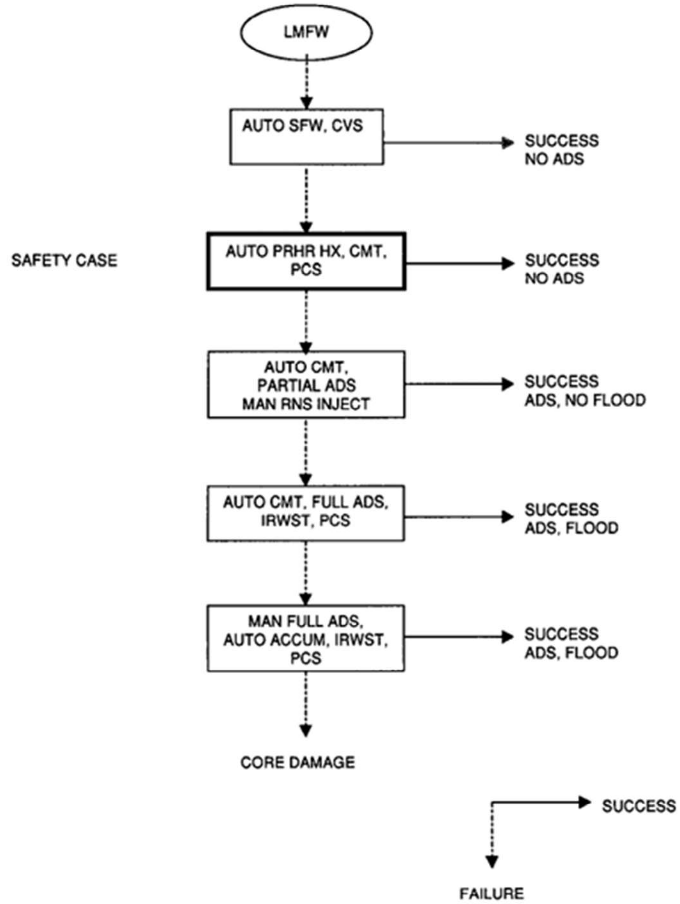
The purpose for the USwC performance indicator (PI), as described in Regulatory Information Summary (RIS) 2007-12 (Reference 7), was stated as follows, "The USwC is designed to identify facilities that are outliers in complications that can elevate the risk of an unplanned manual or automatic reactor trip or scram." RIS 2007-12 also describes the method to identify an USwC as, "The PI will monitor the following six actions or conditions that have the potential to complicate the post trip recovery; reactivity control, pressure control (boiling water reactors)/turbine trip (pressurized water reactors), availability of power to emergency buses, actuation of emergency injection sources, availability of main feedwater, and the use of emergency operating procedures to address complicated scrams."

The "actuation of emergency injection sources" for a Generation II PWR is not the same as AP1000 passive plant response to a safeguards actuation. The AP1000 plant response to a safeguards actuation is dependent on the plant conditions. The initial plant response would typically involve core makeup tank (CMT) outlet valves and PRHR valves opening. This provides for decay heat removal and allows CMT inventory to be available for any loss of RCS inventory. During some events, conditions can progress to the point of Automatic Depressurization System (ADS) actuation. This occurs if CMT level drops to the Low-3 setpoint or if an extended loss of AC power has occurred (i.e., approaching 24 hours). The actuation of ADS would indicate a more complex transient, when additional RCS cooling and injection of additional water from the accumulators and the in-containment refueling water storage tank (IRWST) could be needed. Therefore, the condition of ADS actuation for AP1000 could be considered similar to safety injection actuation for a Generation II plant.

