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U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Duke Energy Carolinas, LLC
Oconee Nuclear Station, Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287,
Renewed Operating Licenses DPR-38, DPR-47, and DPR-55
Tornado and High Energy Line Break (HELB) Mitigation License Amendment Requests (LARs) - Supplemental Responses to Request for Additional Information (RAI) Nos. 61, 62, and 107

By letter dated December 16, 2011, Duke Energy Carolinas, LLC (Duke Energy) responded to a Nuclear Regulatory Commission (NRC) RAI. As described in that response, the following supplemental information was to be provided for the following RAI items:

- RAI 61 Add details on actions required in Mode 3 within 72 hours to transition to Mode 4 including previous licensing references,
- RAI 62 Provide information regarding industry code and standard applicability to the design of the Protected Service Water (PSW) System,
- RAI 107 Include PSW Technical Specification and Bases contingency actions.

The Enclosure to this submittal contains the original and the additional information for each of the aforementioned RAI items. Changed/added text is denoted by revision bars in the right hand margin.

Based on the status of these LARs, Duke Energy believes the next course of action would be to meet with the Staff for follow-up discussions. This meeting should expedite completion of the review of the LARs.

If you have any questions in regard to this letter, please contact Stephen C. Newman, Regulatory Compliance Senior Engineer, Oconee Nuclear Station, at (864) 873-4388.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 20, 2012.

Sincerely,

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Enclosure

A001
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Enclosure

Supplemental Responses to RAI Nos. 61, 62, and 107

RAI 61

Provide a complete event sequence following a tornado and/or an HELB. In your response, provide a summary of the sequence of actions required for achieving each phase of the mitigating strategies showing how MODES of operation 3, 4, and 5 as defined in Table 1.1-1 of your Technical Specifications (TSs) will be achieved. Identify all equipment that will be available following the events to mitigate a tornado and/or an HELB. Identify all required operator actions, and when each operator action has to be completed. Identify all required repairs, and when they must be completed. For any necessary repairs provide the actions required, manpower required, and all equipment and parts that are necessary to complete the repairs. Demonstrate that the sequence of events would not result in unacceptable radiological consequences. Identify and justify the selected acceptance criteria (fission product barriers are maintained, and the spent fuel remains within the licensing basis). Provide a detailed description of the analyses performed to support the conclusions.

Duke Energy Response

HELB

All of the postulated HELBs outside containment are described in ONDS-351, Rev. 2, "Oconee Nuclear Station Units 1, 2, and 3 Analysis of Postulated High Energy Line Breaks (HELBs) Outside of Containment." The high energy systems and their postulated break locations are described in section 4.1 for Unit 1, section 5.1 for Unit 2 and section 6.1 for Unit 3. The interactions with safe shutdown equipment are described in sections 4.2 and 4.3 for Unit 1 HELBs, in sections 5.2 and 5.3 for Unit 2 HELBs and in sections 6.2 and 6.3 for Unit 3 HELBs. A summary of the consequences from the HELB interactions with safe shutdown equipment is contained in Tables 4.2-1 through 4.2-11 for Unit 1 HELBs, Tables 5.2-1 through 5.2-9 and 5.2-11 for Unit 2 HELBs, and Tables 6.2-1 through 6.2-11 for Unit 3 HELBs.

The mitigation of each HELB is dependent upon the HELB itself as well as its interactions with safe shutdown equipment. The transient response and the acceptance criteria for overcooling events (ex. Main Steam line breaks), undercooling events (ex. Main Feedwater line breaks), and excessive reactor coolant leakage (ex. Letdown line breaks) are described in section 7 of ONDS-351, Rev. 2. Section 7 did not provide a detailed listing of all required operator actions, necessary repairs, manpower requirements and the time required to perform these actions.

The tables containing the consequences of the HELB interactions were reviewed to determine if one HELB event could be found that provided the bounding event with respect to operator actions, necessary repairs, manpower requirements and the associated time limits for performing these actions. It was found that HELBs occurring inside the Turbine Building have the potential to create the most bounding scenario involving required operator actions, manpower requirements and damage repairs. Specifically, HELBs that can cause a loss of AC power to all three units coupled with failures to Condenser Circulating Water (CCW) piping resulting in Turbine Building flooding will create the bounding scenario for activities necessary to place the units in Mode 4. The postulated HELBs that create this condition are certain main feedwater line breaks on Units 1 and 2 as listed below:

- HELB 1-FDW-030-R Break 12 or 13
- HELB 2-FDW-008-R Break 6
- HELB 2-FDW-033-R Break 4 or 5
- HELB 2-FDW-035-R02 Break 4 or 5

The above breaks result in a loss of main and emergency feedwater to all three units. No unisolable breaks occur in either main steam lines for these HELB events. Therefore, the acceptance criterion for these events is based on the acceptance criterion for the loss of main feedwater event. The plant transient response for these events is described in section 7.2.1 of ONDS-351 Rev.2 with a general description of the strategy to enable a plant cooldown to cold shutdown. A more detailed description of the mitigation strategy is provided below.

The following acceptance criteria for a loss of main feedwater event ensures that the integrity of the fuel and the RCS remains unchallenged and that the event will not result in unacceptable radiological consequences:

- Peak Reactor Coolant System (RCS) pressure ≤ 2750 psig.
- Reactor Coolant Pump seal cooling restored within 20 minutes.
- Establishing and maintaining the RCS in a subcooled natural circulation condition during all phases of the mitigation strategy.

Mitigation of these postulated HELBs is divided into four distinct phases. Phase 1 is reactor shutdown and the stabilization of the affected unit(s) in Mode 3 with Reactor Coolant (RC) average temperature ≥ 525°F. Phase 2 is the plant cooldown from Mode 3 to Mode 4 (< 250°F). Phase 3 is the assessment and repair of structures, systems, and components (SSCs) required to transition the unit from Mode 4 (< 250°F) to Mode 5 (< 200°F). Phase 4 is the plant cooldown to Mode 5 (< 200°F). The following response is based on approval of the revised mitigation strategy and subsequent completion of committed modifications, and revisions to the mitigation and recovery procedures.

Refer to enclosed flowchart for overview of mitigation strategy addressed in this response.

Phase 1: Reactor Shutdown

The postulated main feedwater HELB event leads to an overheating condition for the Reactor Coolant System (RCS). The Reactor Protection System (RPS) will trip the reactor on the loss of Main Feedwater pumps or on high RCS pressure. The pressurizer code safety valves are credited to relieve pressure to maintain RCS pressure below the safety limit. The Main Steam Relief Valves (MSRVs) are the only credited means of steam release for decay heat removal during this phase.

Operator actions are needed to restore secondary side decay heat removal and reactor coolant pump (RCP) seal cooling to establish a Safe Shutdown Condition. The Standby Shutdown Facility (SSF) and the Protected Service Water (PSW) systems would remain available to establish and maintain safe shutdown for these main feedwater HELBs. Emergency procedures will direct the operators to initiate both pathways in parallel. Actions to place the SSF systems in service are contained in existing emergency procedures and are consistent with the activation of the SSF for other non-HELB events. Actions to place the PSW system (the PSW system is described in section 3.2 of ONDS-351 Rev. 2) in service will be incorporated into plant emergency procedures. The actions taken by the operators have been segregated by the different pathways in which safe shutdown would be achieved and maintained.

Pathway 1: Safe Shutdown Using PSW Systems

1. Operators in the Main Control Room (MCR) emergency start both Keowee Hydro Units (KHUs), if not already started.
2. Operators in the MCR repower the PSW electrical system from one of the two operating KHUs.
3. The control battery chargers automatically swap to the PSW source of power.
4. Operators in the MCR start the PSW pumps and open the main PSW block valve.
5. Operators in the MCR transfer power for one High Pressure Injection (HPI) pump, BWST suction valve (HP-24), HPI header injection valve (HP-26) and the RCS vent valves (RC-155 through RC-160) on each unit from their normal source of power to the PSW power.
6. Operators in the MCR begin feeding each unit's SGs. This activity must be completed within 15 minutes of the loss of main and emergency feedwater.
7. Operators in the MCR reestablish RCP seal cooling by opening HP-24, closing the RCP seal flow control outlet valve (HP-139), starting the HPI pump, and throttling open RCP seal flow control bypass valve (HP-140). This activity must be completed within 20 minutes.
8. Operators in the MCR open RCS vent valves to establish letdown as required to maintain pressurizer level.
9. Operators are dispatched to the East Penetration Room (EPR) to transfer power for selected pressurizer heaters from their normal source of power to the PSW power.
10. Operators in the MCR energize selected pressurizer heaters as necessary to maintain RCS pressure at approximately 2155 psig.

Pathway 2: Safe Shutdown Using SSF Systems

1. Operators are dispatched to the SSF upon recognition of the loss of RCP seal cooling.
2. One operator at the SSF performs breaker transfers for the 600VAC Motor Control Centers (1XSF, 2XSF, and 3XSF)
3. One operator in the SSF Control Room (SSFCR) emergency starts the SSF diesel-generator, starts the diesel engine service water pump, and starts the SSF Auxiliary Service Water (SSF ASW) pump
4. Operators in the SSFCR contact operators in the MCR to determine if actions to place PSW systems in service have been successful.
5. If actions to place PSW in service have been successful, then operators at the SSF would standby awaiting further instructions.
6. If actions to place PSW in service were not successful, then operators at the SSF would continue performing actions as directed by the SSF emergency operating procedure.
7. Operators in the SSFCR start the SSF RC Makeup pump to establish RCP seal cooling as well as RC makeup. This action must be completed within 20 minutes following a loss of all RCP seal cooling.
8. Operators in the SSFCR isolate possible RCS leakage pathways (RC letdown and RCP seal return).
9. Operators in the SSFCR begin feeding the steam generators at a rate necessary to control RCS temperature and pressure. This action must be completed within 14 minutes.
10. Operators in the SSFCR energize pressurizer heaters, when conditions permit, to maintain RCS pressure at approximately 2150 psig.

11. An operator is dispatched to divert the SSF Diesel Service Water discharge to the yard drain after the diesel-generator has been operating for 1 hour and 45 minutes. This action must be completed between 1 hour and 45 minutes and 2 hours following the emergency start of the SSF Diesel Generator.

While the operators are placing the affected unit(s) in Mode 3 with an RC temperature of ≥ 525° F, the Operations Shift Manager initiates the Emergency Plan, and activates the Technical Support Center (TSC) and the Operations Support Center (OSC). Implementation procedure RP/1000/025 directs the OSC to initiate the site coordinating procedure for assessment and damage repair, RP/1000/022. RP/1000/022, in turn, directs Operations and Engineering personnel to assess and identify credited SSCs that are in need of repair.

The postulated main feedwater HELBs (listed above) can result in pipe breaches to the CCW piping. The breaches in CCW piping can create flooding inside the Turbine Building. The source of flooding inside the Turbine Building would need to be isolated. If CCW inlet siphon and gravity flow has been lost, no isolation of the CCW inlet piping would be required to terminate flooding from the CCW intake piping. If lake elevation is below the point needed for gravity induced reverse flow, no isolation of the CCW discharge piping would be required. It is assumed that lake elevation is sufficiently high for gravity induced flow to both CCW inlet and discharge piping in order to maximize the actions needed to terminate TB flooding.

Operators would be dispatched to close the CCW pump discharge valves on each unit at the intake structure to isolate Turbine Building flooding inputs from the CCW inlet because power has been lost. Maintenance would be dispatched to lower the CCW discharge gates to isolate Turbine Building flooding inputs from the CCW discharge. Once the flooding inputs to the Turbine Building have been isolated, the flood waters will gravity drain through the drain located at the south end of the Turbine Building basement.

Operators would begin monitoring the water temperature and water level in the Spent Fuel Pool (SFP) due to the loss of spent fuel cooling as directed by existing emergency procedures. Refill of the SFP is performed using existing site recovery procedures and existing onsite and offsite personnel and resources that provide redundant and diverse methods to refill the SFPs. This is an existing time critical action (TCA) documented in OSS-0254.00-00-4005, Design Basis Specification for the Design Basis Events, Rev. 22, which must be completed no later than 36 hours following start of the SSF RCMU Pump. The acceptance criteria for this activity is that the boron concentration in the SFP is maintained above that required to result in a $K_{eff} \leq 0.95$ and water level is maintained above the fuel to ensure that the integrity of the fuel remains unchallenged and that there are no unacceptable radiological consequences.

A Safe Shutdown condition can be maintained from either the Main Control Room using the PSW and HPI Systems or from the SSF Control Room using the SSF ASW and SSF RCMU Systems. The SSF has an established mission time of 72 hours. There are no required repairs from these postulated HELBs to achieve safe shutdown using either the PSW or SSF systems. However, if makeup to the CCW piping from the CCW intake or discharge via gravity induced flow is not available, a portable pump would need to be installed at the CCW intake to provide replenishment of the water being used by the PSW or SSF systems. Electrical power can be supplied from either the PSW electrical system or the SSF electrical system. The actual required time for installing the portable pump varies with lake level, CCW piping break size and location, and CCW inlet high point air in-leakage. A time critical action (TCA) has been applied to the deployment and placing of the portable pump in operation within 3 hours and 20 minutes

of a loss of forced and gravity CCW system flow. This is an existing TCA documented in OSS-0254.00-00-4005, Design Basis Specification for the Design Basis Events, Rev. 22.

Phase 2: Plant Cooldown to Mode 4 (< 250°F)

A plant cooldown is initiated within 72 hours of event initiation.

