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Docket Nos.: 52-025
52-026

ND-16-1892
10 CFR 50.90
10 CFR 52.63

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

Southern Nuclear Operating Company
Vogtle Electric Generating Plant Units 3 and 4
Request for License Amendment and Exemption Regarding
Passive Core Cooling System (PXS) Condensate Return (LAR-16-026)

Ladies and Gentlemen:

Pursuant to 10 CFR 52.98(c) and in accordance with 10 CFR 50.90, Southern Nuclear Operating Company (SNC) requests an amendment to the combined licenses (COLs) for Vogtle Electrical Generating Plant (VEGP) Units 3 and 4 (License Numbers NPF-91 and NPF-92, respectively). The proposed amendment would revise the licensing basis information to reflect an increase in the efficiency of the return of condensate utilized by the passive core cooling system (PXS) to the in-containment refueling water storage tank (IRWST) to support the capability for long-term cooling.

The requested amendment proposes to depart from approved AP1000 Design Control Document (DCD) Tier 2 information (text, tables, and figures) as incorporated into the Updated Final Safety Analysis Report (UFSAR) as plant-specific DCD information, and also proposes to depart from involved plant-specific Tier 1 information (and associated COL Appendix C information) and from involved plant-specific Technical Specifications (PS-TS) as incorporated in Appendix A of the COL. Associated PS-TS Bases document revisions that will be incorporated coincident with the requested amendment are identified for information.

Pursuant to the provisions of 10 CFR 52.63(b)(1), an exemption from elements of the design as certified in the 10 CFR Part 52, Appendix D, design certification rule is also requested for the plant-specific DCD Tier 1 material departures.

Enclosure 1 provides the description, technical evaluation, regulatory evaluation (including the significant hazards consideration determination), and environmental considerations for the proposed changes.

Enclosure 2 provides the background and supporting basis for the requested exemption.

Enclosure 3 provides the proposed changes to the licensing basis documents.

Enclosure 4 provides the conforming changes to the Technical Specifications Bases for information only.

Enclosure 5 addresses the applicability and endorsement of prior docketed information on this topic.

Enclosure 6 provides the description of proposed changes that differ from the PXS condensate return changes included in the Williams States Lee III submittal for information only.

This letter contains no regulatory commitments.

This letter, including enclosures, has been reviewed and confirmed to not contain security-related information.

SNC requests staff approval of the license amendment and associated exemption by December 15, 2016, to support the 2016 ITAAC initiative. SNC expects to implement the proposed amendment (through incorporation into the licensing basis documents, e.g., the Updated Final Safety Analysis Report) within 30 days of approval of the requested changes.

In accordance with 10 CFR 50.91, SNC is notifying the State of Georgia of this LAR by transmitting a copy of this letter and enclosures to the designated State Official.

Should you have any questions, please contact Mr. Corey Thomas at (205) 992-5221.

Mr. B. H. Whitley states that he is the Regulatory Affairs Director of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

B. H. Whitley

B. H. Whitley

LISA MYRICK SPEARS
MY COMMISSION EXPIRES
June 18, 2019
NOTARY PUBLIC
STATE OF ALABAMA

BHW/ERG/ljs

Sworn to and subscribed before me this 4th day of November, 2016

Notary Public: Lisa Myrick Spears

My commission expires: June 18, 2019

Enclosures:

1) Request for License Amendment Regarding Passive Core Cooling System (PXS) Condensate Return (LAR-16-026)
2) Request for Exemption Regarding Passive Core Cooling System (PXS) Condensate Return (LAR-16-026)
3) Proposed Changes to the Licensing Basis Documents (LAR-16-026)
4) Conforming Changes to the Technical Specifications Bases for Information (LAR-16-026)
5) Applicability and Endorsement of Prior Docketed Information (LAR-16-026)
6) Proposed Changes that Differ from the Duke Submittals for Information (LAR-16-026)

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Southern Nuclear Operating Company

Vogtle Electric Generating Plant Units 3 and 4

ND-16-1892

Enclosure 1

Request for License Amendment Regarding

Passive Core Cooling System (PXS) Condensate Return

(LAR-16-026)

(This Enclosure consists of 23 pages, including this cover page.)

Table of Contents

Pursuant to 10 CFR 52.98(c) and in accordance with 10 CFR 50.90, Southern Nuclear Operating Company (SNC) hereby requests an amendment to Combined License (COL) Nos. NPF-91 and NPF-92, for Vogtle Electrical Generating Plant (VEGP) Units 3 and 4, respectively.

1. Summary Description

The Nuclear Regulatory Commission Staff recommended, in SECY-94-084, that reactor designs utilizing passive safety systems include a residual heat removal system capable of bringing the reactor to a safe shutdown condition of 420°F or lower following non-loss of coolant accident (LOCA) events. According to information in the plant specific Design Control Document (DCD) as incorporated into the Updated Final Safety Analysis Report (the UFSAR) subsection 6.3.1.1.4, "Safe Shutdown," among the licensing performance criterion design bases of the Passive Core Cooling System (PXS) is the capability for the Passive Residual Heat Removal Heat Exchanger (PRHR HX) to cool the Reactor Coolant System (RCS) to the safe shutdown condition of 420°F in 36 hours. To support the capability of the AP1000 design to meet this licensing performance criterion, a shutdown temperature evaluation was performed, which assumed a condensate return fraction for the PXS.

Subsequently, the assumptions regarding condensate return to the In-containment Refueling Water Storage Tank (IRWST) were analyzed. The safety analysis results reported in UFSAR Subsection 19E.4.10.2, "Shutdown Temperature Evaluation," assume 90 percent of the steam discharged to the containment is returned to the IRWST. This requires that 90 percent of steam discharged from the IRWST is condensed on the containment wall surfaces, enters the IRWST gutter, and is returned to the IRWST. Condensation on other surfaces is directed to the containment sump; and is not returned to the IRWST. An investigation was initiated to quantify the returned fraction of condensate to the IRWST. Supplementary testing revealed opportunities to improve the design with regard to the condensate return fraction used to evaluate long-term plant cooldown. In addition, an analysis methodology was applied to characterize both the thermodynamic and the geometric phenomena involved in prolonged non-LOCA events.

The proposed changes revise the Combined Licenses (COLs) in regard to the condensate return function supporting prolonged non-LOCA events. The changes augment the IRWST gutter arrangement by adding a series of safety-related downspouts to collect condensate at intermediate locations from the Polar Crane Girder (PCG) and internal stiffener. In addition, an analysis characterizing condensate return to the IRWST gutter is applied to the shutdown temperature evaluation. The Shutdown Temperature Evaluation in UFSAR Appendix 19E is updated to analyze the PRHR HX performance with the design modifications to confirm it meets its licensing basis performance criterion of cooling the RCS to 420°F within 36 hours and maintaining a safe shutdown condition.

The requested amendment requires changes to the licensing basis documents, including the UFSAR information, and involves changes to plant-specific Technical Specifications (included as Appendix A to the COL), the plant-specific Tier 1 information, and corresponding changes to COL Appendix C, information. (See Section 2 for details.) This enclosure requests approval of the license amendment necessary to implement these changes. Technical Specification Bases changes corresponding to the proposed Technical Specification changes are also included for information in Enclosure 4.

2. Detailed Description

Background

NRC regulation (10 CFR 50 Appendix A, General Design Criterion 34 (GDC-34)) requires that the plant design include a system to remove residual heat from the reactor core so specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. The PXS provides emergency core cooling during transients, accidents, and whenever the normal, nonsafety-related heat removal paths are unavailable. The PXS is capable of bringing the plant to and maintaining the plant in a safe, stable condition following shutdown.