Additionally, unlike Generation II plants which often require operator actions to prevent adverse consequences, the AP1000 is capable of maintaining reactor safety without operator intervention for 72 hours using passive safety systems. The AP1000 response to a plant transient, even with safeguards actuation, confirms the plant design and its ability to automatically respond to any design-basis transient. This is substantially different from Generation II plants that could result in undesirable event sequences if operator action is not taken. For example, a safety injection actuation event in a Generation II PWR design presents a potential challenge of overfilling the pressurizer that could lead to the pressurizer safety valve becoming stuck in the open position and a LOCA developing. In this scenario a time critical operator action may be required to prevent pressurizer overfill. This scenario for Generation II PWR operator action is not applicable to the AP1000.

The following figure outlines the levels of defense included in the design for a general transient event. This example of a general transient event is loss of main feedwater. This same structure of levels of defense is also considered with the structure for a Loss of Offsite power event (Figure 3.1-2 of Reference 8). As outlined in Section 18.12.2 of the VEGP Units 3&4 UFSAR, the AP600 system/event matrix (Reference 8) identifies the post-accident mitigation functions for AP600 to protect the integrity of these fission product barriers. This AP600 document is directly applicable to AP1000.

The abbreviations used in this figure are provided as follows: Loss of Main Feedwater (LMFW), Startup Feedwater (SFW), Chemical and Volume Control System (CVS), Passive Residual Heat Removal Heat Exchanger (PRHR HX), Core Makeup Tank (CMT), Automatic Depressurization System (ADS), Manual (MAN), Normal Residual Heat Removal System (RNS), In-containment Refueling Water Storage Tank (IRWST), Passive Containment Cooling System (PCS), and Accumulator (ACCUM).



During the recent events, the safeguards actuation was an expected plant response. Successful PRHR and CMT actuation places the plant in a safe stable state without additional operator actions or additional mitigation functions as outlined above (i.e., Success – No ADS).

Typical emergency operating procedure response for these types of low-risk events is entry into EOP-E-0, transition to E-1, followed by transition to ES-1.1 to stabilize the plant, placing the RCS on Normal Residual Heat Removal cooling, and cooling down to MODE 5, if needed. In the unlikely event that PRHR were to fail to operate, Startup Feedwater (SFW) could be manually established using applicable emergency operating procedures.

The at-power internal events PRA core damage frequency results for AP1000 identify general transient with main feedwater, general transient without main feedwater, and general transient with a safeguard signal all as having a Risk Achievement Worth (RAW) less than 2. It also identifies them all having a Fussell-Vesely (FV) importance measures all less than 0.02. These RAW values are low when compared to events that may require Automatic Depressurization Actuation like LOCAs and Steam line break events all with RAW values greater than 6.

Therefore, the use of "safeguards actuation" as an input parameter for the AP1000 USwC PI is not appropriate because "safeguards actuation" for AP1000 includes events that are low risk and do not require complex operator response. It is proposed that this be replaced by "Automatic Depressurization

Actuation," which is considered more analogous to the "safety injection actuation" criterion used for Generation II PWRs.

Question 5: Was Main Feedwater unavailable or not recoverable using approved plant procedures during the scram response?

FAQ 21-01 did not modify the guidance for this question in NEI 99-02 for AP1000. Although, there are significant differences between the AP1000 and Generation II PWRs. Specifically, the AP1000 ability to remove decay heat via the Passive Residual Heat Removal (PRHR) system provides safety-related redundancy to the non-safety-related heat removal via the Steam Generators. The AP1000 steam generators can receive feedwater from the Startup Feedwater System (similar to the Emergency Feedwater [EFW] in Generation II PWR plants) and from the Main Feedwater [MFW] System (similar to MFW in Generation II PWR plants). Thus, the AP1000 has three systems for decay heat removal after a reactor trip (PRHR, Startup Feedwater, and MFW), where Generation II PWRs typically only have two systems (EFW and MFW) that can be used for decay heat removal for avoidance of initiation of bleed and feed cooling. Additionally, the AP1000 PRHR system accomplishes the required safety functions by passive operation without operator actions required.