Plant cooldown requires one HPI pump to provide sufficient makeup capability. No damage repairs are needed to perform a plant cooldown to Mode 4. Power is supplied to the HPI pump and the HPI motor operated valves from the PSW electrical system which is not damaged by the HELBs. Cooling water to the HPI pump is provided by the PSW booster pump. Either the PSW System or the SSF ASW System could be used to feed the steam generators to enable a plant cooldown. Since the RCPs have been lost due to the loss of power, a natural circulation cooldown would be required. Any time a natural circulation cooldown is initiated, the Reactor Vessel (RV) Head Vents are required to be opened. Power to operate the RV Head Vents is provided from the PSW electrical system. The RCS is depressurized during plant cooldown by turning the pressurizer heaters off and cycling the Pressurizer Power Operated Relief Valve (PORV) (RC-66). Power to the PORV is supplied from the battery backed buses which will continue to receive power from the battery chargers via the PSW electrical system. Several motor-operated valves would need to be repositioned during the plant cooldown. The Portable Valve Control Panel (PVCP) would be installed to restore power to the decay heat drop line isolation valves (LP-1 and LP-2), and the Core Flood Tank (CFT) outlet valves (CF-1 and CF-2) on each unit since their normal power sources would be lost. The PVCP would be installed in the yard adjacent to each unit's West Penetration Room (WPR). Cables would be pulled from the SSF to the PVCP and connections made at the SSF and the PVCP. Cables would also be pulled from the PVCP to LP-1, LP-2, CF-1, and CF-2 valve electrical penetrations located in the WPR. Connections would be made at the PVCP and at the electrical penetrations for the valves. These activities are completed by Oconee craft personnel with staff augmentation using existing station procedures. Additional personnel would be assembled in accordance with the station emergency plan at the site within 8 hours to begin restoration of power to LP-1, LP-2, CF-1 and CF-2. The estimated time to install the PVCP and repower the valves is an additional 21 hours. Plant cooldown to Mode 4 (< 250°F) would be performed when the valves have been repowered.

Plant Cooldown Sequence to Mode 4:

1. Operators in the MCR start the PSW pumps (if not already operating).
2. Operators in the MCR begin feeding each unit's SGs with PSW (if not already in progress) to maintain SG water levels at or above the levels needed to maintain natural circulation.
3. Operators in the SSFCR would isolate SSF ASW feed to the SGs (if feeding).
4. Operators in the MCR would align HPI pump suction to the BWST and start one HPI pump (if not already operating).
5. Operators in the MCR throttle HP-140 as necessary to maintain RCP seal cooling.
6. Operators at the SSFCR would stop the SSF RCMU Pump (if operating) and isolate the RCS letdown to the Spent Fuel Pool (if open).
7. Operators in the MCR open the RV head vent valves (if not already open for pressurizer level control).
8. Operators in the MCR throttle open the Power-Operated Atmospheric Dump Valves (POADVs) to establish a natural circulation RCS cool down.

9. Operators in the MCR throttle the PSW flow control valves to each SG as necessary to maintain SG water level at the desired level.
10. If additional RCS makeup is needed during plant cooldown, operators in the MCR would throttle open HP-26 as necessary to maintain pressurizer level.
11. Operators at the SSFCR would isolate the RCS letdown to the Spent Fuel Pool (if open).
12. Operators at the SSF would turn off the pressurizer heaters controlled from the SSF.
13. Operators in the MCR turn off the pressurizer heaters controlled from the MCR.
14. Operators in the MCR cycle the Pressurizer PORV to decrease RCS pressure as required to maintain the required RC subcooling margin during plant cooldown.
15. When RCS pressure is approximately 700 psig, the Core Flood Tank Isolation Valves (CF-1 and CF-2) are closed from the PVCP.
16. Operators would be dispatched to throttle open the manually operated ADVs during the latter stages of cooldown to complete the RCS cooldown to Mode 4 (< 250°F).
17. Operators in the MCR would stabilize RCS pressure at approximately 300 psig by energizing pressurizer heaters with RCS temperature < 250°F.
18. At the conclusion of the cooldown, the manually operated ADVs are expected to be fully open with RCS temperature between 212°F and 250°F. Operators would no longer be needed at the ADVs.
19. Operators in the MCR close the RV head vent valves.
20. Operators in the MCR stop the operating HPI Pumps.

In this configuration long term subcooled natural circulation decay heat removal conditions are maintained with RC pressure being controlled by the cycling of the pressurizer heaters and RC temperature being maintained < 250°F by maintaining water level in the SGs at or above the levels needed for natural circulation.

Phase 3: Damage Assessment and Repairs Required to Achieve Mode 5 (< 200°F)

HELB damage assessment is initiated to assess and repair systems needed to allow plant cooldown from Mode 4 (< 250°F) to Mode 5 (< 200°F). Although the assessment may begin in Phase 1, the systems needed to achieve cold shutdown are not required to be repaired prior to initiating a cooldown of the RCS to Mode 4 (< 250°F). The scope of the assessment determines the availability of the CCW system, the Low Pressure Service Water (LPSW) system, the Low Pressure Injection (LPI) system, and the associated electrical power to these systems.

The postulated loss of AC power to all three units would require restoring power to one CCW pump motor, two LPSW pump motors (one shared by Units 1 & 2, and one for Unit 3), and three LPI pump motors (one for each unit). The actions taken to restore power to the pump motors needed for cold shutdown rely on the current Appendix R strategy, procedures and equipment. The necessary electrical equipment has been identified in procedures used for Appendix R and would be available to enable the restoration of power to these motors. The manpower requirements to execute the repairs have been defined in the procedures used for Appendix R. In addition, two LPSW pump motors would need to be replaced due to the effects of Turbine Building Flooding. There are two spare LPSW pump motors that can be installed using existing station procedures. The manpower requirements to execute the repairs have been defined in the procedures used for Appendix R.

The listed HELBs that create the bounding scenario for activities necessary to place the units in Mode 4 (< 250°F), may not establish the bounding time for repairs needed to achieve Mode 5

(< 200°F). HELBs resulting in damage to CCW and LPSW piping systems required for cold shutdown are described in Duke Energy's response to RAI 102. The HELBs identified in that response are expected to create the bounding time for damage repairs needed to proceed to a cold shutdown condition. The postulated line breaks in the CCW and LPSW systems would need to be isolated or repaired as necessary to restore one unit's CCW system to operation, and the LPSW flow paths for the LPI coolers (supply and return headers). The level of effort and the duration to complete piping repairs have not been defined in existing station procedures.

While the units are being maintained in a long term subcooled natural circulation decay heat removal condition, the repairs and system alignments necessary to transition the units from Mode 4 (< 250°F) with decay heat removal via the SGs to Mode 5 (< 200°F) would be completed. The Oconee Emergency Response organization would coordinate these recovery actions augmented by fleet and industry personnel. Duke Energy would obtain materials and resources needed to restore the systems needed to achieve Mode 5, utilizing existing fleet resources and existing relationships with other utilities, suppliers, and manufacturers.

Phase 4: Plant Cooldown to Mode 5 (< 200°F)

When the damage repairs for the CCW and LPSW systems, as well as the system alignments for the CCW, LPSW, and LPI systems have been completed, a plant cooldown would be initiated from Mode 4 to Mode 5.

Plant Cooldown Sequence to Mode 5:

1. Operators would verify at least two condenser outlet valves are open on the unit in which the CCW system would be placed into service. If none are open, maintenance would be contacted to open two condenser outlet valves.
2. An operator would be dispatched to the CCW intake structure to throttle open the discharge valve on the CCW pump to be started.
3. An operator would locally start the one CCW pump at the trailer-mounted (Appendix R) 4160V switchgear.
4. The operator at the CCW intake structure would then fully open the discharge valve on the running CCW pump.
5. Operators would be dispatched to the Turbine Building basement to close the discharge valves on the LPSW pumps to be started. One pump shared by Units 1 and 2 and one pump for Unit 3 would be needed for cooldown to Mode 5.
6. An operator would locally start the desired LPSW pumps at the trailer-mounted (Appendix R) 4160V switchgear.
7. Operators would slowly throttle open the discharge valve on the running LPSW pumps to fill the LPSW system. Eventually, the LPSW pump discharge valves will be fully opened.
8. Operators open LP-1 and LP-2 at the PVCP.
9. An operator would locally start one LPI pump on each unit at the trailer-mounted (Appendix R) 4160V switchgear.
10. Communications would be established between the operators in the MCR and the operators at the LPI cooler outlet LPSW block valve. The LPSW outlet block would be locally throttled open to establish a cooldown to < 200°F.

<u>Tornado</u>

Mitigation of a tornado strike is divided into five distinct phases. Phase 1 is reactor shutdown

and the stabilization of the affected unit(s) in Mode 3 with Reactor Coolant (RC) average temperature ≥ 525°F. Phase 2 is the assessment and repair of SSCs required to transition the unit from Mode 3 to Mode 4. Phase 3 is the plant cooldown from Mode 3 to Mode 4 (< 250°F). Phase 4 is the assessment and repair of structures, systems, and components (SSCs) required to transition the unit from Mode 4 (< 250°F) to Mode 5 (< 200°F). Phase 5 is the plant cooldown to Mode 5 (< 200°F).

The following acceptance criteria for a tornado event ensures that the integrity of the fuel and the RCS remains unchallenged and that the event will not result in unacceptable radiological consequences:

- Peak Reactor Coolant System (RCS) pressure ≤ 2750 psig.
- Reactor Coolant Pump seal cooling restored within 20 minutes
- Establishing and maintaining the RCS in a subcooled natural circulation condition during all phases of the mitigation strategy

The tornado mitigation strategy relies on the strategy, procedures and equipment originally developed to meet the Appendix R license basis with the following exceptions:

- Tornado damage to SSCs needed to transition the affected unit(s) from Mode 3 to Mode 4 and from Mode 4 to Mode 5 will need to be repaired.

- The PSW booster pump is used to provide cooling water to the high pressure injection pump (Appendix R strategy relies upon replacement air cooled motors).

- 4kV PSW power supplies the high pressure injection pump (Appendix R strategy relies upon trailed mounted 4kV switchgear as power source).

The following response is based on the committed modifications being implemented and supersedes responses provided in previously submitted RAI responses due to the change in proposed mitigation strategy. The following response is also based on the proposed revisions to current recovery procedures needed to address repairing tornado damage being implemented.

Refer to enclosed flowchart for overview of mitigation strategy addressed in this response.

Phase 1

Overview

Phase 1 of the event mitigation is the stabilization of the affected unit(s) in hot standby conditions with an RC temperature of ≥ 525° F from the Standby Shutdown Facility (SSF). The SSF Reactor Coolant Makeup (RCMU) Pump, taking suction from the Spent Fuel Pool, maintains the reactor subcritical, maintains adequate RC inventory control and provides reactor coolant pump (RCP) seal cooling. RC pressure is maintained by Pressurizer heaters powered and controlled from the SSF. The SSF Auxiliary Service Water (ASW) Pump, taking suction from the Unit 2 Condenser Circulating Water (CCW) System intake piping, feeds the steam generators to provide SG decay heat removal. Main Steam (MS) pressure is regulated using MS isolation and steam release through the main steam relief valves. The inventory in the Unit 2 CCW intake piping is replenished by gravity induced flow of Lake Keowee directly into the Unit

2 CCW intake piping and gravity induced flow of Lake Keowee indirectly into the Unit 2 CCW intake piping via the CCW discharge piping. An SSF portable pump is deployed and placed into the CCW intake canal following start of the SSF diesel generator but is not immediately placed in operation unless necessary. In the event that gravity flow is interrupted, the discharge hose of the SSF portable pump is aligned to fill the Unit 2 CCW intake piping and the pump is started to maintain inventory in the Unit 2 CCW intake piping for supplying the SSF ASW pump. The SSF ASW Pump, RCMU Pump, HVAC System, SSF portable pump and associated electrical systems are powered from the SSF diesel generator.

Event Sequence

The operating crew is notified of a tornado watch or tornado warning via the weather radio located in the Unit 1 & 2 MCR. Notification of a tornado watch or warning is an entry condition into abnormal operation procedure AP/0/A/1700/006, Natural Disaster. When a tornado warning is in effect AP/06 will direct the operating crew to dispatch one licensed operator and one non-licensed operator to the SSF control room.

Depending upon the severity of the damage when the tornado strikes the station, the reactor(s) may be manually tripped or may automatically trip. If a tornado missile breaches the pressure boundary of steam piping, the Main Steam Isolation Valves (MSIVs) will close and isolate the breach before overcooling of the RCS can occur.

The operating crew on each affected unit will respond to the reactor trip by performing procedure EP/1800/001, Emergency Operating Procedure. The EOP provides guidance to ensure that the reactor and turbine are tripped, and to verify that reactor coolant pump seal cooling is available, that the reactor remains subcritical, that adequate RCS inventory is being provided and that the appropriate amount of steam generator (SG) cooling is being provided. During performance of the immediate manual actions of the EOP the operating crew determines if a loss of RCP seal cooling and SG cooling has occurred and directs the SSF operator to establish RCP seal cooling and SG feed from the SSF.

When directed by the MCR, the operators stationed at the SSF takes the following actions to re-establish RCP seal cooling and SG DHR using AP/1700/025, SSF Emergency Operating Procedure:

1. The non-licensed operator performs breaker transfers for the 600VAC Motor Control Centers (1XSF, 2XSF, and/or 3XSF) in the HVAC room of the SSF to transfer control of the RCS boundary isolation valves, wide range core inlet temperature instruments and pressurizer heaters from the MCR of the affected unit(s) to the SSF control room

2. The licensed operator in the SSF control room emergency starts the SSF diesel-generator, starts the diesel engine service water pump and starts the SSF Auxiliary Service Water (ASW) pump.

3. The licensed operator starts the SSF Reactor Coolant Makeup (RCMU) Pump for each affected unit in the override mode. When started in the override mode, the pump suction and discharge flow paths are automatically aligned and the RCMU pump then automatically starts to provide RCP seal cooling. Restoration of RCP seal cooling is an existing time critical action (TCA) documented in OSS-0254.00-00-4005, Design Basis Specification for the Design Basis Events, Rev. 22, which must be completed within 20

minutes of a loss of seal cooling.

4. The licensed operator aligns SSF ASW to each affected unit with the exception of the SSF ASW header flow regulating valve which will remain closed

5. The licensed operator establishes ASW flow to one of the affected unit's SGs by throttling open the unit's SSF ASW header flow control valve. Restoration of SG feed from the SSF is an existing TCA documented in OSS-0254.00-00-4005, Design Basis Specification for the Design Basis Events, Rev. 22, which must be completed within 14 minutes of a loss of SG cooling and RCP seal cooling.