Operation

The PRHR HX is safety-related and provides emergency core decay heat removal. It is located in the IRWST as shown on UFSAR Figure 6.3-1, sheets 1-3. The heat exchanger is used in non-LOCA transients and also in LOCA events until voiding begins in the RCS Hot Leg. For any non-LOCA event, the PRHR HX plays an integral role in decay heat removal, as opening one of the two outlet isolation valves initiates natural circulation of the heat exchanger, transferring heat from the RCS into the IRWST. This transfer of heat from the RCS to the IRWST causes the water in the tank to heat up, eventually become saturated, and begin to steam from the tank. The evaluation of the PRHR HX operation during LOCA events is not impacted by the following proposed changes because the transition to open-loop cooling occurs much faster than in non-LOCA transients and provides another means of recirculation of water through containment.

The steam generated during the non-LOCA events discharges through a series of vents located near the steam generator compartment at the roof of the IRWST. The steam generator wall vents open with a slight pressure differential between the IRWST and containment, providing a path to vent steam produced by the PRHR HX into the containment atmosphere. The steam generator wall vents open at a lower differential pressure than the IRWST hood vents located near the containment wall so that the steam generator wall vents open first. The location of the steam generator wall vents (near the center of containment) contributes to mixing of the containment atmosphere. The steam released from the IRWST condenses on passive heat sinks within the containment, such as the containment vessel wall, concrete, piping, components, or any other subcooled surface until these passive heat sinks reach saturation temperature. Condensation on the inside of the containment vessel wall forms a thin fluid film and runs down the containment wall surface. Provisions are made to collect and channel condensate to the IRWST.

The PCG and internal stiffener are horizontal, circumferential attachments to the containment sidewalls that interrupt condensate flow. The PCG and internal stiffener increase the radial and rotational stiffness of the containment vessel, and are designed to allow condensate to drain back to the IRWST gutter. The PCG also supports the polar crane.

The PCG is a box girder consisting of 80 enclosed boxes; and is shown in UFSAR Figure 3.8.2-1 (Sheet 3 of 3). The front face of each box (facing into containment) has a 2-foot diameter opening. The rear face of each box is the containment wall. The PCG is constructed with chamfers and fabrication holes to allow condensate to drain past the PCG to the internal stiffener.

The internal stiffener contains fabrication holes to allow condensate to drain past it to the IRWST gutter. Condensate is also collected directly in the IRWST gutter, which extends around the circumference of containment and returns condensate to the IRWST.

Upon actuation of the PRHR HX, two air-operated valves in series are actuated to isolate the normal gutter drain path to the Liquid Radwaste System, and divert condensate to the IRWST. It is important that sufficient condensate return is achieved during non-LOCA PRHR HX operation. The ability to maintain closed-loop PRHR HX cooling for long periods minimizes the probability that open-loop cooling is needed. Although maintaining IRWST level above the top of the HX tubes is not a prerequisite for maintaining adequate decay heat removal, reduction of IRWST level to below the top of the tubes begins to degrade the heat exchanger performance.

Safety Analyses

The analysis described in UFSAR Subsection 19E.4.10.2, “Shutdown Temperature Evaluation,” assumes a constant portion of steam discharged to the containment is returned back to the IRWST. The decision was made to further investigate the efficiency of the PXS condensate return function by conducting testing and analyzing the plant performance with a series of calculations that included time-dependent quantification of steaming from the IRWST and the portion of that steam that condenses and returns to the IRWST.

Testing results showed that the prior design of the PCG, internal stiffener, and IRWST gutter contributed to losses at each location, which were larger than assumed. In addition to the losses due to the physical geometry of containment, there were also losses due to pressurization and heat-up of containment structures that were not previously considered. These losses indicated the constant condensate return assumed in the safety analyses was not conservative. The PCG and internal stiffener are modified to improve condensate return rate and the analysis was modified to assume higher condensate losses (a bounding rate of losses, proved by the experimental test data) such that the acceptance criterion for design basis analysis (UFSAR Chapter 15) events are met for closed loop cooling.

GDC-34 compliance is shown by a set of analyses that perform three evaluations (refer to Figure 3.1 in the Technical Evaluation section) using the LOFTRAN computer code. The limiting transient for removal of core decay heat (for the set of analyses) is the loss of normal feedwater with coincident loss of AC power, which is used to demonstrate that the passive safety systems can bring the plant to a safe, stable condition. The design basis extension analysis is documented in Chapter 6 of the UFSAR and extends beyond the end of the analyses described in Chapter 15 of the DCD out to 72 hours. This evaluation demonstrates that the Chapter 15 non-LOCA acceptance criteria are met for 72 hours (including maximum primary and secondary pressures and minimum departure from nucleate boiling (DNB)). The second evaluation is documented in Appendix 19E of the UFSAR and, using conservative, non-bounding assumptions, demonstrates the capability of the PRHR HX to reduce the core average

temperature to 420°F in 36 hours after shutdown following the limiting decay heat removal transient. The third evaluation, also in Appendix 19E of the UFSAR, is an extension of the second evaluation and, using conservative, non-bounding assumptions, evaluates the duration that safe shutdown can be maintained. The results show the AP1000 plant can maintain safe shutdown for greater than 14 days, using conservative, non-bounding assumptions.

Condensation

As steaming to the containment begins following PRHR HX operation and saturation of the IRWST, there are a number of mechanisms, both thermodynamic and geometric, that can prevent the condensed steam from returning to the IRWST. The mechanisms are as follows:

1) Steam to pressurize the containment
2) Steam condensation on passive heat sinks
3) Raining from the containment roof
4) Losses at the PCG and internal stiffener
5) Losses at support plates attached to the containment vessel
6) Losses at the Equipment Hatch and Personnel Air Lock
7) Losses at entry to the IRWST gutter

Condensation losses were evaluated by calculations and prototype testing. The losses due to pressurization, raining, and condensation on passive heat sinks were quantified with a revision to existing calculations and the development of three new calculations.

Structure, System, Component, and Analysis Descriptions

1) PXS Downspout Piping

A downspout piping network is added to collect and transport condensation from the PCG and internal stiffener to PXS Collection Boxes. The downspouts consist of two downspout branches, each with two connections to the top of the PCG, two connections to the bottom of the PCG and two connections to the internal stiffener. In each branch, the four connections from the PCG join together into a common header which extends below the internal stiffener. The two connections from the internal stiffener join together into a common line, which connect to the header below the internal stiffener. The header is routed to one of the two IRWST gutter collection boxes at either side of the IRWST. The downspouts are situated with approximate symmetry around the circumference of containment. The common header for each branch passes through the internal stiffener. These pass-through locations include penetration sleeves to allow sufficient depth for collection at the internal stiffener downspout inlets.

The configuration of the collection boxes is modified to accommodate the additional downspouts. The PCG boxes are modified to allow condensate to drain from inside the PCG. The piping is constructed of materials approved for use inside containment, consistent with UFSAR Subsection 6.1.1.4. The downspouts have tag numbers PXS-L301A/B to PXS-L310A/B. The downspouts are Safety Class C, seismic Category I.

Pipe sizes are selected to prevent pipes from running full of water. The pipe sizes are selected to accommodate a single failure (blockage) of one of the screens over the inlet to the downspouts. The PCG and internal stiffener are integral to the containment vessel and are included in the containment vessel inspection program. The PCG and internal stiffener are high in the containment away from the operating deck, and are not expected to be subject to foreign material like the Containment Recirculation Screens or IRWST Screens might be after a LOCA event, or the IRWST gutter might be during refueling activities. The horizontally routed sections of piping are sloped 1/8 inch per foot or greater downward toward each of the respective collection boxes. The PXS piping and instrumentation diagrams are revised accordingly.