The NEI 99-02 basis information for Question 5 states, "Since all PWR designs have an emergency Feedwater system that operates, if necessary, the availability of the normal or main Feedwater systems as a backup in emergency situations can be important for managing risk following a reactor scram." This statement is not correct for AP1000. Specifically, the PRHR system is the design basis safety-related method of decay heat removal. The Startup and Main Feedwater Systems are non-safety-related alternatives for decay heat removal via the steam generators. This shows that the AP1000 design does not correspond with the basis for Question 5.

Therefore, Question 5 should be changed to account for the AP1000 design that has a Startup Feedwater system, a Main Feedwater system, and a safety-related PRHR system. Specifically, the NEI 99-02 guidance should be changed to state that an event is to be counted as an USwC if both the Startup Feedwater system and the Main Feedwater system are unavailable or not recoverable using approved plant procedures during the scram response. This includes allowance for recovery after safeguards actuation has occurred, based on reset of the safeguards actuation during the scram response which allows the recovery of the startup feedwater and main feedwater systems.

Question 6: Was the scram response procedure unable to be completed without entering another EOP?

The basis for Question 6 in NEI 99-02 states, "The criteria in this question is used to verify there were no other conditions that developed during the stabilization of the plant in the scram response that required re-entry into the EOPs or transition to a follow on EOP." The transitions to additional EOPs for Generation II plants are typically associated with the diagnosis of plant conditions that require additional operator actions. As previously noted in the evaluation of Question 4, the AP1000 is capable of maintaining reactor safety without operator intervention for 72 hours using passive safety systems. Therefore, EOP transitions for the AP1000 are not a clear indication of risk significance or complication of the unplanned scram response.

VEGP Units 3&4 UFSAR Section 7.4.1 describes the available methods for safe shutdown of the AP1000. It is expected that for most events where there is no significant loss of RCS inventory, the PRHR system can achieve and maintain safe shutdown with no operator action. As noted in the discussion of Question 4 above, operation of the ADS could be used as a criterion for determining if complications have occurred after a scram. For AP1000, the EOP-ES-1.3 is entered when Stage 1, 2, and 3 ADS actuation occurs and EOP-ES-1.4 is entered when Stage 4 ADS actuation occurs.

Therefore, the appropriate EOP criterion for this question would be entry into EOP-ES-1.3 or EOP-ES-1.4. Although, this criterion would be redundant to Question 4, regarding ADS actuation. Hence, the deletion of this question for AP1000 would be appropriate.

Conclusion:

Based on the preceding, the revised USwC performance indicator screening questions for the AP1000 are proposed as follows:

1. (Unchanged) Did two or more control rods fail to fully insert?
2. (Unchanged) Did the turbine fail to trip?
3. (Unchanged) Was power lost to any battery backed Class 1E DC and UPS System (IDS) bus (For AP1000 only)?
4. (Changed) Was an Automatic Depressurization Actuation signal received? (For AP1000 only)
5. (Changed) Were both Main Feedwater and Startup Feedwater unavailable or not recoverable using approved plant procedures during the scram response?
6. (Deleted) ~~Was the scram response procedure unable to be completed without entering another EOP?~~

The preceding information provides proposed changes to the USwC performance indicator for the AP1000. These proposed changes are based on recent operating experience and recognition of the substantial differences between Generation II PWRs and AP1000 event response and capabilities. These changes are expected to establish criteria that will achieve the goal of the USwC performance indicator, which is identification of AP1000 initiating event performance that should be subject to additional regulatory oversight.