While the operator stationed at the SSF is placing the SSF in operation, additional licensed operators are dispatched to the SSF from the MCR of any other affected unit. Three available exits from the Auxiliary Building to the SSF and two entrances into the SSF provides reasonable assurance that the additional licensed operators can access the SSF control room and establish SSF ASW flow within 14 minutes. Walkdowns demonstrate that an operator can travel from each of the MCRs, exit the Auxiliary Building through any of the three available exits after first determining that the other two are blocked, enter the SSF through the most distant entrance and establish auxiliary service water flow within 14 minutes.

Upon arriving in the SSF control room these additional licensed operators establish feed to the SGs on the remaining affected units by throttling open the closed flow regulating valve. The licensed operators in the SSF control room then isolate possible RCS leakage pathways (RC letdown and RCP seal return) and energize pressurizer heaters, when conditions permit, to maintain RCS pressure at approximately 2150 psig.

The Maintenance Department is notified to deploy the SSF portable pump and prepare it for operation per AP/1700/25. If the gravity flow paths used to replenish the Unit 2 CCW intake piping are unavailable, the discharge piping of the SSF portable pump is aligned to the Unit 2 CCW intake piping and the pump is placed in operation. The deployment and placing of the SSF portable pump in operation is an existing TCA documented in OSS-0254.00-00-4005, Design Basis Specification for the Design Basis Events, Rev. 22, which must be completed within 3 hours and 20 minutes of a loss of forced and gravity Condenser Circulating Water (CCW) system flow.

One hour and forty five minutes after the SSF diesel generator is started the non-licensed operator locally diverts the discharge from the diesel generator heat exchangers to the yard drains per AP/25. This is an existing TCA documented in OSS-0254.00-00-4005, Design Basis Specification for the Design Basis Events, Rev. 22, which must be completed between one hour forty five minutes and two hours following the start of the SSF diesel generator.

The licensed operators stationed at the SSF maintain the affected unit(s) in hot standby subcooled natural circulation conditions with an RC temperature of ≥ 525° F for up to 72 hours while the assessment and repair of components credited for the transition of the units from Mode 3 to Mode 4 are completed.

While the operators at the SSF are placing the affected unit(s) in Mode 3 with an RC temperature of ≥ 525° F, the Operations Shift Manager initiates the Emergency Plan, and activates the Technical Support Center (TSC) and the Operations Support Center (OSC). Implementation procedure RP/1000/025 directs the OSC to initiate the site coordinating

procedure for assessment and damage repair, RP/1000/022. RP/1000/022, in turn, directs Operations and Engineering personnel to assess and identify credited SSCs that are in need of repair.

Phase 2

Local operation of a train of High Pressure Injection (HPI) including RCP seal cooling, local operation of the Protected Service Water (PSW) booster pump, local operation of the tornado protected MS Atmospheric Dump Valves (ADVs) and SG cooling via the SSF ASW System is credited for transitioning the affected unit(s) from Mode 3 to Mode 4. In addition, operation of the reactor vessel head vent valves, Pressurizer (PZR) PORV and the core flood tank isolation valves via a portable valve control panel is required.

Phase 2 of the event mitigation consists of performing assessments, repairs and alignment of these SSCs. These assessment and repair activities are completed within 72 hours of event initiation.

Event Sequence

The assessment, repair and alignment activities consist of:

1. Assessing the availability of power to the Protected Service Water (PSW) Building from either a Keowee Hydro Unit or the PSW substation. If neither source of power is available, the 13.8kV transmission line from the PSW substation to the PSW building will be restored to service and any faults on the 100kV line to the PSW substation will be isolated. These repairs will be completed using readily available commercial resources and personnel from the fleet transmission organization.

2. Manually restoring the PSW building electrical systems and ventilation systems. This activity is completed following restoration of 13.8kV power to the PSW building using site damage repair procedures, personnel and resources.

3. Configuring the PSW booster pump motor for local start. The PSW pump, motor. 4kV breaker, control power, cabling, and piping are tornado protected. This activity is completed using site damage repair procedures, personnel and resources.

4. Configuring the High Pressure Injection (HPI) pump motor for local start. The HPI pump, motor, 4kV breaker, control power, cabling, pump suction piping and pump discharge piping below ground elevation is tornado protected. This activity is completed using site damage repair procedures, personnel and resources.

5. Assessing and replacing damaged HPI discharge piping, including seal injection piping, located above grade in the East and West Penetration Rooms. This activity is completed using site damage repair procedures, personnel and resources.

6. Installing the existing Appendix R Portable Valve Control Panel (PVCP) in the yard adjacent to each affected unit(s) West Penetration Room, pulling 600V power cables and 120V control cables from the SSF to the PVCP and making associated connections and pulling 600 V power cables and 120V control cables from the PVCP to the reactor head vent valves (RC-159/RC-160), the Pressurizer PORV (RC-66), the Core Flood

Tank Isolation Valves (CF-1/CF-2) the LPI Return Block from RCS Valves (LP-1/LP-2) penetrations located in the Penetration Rooms, and making associated connections. This activity is completed using existing procedures.

7. Aligning power to the SSF from the PSW 4kV switchgear and shutting down the SSF D/G. This activity is completed by Operations personnel from the SSF control room.

8. Initiate monitoring the water temperature and water level in the Spent Fuel Pool (SFP) due to the loss of spent fuel cooling as directed by existing emergency procedures. Refill of the SFP is performed using existing site recovery procedures and existing onsite and offsite personnel and resources that provide redundant and diverse methods to refill the SFPs. This is an existing time critical action (TCA) documented in OSS-0254.00-00-4005, Design Basis Specification for the Design Basis Events, Rev. 22, which must be completed no later than 36 hours following start of the SSF RCMU Pump. The acceptance criteria for this activity is that the boron concentration in the SFP is maintained above that required to result in a $K_{eff} \leq 0.95$ and water level is maintained above the fuel to ensure that the integrity of the fuel remains unchallenged and that there are no unacceptable radiological consequences.

The previous paragraphs provide a complete event sequence following a tornado. The following paragraphs provide a detailed description of the personnel, equipment and parts required to complete repairs to the SSCs described in phase 2 that are credited for a plant cool down from Mode 3 with average reactor coolant temperature at or above 525°F to Mode 4.

The scope of the needed repair activities is dependent upon the magnitude and location of the tornado. The estimates used for the following repair activities assume a complete loss of the 13.8kV transmission line between the PSW substation and the PSW building, a complete loss of all credited High Pressure Injection piping located in the East and West Penetration Rooms of all three units and a complete loss of all of the power and control cables for the valves located inside containment credited for cooling down all three units from Mode 3 to Mode 4. This is considered to be a conservative approach in that it is unlikely that a tornado will result in this scope of damage.

These repair activities are completed by Oconee resources that are augmented by personnel, equipment and materials supplied from the fleet organization. Equipment and materials not readily available in the Duke Energy fleet will be maintained onsite in a tornado protected structure or stored at an offsite location.

Resource loaded tornado damage repair procedures containing personnel, equipment and material requirements will be developed and time validated as an implementation item in support of license issuance.

The repair activities consist of:

1. Restoring the 13.8kV transmission line from the PSW substation to the PSW building.

   To restore the 13.8kV transmission line to its as built configuration it is estimated that the following personnel, equipment and materials are required. This is considered to be a conservative estimate as a more direct routing of the transmission line may be used during emergency repairs.

- 27 power poles
- 17,120 feet of 13.8kV transmission line
- Associated cross ties, insulators, arrestors, fasteners, etc.
- 25 bucket & line trucks
- A crew of 70 to 75 personnel

These repairs are completed using readily available resources, equipment and materials obtained from the local Duke Energy service area and are estimated to be completed within 60 hours of the event.

2. Restoring the PSW building electrical and ventilation systems.

   Restoration of the PSW building electrical and ventilation systems consists of the following activities and these activities are performed in the PSW building. The restoration of these systems is completed in parallel with the restoration of 13.8kV power to the PSW building. With an expected total completion time of approximately seven hours, the time to complete these activities is bounded by restoration of the 13.8kV line.

   a. Isolate the PSW 4kV B6T and B7T Switchgear by locally opening the four incoming 13.8kV incoming breakers. This activity is estimated to be completed within two hours by one operator. This activity is completed prior to disconnecting the damaged 13.8kV line from the incoming feeder breakers to B6T Switchgear.
   b. Isolate 600V loads that are not credited for recovery by locally opening six feeder breakers. This activity is estimated to be completed within two hours by one operator and is completed prior to re-energizing B6T Switchgear.
   c. Isolate control power to the B6T Switchgear 13.8kV incoming feeder breaker from the PSW substation and install a prefabricated local control switch. This activity is estimated to be completed within two hours by one electrician and is completed prior to re-energizing B6T Switchgear.
   d. Re-energize B6T Switchgear by closing in the 13.8kV incoming feeder breaker from the PSW substation using the local control switch. This activity is completed after the 13.8kV PSW transmission line has been repaired and is estimated to be completed within one hour by one operator.

3. Configuring the PSW booster pump motor for local start.

   This activity consists of isolating control power to the 4kV PSW booster pump motor breaker and installing a prefabricated local control switch. This activity is performed in the PSW building and is estimated to be completed within two hours by one electrician. This activity is completed in parallel with the restoration of the PSW building electrical and ventilation systems and is completed prior to re-energizing B6T Switchgear.

4. Configuring the High Pressure Injection (HPI) pump motor for local start.

   Configuring the High Pressure Injection (HPI) pump motor for local start consists of the following activities and these activities are completed prior to re-energizing B6T Switchgear. These activities are completed in parallel with the restoration of 13.8kV power to the PSW building. With an expected total completion time of approximately six

hours, the time to complete these activities is bounded by restoration of the 13.8kV line.

a. Isolate control power to each affected unit's PSW 4kV HPI pump motor breaker and install a prefabricated local control switch. This activity is estimated to be completed within six hours by one electrician (two hours for each HPI pump switch). This activity is performed in the PSW building.
b. Locally align the credited HPI pump on each affected unit to receive 4kV power from the PSW supply breaker using the transfer switch located at elevation 775 in the Auxiliary Building. This activity is estimated to be completed within three hours by one operator (one hour per HPI pump transfer switch) and is performed in parallel with installation of the local control switch.

5. Replacing damaged High Pressure Injection (HPI) discharge piping, including seal injection piping, located above grade in the East and West Penetration Rooms.

   To restore the 'A' HPI discharge header and all Reactor Coolant Pump seal injection piping located in the East and West Penetration Rooms on all three units to its as built configuration it is estimated that the following personnel, equipment and materials are required. This is considered to be a conservative estimate as it is unlikely that 100% of the credited piping is damaged and a more direct routing of the HPI piping may be used during emergency repairs. The restoration of this system is completed in parallel with the restoration of 13.8kV power to the PSW building. With an expected total completion time of approximately 36 hours, the time to complete this activity is bounded by restoration of the 13.8kV line.

   - A total of 1800 feet of 1.5 inch schedule 160 stainless steel piping stored in prefabricated sections
   - A total of 250 feet of 4 inch schedule 160 stainless steel piping stored in prefabricated sections
   - Associated pipe tees, elbows, caps, couplings, supports, etc.
   - One boom truck, one mobile crane and one front end loader
   - Twelve self powered portable welding machines
   - Three portable air compressors, 600 feet of hose and six remote supply tanks
   - Self powered portable lighting
   - Four 12 person crews per unit

6. Installing the existing portable valve control panel in the yard adjacent to each affected unit(s) West Penetration Room and connecting it to the valves credited for cooling the unit down from Mode 3 to Mode 4.

   Installation of the portable valve control panel will be performed in accordance with site procedure. Details of the procedure and the associated equipment, materials and resources were reviewed and inspected as part of 10CFR50 Appendix R implementation in conjunction with the April 28, 1983 SSF Safety Evaluation Report and found to be satisfactory. The NRC inspection reports include Inspection Reports 50-269/87-02, 50-270/87-02, 50-287/87-02; 50-269/02-03, 50-270/02-03, 50-287/02-03 (ML020790521); and 05000269/2008007, 05000270/2008007, 05000271/2008007 (ML082740055). While these inspections were focused on damage repair relative to Appendix R, the damage repair procedure is also employed for damage repair activities following a tornado.

7. Aligning power to the SSF from the PSW 4kV switchgear.

   The SSF is aligned to receive power from the PSW 4kV switchgear after PSW 4kV B6T Switchgear has been re-energized. No damage repairs are required and this activity consists of local operation of one breaker in the PSW building and two breakers in the SSF control room. This activity is estimated to be completed within one hour by two operators.

8. Monitoring the temperature and level of the water in the Spent Fuel Pool (SFP) and refilling as necessary.

   The monitoring and refilling of the SFP consists of the following activities and is completed in parallel with the restoration of 13.8kV power to the PSW building:

   a. Locally monitor the temperature and level of the water in each SFP.
   b. Estimate time that 212°F is reached in the SFPs based upon rate of temperature increase.
   c. Install SFP spray manifolds (Boggs Box) and route manifold supply hoses from each SFP to the plant yard area adjacent to the fuel receiving area doors. This activity is estimated to be completed within four hours (two hours for each SFP) by two operators.
   d. Align pumper truck provided by local fire department to take suction on Lake Keowee and discharge into SFPs via dedicated emergency SFP fill headers. This activity requires ten technicians and operators, and refill of the SFPs is initiated within 36 hours of the event.
   e. In the event that the dedicated emergency SFP fill header is damaged beyond repair the discharge of the pumper truck is aligned to supply the SFP spray manifolds and refill of the SFPs is initiated within 36 hours of the event.