2) Downspout Screens

The original IRWST gutter design includes an expanded metal flat screen which is fastened over the entrance to the gutter. The primary focus of the metal screen is to prevent larger debris from entering the gutter and potentially interfering with flow into the gutter or piping from the PXS collection boxes. Similarly, at the entrance of each of the downspouts from the top of the PCG and from the internal stiffener, a screen is needed for the same function – to prevent any larger debris from blocking the downspout piping.

Eight new PXS downspout screens are added. The screens have the tag numbers PXS-MY-Y81 to PXS-MY-Y88. The screen at each downspout entrance is Safety Class C, seismic Category I. The screens are constructed of materials compatible with the post-accident environment, consistent with UFSAR Subsection 6.1.1.4. Aluminum is not used for these components. The screens are designed to allow small debris to pass through; and provide sufficient flow area to accommodate design basis flow rates at the PCG and internal stiffener locations. The holes in the screens are small enough to aid in capturing large size debris from clogging the downspout piping. The screens are high in the containment away from the operating deck, and are not expected to be subject to foreign material like the Containment Recirculation Screens or IRWST Screens might be after a LOCA event, or the IRWST gutter might be during refueling activities.

3) Shutdown Temperature Evaluation

The Shutdown Temperature Evaluation summarized in UFSAR Subsection 19E.4.10.2 is updated. The analysis is performed using the LOFTRAN computer code, as before, with a more detailed input for the condensate return fraction. Condensate return is affected by the containment pressure, which determines the PRHR HX heat sink (IRWST water) temperature. The WGOTHIC containment model described in UFSAR Subsection 6.2.1.1.3 is used to model the peak containment pressure during loss of coolant accidents, with limited changes to the model to maximize condensate losses (as opposed to maximizing peak pressure). The WGOTHIC model is used to calculate thermodynamic condensate losses due to containment pressurization, containment leakage, and passive heat sink saturation. A bounding condensate loss percentage from the containment shell calculation is used as input to the WGOTHIC model, which in turn, outputs a condensate return flow and an IRWST steaming rate. Subsequently, these two outputs are combined to develop a time dependent condensate return fraction, which is input to the downstream LOFTRAN calculation. The condensate loss calculation confirms that sufficient condensate is returned to the IRWST.

The time-dependent condensate return fraction is input into the LOFTRAN code to demonstrate the ability of the PRHR HX to cool the core temperature in order to meet three sets of acceptance criteria. The first, the PRHR HX meets the success criteria of Chapter 15 non-LOCA events for greater than 72 hours in a design basis extension analysis (this includes acceptable primary and secondary side pressures and minimum DNBR). The second, to maintain compliance with SECY-94-084 recommendations, LOFTRAN demonstrates the PRHR HX cools the core temperature to 420°F within 36 hours in a closed-loop mode of operation using conservative, non-bounding assumptions. This analysis demonstrated that the addition of downspouts to channel condensate that reaches the PCG and internal stiffener back to the IRWST maintains sufficient IRWST inventory. The analysis shows the plant can successfully meet this licensing performance criterion. Finally, the third evaluation demonstrates that the PRHR, using conservative, non-bounding acceptance criteria, maintains safe shutdown of the AP1000 plant for greater than 14 days. The conservative, non-bounding evaluations are documented in Appendix 19E of the UFSAR.

Based on the updates to the safe shutdown evaluation, the supporting licensing basis tables and figures need to be updated to reflect the proposed design. Subsections 19E.2.3.2.6 and 19E.4.10.2, Table 19E.4.10-1, and the supporting figures, Figure 19E.4.10-1, -2, -3, and -4 are updated by this license amendment request.

Licensing Basis Change Descriptions

Condensate return to the IRWST is discussed widely throughout the UFSAR in conjunction with PRHR HX operation. Though the changes previously described do not change the condensate return concept, the licensing basis changes proposed herein provide additional piping, components, and adjustments to the descriptions of the condensate return provisions and provide revised descriptions of the analysis methodology in the UFSAR.

UFSAR Chapter 1

The AP1000 design response for Generic Safety Issue A-31 (Residual Heat Removal Requirements) is updated in UFSAR Subsection 1.9.4.2.2 to provide consistent language with other portions of the UFSAR that discuss the condensate return scenario. The statement regarding dependency on open-loop cooling is eliminated because open loop cooling may eventually be required. The pointers to safe shutdown in UFSAR Subsections 7.4 and 6.3 in this section provide more details regarding expected duration of closed-loop cooling. In addition, a pointer is added to Subsection 19E.4.10.2.

Subsection 1.9.5.1.5, “Station Blackout,” is updated to keep the language consistent with other areas of the UFSAR.

UFSAR Chapter 3

The new PXS downspout screens are Safety Class C and seismic Category I components. These components meet the quality assurance requirements of 10 CFR 50, Appendix B. Additionally, the screens must be demonstrated to have no functional damage following a seismic ground motion exceeding the operating basis earthquake ground motion before resuming operations in accordance with 10 CFR Part 50, Appendix S. The screens are added to Table 3.2-3 to capture these requirements.

Chapter 5

- Subsection 5.4.5.2.1 is updated to add a discussion about the metallic reflective insulation (MRI) installed on the pressurizer, specifically what the expected heat transfer coefficient through the MRI is during normal operations.
- Subsection 5.4.11.2 is revised such that a cross reference to Figure 6.3-2 is changed to Figure 6.3-1 for consistency across the chapters.
- Subsection 5.4.14.1 is updated to discuss the capabilities of the PRHR HX.

Chapter 6

To reflect the changes to the PXS system, the additional downspout piping is captured in the gutter discussions of UFSAR Section 6.3 and on a new sheet of the PXS piping and instrumentation diagrams (P&IDs). In order to add the new P&ID sheet to the licensing basis, Figure 6.3-1 is expanded to include all sheets of the PXS P&IDs and Figure 6.3-2 is not used. In addition to those changes, additional discussion about the PRHR HX is added to Chapter 6. The changes to UFSAR Chapter 6 are as follows.

- Subsection 6.3.1.1.1 is updated to describe the downspouts in the safety-related design criteria and to describe the operation and design capability of the PRHR HX in events requiring emergency core decay heat removal.
- Subsection 6.3.1.1.4 is updated to add discussion about the safe shutdown process.
- Subsection 6.3.1.1.6 is updated to reflect revised section numbering.
- Subsection 6.3.1.2 is renamed to “Nonsafety Design Basis” and “Power Generation Design Basis” is renumbered as Subsection 6.3.1.3.
- Subsection 6.3.1.2.1 is added to discuss Long Term Core Decay Heat Removal for a period of 36 hours and a period of greater than 14 days.
- Subsection 6.3.2.1 is updated to remove reference to Figure 6.3-2, which is no longer being used.
- Subsection 6.3.2.1.1 is updated to include the intermediate collection points of the safety-related gutter arrangement, a discussion of how condensation is formed inside of containment, and the duration of operation of the PRHR HX.
- Subsections 6.3.2.2.7 and 6.3.2.2.7.1 are updated to clarify the number of screen sets in the PXS and to which set of screens the criteria in this section apply.
- Subsection 6.3.2.2.7.2 is updated to clarify the condensate return gutter arrangement related to LOCA operation.
- Subsection 6.3.2.8 is updated to discuss operator action to block the automatic depressurization system.
- Subsection 6.3.3 is updated to add discussion about the operation of the PRHR HX.
- Subsection 6.3.3.2.1.1, “Loss of AC Power to the Plant Auxiliaries” is added.
- Subsection 6.3.3.4.1 is updated to show long term core cooling for greater than 14 days.