References:

1. NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 8, October 2024.
2. SECY-18-0091, "Recommendations for Modifying the Reactor Oversight Process for New Large Light Water Reactors with Passive Safety Systems Such as the AP1000 (Generation Ill+ Reactor Designs)," September 12, 2018.
3. Reactor Oversight Process Whitepaper – Modification of the Description of Unplanned Scrams with Complications Performance Indicator to Reflect AP1000 Design, Revision 1, January 5, 2024 (NRC Accession No. ML24023A026).
4. FAQ 21-01: Unplanned Scrams with Complications for AP1000 Final Approved (NRC Accession No. ML21075A282).
5. LER 2024-002-00, "Manual Reactor Protection System and Automatic Safeguards Actuation Due to an Unexpected Change in Position of a Main Feedwater Pump Minimum Flow Control Valve," September 5, 2024.
6. LER 2024-003-00, "Automatic Reactor Protection System and Manual Safeguards Actuation Due to a Failed Open Flow Control Valve," November 18, 2024.
7. RIS 2007-12, "Changes to the Unplanned Scrams with Loss of Normal Heat Removal Performance Indicator," June 26, 2007.
8. WCAP-13793, "The AP600 System/Event Matrix," June 1994.

Appendix: Background Information for AP1000 Unplanned Scrams with Complications White Paper

Event Descriptions:

Two events occurred at Vogtle Unit 3 in July and September of 2024, which resulted in “white” performance indicator (PI) under the Initiating Event IE04 category, “Unplanned Scrams with Complications.” The first event occurred on July 8, 2024, and involved the failure of the main feedwater pump (MFWP) minimum flow control valve (FCV). The second event occurred on September 17, 2024, and involved one of the passive residual heat removal (PRHR) heat exchanger (HX) outlet FCVs failing open.

The event involving the MFWP minimum FCV is detailed in LER (Licensee Event Report) 2024-002-00 (Reference 1). On July 8, 2024, at 2125 EDT, the Vogtle Unit 3 reactor protection system (RPS) was manually actuated, followed by an automatic safeguards actuation. The cause of this event was the failure of the MFWP minimum FCV, which failed from its normally closed position to its fail-safe, fully open position. The change in valve position resulted in a decrease in flow to the steam generators (SGs). Operators tried to manually close the MFWP FCV from the control room but observed no response. The continued reduction in flow to the SGs caused water levels in the SG to fall below the threshold for a manual reactor trip. Immediately following the manual reactor trip, the SG #2 wide range level briefly dropped below the setpoint to automatically actuate the PRHR HX. Actuation of the PRHR HX caused rapid cooldown of the reactor coolant, resulting in the reactor coolant temperature dropping below the threshold for automatic safeguards actuation on low reactor coolant temperature. There were no safety consequences as a result of this safeguards actuation, the plant was stabilized in timely fashion and decay heat was removed by the PRHR HX. No safety related SSCs were inoperable at the beginning of or contributed to the event and all safety systems responded to the event as expected.

The event involving the PRHR HX FCV is detailed in LER 2024-003-00 (Reference 2). On September 27, 2024, at 0127 EDT, the Vogtle Unit 3 RPS was automatically actuated, followed by a manual safeguards actuation. The cause of the event was a loss of power to the air-operated solenoid on one of the PRHR HX FCV, causing the valve to transition from its normally closed position to its full-open, fail-safe position. The change in valve position caused an automatic trip of the reactor and PRHR HX actuation. The resulting rapid cooldown of reactor coolant resulted in the pressurizer water level falling below the limit to require manual actuation of the safeguards. There were no safety consequences as a result of this safeguards actuation, the plant was stabilized in timely fashion and decay heat was removed by the PRHR HX. No safety related SSCs were inoperable at the beginning of or contributed to the event and all safety systems responded to the event as expected.

Both of these events were determined to meet the criteria for USwC, which exceeded the Green/White threshold for the IE04 performance indicator. The criteria for USwC are described in NEI 99-02 (Reference 3), as follows:

- Did two or more control rods fail to fully insert?
    - Per NEI 99-02, “for AP1000 plants, Gray Rod Cluster Assemblies (GRCAs) are not considered when answering this question. ES-0.1 Step 10 Action/Expected Response (AER) checks all Rod Cluster Control Assemblies (RCCAs) fully inserted and uses this to determine whether or not to borate. However, the GRCAs are not considered when determining the need to borate, because GRCAs are not considered in the Shutdown Margin (SDM) calculation while in MODE 1 as displayed on On-Line Power

Distribution Monitoring System (OPDMS) (OPDMS displays Xe-free conditions with one stuck rod). For AP1000, the question would be 'Did two or more Black Control Rods (RCCAs) (excluding GRCAs) fail to fully insert?'" Neither of the events involved rods failing to fully insert.