Phase 3

Overview

Phase 3 of event mitigation consists of a cooldown of the affected unit(s) from Mode 3 to Mode 4 (< 250°F) by performing a subcooled natural circulation cool down. Local operation of the PSW booster pump provides cooling water to the High Pressure Injection Pump. Local operation of the PSW powered HPI Pump taking suction from the Borated Water Storage Tank maintains the reactor subcritical, maintains adequate RC inventory control and provides reactor coolant pump seal cooling. Remote operation of the SSF Auxiliary Service Water (ASW) Pump taking suction from the Unit 2 Condenser Circulating Water (CCW) System intake piping feeds the steam generators to provide SG decay heat removal. MS pressure and flow is regulated during cool down by local operation of tornado protected ADVs. Gravity induced CCW flow or the SSF portable pump maintains inventory in the CCW system for supplying the SSF ASW pump. The SSF ASW Pump, HVAC system, portable pump and associated electrical systems are powered from the SSF via the PSW Power System.

Phase 3 is initiated within 72 hours of event initiation.

Event Sequence

Cooldown of the affected unit(s) from Mode 3 to Mode 4 is performed by Operations personnel using OP/0/A/1102/025, Cooldown Following Major Site Damage

These activities consist of:

1. Locally starting the PSW booster pump to provide cooling water to the HPI Pump
2. Aligning HPI Pump suction from BWST and locally starting an HPI Pump on each affected unit
3. Locally throttling open the RCP seal injection total flow control valve to establish RCP seal cooling
4. Securing the SSF RCMU Pump from the SSF control room
5. Locally opening the reactor head vent valves from the PVCP
6. Locally throttling open the ADVs to establish a natural circulation RCS cool down.
7. Throttling the SSF ASW flow control valves from the SSF control room to each SG as necessary to maintain SG level at the desired level during plant cooldown.
8. Locally throttling open RC makeup valve as necessary to maintain Pressurizer at desired level during plant cooldown.
9. Isolating RC letdown to the Spent Fuel Pool from the SSF control room
10. Securing the Pressurizer heaters from the SSF control room
11. Locally cycling the Pressurizer PORV from the PVCP to decrease RCS pressure.
12. Locally closing the Core Flood Tank Isolation Valves (CF-1/CF-2) from the PVCP when RCS pressure is approximately 700 psig.
13. RCS pressure is stabilized at approximately 300 psig and RCS temperature is stabilized at less than 250 deg. F (Mode 4).
14. The reactor head vent valves are closed from the PVCP and the HPI Pump is secured locally.

In this configuration long term subcooled natural circulation decay heat removal conditions are maintained with RC pressure being controlled by the cycling of PZR heaters from the SSF and RC temperature being maintained < 250° F by the feeding and steaming of the SGs.

Phase 4: Damage Assessment and Repairs required to achieve Mode 5 (< 200°F)

While the affected unit(s) are being maintained in a long term subcooled natural circulation decay heat removal condition a tornado damage assessment will be initiated to assess and repair systems needed to allow plant cooldown from Mode 4 (< 250°F) to Mode 5 (< 200°F). Although the assessment may begin in Phase 1, the systems needed to achieve cold shutdown are not required to be repaired prior to initiating a cooldown of the RCS to Mode 4 (< 250°F). The scope of the assessment determines the availability of the CCW system, the Low Pressure Service Water (LPSW) system, the Low Pressure Injection (LPI) system, and the associated electrical power to these systems.

The existing procedures originally developed to meet the Appendix R license basis provide the bases for these tornado mitigation assessments, repairs and system alignments. In addition, damage to piping on the SSCs needed to achieve Mode 5 will need to be repaired.

The Oconee Emergency Response organization will coordinate these recovery actions augmented by fleet and industry personnel. Duke Energy would obtain materials and resources needed to restore the systems needed to achieve Mode 5, utilizing existing fleet resources and existing relationships with other utilities, suppliers, and manufacturers.

The timeline and resource requirements for repair of equipment is unknown due to the unpredictable scope of damage that may occur to SSCs needed to establish cooling via the Low Pressure Injection System.

Phase 5: Plant Cooldown to Mode 5 (< 200°F)

When damage repair and system alignments have been completed for the CCW, LPSW and LPI systems, a plant cooldown is initiated from Mode 4 to Mode 5.

Plant Cooldown Sequence to Mode 5:

1. Operators verify at least two condenser outlet valves are open on the unit in which the CCW system would be placed into service. If none are open, maintenance personnel are contacted to open two condenser outlet valves.

2. An operator is dispatched to the CCW intake structure to throttle open the discharge valve on the CCW pump to be started.

3. An operator locally starts the one CCW pump.

4. The operator at the CCW intake structure fully opens the discharge valve on the running CCW pump.

5. Operators locally close the discharge valves on the LPSW pumps to be started. One pump shared by Units 1 and 2 and one pump for Unit 3 is needed for cooldown to Mode 5.

6. An operator locally starts the desired LPSW pumps.

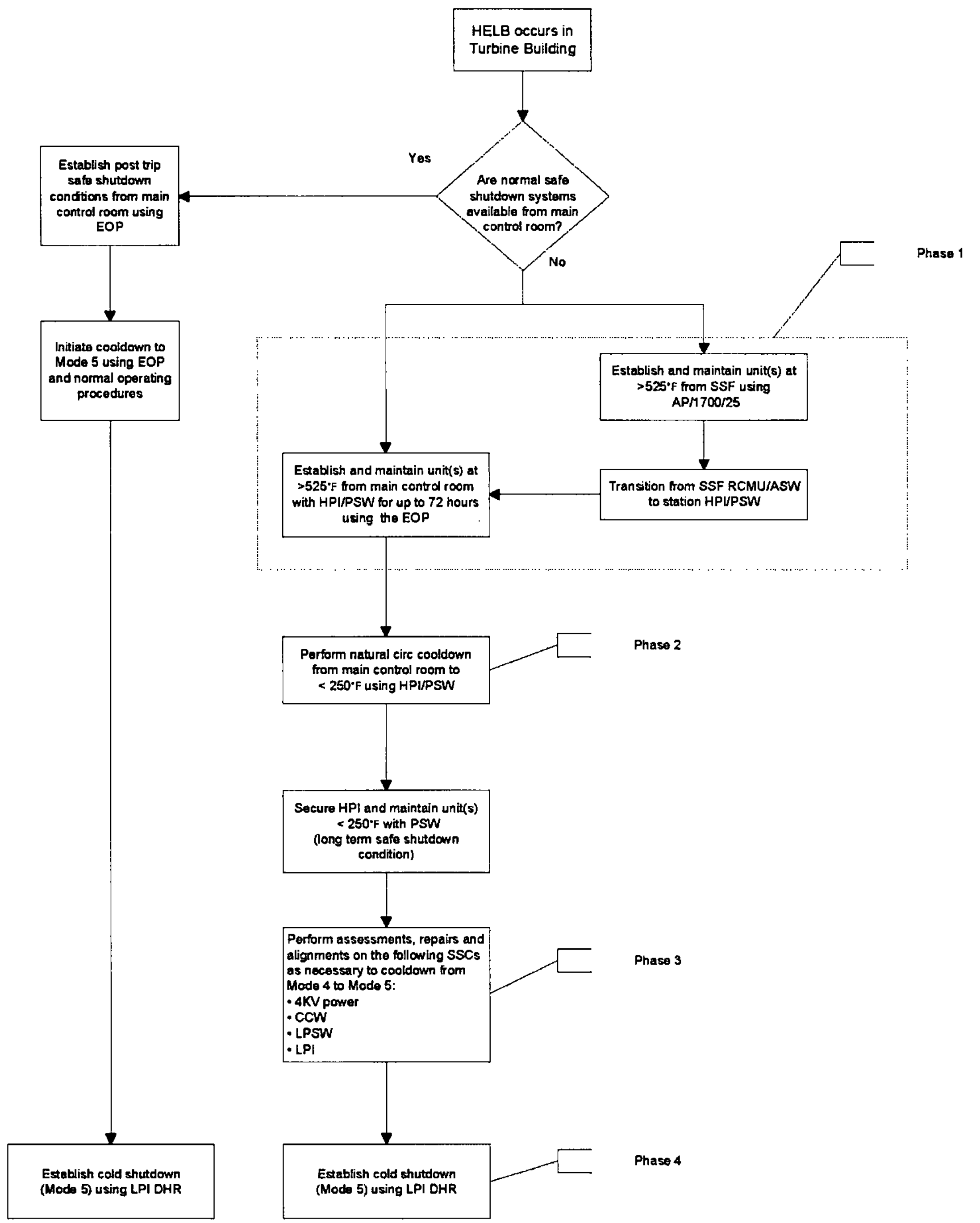
7. Operators slowly throttle open the discharge valve on the running LPSW pumps to fill the LPSW system.

8. Operators locally open LP-1 and LP-2 at the PVCP.

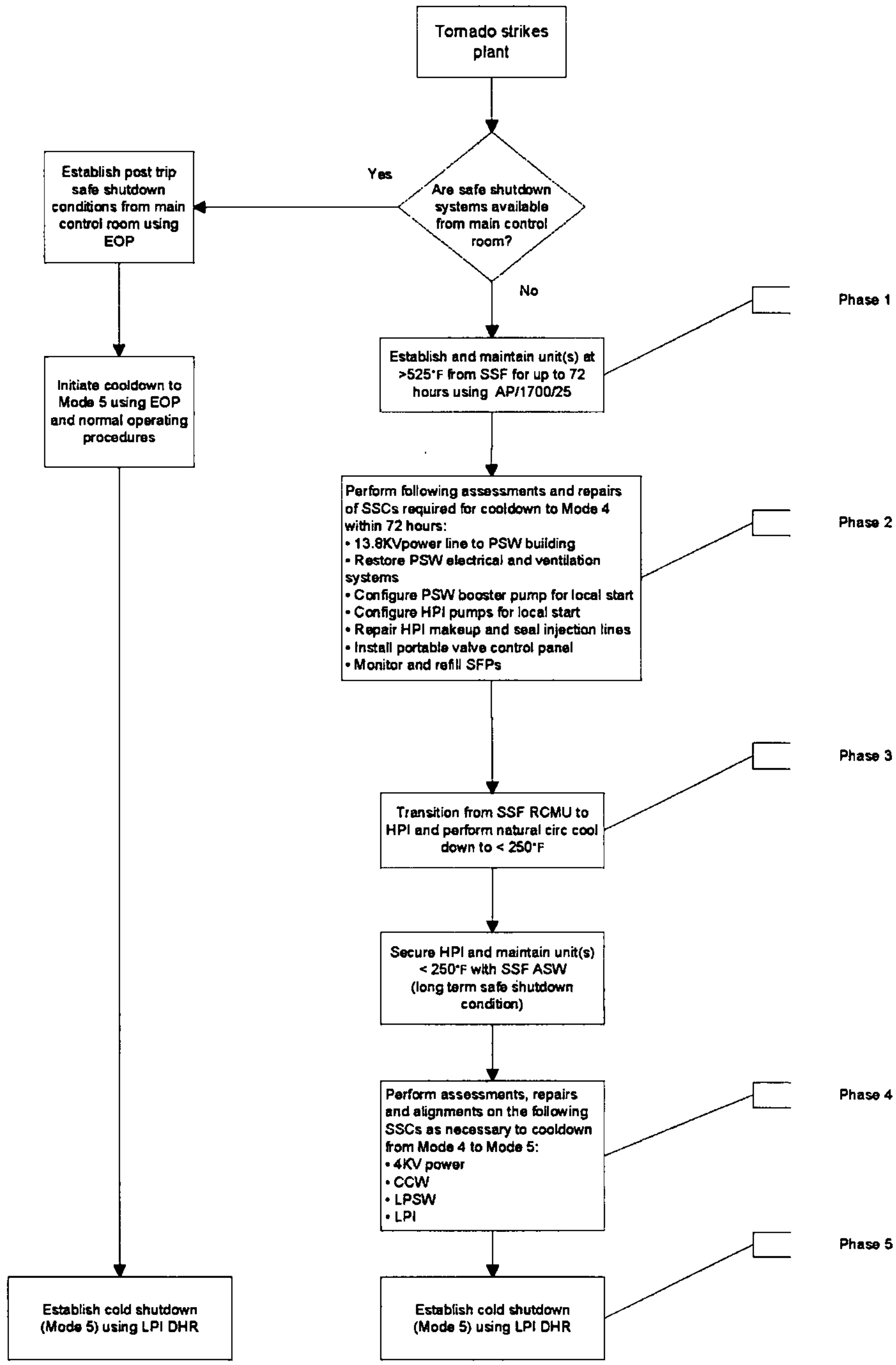
9. An operator locally starts one LPI pump on each unit.

10. An operator locally throttles open the LPI cooler outlet LPSW block valve to establish a cooldown to < 200°F.

RAI 61 HELB Event Overview



RAI 61 Tornado Event Overview



RAI 62

Provide a complete list of all modifications, design packages and supporting calculations which require prior NRC review and approval prior to changing the UFSAR associated with the LARs. In addition provide an executive summary for each calculation. The summary should include initial conditions, all assumptions, analyses performed, acceptance criteria, and a justification for the conclusion reached.

Duke Energy Response

There are two modifications that need NRC approval associated with the tornado and HELB license amendments. They are the Main Steam Isolation Valve (MSIV) modification and the Protected Service Water (PSW) system modification.

Design of the MSIV modification is in progress and submittal of a separate license amendment request addressing MSIV installation is planned to occur in 2013. The tornado and HELB license amendment requests commit to and rely on the MSIV modification. These license amendment requests do not include the necessary information to support licensing the MSIVs.

The PSW modification requires NRC review and approval. Some PSW related design change packages have been implemented using 10CFR 50.59 as listed in the table given later in this response. The scope of the PSW modification within these design change packages does not require NRC review and approval.

Summary of Protected Service Water Modification Response Contents

During a conference call with members of the NRC Staff on November 21, 2011, Duke Energy outlined the proposed structure of the response to this RAI. The proposed response structure was considered acceptable. In order to provide the information in the most efficient means possible, the Duke Energy response includes the following:

1. A System Description of the Protected Service Water System.

2. A discussion of the design of the Protected Service Water System and the application of industry codes and standards to the design.