- Figure 6.3-1 is relabeled as "Figure 6.3-1 (Sheet 1 of 3)." This editorial change is the only change made to this figure. No technical changes are made.
- Figure 6.3-2 is relabeled "Figure 6.3-1 (Sheet 2 of 3)." On relabeled Sheet 2, the IRWST gutter is relocated to a new sheet 3 of the PXS P&IDs. Sheet 2 is modified to include continuation flags for condensate returning to the IRWST originating from PXS Collection Boxes A and B in the IRWST gutter.
- Figure 6.3-1 (Sheet 3 of 3) is a new P&ID sheet and is added to the licensing basis. This new figure shows the relocated IRWST gutter and the screens and piping comprising the PXS downspouts originating from the PCG and internal stiffener.

<u>Chapter 7</u>

- Section 7.4 is updated to discuss the long term operation of the PRHR HX.
- Subsection 7.4.1.1 is updated to discuss PRHR capability to maintain safe shutdown.

<u>Chapter 9</u>

- Table 9.5.1-1 is updated to make reference to the ability for the plant to maintain safe shutdown for at least 14 days.

<u>Chapter 14</u>

- In Table 14.3-2, cross reference to Figure 6.3-2 is changed to Figure 6.3-1 for consistency across the chapters.

<u>Chapter 15</u>

- Subsection 15.0.13 is updated with the safety related mission time of the PRHR HX (72 hours) and other editorial changes.
- Subsection 15.2 is updated to provide additional clarification regarding the measurement of performance of the PRHR HX during design basis events and point to Chapter 6, Subsection 6.3.1.1.1.
- Subsection 15.2.6.1 is updated with regard to functionality and timing for the PRHR and PCS to provide cooling and to maintain RCS system conditions to satisfy the evaluation criteria as described in Chapter 6.

<u>Chapter 19</u>

Two portions of Chapter 19 were impacted. Table 19.59-18 is updated to add clarification to the systems involved in core cooling for 14 days during a non-LOCA event where ac power is lost. The second portion is in Appendix 19E.

Per SECY-94-084, the NRC recommends the requirement of 420°F or below as a safe, stable shutdown condition. A shutdown temperature evaluation is performed to demonstrate the adequacy of the PRHR HX to reduce the core average temperature to 420°F within a

reasonable period of time after shutdown (which is taken to be 36 hours) following a loss of normal feedwater coincident with loss of ac power event. The results of the shutdown temperature evaluation are presented in UFSAR Subsections 19E.2.3.2.6 and 19E.4.10.2, Table 19E.4.10-1 and Figures 19E.4.10-1 through 19E.4.10-4. Changes to Chapter 19 include changes to Subsection 19E.4.10.2, Shutdown Temperature Evaluation, to describe the analysis methodology for a non-LOCA shutdown event, the time for the cold leg and core average temperatures to reach the specified safe, stable condition after shutdown following a loss of ac power event, and updates to the corresponding tables and figures, which further detail the sequence of events. Finally, UFSAR Subsection 19E.9 is updated to delete Reference 14, SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs, March 28, 1994," since the reference is removed from the text of the UFSAR.

Plant-Specific Tier 1 (and corresponding COL Appendix C) Information

The added components of the PXS are integral to providing safety-related core decay heat removal during non-LOCA events. Therefore, it is appropriate to apply inspections, tests, analyses, and acceptance criteria to the added PXS components to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the applicable design criteria, codes and standards.

The downspout screens support the capability of the PRHR HX to maintain the reactor in a safe shutdown condition by preventing large objects from entering the downspout piping. As required by General Design Criterion 2 of Appendix A to 10 CFR Part 50, the PXS is designed to withstand the effects of natural phenomena and normal and accident conditions without loss of capability to perform its safety functions. The PXS downspout screens are safety-related, located on the Nuclear Island; and required to withstand design basis seismic and post-accident operating loads without losing the capability to perform their safety function. To provide assurance these ITAAC design commitments would be met, plant-specific Tier 1 Table 2.2.3-1 and the corresponding COL Appendix C Table 2.2.3-1 are updated to include eight new downspout screens.

The downspout piping supports the capability of the PRHR HX to maintain the reactor in a safe shutdown condition by inhibiting containment floodup during PRHR HX operation and delaying the need for containment recirculation following RCS depressurization. As required by General Design Criterion 4 of Appendix A to 10 CFR Part 50, the PXS containment downspout piping is safety-related and required to withstand normal and seismic design basis loads without losing functional capability. To provide assurance these ITAAC design commitments are met, plant-specific Tier 1 Table 2.2.3-2 and the corresponding COL Appendix C Table 2.2.3-2 are updated to include the new PXS pipe lines.

Plant-Specific Technical Specifications (COL Appendix A)

- The Surveillance Requirement (SR) 3.5.4.7 is updated to include the downspout screens in the surveillance.

Plant-Specific Technical Specification Bases

The following associated Technical Specification Bases are provided for information to reflect the proposed Technical Specification change to include the downspouts and screens in the descriptions of the gutter arrangement.

- The Bases for LCO 3.3.17 are updated to reflect the addition of downspouts.
- The Bases for SR 3.5.4.7 are updated to encompass the entire gutter arrangement, including the downspout screens, in the surveillance.
- The Bases for TS 3.5.4, Background, is updated to reflect the addition of downspouts.

3. Technical Evaluation

The proposed changes provide for sufficient condensate return to retain the optimal PRHR HX performance needed to cool the RCS to 420°F within 36 hours as delineated in the PXS licensing basis performance criteria. The design changes also maintain the efficiency of the condensate return function such that the IRWST water level remains above the top of the PRHR HX tubesheet for the time durations considered in the applicable UFSAR Chapter 15 safety analyses.

Chapter 15 design basis transients that credit PRHR HX operation, along with the analysis run time, are listed in Table 1. In these analyses, a constant condensate return fraction was used for the safety analysis models supporting Chapter 15. However, though the condensate return fraction has changed, the transient analyses in Chapter 15 bound the plant response expected as a result of the proposed design changes. During the transients which credit PRHR HX operation, there is no impact to the heat transfer rate of the heat exchanger until the point that the water level in the IRWST drops below the top of the tube sheet, reducing the available heat transfer area. For the transient analyses in Chapter 15, the response does not change because even if the time-dependent condensate return fraction is applied, the PRHR HX remains submerged well beyond the duration of the relevant design basis analyses listed in Table 1. In order to demonstrate that there is no impact to Chapter 15, an extended 72-hour case of the limiting event was completed. This analysis shows the Chapter 15 acceptance criteria are met through the extended duration of the analysis.

Table 1
UFSAR Chapter 15 Non-LOCA Design Basis Accidents Crediting PRHR HX Operation

UFSAR Subsection	Transient Name	Run Time
15.2.2	Loss of external electrical load[1]	(2)
15.2.3	Turbine trip[1]	<1 minute
15.2.6	Loss of ac power to the plant auxiliaries	<6.2 hours
15.2.7	Loss of normal feedwater flow	<5.5 hours
15.2.8	Feedwater system pipe break	<3.2 hours
15.5.1	Inadvertent operation of the core makeup tanks during power operation	<8.6 hours
15.5.2	Chemical and volume control system malfunction that increases reactor coolant Inventory	<5.7 hours
15.6.3	Steam generator tube rupture	<6.7 hours

1. PRHR HX is not specifically credited in this analysis; but could be relied upon in the long term to support long-term recovery.
2. This transient is bounded by the turbine trip event.