- Did the turbine fail to trip?
  - Neither of the events involved the turbine failing to trip.
- Was power lost to any battery backed Class 1E DC and UPS System (IDS) bus (For AP1000 only)?
  - Power was not lost to any battery backed IDS bus.
- Was a Safeguards Actuation signal received? (For AP1000 only)
  - During both of these events, a safeguards actuation signal was received following the rapid cooldown of the reactor coolant resulting from PRHR HX actuation.
- Was Main Feedwater unavailable or not recoverable using approved plant procedures during the scram response?
  - Main feedwater was isolated based on the safeguards actuation. Although main feedwater would have been recoverable using approved plant procedures after reset of safeguards.
- Was the scram response procedure unable to be completed without entering another EOP?
  - Transition into the EOP-E-1 procedure was required for these events due to the safeguards actuation.

ADS Information:

The Automatic Depressurization System (ADS) functions to depressurize the Reactor Coolant System (RCS) to allow safety injection from the Passive Core Cooling System (PXS). Section 5.4.6 of the VEGP 3&4 UFSAR states, "Opening of the automatic depressurization system valves is required for the passive core cooling system to function as required to provide emergency core cooling following postulated accident conditions." ADS actuation is an important component of safety injection from the PXS. VEGP 3&4 UFSAR Section 6.3.3 states, "As the core makeup tanks drain down, the automatic depressurization system valves are sequentially actuated. The depressurization sequence establishes reactor coolant pressure conditions that allow injection from the accumulators, and then from the in-containment refueling water storage tank and the containment recirculation path. Therefore, an injection source is continually available." ADS is aligned with PXS to ensure passive injection is available throughout an emergency. However, safeguards actuation does not always lead to ADS actuation depressurizing the RCS. The ADS actuation is based on reduction in CMT volume. In recent events, ADS actuation did not occur because core makeup tank level did not drop below the Low-3 setpoint.

As stated above, one of the current criteria for an AP1000 plant to experience a countable USwC for PI IE04, as described in NEI 99-02, is to receive a safeguards actuation signal. NEI 99-02 states that the purpose of this criteria "is to determine if the operator had to respond to an abnormal condition that required passive safety injection or respond to the actuation of additional equipment that would not normally actuate on an uncomplicated scram. This question would include any condition that challenged Reactor Coolant System (RCS) inventory, pressure, or temperature severely enough to require passive safety injection” (Reference 3). The RCS pressure boundary was not challenged in these recent events and ADS was not actuated.

Inadvertent Safeguards Actuation:

Inadvertent safeguards actuation transients have a minimal impact on the RCS. A good example of this is on a loss of normal feedwater, where there is a decrease in heat removal by the secondary system. UFSAR Section 15.2.7.1 states, “A loss of normal feedwater (from pump failures, valve malfunctions, or loss of ac power sources) results in a reduction in the capability of the secondary system to remove the heat generated in the reactor core. If startup feedwater is not available, the safety-related PRHR heat exchanger is automatically aligned by the protection and safety monitoring system to remove decay heat.”

Specifically, according to UFSAR Section 15.2.7.1, a loss of normal feedwater results in a steam generator water inventory decrease as a consequence of the continued steam supply to the turbine. The mismatch between the steam flow to the turbine and the feedwater flow leads to the reactor trip on a Low-2 steam generator narrow range water level signal. The same signal also actuates the startup feedwater system. If the startup feedwater is not available, the PRHR HX is actuated on a Low-2 SG narrow range water level (coincident with a Low-2 startup feedwater flow rate signal) or a Low-2 SG wide range water level signal (UFSAR Section 15.2.7.1).