3. Simplified mechanical, electrical, and civil drawings that illustrate the individual design packages that constitute the PSW modification.

4. Final Scope Documents for Design Change Packages that have been released to the field for implementation.

5. A list of calculations developed in support of, or impacted, and revised by the design changes identified in item 4, arranged by design change package. Copies of the associated, completed calculations are also included.

Items 1 and 2 are included in this response.

Items 3, 4, and 5 have been made available electronically to the NRC Staff for audit, or will be made available as the design change packages are issued for field implementation. All information is scheduled to be available for NRC Staff audit no later than March 2, 2012.

1. Protected Service Water System Description

The Protected Service Water (PSW) system is designed as a standby system for use under emergency conditions. The PSW System includes a dedicated power system. The PSW System provides added "defense-in-depth" protection by serving as a backup to existing safety systems and as such, the system is not required to comply with single failure criteria. The PSW system is provided as an alternate means to achieve and maintain a stable RCS pressure and temperature for one, two, or three units following postulated event scenarios that result in the loss of 4160V essential power.

The PSW System is also capable of cooling the RCS to 250 °F and maintaining this condition until damage repairs can be implemented to proceed to cold shutdown. Failures in the PSW system will not cause failures or inadvertent operations in existing plant systems. The PSW system is fully controllable from the main control rooms and will be activated when existing redundant emergency systems are not available.

The PSW System can maintain these conditions for all three units for an extended period of operation during which time other plant systems required to cool down to Mode 5 conditions will be restored and brought into service as required.

The mechanical portion of the PSW system is designed to provide decay heat removal by feeding Keowee Lake water to the secondary side of the steam generators. The system, consisting of one booster pump, one high head pump and a portable pump, shall be capable of providing 375 gpm per unit at 1082 psig within 15 minutes following the initiating event. In addition, the system is designed to supply Keowee Lake water at 10 gpm per unit to the HPI pump motor coolers.

The PSW system utilizes the inventory of lake water contained in the plant Unit 2 Condenser Circulating Water (CCW) embedded piping. The PSW pumps are located in the Auxiliary Building at Elev.771' (except the portable pump) and take suction from the Unit 2 CCW embedded piping and discharge into the steam generators of each unit via separate lines into the emergency feedwater headers. The raw water is vaporized in the steam generator removing residual heat and is dumped to atmosphere. The Unit 2 CCW embedded piping is interconnected with Units 1 & 3. For extended operation, the PSW portable pump with a flow path capable of taking suction from the intake canal and discharging into the Unit 2 CCW line, is designed to provide a backup supply of water to the PSW system in the event of loss of CCW and subsequent loss of CCW siphon flow. The PSW portable pump is installed manually according to procedures.

The piping system has pump minimum flow lines that discharge back into the Unit 2 CCW embedded piping. For flow testing to the steam generators, the system is connected to a condensate water source located in the Turbine Building that is normally isolated using valves in the Auxiliary Building.

The PSW pumps and motor operated and solenoid valves required to bring the system into service are controlled from the main control rooms. Check valves and manual handwheel operated valves are used to prevent back-flow, accommodate testing, or are used for system isolation. Periodic testing of the PSW valves and pumps (except the portable pump) will be performed in accordance with the Inservice Testing (IST) program.

A separate PSW electrical equipment structure is provided for major PSW electrical equipment. Power is provided from the KHU via a tornado protected underground path. The PSW building is designed for tornado related loads (wind, ΔP, missiles) in accordance with Regulatory Guide 1.76, Rev.1, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants." The structure is also designed for seismic bedrock input accelerations in accordance with the current licensing basis of the Oconee station. Alternate power is provided by a transformer connected to a 100kV overhead transmission line that receives power from the Central Tie Switchyard located approximately 8 miles from the plant. These external power sources provide power to transformers, switchgear, breakers, load centers, batteries, and battery chargers located in the PSW electrical equipment structure.

The PSW HVAC is designed to maintain the Transformer Space (main equipment area) and the Battery rooms within their design temperature range. There are two redundant battery systems inside the PSW Building. The redundant battery banks are located in different rooms separated by fire rated walls. One HVAC system is QA-1; the other is non-QA. The hydrogen removal fans shall maintain the hydrogen in the Battery rooms below 2% in accordance with IEEE 484-2002.

2. Design of the Protected Service Water System and the Application of Industry Codes and Standards to the Design.

The PSW System requires not only the installation of new equipment and structures but also requires modification of existing systems and structures which have been constructed in accordance with various industry codes and standards effective dates during the past 40 years of ONS operation. Depending on the scope of work, location of work, new installation or modification of existing equipment and structures, or effective design date, the appropriate revision and date of the industry code or standard is utilized.

The PSW Project is being implemented via thirty nine (39) Design Change Packages (DCPs) with associated Engineering Changes (ECs) as tabulated below.

Design Change Package / Title / Engineering Change #	Status
OD100937 U1 VITAL I&C CABLE REROUTE EC91829	Field Issued
OD100941 U1 MCR ADDITIONS EC91830	Field Issued
OD100950 U1 POWER TO HPI EC91834	Field Issued
OD200925 U2 PZR HTRS & BATT CHRGR PWR EC91849	Field Issued
OD200934 U2 CCW MINI-FLOW EC 91850	Field Issued
OD200938 U2 VITAL I&C CABLE REROUTE EC91851	Field Issued
OD200942 U2 MCR ADDITIONS Pre-Outage EC91853	Field Issued
OD200945 U2 MCR ADDITIONS Outage EC91852	Field Issued
OD200953 U2 POWER TO HPI OUTAGE EC91857	Field Issued
OD200954 U2 POWER TO HPI Pre-Outage EC91858	Field Issued
OD300935 U3 CONDENSATE SUCTION EC91860	Field Issued

Design Change Package / Title / Engineering Change #	Status
OD300939 U3 VITAL I&C CABLE REROUTE EC91861	Field Issued
OD300943 U3 MCR ADDITIONS Pre-Outage EC91863	Field Issued
OD300955 U3 MCR ADDITIONS Outage EC91866	Field Issued
OD300958 U3 POWER TO HPI Pre-Outage EC91869	Field Issued
OD500920 PSW BUILDING ERECTION EC91870	Field Issued
OD500921 PSW BUILDING EQUIP Part A EC91871	Field Issued
OD500921 PSW BUILDING EQUIP Part B EC91833	Field Issued
OD500922 POWER FEED TO PSW BLDG EC91873	Field Issued
OD500923 13.8kV PSW FEED FROM 100kV EC91874	Field Issued
OD500928 SSF FEED FRM PSW EC91876	Field Issued
OD500932 HEADER PIPING - PUMP TO SG'S EC91877	Field Issued
OD500936 AUX BLDG CABLE TRAY EC91879	Field Issued
OD500940 KEOWEE EMER START REROUTE EC91880	Field Issued
OD500947 UNDERGROUND DUCT BANKS EC91881	Field Issued
OD100924 U1 PZR HTRS & BATT CHRGR PWR EC91826	Design In-Progress
OD300926 U3 PZR HTRS & BATT CHRGR PWR EC91859	Design In- Progress
OD300957 U3 POWER TO HPI Outage EC91868	Design In-Progress
OD500921 PSW BUILDING EQUIP Part C EC91856	Design In-Progress
OD500927 KEOWEE AC POWER TIE-INS EC91875	Design In-Progress
- PUMP TIE-IN CABLE PACKAGE EC106526	Design In-Progress
OD500933 NEW PSW PUMP Tie-IN EC91878	Design In-Progress
OD100929 U1 S/G PIPING TIE-INS EC91884	Installation under 50.59
OD500927 KEOWEE AC POWER Mech EC106349	Installation under 50.59
OD500944 ASW DEMOLITION (ALL UNITS) EC95601	Installation under 50.59
OD500946 KEOWEE DEMOLITION EC92519	Installation under 50.59
OD500948 UNDERGROUND RELOCATION EC91832	Installation under 50.59
OE200930 U2 S/G PIPING TIE-INS EC91885	Installation under 50.59
OE300931 U3 S/G PIPING TIE-INS EC91887	Installation under 50.59

Scope descriptions and calculations for each of the above twenty five (25) completed DCPs/ECs which have been field issued are available for NRC review and audit. These scope descriptions and calculations contain executive summaries/problem statements, applicable QA conditions, applicable codes and standards, assumptions and analyses. Based on discussions with the NRC Staff and the volume of this material (25 DCPs/ECs and associated scope

descriptions with 400+ calculations for those DCPS/ECs), Duke Energy made PDF files with this information available electronically for audit by the NRC Staff. The electronic files also include a PSW Master AC Power Diagram (Electrical), PSW System (Flow Diagram) In Standby (Mechanical) and a PSW Duct bank and Manhole Location Plan (Civil) to provide summary pictorial scope descriptions to facilitate NRC review.

The attached table contains the industry codes and standards and their effective dates which are being used for PSW System design:

CIVIL	
Installation of New Safety Related Structures	
Protected Service Water (PSW) Building	ACI 349-97, Code Requirements for Nuclear Safety Related Concrete Structures (and its supplements) except Appendix B.
	AISC, Manual of Steel Construction, 13th Edition, 2006
	ANSI/AISC N690-1984, Specifications for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities.
	Regulatory Guide 1.122, Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components, Revision 1, February 1978
	Regulatory Guide 1.142, Safety-Related Concrete Structures for Nuclear Power Plants, Revision 2, November 2001
	Regulatory Guide 1.199, Anchoring Components and Structural Supports in Concrete, November 2003.
	Regulatory Guide 1.76, Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants, Revision 1, March 2007.
	Topical Report BC-TOP-9A, Design of Structures for Missile Impact, Revision 2, Bechtel Power Corp, 1974.
Conduit Duct Banks and Manholes Connecting New PSW Building to Existing Keowee Trench	ACI 349-97, Code Requirements for Nuclear Safety Related Concrete Structures (and its supplements) except Appendix B.
	AISC, Manual of Steel Construction, 13th Edition, 2006
	ANSI/AISC N690-1984, Specifications for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities.
	Regulatory Guide 1.142, Safety-Related Concrete Structures for Nuclear Power Plants, Revision 2, November 2001
	Regulatory Guide 1.76, Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants, Revision 1, March 2007.
	Topical Report BC-TOP-9A, Design of Structures for Missile Impact, Revision 2, Bechtel Power Corp, 1974.

• Conduit Duck Banks, Cable Trenches, Elevated Raceway and Manhole Connecting New PSW Building to Existing Auxiliary Building	ACI 318-63, Building Code Requirements for Reinforced Concrete
• Conduit Ductbanks Connecting New PSW Building to Existing Standby Shutdown Facility (SSF) Cable Trench and SSF Building	ACI 318-71, Building Code Requirements for Reinforced Concrete
	ACI 349-97, Code Requirements for Nuclear Safety Related Concrete Structures (and its supplements) except Appendix B.
	AISC, Manual of Steel Construction, 13th Edition, 2006
	ANSI/AISC N690-1984, Specifications for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities.
	Regulatory Guide 1.142, Safety-Related Concrete Structures for Nuclear Power Plants, Revision 2, November 2001
	Regulatory Guide 1.76, Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants, Revision 1, March 2007.
	Topical Report BC-TOP-9A, Design of Structures for Missile Impact, Revision 2, Bechtel Power Corp, 1974.
• Cable Tray System Within the PSW Building **• Cable Tray System Within the Auxiliary Building** **• Cable Tray System Within the Keowee Hydro Station**	Seismic Qualification Utility Group (SQUG), Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment, Revision 3A.
Installation of Equipment	
Attachment of New Equipment Housed Within the New PSW Building	AISC, Manual of Steel Construction, 13th Edition, 2006
	AISI, North American Specification for the Design of Cold-Formed Steel Structural Members, 2001 Edition
	ANSI/AISC N690-1984, Specifications for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities
	Regulatory Guide 1.122, Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components, Revision 1, February 1978
	Regulatory Guide 1.142, Safety-Related Concrete Structures for Nuclear Power Plants, Revision 2, November 2001
	Regulatory Guide 1.199, Anchoring Components and Structural Supports in Concrete, November 2003

Attachment of New Equipment Housed Within the Auxiliary Building	ACI 318-63, Building Code Requirements for Reinforced Concrete
	AISC, Manual of Steel Construction, 13th Edition, 2006
Attachment of New and Relocated Equipment Within the Standby Shutdown Facility	AISC, Manual of Steel Construction, Seventh Edition, 1969
Attachment of New Equipment Within the Keowee Hydro Station(i.e. Keowee Powerhouse, Keowee Service Building Substructure and Keowee Breaker Vault)	AISC, Manual of Steel Construction, Eighth Edition, 1980
Modification of Existing Structures	
• Core Bores within the Keowee Hydro Station **• Penetrations Within the Auxiliary Building (i.e. Ductbank through Unit 3 South Wall, Core Bores, HVAC Penetrations in Pump Room)** **• Core Bores Within the SSF Building**	ACI 318-63, Building Code Requirements for Reinforced Concrete
Installation of Elevated Walkway Within Auxiliary Building	AISC, Manual of Steel Construction, Eighth Edition, 1980
Conduit Ductbank Penetration into and out of SSF Trench and Upgrade of Existing Cable Trench South of SSF Trench to Safety Related Status	ACI 349-01, Code Requirements for Nuclear Safety Related Concrete Structures
	AISC, Manual of Steel Construction, Ninth Edition, 1989
	AISC, Manual of Steel Construction, 13th Edition, 2006
	Regulatory Guide 1.199, Anchoring Components and Structural Supports in Concrete, November 2003
	Regulatory Guide 1.76, Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants, Revision 1, March 2007
	Topical Report BC-TOP-9A, Design of Structures for Missile Impact, Revision 2, Bechtel Power Corp, 1974