In addition, the WGOTHIC peak containment pressure analysis is considered during the course of testing and analysis for this change; and is determined not to be affected by this change for the following reasons. With regard to peak containment pressure, the limiting design basis event is a double-ended cold leg guillotine break (DECLG) LOCA. The containment peak pressure for the DECLG LOCA case is not sensitive to the time-dependent condensate return, as the peak pressure is reached well before condensate return plays a factor in the event. Additionally, in the later stages of the transient (24 hours to 72 hours) the beneficial effects of condensate return are not considered in the containment peak pressure and temperature analysis. The WGOTHIC containment response model assumes condensate that reaches the polar crane girder and internal stiffener is deposited in the containment sump and no longer contributes to the film thickness at lower elevations of the containment wall. Therefore, the containment analysis methodology remains bounding and is consistent with the modified design.

The proposed changes do not adversely affect compliance with any design code limit (allowable value), safety-related function or design analysis, nor do they adversely affect any UFSAR Chapter 6 or Chapter 15 safety analysis result or safety margin. The proposed changes do not adversely affect the prevention and mitigation of accidents or their safety analyses. No safety-related structure, system, component (SSC) or function is adversely affected. The proposed changes do not affect any SSC accident initiator or initiating sequence of events. Thus, the probabilities of the accidents evaluated in the UFSAR are not affected. The proposed changes do not result in a new failure mode, malfunction, or sequence of events that could adversely affect a radioactive material barrier or safety-related equipment. The proposed changes do not allow for a new fission product release path, result in a new fission product barrier failure mode, or create a new sequence of events that results in significant fuel cladding failures. The proposed changes do not affect the radiological source terms (i.e., amounts and

types of radioactive materials released, their release rates and release durations) used in the accident analyses, thus the consequences of accidents are not affected.

The new components are classified and supplied as Safety Class C, seismic Category I designs. The added components are constructed of only those materials appropriately suited for exposure to the reactor coolant environment as described in UFSAR Subsection 6.1.1.4. No exposed aluminum is permitted to be used in the construction of these components such that they do not contribute to hydrogen production in containment. No system or design function or equipment qualification is adversely impacted by the proposed changes. The locations of the modifications and additional components allow for appropriate inspections during installation, and for periodic post-installation inspections. The added and affected SSCs do not adversely affect safety-related equipment or equipment whose failure could initiate an accident. The proposed changes do not adversely interface with a radioactive material barrier.

As previously stated in SECY-94-084, the NRC recommended the requirement of 420°F or below as a safe, stable shutdown condition. The results of the original shutdown temperature evaluation, are represented in UFSAR Subsection 19E.4.10.2, Table 19E.4.10-1, and Figures 19E.4.10-1 through 19E.4.10-4. The original evaluation was performed with realistic assumptions, with a number of conservatisms maintained, and assumed a constant condensate return fraction. The plant response after shutdown following non-LOCA events is reanalyzed with a series of interdependent calculations. The information flow between these calculations is illustrated in Figure 3.1.

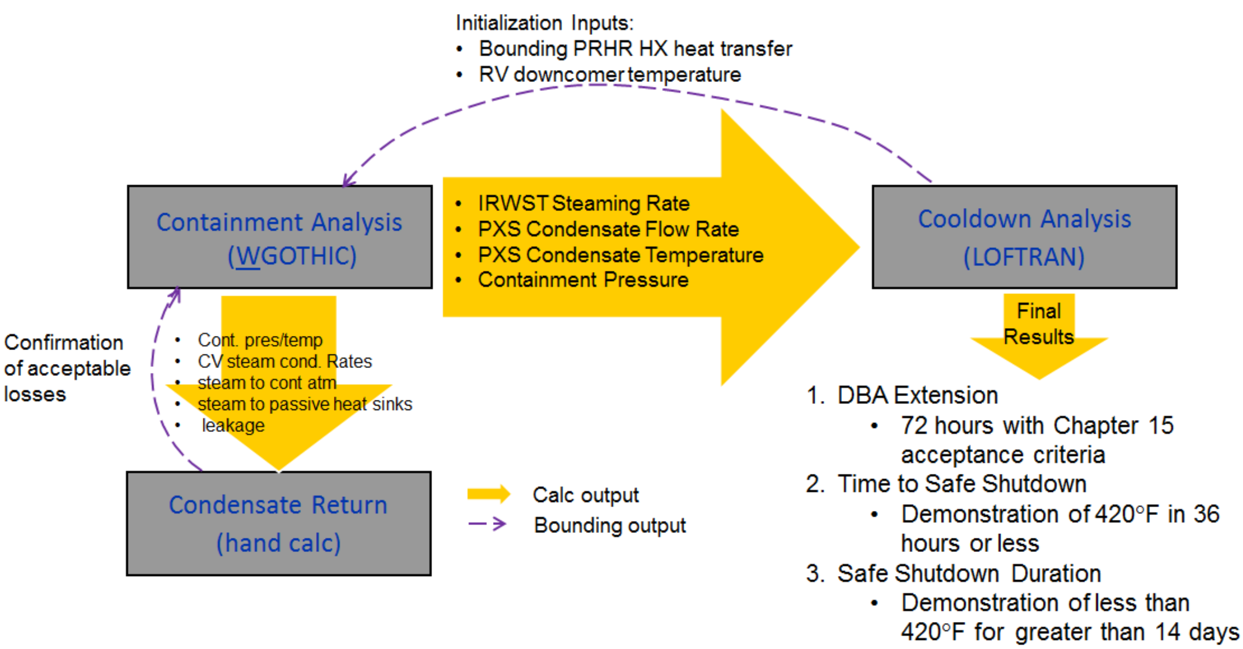


Figure 3.1: Calculation inter-relationships

The design changes described provide sufficient condensate return to the IRWST to preserve PRHR HX performance after shutdown following a non-LOCA event. To verify the effectiveness of the proposed changes to the PXS system, several analyses are performed, which incorporate the lessons learned about condensate return from design review and testing. As described in the Shutdown Temperature Evaluation, a loss of normal feedwater coincident with loss of ac power event is identified to be the most limiting transient with regard to PRHR HX performance. The Shutdown Temperature Evaluation described in Subsection 19E.4.10.2 is performed to demonstrate the adequacy of the PRHR HX to reduce the core average temperature to 420°F within 36 hours after shutdown following a loss of normal feedwater coincident with loss of ac power event. The containment peak pressure and temperature design limits are not challenged by the long-term loss of normal feedwater with loss of ac power event that forms the limiting basis for the shutdown temperature evaluation, as the maximum pressure reached during the loss of normal feedwater coincident with loss of ac power event does not exceed the containment design pressure.

Following a loss of ac power event, reactor coolant system energy is slowly transferred to the IRWST following actuation of the PRHR HX. The water in the IRWST begins to heat up, eventually coming to a boil. The steam released by boiling of the IRWST causes the containment temperature and pressure to increase. To evaluate the containment response to IRWST steaming and time-dependent condensate return on PRHR HX performance, minor modifications are made to the approved WGOTHIC containment response model to increase condensation and produce conservative results. The Containment Analysis performed with the WGOTHIC model calculates several key inputs to both the Condensate Return calculation and the Cooldown Analysis (LOFTRAN). The Containment Analysis (WGOTHIC) provides the containment pressure and temperature, containment vessel steam condensate rates, steam in the containment atmosphere, transient mass of condensate on the passive heat sinks, and steam lost to containment leakage. The Condensate Return calculation determines the percentage losses as a function of flow from the containment vessel shell and this percent loss is bounded as an input to the WGOTHIC model. The Containment Analysis also provides the IRWST steaming rate, the PXS condensate flow rate and temperature, and containment pressure to the shutdown temperature evaluation. This analysis performs the shutdown temperature evaluation using the modified LOFTRAN computer code (described in UFSAR Subsection 15.0.11.2).