The analysis described in UFSAR Section 15.2.7.3 concludes, “…a loss of normal feedwater or a loss of normal feedwater with a consequential loss of ac power to the plant auxiliaries do not adversely affect the core, the reactor coolant system, or the steam system. The heat removal capacity of the PRHR heat exchanger and the steam generator safety valves is such that reactor coolant water is not relieved from the pressurizer safety valves. DNBR always remains above the design limit values, and reactor coolant system and steam generator pressures remain below 110 percent of their design values.”

The actuation of the PRHR HX does not mean there is a loss of coolant accident. UFSAR Section 6.3.1.1.4 states, “The functional requirements for the passive core cooling system specify that the plant be brought to a safe, stable condition using the passive residual heat removal heat exchanger for events not involving a loss of coolant.” The PRHR HX is credited with removing decay heat in non-LOCA scenarios and “in most sequences, the operators would return the plant to normal system operations and terminate passive system operation within several hours in accordance with the plant emergency operating procedures” (UFSAR Section 6.3.1.1.4).

The inadvertent actuation of the PRHR HX is a scenario covered in the UFSAR Section 15 Accident Analyses. UFSAR Section 15.1.6 states, “The inadvertent actuation of the PRHR heat exchanger could be caused by operator error or a false actuation signal, or by malfunction of a discharge valve. Actuation of the PRHR heat exchanger involves opening one of the isolation valves, which establishes a flow path from one reactor coolant system hot leg, through the PRHR heat exchanger, and back into its associated steam generator cold leg plenum.” Once either PRHR HX valve is not fully closed the reactor trip setpoint is reached, this is followed by rod motion and full rod insertion (UFSAR Table 15.1.2-1). The result of this sequence is that the reactor is tripped before the increase in reactivity resulting from the cooling of the RCS by the PRHR HX. Further, UFSAR Section 15.1.6.2 states, “Since a reactor trip is initiated as soon as the PRHR discharge valves are not fully closed, this event is essentially a reactor trip from the initial condition and requires no separate transient analysis,” and Section 15.1.6.3 states, “Inadvertent actuation of the PRHR does not result in violation of the core thermal design limits (DNB and linear power generation) or RCS overpressure.”

Safeguards Actuation Transients:

Per UFSAR Section 3.9.1.1 Design Transients, both of these events fall within Level B events. Per Section 3.9.1.1, Level B transients are upset conditions or incidents of moderate frequency which "include any deviations from Level A service conditions anticipated to occur often enough that the design includes a capability to withstand the conditions without operational impairment."

The event involving the MFWP minimum FCV falls under the UFSAR Section 3.9.1.1.2.4 Case C - Reactor Trip with Cooldown and Safeguards Actuation. UFSAR Section 3.9.1.1.2.4 Case C, states, "…it is assumed that the steam generator secondary side shrinkage is sufficient to actuate the passive residual heat removal heat exchanger in addition to the PLS starting of the startup feedwater pumps. The core makeup tanks are then actuated on Low-2 pressurizer water level signal. Main and startup feedwater are assumed to continue until isolated on Low-2 T_{cold}. After 30 minutes, the plant is expected to be manually controlled to either cold shutdown or returned to the no-load conditions." This transient is assumed to have 20 occurrences.

The event involving the failure of the PRHR FCV falls under UFSAR Section 3.9.1.1.2.7 Inadvertent Safeguards Actuation. UFSAR Section 3.9.1.1.2.7 states, "A spurious actuation of the passive residual heat removal heat exchanger isolation valves or the core makeup tank valves causes cold reactor coolant to flow into the reactor coolant system. Rapid changes in the temperature of the core makeup tank or passive residual heat removal heat exchanger and associated piping occur. Ten events of this limited transient are postulated."