MECHANICAL	
Installation of New and /or Modification of Existing Piping for ASW/PSW Systems	ASME Section XI 1998 Edition, 2000 Addenda Rules for Inservice Inspection of Nuclear Power Plant Components
	ASME (USAS) B31.1.0, 1967 Edition Power Piping
Installation of New and /or Modification of Existing Piping Supports (all Locations)	AISC Manual of Steel Construction, 6th Edition (1963) with some member properties taken from later editions.
New Heating / Ventilation System Housed Within the New PSW Building	ASME AG-1 2003 Code on Nuclear Air and Gas Treatment
Installation of New and /or Modification of Existing HVAC Supports (all Locations)	AISC Manual of Steel Construction, 6th Edition (1963) with some member properties taken from later editions.
• **New Valves Installed in New or Modified Piping Systems** • **New Pumps Installed in New or Modified Piping Systems**	ASME Section III Class 3 1998 Edition, 2000 Addenda Rules for Construction of Nuclear Facility Components

ELECTRICAL	
Medium Voltage Switchgear (13.8 kV / 4.16kV)	
• **13.8kV / 4.16kV 10MVA Transformer (PSW Building)** • **13.8kV Switchgear (PSW Building)** • **13.8kV Manual Disconnect Switch (PSW Building)** • **4.16kV Switchgear (SSF & PSW Building)**	IEEE Standard 308-2001: Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
	IEEE Standard 323-1983: Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations
	IEEE Standard 344-1975: Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
	IEEE Standard C37.82-1987: Standard for the Qualification of Switchgear Assemblies for Class 1E Applications in Nuclear Power Generating Stations
13.8kV Switchgear (Keowee)	IEEE Standard 308-2001: Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
	IEEE Standard 323-1983: Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations
13.8kV Transition Junction Box (Keowee)	IEEE Standard 344-1975: Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
• **5kV Manual Disconnect Alignment Switch (Aux Building)** • **5kV Motor Operated Manual Transfer Switch (Aux Building)**	IEEE Standard 308-2001: Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
	IEEE Standard 323-1983: Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations
	IEEE Standard 344-1975: Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations

Low Voltage Switchgear (600V)	
• 4.16kV / 600V Transformer (PSW Building) • 600V Load Center (PSW Building)	IEEE Standard 308-2001: Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
	IEEE Standard 323-1983: Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations
	IEEE Standard 344-1975: Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
	IEEE Standard C37.82-1987: Standard for the Qualification of Switchgear Assemblies for Class 1E Applications in Nuclear Power Generating Stations
600V Motor Control Center (Aux. Building & PSW Building)	IEEE C37.82-1987: Standard for the Qualification of Switchgear Assemblies for Class 1E Applications in Nuclear Energy Generating Stations
	IEEE Standard 308-2001: Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
	IEEE Standard 323-1983: Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations
	IEEE Standard 344-1975: Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
	IEEE Standard 649-2006: Standard for Qualifying Class 1E Motor Control Centers for Nuclear Energy Generating Stations
	UL 489-2002: Molded-Case Circuit Breakers, Molded-Case Switches and Circuit-Breaker Enclosures
• 600V Auto Transfer Switch (Aux. Building) • 600V Manual Transfer Switch (Aux. Building & PSW Building)	IEEE Standard 308-2001: Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
	IEEE Standard 323-1983: Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations
	IEEE Standard 344-1975: Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
600V Manual Transfer Switch (Aux. Bldg PSW Pump Room)	IEEE Standard 308-2001: Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
	IEEE Standard 323-1974: Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations
	IEEE Standard 344-1975: Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations

125V DC Power Source	
125V DC Batteries (PSW Building)	IEEE Standard 308-1991: Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
	IEEE Standard 323-1974: Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations
	IEEE Standard 344-1975: Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
	IEEE Standard 450-2002: IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications
	IEEE Standard 484-2002: IEEE Recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations
	IEEE Standard 485-1997 (R2003): IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications
	IEEE Standard 535-2006: Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations
125V DC Battery Charger (PSW Building)	IEEE Standard 308-1991: Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
	IEEE Standard 323-1974: Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations
	IEEE Standard 344-1975: Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
	IEEE Standard 650-2006: Standard for Qualification of Class 1E Static Battery Chargers and Inverters for Nuclear Power Generating Stations
	IEEE Standard 946-2004: Recommended Practice for the Design of DC Auxiliary Power System for Generating Stations
• 125V DC Distribution Center (PSW Building) **• 125V DC Panel Boards (Aux. Building & PSW Building)**	IEEE Standard 308-2001: Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
	IEEE Standard 323-1983: Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations
	IEEE Standard 344-1975: Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
	IEEE Standard 946-2004: Recommended Practice for the Design of DC Auxiliary Power System for Generating Stations
	UL 489-2002: Molded-Case Circuit Breakers, Molded-Case Switches and Circuit-Breaker Enclosures

Cables	
Cables (13.8kV, 4.16kV, 600V, 125V PSW Bldg, Aux Bldg, SSF)	IEEE 383-1974: Standard for Type Test of Class 1E Electric Cables, field Splices, and Connections for Nuclear Power Generating Stations
	IEEE Standard 323-1974: Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations
	IEEE 1202-1991: Standard for Flame Testing of Cables for Use in Cable Tray in Industrial and Commercial Occupancies.
13.8kV Cables (Keowee)	IEEE 383-1974: Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations
	IEEE Standard 323-1974: Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations
Cables (Instrumentation & Control)	IEEE 383-1974: Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations
	IEEE Standard 323-1974: Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations

RAI 107

To ensure licensing-basis clarity and component operability, Technical Specifications (TSs) need to properly address the PSW system in a manner that is consistent with the Standard TS requirements that have been established for the functions that are being performed by similar systems. For example, the minimum required mission time should be 7 days and the completion times should be limited to 72 hours in most cases. The proposed TS for the PSW system allow the system to be out of service for up to 45 days while maintenance is being performed on the system. The proposed TS does not put restrictions on other diverse systems (SSF) that are also used for tornado, HELB and fire mitigation while the PSW system is out of service. The suction source for the PSW system and the SSF are the same (Unit 2 circulation cooling water piping (CCW)). When the Unit 2 CCW piping is dewatered both the PSW system, and the SSF are out of service and cannot perform their intended functions. The proposed PSW TSs does not address this situation. Please address each of the above concerns.

Duke Energy Response

Duke Energy reviewed 44 other plant TSs and TS Bases to ascertain whether these plants had similar Criterion 4 systems that could be used to correlate Conditions, Required Actions, and Completion Times. Duke Energy also reviewed the BWOG, CEOG, WOG, and BWR STS. The review did not identify any Technical Specifications directly comparable to PSW. The closest comparison was the Safe Shutdown Makeup Pump System at Quad Cities. Also, the BWR plants have a single train RCIC system. These systems have a 14-day restoration time.

As concluded from the information given above, the STS emergency core cooling system guidelines are not directly applicable to the PSW system; however, Duke Energy determined that changes were warranted for the PSW system TSs. Specifically, Action A, completion time (CT) was revised to 14 days which is reasonable based on the SSF Auxiliary Service water and Reactor Coolant Makeup systems being operable and the low probability of a tornado or HELB event occurring during this time period. For Action C, the CT was revised to 30 days and the note was removed. Surveillance requirements were added for the battery charger (aligned with STS) and the PSW portable pump. The PSW Battery Parameters TS was revised to align with STS. As such, revised proposed PSW TSs are provided. In addition, Duke Energy proposes the following as a license condition to support implementation of the proposed TS changes:

"Upon implementation of the Amendment, TS SR 3.7.10.6 and SR 3.7.10.10 shall be considered met based on appropriate post modification testing. Following implementation, the first performance of SR 3.7.10.6 and SR 3.7.10.10 is due at the next refueling outage of each Oconee unit after implementation of this amendment."

The effect of having both the PSW and SSF systems out of service at the same time (TS Condition B) was evaluated to determine whether or not the condition represented a loss of safety function (LOSF).

The LOSF evaluation considered:

1. 10CFR50.36 requires a technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

(A) *Criterion 1.* Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

(B) *Criterion 2.* A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(C) *Criterion 3.* A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(D) *Criterion 4.* A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Because of the risk reduction benefits provided, the new PSW system satisfies only Criterion 4 of 10 CFR 50.36 and thus the condition does not represent a LOSF.

2. The PSW System is not an engineered safety feature system and no credit is taken in the safety analyses for PSW system operation. The PSW provides additional "defense in-depth" protection by serving as a backup to existing safety systems. PSW provides an alternate means to achieve and maintain a stable RCS pressure and temperature for one, two, or three units following postulated event scenarios, e.g., tornado and HELB events, and a loss of Lake Keowee event.

   Likewise, the SSF serves as a backup for existing safety systems to provide an alternate and independent means to achieve and maintain one, two, or three Oconee units in MODE 3 with average RCS temperature ≥ 525°F (unless the initiating event causes the unit to be driven to a lower temperature) for up to 72 hours following 10CFR50 Appendix R fire, a Turbine Building flood, sabotage, Station Blackout (SBO), or tornado missile events.

   The PSW System is capable of cooling the RCS to approximately 250°F and maintaining this condition until damage repairs can be implemented to proceed to cold shutdown. Failures in the PSW system will not cause failures or inadvertent operations in existing plant systems. The PSW system is fully controllable from the main control rooms and will be activated when existing redundant emergency systems are not available.

   For a tornado, the overall objective is to utilize the tornado-protected SSF system to maintain the units in Mode 3 for up to 72 hours while damage control measures are completed to restore any unavailable PSW system equipment needed to cooldown the units to approximately 250°F. This temperature is the least that can be attained using the Steam Generators (SGs) for cooldown. The PSW system can be used to maintain these conditions while additional repairs to systems, structures, and components (SSCs) required to transition the units to Mode 5 are completed.

   For HELBs inside the Turbine Building resulting in loss of 4160 essential power, either the SSF or the PSW systems are used for safe shutdown. The SSF System is limited to 72 hours and is not credited for cooling the units beyond Mode 3 conditions (unless the initiating event causes the unit to be driven to a lower temperature). The PSW system is designed to further cooldown to approximately 250°F and for longer-term operation beyond the initial 72 hours.

Therefore, the concurrent loss of both systems does not result in a loss of a safety function needed to mitigate a design basis accident.

As mentioned previously, Duke Energy's review of other plant TSs and TS Bases and STS did not identify any TSs directly comparable to PSW. The closest comparison was the Safe Shutdown Makeup Pump (SSMP) System at Quad Cities (QC). Also, the BWR plants have a single train RCIC system. These systems (QC SSMP & RCIC) have a 14-day restoration time. The QC SSMP System, which has a 14-day AOT, is described as a backup to the Unit 1 and 2 RCIC systems. The BWR RCIC system is considered backup to the high pressure core spray system (HPCS). The allowed outage time for the HPCS, RCIC, or SSMP systems are not dependent upon the operability of the other systems. Duke Energy is proposing to reduce the allowed outage time for an inoperable PSW System from 14 to 7 days when the SSF is inoperable concurrently. Duke Energy considers it as prudent and appropriate to match the AOT of the more restrictive system when both (partially redundant systems) are concurrently inoperable. Any further reduction is not warranted since these systems are not used to mitigate design basis accidents.

3.7 PLANT SYSTEMS

3.7.10 Protected Service Water (PSW) System

LCO 3.7.10 The PSW System shall be OPERABLE

APPLICABILITY: MODES 1, 2, and 3
MODE 4 when steam generators are relied upon for heat removal.

ACTIONS

------------------------------NOTE------------------------------
LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. PSW System is inoperable.	A.1 Restore PSW System to OPERABLE status.	14 days
B. PSW System is inoperable. <u>AND</u> SSF Systems are inoperable.	B.1 Restore PSW System to OPERABLE status.	7 days
C. -----------NOTE----------- Condition may only be entered when contingency actions for the KHU Emergency Power, Emergency Feedwater, High Pressure Injection, Elevated Water Storage Tank, and 230kV Switchyard have been implemented. ------------------------------ Required Action and associated Completion Time of Condition A or B not met when PSW inoperable due to maintenance.	C.1 Restore to OPERABLE status.	30 days from discovery of initial inoperability.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met. OR Required Action and associated Completion Time of Condition A or B not met for reasons other than Condition C.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 84 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Verify required PSW battery terminal voltage is ≥ 125 VDC on float charge.	7 days
SR 3.7.10.2	Verify that the KHU underground can be aligned to and power the PSW electrical system.	92 days
SR 3.7.10.3	--------------NOTE-------------- Not applicable to the PSW portable pump. -------------------------------- Verify that the developed head of the PSW pumps at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.7.10.4	Verify battery capacity of required battery is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	24 months

(continued)

SURVEILLANCE		FREQUENCY
SR 3.7.10.5	Verify each PSW battery charger supplies ≥ 300 amps at greater than or equal to the minimum established float voltage for ≥ 8 hours. <u>OR</u> Verify each battery charger can recharge the battery to the fully charged state within 24 hours while supplying the largest combined demands of the various continuous steady state loads, after a battery discharge to the bounding design basis event discharge state.	24 months
SR 3.7.10.6	Verify that the PSW switchgear can be aligned and power either the "A" or "B" HPI pumps.	24 months
SR 3.7.10.7	Verify that the switches used for the "A" or "B" HPI pumps, pressurizer heaters, PSW control and electrical panels, and miscellaneous valves, are OPERABLE.	24 months
SR 3.7.10.8	Verify that the PSW pumps can be used to provide adequate cooling water flowrate to the HPI pump motor coolers.	In accordance with the Inservice Testing Program
SR 3.7.10.9	Verify the developed head of the PSW portable pump at the flow test point is greater than or equal to the required developed head.	24 months
SR 3.7.10.10	Verify that the PSW pumps can be aligned and provide flow to each unit's Steam Generator (SG).	24 months

B 3.7 PLANT SYSTEMS

B 3.7.10 Protected Service Water (PSW) System

BASES

BACKGROUND

The Protected Service Water (PSW) system is designed as a standby system for use under emergency conditions. The PSW System includes a dedicated power system. The PSW System provides added "defense in-depth" protection by serving as a backup to existing safety systems and as such, the system is not required to comply with single failure criteria. The PSW system is provided as an alternate means to achieve and maintain a stable RCS pressure and temperature for one, two, or three units following postulated event scenarios, e.g., tornado and high energy line break (HELB) events, and a loss of Lake Keowee event.