In order to confirm that LOFTRAN is an acceptable code to evaluate extended station blackout (SBO) scenarios, additional evaluation of the performance of the PRHR HX under saturation conditions is performed. The engineering review examined two limitations of the LOFTRAN computer code:

- The ability to mechanistically account for uncovery of the lower PRHR HX tube region considering LOFTRAN's single flow path nodalization in the PRHR loop circuit.
- The ability to account for the effects of the loss of sub-cooling in the RCS when the extent of voiding extends beyond the pressurizer and upper head of the reactor vessel.

The conclusion of this review is that LOFTRAN is an appropriate code and the limitations defined lead to a representative prediction of Tavg for extended SBO scenarios. This conclusion is reached through a review of a supporting RELAP analysis that models RCS heat losses to understand the impact of the loss of sub-cooling on PRHR HX operations. The

confirmatory RELAP analyses demonstrates that a loss of RCS sub-cooling does not result in degradation of PRHR performance, and that modeling ambient heat losses in the RCS yields lower values of Tavg than that of adiabatic conditions. The RELAP evaluation also demonstrates that, consistent with LOFTRAN, the lower PRHR HX tube bundle does not begin to uncover during the 14 day mission time of the PRHR HX. Therefore, LOFTRAN's treatment of the lower PRHR HX tube bundle uncovery is not pertinent to analysis results.

The purpose of the shutdown temperature evaluation (modeled with LOFTRAN) is twofold; an evaluation that demonstrates compliance with SECY-94-084 and a safe shutdown duration evaluation to evaluate the duration the AP1000 plant can maintain safe shutdown. The shutdown temperature evaluation and its analytical inputs implementing the variable condensate return fraction produced by the proposed changes demonstrate the efficacy of the proposed changes in helping to bring the core average temperature to 420°F in less than 36 hours. The extension of this analysis, also performed with LOFTRAN, demonstrates that the AP1000 plant can maintain safe shutdown for greater than 14 days. Therefore, the plant continues to meet its licensing performance criterion and the requirements of SECY-94-084.

The proposed changes associated with this license amendment request do not involve or affect the containment, control, channeling, monitoring, processing or releasing of radioactive or non-radioactive materials. No effluent release path is affected. The types and quantities of expected effluents are not changed, and no effluent release path is affected by the proposed changes. Therefore, radioactive and non-radioactive material effluents are not affected by the proposed changes.

Plant radiation zones (as described in UFSAR Section 12.3), controls under 10 CFR 20, and expected amounts and types of radioactive materials are not affected by the proposed changes. Therefore, individual and cumulative radiation exposures are not affected.

Summary

The proposed changes modify the PCG and internal stiffener to augment the IRWST gutter arrangement to capture condensate at the PCG and internal stiffener locations. In addition, the proposed changes provide additional, detailed discussion about the function of the PRHR HX in conjunction with the condensate return scenario to provide a better basis for operation in closed-loop cooling after 22 hours. Those changes affect material in the UFSAR (Tier 2) and involve changes to the plant-specific Tier 1 information (and the corresponding COL Appendix C Table 2.2.3-1 and Table 2.2.3-2), and also involve the plant-specific Technical Specifications (included in COL Appendix A).

The design basis extension analysis shows that the PRHR HX functions to remove decay heat and maintain safe, stable conditions in the RCS for greater than 72 hours during a design basis event. The updated shutdown temperature evaluation demonstrates that; the changes provide for sufficient condensate return to retain the optimal PRHR HX performance needed to cool the RCS to 420°F within 36 hours in a closed-loop mode of operation as delineated in the PXS licensing performance criteria and the changes provide sufficient condensate return to maintain safe shutdown for greater than 14 days in a closed-loop mode of cooling. The proposed changes do not adversely affect any safety-related equipment or function, design function, radioactive material barrier or safety analysis.

4. Regulatory Evaluation

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.46 as it relates to the PXS performance analyses requires that the analyses use an acceptable evaluation model and provide results that meet the applicable regulatory acceptance criteria. The proposed design and licensing basis changes reflect that the Chapter 6 and Chapter 15 safety analyses are not affected and remain bounding. The design basis analysis methods used to evaluate performance of the PXS include only methods approved for use by the Commission.

10 CFR 50, Appendix A, General Design Criterion 2, "Design bases for protection against natural phenomena," requires that the PXS be designed to withstand the effects of natural phenomena and normal and accident conditions without loss of capability to perform its safety functions. The PXS, including the additional PXS components added for the condensate return function, is designed to meet seismic Category I design requirements; and is protected from the effects of external events such as earthquakes, tornadoes, and floods.

10 CFR 50, Appendix A, General Design Criterion 4, "Environmental and dynamic effects design bases," requires that the PXS be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. The PXS is designed to accommodate the environmental conditions associated with all modes of operation, and to prevent excessive dynamic events. Additionally, piping line sizes are selected to prevent steam flashing in the downspout piping. The piping and screens are constructed of only those materials compatible with the post-accident environment, consistent with UFSAR Subsection 6.1.1.4.

10 CFR 50, Appendix A, General Design Criterion 5, "Sharing of structures, systems, and components," specifies that the PXS is prohibited from being shared among nuclear power units unless it can be demonstrated that sharing will not impair their ability to perform their safety function. The PXS contains no components that are shared between nuclear power units. Therefore, the PXS changes meet the requirements of General Design Criterion 5.

10 CFR 50, Appendix A, General Design Criterion 17, "Electric power systems," specifies that an onsite electric power system and an offsite electric power system be provided to provide sufficient capacity to ensure that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded and that the core is cooled during anticipated operational occurrences and accident conditions. The plant does not require ac power sources to mitigate design-basis events. Likewise, the PXS condensate return design relies on natural forces and does not require power sources to perform its safety-related functions. The components added are passive components maintained in their safety-related configuration for the duration of operation. Therefore, the design continues to support the approved exemption from the requirements of General Design Criterion 17.

<u>10 CFR 50, Appendix A, General Design Criterion 27,</u> "Combined reactivity control systems capability," requires that the PXS be designed to have a combined capability, in conjunction with poison addition, of reliably controlling reactivity changes to assure that, under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained. The proposed changes do not affect the capability of the PXS to control core reactivity with poison addition. The proposed changes do affect the ability of the PXS to provide adequate core cooling by increasing the fraction of condensate returned to the IRWST during an event where steaming from the IRWST to containment occurs.

<u>10 CFR 50, Appendix A, General Design Criterion 34,</u> "Residual heat removal" requires that the plant be designed with a residual heat removal system to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the RCPB are not exceeded. The PRHR HX is capable of cooling the RCS in accordance with the provisions of SECY-94-084. The changes proposed provide a fractional condensate returned to the IRWST over time that exceeds the return fraction necessary to ensure adequate PRHR HX performance. With the proposed changes, the updated safe shutdown analysis continues to demonstrate the plant complies with its licensing performance criteria to cool the RCS to 420ºF within 36 hours. In addition, the extended design basis accident evaluation confirms compliance with the Chapter 15 non-LOCA acceptance criteria for 72 hours during the most limiting event.