Both of the Vogtle Unit 3 events which triggered the USwC performance indicator criteria were included in AP1000 plant design, and while they are not expected to occur frequently, the plant was designed so that it could be recovered from Level B transients without impairment.

Generation II PWR vs. AP1000 Plant Design Features:

One of the main differences between Generation II plant and the AP1000 plant design is the function of the core and containment cooling systems, which are active systems in operating plants but passive systems in AP1000 plants.

At Generation II PWR plants, there are varying methods for containment heat removal. The safety functions of the systems responsible for containment heat removal are manually actuated by operators upon receiving a safety injection (SI) signal.

The system credited with core cooling for a Generation II PWR is the emergency core cooling system (ECCS). The ECCS is designed to cool the reactor core and provide additional shutdown capability during loss-of-coolant accidents (LOCA), loss-of-secondary accidents, and a SG tube rupture accident. The accidents that result in actuation of the ECCS are further broken down into the following:

- Increase in heat removal by the secondary system – resulting from an inadvertent opening of a SG safety valve or power-operated relief valve, or a steam system piping failure

- Decrease in heat removal by the secondary system – resulting from a feedwater system piping failure

- Decrease in RCS inventory – resulting from SG tube rupture, LOCA, or loss of coolant due to a rod cluster control assembly ejection accident

Each of these accidents results in the generation of an SI signal and ECCS operation. An SI signal is generated upon pressurizer low pressure, steam line low pressure, containment high pressure, or manual actuation.

For AP1000, the passive containment cooling system (PCS) is credited with containment heat removal. The PCS is an engineered safety features system which reduces containment temperature and pressure following a LOCA or main steam line break accident. The PCS also serves as the means of transferring heat to the safety-related ultimate heat sink for other events involving significant pressure and temperature change.

The VEGP Units 3&4 UFSAR Section 6.2.2.2.2, states, "The major components of the passive containment cooling system are: the passive containment cooling water storage tank (PCCWST) which is incorporated into the shield building structure above the containment; an air baffle, located between the steel containment vessel and the concrete shield building, which defines the cooling air flowpath; air inlets and an air exhaust, also incorporated into the shield building structure; and a water distribution system, mounted on the outside surface of the steel containment vessel, which functions to distribute water flow on the containment. A passive containment cooling ancillary water storage tank and two recirculation pumps are provided for onsite storage of additional passive containment cooling system cooling water, to transfer the inventory to the passive containment cooling water storage tank, and to provide a back-up supply to the fire protection system (FPS) seismic standpipe system." Additionally, the VEGP Units 3&4 UFSAR Section 6.2.2.2.4, states that the operation of the PCS is initiated upon receiving 2 out of 4 of the High-2 containment pressure signals or upon manual initiation from the operator. System actuation involves opening the passive containment cooling water storage tank isolation valves, allowing water from the tank to be distributed over the top of the containment shell. The flow of water from the tank is completely driven by gravity.

At Vogtle Units 3&4, the passive core cooling system (PXS) is credited with emergency core decay heat removal, RCS emergency makeup and boration, safety injection, and containment pH control. Per VEGP Units 3&4 UFSAR Section 6.3, the PXS is designed to operate without the use of active equipment. The events that lead to PXS actuation are broken down in Section 6.3.3:

- Increase in heat removal by the secondary system – resulting from inadvertent opening of a SG power-operated atmospheric steam relief or safety valve or from a steam system piping failure

- Decrease in heat removal by the secondary system – resulting from loss of main feedwater flow or from a feedwater system piping failure

- Decrease in RCS inventory – resulting from an SG tube rupture, a loss of coolant due to rod cluster control assembly ejection accident, or a LOCA

- Shutdown events – resulting from loss of startup feedwater, loss of normal residual heat removal system (RNS) with RCS pressure boundary intact, loss of RNS during mid-loop operation, or loss of RNS with refueling cavity flooded