The PSW System is also capable of cooling the RCS to 250 °F and maintaining this condition until damage repairs can be implemented to proceed to cold shutdown. Failures in the PSW system will not cause failures or inadvertent operations in existing plant systems. The PSW system is fully controllable from the main control rooms and will be activated when existing redundant emergency systems are not available.

For a tornado event, the overall objective is to utilize the tornado-protected SSF system to maintain the units in Mode 3 for up to 72 hours while damage control measures are completed to restore any unavailable PSW/HPI system equipment needed to cooldown the units to approximately 250 °F. This temperature is the least that can be attained using the Steam Generators (SGs) for cooldown. The PSW system can be used for an extended period of operation while additional repairs to systems, structures, and components (SSCs) required to transition the units to Mode 5 are completed.

For HELBs inside the Turbine building resulting in loss of 4160 essential power, either the SSF or the PSW system are used for safe shutdown. The SSF System is limited to 72 hours and is not credited for cooling the units beyond Mode 3 conditions (unless the initiating event causes the unit to be driven to a lower temperature). The PSW system is designed to further cooldown to approximately 250 °F and for longer-term operation beyond the initial 72 hours. The PSW System can maintain

BACKGROUND (continued)

these conditions for all three units for an extended period of operation during which time other plant systems required to cool down to Mode 5 conditions will be restored and brought into service as required.

The mechanical portion of the PSW system is designed to provide decay heat removal by feeding Keowee Lake water to the secondary side of the steam generators. The system, consisting of one booster pump, one high head pump and a portable pump , shall be capable of providing 375 gpm per unit at 1082 psig within 15 minutes following the initiating event. In addition, the system is designed to supply Keowee Lake water to the HPI pump motor coolers.

The PSW system utilizes the inventory of lake water contained in the plant Unit 2 CCW embedded piping. The PSW pumps are located in the Auxiliary Building at Elev.771' (except the portable pump) and take suction from the Unit 2 CCW embedded piping and discharges into the steam generators of each unit via separate lines into the emergency feedwater headers. The raw water is vaporized in the steam generator removing residual heat and is dumped to atmosphere. The Unit 2 CCW embedded piping is interconnected with Units 1 & 3. For extended operation, the PSW portable pump with a flow path capable of taking suction from the intake canal and discharging into the Unit 2 CCW line is designed to provide a backup supply of water to the PSW system in the event of loss of CCW and subsequent loss of CCW siphon flow. The PSW portable pump is installed manually according to procedures.

The piping system has pump minimum flow lines that discharge back into the Unit 2 CCW embedded piping. For flow testing to the steam generators, the system is connected to a condensate water source located in the Turbine Building that is normally isolated using valves in the Auxiliary Building.

The PSW pumps and motor operated and solenoid valves required to bring the system into service are controlled from the main control rooms. Check valves and manual handwheel operated valves are used to prevent back-flow, accommodate testing, or are used for system isolation. Periodic testing of the PSW valves and pumps (except the portable pump) will be performed in accordance with the Inservice Testing (IST) program.

The PSW electrical system is designed to provide power to PSW mechanical and electrical components as well as other system components needed to establish and maintain a safe shutdown condition. A separate PSW electrical equipment structure is provided for major PSW electrical equipment. Power is provided from the KHU via a

BASES

BACKGROUND (continued)

tornado protected underground path. Alternate power is provided by a transformer connected to a 100 kV overhead transmission line that receives power from the Central Tie Switchyard located approximately 8 miles from the plant. These external power sources provide power to transformers, switchgear, breakers, load centers, batteries, and battery chargers located in the PSW electrical equipment structure.

The PSW HVAC is designed to maintain the Transformer Space (main equipment area) and the Battery Rooms within their design temperature range. There are two redundant battery systems inside the PSW Building. The redundant battery banks are located in different rooms separated by fire rated walls. The HVAC System consists of two (2) systems; One (1) system (non-QA) runs continually to maintain the Transformer Space and Battery Rooms environmental profile; the other system (QA-1) is designed to actuate whenever the non-QA system is not providing its design function (i.e., following a tornado, component, or power failure). Both systems have redundant features that provide increased reliability. The hydrogen removal fans shall maintain the hydrogen in the Battery rooms below 2% in accordance with IEEE 484-2002. There are multiple thermostats in each Battery Room to ensure temperatures are maintained within acceptable limits.

APPLICABLE SAFETY ANALYSES

The safety function of the PSW system is to supply cooling water for secondary side decay heat removal at full system pressure to all six (6) steam generators (SGs) following postulated event scenarios. A secondary safety function of the PSW system is, in combination with the HPI System, to provide borated water to the RCS pump seals and to provide primary RCS makeup. Two redundant sources of electrical power serve the PSW electrical switchgear.

Because portions of the PSW System are not completely protected from the effects of a tornado, the system is not credited during the initial 72 hours after a tornado strike to the station. During the first 72 hours, the SSF will be utilized until damaged portions of the PSW system, which would be required for continued cooldown of the units to approximately 250 °F are repaired. For HELBs occurring in the Turbine Building, the PSW System can be used as long as there is water contained in the underground CCW piping or until restorations are made to additional systems needed to cool down the units to Mode 5 conditions.

The PSW System is designed to mitigate the consequences of a loss of Lake Keowee event by providing emergency cooling water to one or more of the three Oconee Units' SGs and HPI pump motor coolers.

BASES

LCO

The PSW System is considered to be OPERABLE when its mechanical and electrical equipment, as well as associated support equipment, are OPERABLE. The system is designed to adequately perform these functions for one, two, or all three units concurrently. The system is aligned, controlled, and monitored, from the main control rooms.

For OPERABILITY, the following are required:

- One (1) booster pump and one (1) high head pump.
- A viable suction source from the embedded Unit 2 CCW piping to the PSW pumping system.
- Five (5) of the six (6) 125 VDC Vital I&C Normal Battery Chargers

The following are required to be powered from PSW (each unit):

- Either the "A" or "B" High Pressure Injection Pump.
- HPI valve needed to align the HPI pumps to the Borated Water Storage Tanks (HP-24).
- HPI valves and instruments that support RCP seal injection and RCS makeup.
- PSW booster pump flow to an HPI pump motor cooler.
- Pressurizer Heaters ($\geq$ 400 kW).
- RCS and Reactor Vessel Head high point vent valves
- PSW electrical system from either the KHU underground or 13.8 kV overhead power paths to support secondary side decay heat removal (SSDHR) and reactor coolant make-up (RCMU) functions.

PSW system dedicated instrumentation and controls located in each main control room:

- Two (2) high flow controllers (one per SG).
- Two (2) low flow controllers (one per SG).
- One (1) flow indicator (per SG).
- One (1) SG header isolation valve (one per unit).
- Two (2) HPI pump power transfer switches per unit.
- Power transfer switches to HPI valves needed to align the BWST to the HPI pumps.

BASES

APPLICABILITY

In MODES 1, 2, and 3, the PSW System is required to be OPERABLE and to function in the event that all normal and emergency feedwater systems are lost. In MODE 4, with RCS temperature above 212 °F, the PSW System may be used for heat removal via the steam generators. In MODE 4, the steam generators are used for heat removal unless this function is being performed by the Low Pressure Injection System.

In MODE 4 steam generators are relied upon for heat removal whenever an RCS loop is required to be OPERABLE or operating to satisfy LCO 3.4.6, "RCS Loops – Mode 4."

In MODES 5 and 6, the steam generators are not used for SSDHR and the PSW System is not required.

ACTIONS

The exception for LCO 3.0.4, provided in the Note of the Actions, permits entry into MODES 1, 2, 3 or 4 with the PSW not OPERABLE. This is acceptable because the PSW is not required to support normal operation of the facility or to mitigate a design basis accident.

A.1

With the PSW system inoperable, action must be taken to restore the system to OPERABLE status within 14 days. The 14-day Completion Time is reasonable based on the SSF Auxiliary Service Water and RCMU systems being OPERABLE and a low probability of a tornado or HELB event occurring that would require the PSW System during the 14 day time period.

B.1

With both the PSW and SSF Systems inoperable, action must be taken to restore the PSW system to OPERABLE status within 7days. The required action is not intended for voluntary removal of both systems from service that provide alternate means for safe shutdown. This required action is applicable if the PSW system is inoperable for reasons other than maintenance and the SSF is found to be inoperable, or if both the PSW and the SSF systems are found to be inoperable at the same time. The 7 day Completion Time is based on the redundant heat removal capabilities afforded by other safety systems, reasonable times for repairs, and the low probability of a tornado or HELB event occurring that would require the PSW System during this time period.

ACTIONS (continued)

C.1

If the Required Action and associated Completion Time of Condition A or B is not met when the PSW System is inoperable due to maintenance (e.g., dewatering of the Unit 2 CCW underground piping or repair of the PSW pump), action must be taken to restore the PSW System to OPERABLE status within 30 days. Operation for up to 30 days is permitted if the following risk-lowering contingency measures are taken. The 30 days is from the time of discovery of initial inoperability.

The condition contains a note indicating that contingency measures must be in place for specified equipment prior to entry. This provides additional assurance that key equipment (Keowee Hydroelectric Units (KHUs), Emergency Feedwater (EFW) pump, High Pressure Injection (HPI) pump, Elevated Water Storage Tank (EWST), and 230kV switchyard) that could be needed to mitigate the adverse effects of postulated events during the extended outage time period are available. The inoperability of this key equipment (KHU, EFW pump, HPI pump, EWST, and 230kV switchyard) does not preclude the entry into the condition nor does it require any action by this TS. Rather the appropriate actions for this equipment are specified in the applicable TS or SLC for the inoperable equipment. For example, if the 1A HPI pump becomes inoperable before entry or becomes inoperable after entry, only TS 3.5.2 (HPI), Condition A shall be entered for Unit 1 and the appropriate actions taken until the pump is restored. There is no impact on the TS 3.7.10 Condition C entry.

The following contingency measure strategy will be employed:

- Defer non-essential surveillances or other maintenance activities in the 230kV switchyard where human error could contribute to the likelihood of a loss of offsite power (LOOP). Technical Specification required surveillances and corrective maintenance of risk important equipment are examples of essential activities.
- Defer non-essential surveillances or other maintenance activities on risk significant equipment. This equipment includes the KHU emergency power system and both the associated unit's EFW motor-driven and turbine-driven pumps. Also included, the HPI pumps and associated equipment, and the EWST that provides backup cooling water to the HPI pump motor.

The following specific contingency measures are being taken to reduce the plant risk:

- No non-essential surveillances or other maintenance activities, or testing, will be conducted in the 230kV switchyard.

ACTIONS (continued)

- No non-essential surveillances or other maintenance activities, or testing, will be conducted on the Keowee Hydro Units' emergency power system and associated power paths.
- No non-essential surveillances or other maintenance activities, or testing, will be conducted on the unit's EFW motor-driven and turbine-driven pumps and associated equipment.
- No non-essential surveillances or other maintenance activities, or testing, will be conducted on the unit's HPI pumps and associated equipment.
- No non-essential surveillances or other maintenance activities, or testing, will be conducted on the EWST.

D.1 and D.2

If the Required Action and associated Completion Times of either Condition C or Conditions A or B are not met for reasons other than Condition C, the unit(s) must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and MODE 4 within 84. The allowed Completion Times are appropriate to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems, considering a three unit shutdown may be required.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or a battery cell) in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltage assumed in the battery sizing calculations. The 7 day frequency is consistent with manufacturer recommendations and IEEE-450.

SR 3.7.10. 2

This SR verifies the availability of the KHU associated with the underground power path to the PSW electrical system. Power path

SURVEILLANCE REQUIREMENTS (continued)

verification is included to demonstrate breaker OPERABILITY from the KHU to the PSW electrical system. This is accomplished by closing the Keowee to PSW Feeder Breakers. The 92 day Frequency is adequate based on operating experience to provide reliability verification without excessive equipment cycling for testing.

SR 3.7.10.3

This SR requires the PSW pumps be tested in accordance with the IST Program. The IST verifies the required flow rate at a discharge pressure to verify OPERABILITY. The SR is modified by a note indicating that it is not applicable to the PSW portable pump.

The specified Frequency is in accordance with the IST Program requirements. Operating experience has shown that these components usually pass the SR when performed at the IST Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.7.10.4

A battery service test is a special test of the battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length correspond to the design duty cycle requirements.

The Surveillance Frequency for this test is 24 months which is consistent with expected fuel cycle lengths.

SR 3.7.10.5

This SR verifies the design capacity of the battery charger. According to Regulatory Guide 1.32, the battery charger supply is recommended to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensure that these requirements can be satisfied.

This SR provides two options. One option requires that each battery charger be capable of supplying 300 amps at the minimum established float voltage for 8 hours. The ampere requirements are based on the

SURVEILLANCE REQUIREMENTS (continued)

output rating of the charger. The voltage requirements are based on the charger voltage level after a response to a loss of AC power. The time period if sufficient for the charger temperature to have stabilized and to have been maintained for at least 2 hours.

The other option requires that the battery charger be capable of recharging the battery after a service test coincident with supplying the largest coincident demands of the various continuous steady state loads (irrespective of the status of the plant during which these demands occur). This level of loading may not normally be available following the battery service test and will need to be supplemented with additional loads. The duration for this test may be longer than the charger sizing criteria since the battery recharge is affected by float voltage, temperature, and the exponential decay in charging current. The battery is recharged when the measured charging current is ≤ 2 amps. The Surveillance Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these 24 month intervals.

SR 3.7.10.6

This SR verifies that the PSW switchgear can be aligned and power either the “A” or “B” HPI pumps once every 24 months.

SR 3.7.10.7

This SR verifies that the power transfer switches for the HPI pumps, pressurizer heaters, PSW control and electrical panels, and miscellaneous valves, are functional every 24 months.