<u>10 CFR 50, Appendix A, General Design Criterion 35,</u> "Emergency core cooling," requires that the PXS be able to provide an abundance of core cooling to transfer heat from the core at a rate so fuel and clad damage will not interfere with continued effective core cooling. The functionality of components of the PXS providing direct injection to the RCS for emergency core cooling is not affected by the proposed changes. The changes described herein ensure the PRHR HX can provide adequate core cooling during non-LOCA events, in conjunction with core makeup tank and accumulator operation. Therefore, the PXS continues to satisfy General Design Criterion 35.

<u>10 CFR 50, Appendix A, General Design Criterion 36,</u> "Inspection of emergency core cooling system," requires that the PXS be designed to permit appropriate periodic inspection of important components. The proposed modifications are accessible to periodic inspections. The proposed piping and downspout screens are accessible for inspection and maintenance as necessary. The PXS continues to comply with General Design Criterion 36.

<u>10 CFR 50, Appendix A, General Design Criterion 37,</u> "Testing of emergency core cooling system," requires that the PXS be designed to permit appropriate periodic pressure and functional testing. The proposed modifications do not affect the ability to periodically test the emergency core cooling capability of the PXS. The periodic inspection and testing program for the PXS does not include requirements specifically for testing condensate return to the IRWST, because steaming the containment is not practical. However, the added components are accessible for periodic inspection to confirm structural integrity and may be flow tested to confirm overall operability.

<u>10 CFR 52.98(f)</u> requires NRC approval for any modification to, addition to, or deletion from the terms and conditions of a COL. This activity involves a departure from plant-

specific Tier 1 information, and a corresponding change to COL Appendix C, Inspections, Tests, Analyses, and Acceptance Criteria information, and also involves a change to the plant-specific Technical Specifications as provided in COL Appendix A; therefore, this activity requires a proposed amendment to the COL. Accordingly, NRC approval is required prior to making the plant-specific changes in this license amendment request.

10 CFR 52, Appendix D, Section VIII.B.5.a allows an applicant or licensee who references this appendix to depart from Tier 2 information, without prior NRC approval, unless the proposed departure involves a change to or departure from Tier 1 information, Tier 2* information, or the Technical Specifications, or requires a license amendment under paragraphs B.5.b or B.5.c of the section. This change involves a revision to plant-specific Tier 1 information (and corresponding COL Appendix C information) and also involves a change to the plant specific Technical Specifications as provided in COL Appendix A, and thus requires NRC approval for the proposed changes to the licensing basis documents.

4.2 Precedent

Duke Energy Florida and Duke Energy Carolinas have previously submitted similar information. See Enclosures 5 and 6 for further information on applicability and differences.

4.3 Significant Hazards Consideration

The proposed changes revise the Combined Licenses (COLs) in regard to the addition of downspouts to capture condensate at the PCG and internal stiffener locations and return it to the IRWST.

The requested amendment requires changes to Updated Final Safety Analysis Report (UFSAR) Tier 2 information, which involve changes to plant-specific Tier 1 and corresponding changes to COL Appendix C information, and also involve a change to the plant-specific Technical Specifications as provided in COL Appendix A.

An evaluation to determine whether or not a significant hazards consideration is involved with the proposed amendment was completed by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

4.3.1 Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed containment condensate flow path changes provide sufficient condensate return flow to maintain In-containment Refueling Water Storage Tank (IRWST) level above the top of the Passive Residual Heat Removal Heat Exchanger (PRHR HX) tubes long enough to prevent PRHR HX performance degradation from that considered in the UFSAR Chapter 15 safety analyses. The added components are seismically qualified and constructed of only those materials appropriately suited for exposure to the reactor coolant environment as described in UFSAR Section 6.1.

No aluminum is permitted to be used in the construction of these components so that they do not contribute to hydrogen production in containment.

The proposed changes clarify the design basis for the PRHR HX, which removes decay heat from the Reactor Coolant System (RCS) during a non-loss of coolant accident (non-LOCA). With operator action to avoid unnecessary Automatic Depressurization System (ADS) actuation based on RCS conditions, PRHR HX operation can be extended longer than is maintained automatically by the protection and safety monitoring system. Though analysis shows significantly greater capacity, the extent of capability of the PRHR HX in the licensing basis is changed from operating indefinitely to operating for at least 72 hours. If PRHR HX capability was exhausted after 72 hours, the ADS is actuated, which could result in significant containment floodup. However, the probabilistic analysis shows that the probability of design basis containment floodup after PRHR HX operation during a non-LOCA event is significantly lower than the probability of a small break LOCA, for which comparable containment floodup is anticipated. Therefore, the probability of significant containment floodup is not increased.

The proposed changes do not affect components whose failure could initiate an event, thus the probabilities of the accidents previously evaluated are not affected. The affected equipment does not adversely affect or interact with safety-related equipment or another radioactive material barrier. The proposed changes clarify the post-accident performance requirements for the PRHR HX. However, the proposed changes do not prevent the engineered safety features from performing their safety-related accident mitigating functions. The radioactive material source terms and release paths used in the safety analyses are unchanged, thus the radiological releases in the UFSAR accident analyses are not affected.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

4.3.2 Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The long-term safe shutdown analysis results show that the PRHR HX continues to meet its acceptance criterion, i.e., to cool the Reactor Coolant System (RCS) to below 420ºF in 36 hours. The added equipment does not adversely interface with any component whose failure could initiate an accident, or any component that contains radioactive material. The modified components do not incorporate any active features relied upon to support normal operation. The downspout and gutter return components are seismically qualified to remain in place and function during seismic and dynamic events. The containment condensate flow path changes do not create a new fault or sequence of events that could result in a radioactive material release.

The proposed change quantifies the duration that the PRHR HX is capable of maintaining adequate core cooling, and specifies that if the PRHR HX cooling capability is exhausted, the ADS is actuated. This involves the possibility of opening

the ADS valves after the IRWST water level has decreased below the spargers, which promote steam condensation in the IRWST. During this condition, the loads on the IRWST, spargers, and any internal structures or components in the IRWST are still less than their limiting loads, and these SSCs are not adversely affected or cause a different mode of operation. Therefore, no new type of accident could be created by this condition.

Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

4.3.3 Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No

The proposed changes do not reduce the redundancy or diversity of any safety-related function. The added components are classified as safety-related, seismically qualified, and are designed to comply with applicable design codes. The proposed containment condensate flow path changes provide sufficient condensate return flow to maintain adequate IRWST water level for those events using the PRHR HX cooling function. The long-term Shutdown Temperature Evaluation results in UFSAR Appendix 19E show the PRHR HX continues to meet its acceptance criterion. The UFSAR Chapters 6 and 15 analyses results are not affected, thus margins to their regulatory acceptance criteria are unchanged. The former design basis, which stated the PRHR HX could bring the plant to 420°F within 36 hours is changed to state the heat exchanger can establish safe, stable conditions in the reactor coolant system after a design basis event. Such safe, stable conditions may not coincide with a core average temperature of 420°F. However, the PRHR HX is able to bring the RCS to a sufficiently low temperature such that RCS conditions are comparable to those achieved at 420°F – peak cladding temperatures and departure from nucleate boiling are maintained within acceptable limits of the evaluation criteria with adequate margin. No safety analysis or design basis acceptance limit/criterion is challenged or exceeded by the proposed changes, thus no margin of safety is reduced.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above, it is concluded the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. Pursuant to 10 CFR 50.92, the requested change does not involve a Significant Hazards Consideration.