PXS operation varies depending on the severity and type of event. UFSAR Section 6.3.3 states, "For non-LOCA events, the passive residual heat removal heat exchanger is actuated so that it can remove core decay heat. The passive residual heat removal heat exchanger can operate for at least 72 hours after initiation of a design basis event to satisfy Condition I, II, III, and IV safety evaluation criteria described in the relevant safety analyses." Where Condition I is normal operation and operational transients, Condition II are faults of moderate frequency, Condition III are infrequent faults, and Condition IV are limiting faults (UFSAR Section 15.0.1). UFSAR Section 6.3.3 also states, “For loss of coolant accidents, the core makeup tanks deliver borated water to the reactor coolant system via the direct vessel injection nozzles. The accumulators deliver flow to the direct vessel injection line whenever reactor coolant system pressure drops below the tank static pressure. The in-containment refueling water storage tank provides gravity injection once the reactor coolant system pressure is reduced to below the injection head from the in-containment refueling water storage tank. The passive core cooling system flow rates vary depending upon the type of event and its characteristic pressure transient.” The type of event being experienced by the plant dictates the PXS response ranging from PRHR HX activation for a non-LOCA event to full safety injection for a severe LOCA event.

Relating to PRHR HX activation, Section 6.3.3 states that the PRHR HX is actuated for non-LOCA events on the following setpoints:

- Low-2 steam generator narrow range water level, coincident with startup feedwater Low-2 flow
- Low-2 steam generator wide range water level
- Core makeup tank actuation
- Automatic depressurization actuation
- Pressurizer water level - High 3
- Manual actuation.

For non-AP1000 operating plants, manual operator action is normally required following events where a safety injection signal occurs. The AP1000 design allows for manual operator action to block actuation of the automatic depressurization system (ADS) should actuation be deemed unnecessary based on reactor coolant system conditions. Otherwise, the ADS would be automatically actuated after about 22 hours, which is prior to depletion of the 24-hour Class 1E batteries, if AC power is not available during a non-LOCA event (UFSAR Section 6.3.2.8).

Unplanned Scrams with Complications:

The following is the summary of the purpose for Unplanned Scrams with Complications (USwC) from Section 2.1 of NEI 99-02 (Reference 3):

> Purpose: This indicator monitors that subset of unplanned automatic and manual scrams that either require additional operator actions beyond that of the normal scram or involve the unavailability of or inability to recover main feedwater. Such events or conditions have the potential to present additional challenges to the plant operations staff and therefore, may be more risk-significant than uncomplicated scrams.

The following is the indicator definition for Unplanned Scrams with Complications (USwC) from Section 2.1 of NEI 99-02 (Reference 3):

> Indicator Definition: The UswC indicator is defined as the number of unplanned scrams while critical, both manual and automatic, during the previous four quarters that require additional operator actions or involve the unavailability of or inability to recover main feedwater as defined by the applicable flowchart (Figure 2) during the scram response (see definition of scram response in the Definitions of Terms section) and the associated flowchart questions.

Both events did not present the additional challenges of ADS actuation to the plant operations staff. Both events did not result in a risk-significant initiating event. In both events decay heat removal was achieved. During both events the levels of defense available to prevent core damage were maintained.

Therefore, the criteria for AP1000 USwC should be appropriately modified as proposed in this white paper.

Appendix References:

1. Licensee Event Report 2024-002-00, “Manual Reactor Protection System and Automatic Safeguards Actuation Due to an Unexpected Change in Position of a Main Feedwater Pump Minimum Flow Control Valve,” September 2024.
2. Licensee Event Report 2024-003-00, “Automatic Reactor Protection System and Manual Safeguards Actuation Due to a Failed Open Flow Control Valve,” November 2024.
3. NEI 99-02, Revision 8, “Regulatory Assessment Performance Indicator Guideline,” October 2024.
4. VEGP 3&4 USFAR, Revision 12.2, July 2024.