SR 3.7.10.8

This SR verifies that the PSW pumps can supply Keowee Lake water to the "A" or "B" HPI pump motor coolers per the IST program.

SR 3.7.10.9

This SR requires the PSW portable pump to be tested on a 24 month frequency and verifies the required flow rate at a discharge pressure to verify OPERABIITY.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The specified frequency is based on the pump being not QA grade and on operating experience that has shown it usually passes the SR when performed at the 24 month frequency.

SR 3.7.10.10

The ability to align, start, and control flow of the PSW system to each unit must be verified every 24 months. This includes verification that the PSW header isolation valves to each unit's SGs open upon demand and that flow can be throttled to each SG through the full range of operation.

REFERENCES

1. Nuclear Station Report ONDS-351, "Analysis of Postulated High Energy Line Breaks (HELBs) Outside of Containment," dated May 20, 2008.
2. IEEE-450-1995.
3. Regulatory Guide 1.32, February 1977.
4. Regulatory Guide 1.129, December 1974.

3.7 PLANT SYSTEMS

3.7.10a Protected Service Water (PSW) Battery Parameters

LCO 3.7.10a Battery parameters for the PSW batteries shall be within limits.

APPLICABILITY: When the PSW system is required to be OPERABLE.

ACTIONS

--NOTES--
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One battery on one train with one battery cell float voltage < 2.07 V.	A.1 Perform SR 3.7.10.1	2 hours
	<u>AND</u>	
	A.2 Perform SR 3.7.10a.1.	2 hours
	<u>AND</u>	
	A.3 Restore affected cell voltage ≥ 2.07 V.	24 hours
B. One battery on one train with float current > 2 amps.	B.1 Perform SR 3.7.10.1	2 hours
	<u>AND</u>	
	B.2 Restore battery float current to ≤2 amps.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
------------NOTE------------ Required Action C.2 shall be completed if electrolyte level was below the top of plates. C. One battery on one train with one or more cells electrolyte level less than minimum established design limits.	-------------NOTE--------------- Required Actions C.1 and C.2 are only applicable if electrolyte level was below the top of plates. C.1 Restore electrolyte level to above top of plates. AND C.2 Verify no evidence of leakage. AND C.3 Restore electrolyte level to greater than or equal to minimum established design limits.	 8 hours 12 hours 31 days
D. One battery on one train with pilot cell electrolyte temperature less than minimum established design limits.	D.1 Restore battery pilot cell temperature to greater than or equal to minimum established design limits.	12 hours
E. One battery with battery parameters not within limits.	E.1 Restore battery parameters for battery within limits.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met. OR One battery on one train with one or more battery cells float voltage < 2.07 V and float current > 2 amps.	F.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10a.1	--------------------NOTE-------------------- Not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.7.10.1. -- Verify battery float current is ≤2 amps.	7 days
SR 3.7.10a.2	Verify battery pilot cell voltage is ≥-2.07 V.	31 days
SR 3.7.10a.3	Verify battery connected cell electrolyte level is greater than or equal to minimum established design limits.	31 days

(continued)

ACTIONS (continued)

SR 3.7.10a.4	Verify battery pilot cell temperature is greater than or equal to minimum established design limits.	31 days
SR 3.7.10a.5	Verify battery connected cell voltage is ≥ 2.07 V.	92 days
SR 3.7.10.a.6	-----------------------NOTE--------------------------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. --- Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	60 months <u>AND</u> 12 months when battery shows degradation or has reached 85% of the expected life with capacity < 100% of manufacturer's rating <u>AND</u> 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

B 3.9 PLANT SYSTEMS

B 3.7.10a PSW Battery Parameters

BASES

BACKGROUND	This LCO delineates the limits on battery float current as well as electrolyte temperature, level, and float voltage for the PSW Power system batteries. In addition to the limitations of this Specification, the PSW Battery Monitoring and Maintenance Program specified in Specification 5.5.xx for monitoring various battery parameters that is based on the recommendations of IEEE Standard 450-1995, "IEEE Recommended Practice For Maintenance, Testing, And Replacement Of Vented Lead- Acid Batteries For Stationary Applications." The battery cells are of flooded lead acid construction with a nominal specific gravity of 1.215. This specific gravity corresponds to an open circuit battery voltage of approximately 120 V for 58 cell battery (i.e., cell voltage of 2.065 volts per cell (Vpc)). The open circuit voltage is the voltage maintained when there is no charging or discharging. Once fully charged with its open circuit voltage < 2.065 Vpc, the battery cell will maintain its capacity for 30 days without further charging per manufacturer's instructions. Optimal long term performance however, is obtained by maintaining a float voltage 2.20 to 2.2 Vpc. This provides adequate over-potential which limits the formation of lead sulfate and self discharge. The nominal float voltage of 2.22 Vpc corresponds to a total float voltage output of 128.8 V for a 58 cell battery.
APPLICABLE SAFETY ANALYSES	The safety function of the PSW system is to supply cooling water for the secondary side decay heat removal at full system pressure to all six (6) steam generators (SGs) following postulated event scenarios. A secondary safety function of the PSW system is, in combination with the HPI System, to provide borated water to the RCS pump seals and to provide primary RCS makeup. Two redundant sources of electrical power serve the PSW electrical switchgear. Because portions of the PSW System are not completely protected from the effects of a tornado, the system is not credited during the initial 72 hours after a tornado strike to the station. During the first 72 hours, the SSF will be utilized until damaged portions of the PSW system, which would be required for continued cooldown of the units to approximately

BASES

APPLICABLE SAFETY ANALYSES (continued)

250°F, are repaired. For HELBs occurring in the Turbine Building, the PSW System can be used as long as there is water contained in the underground CCW piping or until restorations are made to additional systems needed to cool down the units to Mode 5 conditions.

The PSW System is designed to mitigate the consequences of a loss of Lake Keowee event by emergency cooling water to one or more of the three Oconee Units' SGs and HPI pump motor coolers.

LCO

PSW Battery parameters must remain within acceptable limits to ensure availability of the required PSW DC power system to shut down the reactor and maintain it in a safe condition after an occurrence of a Tornado or High Energy Line Break inside the Turbine Building. Battery parameter limits are conservatively established, allowing continued PSW DC electrical system function even with limits not met. Additional preventative maintenance, testing, and monitoring performed in accordance with the PSW Battery Monitoring and Maintenance Program is conducted as specified in Specification 5.5.xx.

APPLICABILITY

The battery parameters are required solely for the support of the associated PSW electrical power systems. Therefore, battery parameter limits are only required when the PSW DC power source is required to be OPERABLE.

ACTIONS

A.1, A.2, and A.3

With one or more cells in a battery < 2.07 V, the battery cell is degraded. Within 2 hours verification of the required battery charger OPERABILITY is made by monitoring the battery terminal voltage (SR 3.7.10.1) and of the overall battery state of charge by monitoring the battery float charge current (SR 3.7.10a.1). This assures that there is still sufficient battery capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of one or more cells in a battery < 2.07 V, and continued operation is permitted for a limited period up to 24 hours.

Since the Required Actions only specify "perform," a failure of SR 3.7.10.1 or SR 3.7.10a.1 acceptance criteria does not result in this Required Action not met. However, if one of the SRs is failed the appropriate Condition(s), depending on the cause of the failures, is entered. If SR 3.7.10a.1 is failed then there is no assurance that there is still sufficient battery capacity to perform the intended function and the battery must be declared inoperable immediately.

BASES

ACTIONS (continued)

B.1 and B.2

One battery with float current >2 amps indicates that a partial discharge of the battery capacity has occurred. This may be due to a temporary loss of a battery charger or possibly due to one or more battery cells in a low voltage condition reflecting some loss of capacity. Within 2 hours verification of the required battery charger OPERABILITY is made by monitoring the battery terminal voltage. If the terminal voltage is found to be less than the minimum established float voltage there are two possibilities, the battery charger is inoperable or is operating in the current limit mode. Condition A addresses charger inoperability. If the charger is operating in the current limit mode after 2 hours that is an indication that the battery has been substantially discharged and likely cannot perform its required design functions. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within 12 hours (Required Action B.2). The battery must therefore be declared inoperable.

If the float voltage is found to be satisfactory but there are one or more battery cells with float voltage less than 2.07 V, the associated "OR" statement in Condition F is applicable and the. battery must be declared inoperable immediately. If float voltage is satisfactory and there are no cells less than 2.07 V there is good assurance that, within 12 hours, the battery will be restored to its fully charged condition (Required Action B.2) from any discharge that might have occurred due to a temporary loss of the battery charger.

A discharged battery with float voltage (the charger setpoint) across its terminals indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within 12 hours, avoiding a premature shutdown with its own attendant risk.

If the condition is due to one or more cells in a low voltage condition but still greater than 2.07 V and float voltage is found to be satisfactory, this is not indication of a substantially discharged battery and 12 hours is a reasonable time prior to declaring the battery inoperable.

BASES

ACTIONS

B.1 and B.2 (continued)

Since Required Action B.1 only specifies "perform," a failure of SR 3.7.10.1 acceptance criteria does not result in the Required Action not met. However, if SR 3.7.10.1 is failed, the appropriate Condition(s), depending on the cause of the failure, is entered.

C.1, C.2, and C.3

With one battery with one or more cells electrolyte level above the top of the plates, but below the minimum established design limits, the battery still retains sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of electrolyte level not met. Within 31 days the minimum established design limits for electrolyte level must be re-established.

With electrolyte level below the top of the plates there is a potential for dryout and plate degradation. Required Actions C.1 and C.2 address this potential (as well as provisions in Specification 5.5.xx, PSW Battery Monitoring and Maintenance Program). They are modified by a Note that indicates they are only applicable if electrolyte level is below the top of the plates. Within 8 hours level is required to be restored to above the top of the plates. The Required Action C.2 requirement to verify that there is no leakage by visual inspection and the Specification 5.5.xx. item to initiate action to equalize and test in accordance with manufacturer's recommendation are taken from Annex D of IEEE Standard 450. They are performed following the restoration of the electrolyte level to above the top of the plates. Based on the results of the manufacturer's recommended testing the battery may have to be declared inoperable and the affected cell[s] replaced.

D.1

With one battery with pilot cell temperature less than the minimum established design limits, 12 hours is allowed to restore the temperature to within limits. A low electrolyte temperature limits the current and power available. Since the battery is sized with margin, while battery capacity is degraded, sufficient capacity exists to perform the intended function and the affected battery is not required to be considered inoperable solely as a result of the pilot cell temperature not met.

BASES

ACTIONS (continued)

E.1

With one battery with parameters not within limits there is not sufficient assurance that battery capacity has not been affected to the degree that the battery can still perform their required function. The longer Completion Times specified for battery parameters not within limits are therefore not appropriate, and the parameters must be restored to within limits within 2 hours.

F.1

With one battery with any battery parameter outside the allowances of the Required Actions for Condition A, B, C, or D, sufficient capacity to supply the maximum expected load requirement is not assured and must be declared inoperable. Additionally, discovering one battery with one or more battery cells float voltage less than 2.07 V and float current greater than 2 amps indicates that the battery capacity may not be sufficient to perform the intended functions. The battery must therefore be declared inoperable immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.10a.1

Verifying battery float current while on float charge is used to determine the state of charge of the battery. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a charged state. The float current requirements are based on the float current indicative of a charged battery. Use of float current to determine the state of charge of the battery is consistent with IEEE-450. The 7 day Frequency is consistent with IEEE-450.

This SR is modified by a Note that states the float current requirement is not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.7.10.1. When this float voltage is not maintained, actions should be taken to provide the necessary and appropriate verifications of the battery condition. Furthermore, the float current limit of 2 amps is established based on the nominal float voltage value and is not directly applicable when this voltage is not maintained.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10a.2 and SR 3.7.10a.5

Optimal long term battery performance is obtained by maintaining a float voltage greater than or equal to the minimum established design limits provided by the battery manufacturer, which corresponds to 130.5 V at the battery terminals, or 2.25 Vpc. This provides adequate over-potential, which limits the formation of lead sulfate and self discharge, which could eventually render the battery inoperable. Float voltages in this range or less, but greater than 2.07 Vpc, are addressed in Specification 5.5.xx. SRs 3.7.10a.2 and 3.7.10a.5 require verification that the cell float voltages are equal to or greater than the short term absolute minimum voltage of 2.07 V. The Frequency for cell voltage verification every 31 days for pilot cell and 92 days for each connected cell is consistent with IEEE-450.

SR 3.7.10a.3

The limit specified for electrolyte level ensures that the plates suffer no physical damage and maintains adequate electron transfer capability. The Frequency is consistent with IEEE-450.

SR 3.7.10a.4

This Surveillance verifies that the pilot cell temperature is greater than or equal to the minimum established design limit (i.e., 40°F). Pilot cell electrolyte temperature is maintained above this temperature to assure the battery can provide the required current and voltage to meet the design requirements. Temperatures lower than assumed in battery sizing calculations act to inhibit or reduce battery capacity. The Frequency is consistent with IEEE-450.

SR 3.7.10a.6

A battery performance discharge test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.7.10a.6; however, only the modified performance discharge test may be used to satisfy the battery service test requirements of SR 3.7.10.4.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.10a.6 (continued)

A modified discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test.

It may consist of just two rates; for instance the one minute rate for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test must remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test.

The acceptance criteria for this Surveillance are consistent with IEEE-450 and IEEE-485. These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements. Furthermore, the battery is sized to meet the assumed duty cycle loads when the battery design capacity reaches this 80 percent limit.

The Surveillance Frequency for this test is normally 60 months. If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity ≥ 100% of the manufacturer's ratings. Degradation is indicated, according to IEEE-450, when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is ≥10% below the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE-450.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.10a.6 (continued)

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1, 2, 3, or 4 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1, 2, 3, or 4. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

REFERENCES

1. IEEE-450-1995

2. UFSAR, Chapter 15.

3. IEEE-485-1983, June 1983.