5. Environmental Consideration

The details of the proposed changes are provided in Sections 2 and 3 of this license amendment request.

The proposed changes revise the Combined Licenses (COLs) in regard to design changes for providing adequate post-accident condensate return flow to the In-containment Refueling Water Storage Tank (IRWST). The proposed changes involve modifications to the Polar Crane Girder (PCG), internal stiffener, and Passive Core Cooling System (PXS) IRWST gutter, and an addition of a downspout system to capture condensate at the PCG and internal stiffener locations.

The requested amendment requires changes to Updated Final Safety Analysis Report (UFSAR) information, which involve changes to the plant-specific Tier 1 and corresponding changes to COL Appendix C information, and also involve a change to the plant-specific Technical Specifications as provided in COL Appendix A. (See Section 2 for details.)

This review has determined that the proposed change requires an amendment to the COL; however, a review of the anticipated construction and operational effects of the proposed amendment has determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9), in that:

(i) There is no significant hazards consideration.

As documented in Section 4.3, Significant Hazards Consideration, of this license amendment request, an evaluation was completed to determine whether or not a significant hazards consideration is involved by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of amendment.” The Significant Hazards Consideration determined that (1) the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) the proposed amendment does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of “no significant hazards consideration” is justified.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed changes in the requested amendment are to improve condensate flow paths within the containment. The proposed changes are unrelated to any aspect of plant construction or operation that would introduce any change to effluent types (e.g., effluents containing chemicals or biocides, sanitary system effluents, and other effluents), or affect any plant radiological or non-radiological effluent release quantities. Furthermore, the proposed changes do not affect any effluent release path or diminish the functionality of any design or operational features that are credited with controlling the release of effluents during plant operation. Therefore, it is concluded that the proposed amendment does not involve a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite.

(iii) *There is no significant increase in individual or cumulative occupational radiation exposure.*

The proposed changes modify the condensate flow paths within the containment. Plant radiation zones (addressed in UFSAR Section 12.3) are not affected, and controls under 10 CFR 20 preclude a significant increase in occupational radiation exposure. Therefore, the proposed amendment does not involve a significant increase in individual or cumulative occupational radiation exposure.

Based on the above review of the proposed amendment, it has been determined that the anticipated construction and operational effects of the proposed amendment do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment of the proposed exemption is not required.

6. References

None

Southern Nuclear Operating Company

Vogtle Electric Generating Plant Units 3 and 4

ND-16-1892

Enclosure 2

Request for Exemption

Regarding Passive Core Cooling System (PXS) Condensate Return

(LAR-16-026)

(This Enclosure consists of 7 pages, including this cover page.)

1.0 Purpose

Southern Nuclear Operating Company (the Licensee) requests a permanent exemption from the provisions of 10 CFR 52, Appendix D, Section III.B, "Design Certification Rule for the AP1000 Design, Scope and Contents," to allow a departure from elements of the certification information in Tier 1 of the Generic DCD. The regulation, 10 CFR 52, Appendix D, Section III.B, requires an applicant or licensee referencing Appendix D to 10 CFR Part 52 to incorporate by reference and comply with the requirements of Appendix D, including certification information in DCD Tier 1. The Tier 1 information for which a plant-specific departure and exemption is being requested includes changes to improve the condensate return for the Passive Core Cooling System (PXS).

This request for exemption applies the requirements of 10 CFR 52, Appendix D, Section VIII.A.4 to allow departures from generic Tier 1 information due to the following proposed additions to the system-based Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for the Passive Core Cooling System as identified in Table 2.2.3-1 and Table 2.2.3-2.

The added components of the PXS are integral to providing safety-related core decay heat removal during non-LOCA events. Therefore, it is appropriate to apply inspections, tests, analyses, and acceptance criteria to the added PXS components to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the applicable design criteria, codes and standards.

The downspout screens support the capability of the passive residual heat removal heat exchanger (PRHR HX) to maintain the reactor in a safe shutdown condition by preventing large objects from entering the downspout piping. As required by General Design Criterion 2 of Appendix A to 10 CFR Part 50, the PXS is designed to withstand the effects of natural phenomena and normal and accident conditions without loss of capability to perform its safety functions. The PXS downspout screens are safety-related; located on the Nuclear Island; and required to withstand design basis seismic and post-accident operating loads without losing the capability to perform their safety function. To provide assurance these ITAAC design commitments are met, plant-specific Tier 1 Table 2.2.3-1 is updated to include eight new downspout screens.

The downspout piping supports the capability of the PRHR HX to maintain the reactor in a safe shutdown condition by inhibiting containment flood-up during PRHR HX operation and delaying the need for containment recirculation following RCS depressurization. As required by General Design Criterion 4 of Appendix A to 10 CFR Part 50, the PXS containment downspout piping is safety-related and required to withstand normal and seismic design basis loads without losing functional capability. To provide assurance these ITAAC design commitments are met, plant-specific Tier 1 Table 2.2.3-2 is updated to include the new PXS pipe lines.

2.0 Background

The Licensee is the holder of Combined License Nos. NPF-91 and NPF-92, which authorize construction and operation of two Westinghouse Electric Company AP1000 nuclear plants, named Vogtle Electric Generating Plant (VEGP) Units 3 and 4, respectively.

The Updated Final Safety Analysis Report (UFSAR), Subsection 6.3.1.1.1, “Emergency Core Decay Heat Removal,” identifies the safety-related design bases of the Passive Core Cooling System (PXS) including the capability for the Passive Residual Heat Removal Heat Exchanger (PRHR HX) to cool the Reactor Coolant System (RCS) to the safe shutdown condition of 420°F in 36 hours. The Nuclear Regulatory Commission Staff recommended, in SECY-94-084, that reactor designs utilizing passive safety systems include a residual heat removal system capable of bringing the reactor to a safe shutdown condition of 420°F or lower following non-loss of coolant accident (non-LOCA) events. To support the capability of the AP1000 design to meet this design criterion, a safe shutdown temperature evaluation was performed, which assumed a specific condensate return fraction for the PXS.

Through a series of design reviews, the efficiency of the condensate return to the In-Containment Refueling Water Storage Tank (IRWST) was further evaluated. Testing results showed that the current design could have an efficiency for condensate return lower than initially assumed. These evaluations were initiated to investigate and better quantify the returned fraction of condensate to the IRWST. Supplementary testing revealed opportunities to improve the design with regard to the condensate return fraction used to evaluate long-term plant cooldown. In addition, an analysis methodology was applied to characterize both the thermodynamic and the geometric phenomena involved in prolonged non-LOCA events.

3.0 Technical Justification of Acceptability

General Design Criteria 34 and 35 require that the PXS be capable of removing core decay and residual heat, and provide an abundance of core cooling such that fuel design limits and the RCS design conditions are not exceeded. As the PXS provides core decay heat removal during design basis events, performance of this safety-related function is confirmed through ITAAC 2.2.3, design commitment 8.b. The changes described herein do not change the commitment to complete the performance test of the PRHR HX.

Additional detail for justification for this exemption is provided in Section 2 of the accompanying License Amendment Request in Enclosure 1.

4.0 Justification of Exemption

10 CFR Part 52, Appendix D, Section VIII.A.4 and 10 CFR 52.63(b)(1) govern the issuance of exemptions from elements of the certified design information for AP1000 nuclear power plants. Because the Licensee has identified changes to the Tier 1 information related to the Tier 2 departure discussed in Enclosure 1 of the accompanying License Amendment Request, an exemption from the certified design information in Tier 1 is needed.

10 CFR Part 52, Appendix D, and 10 CFR §§ 50.12, 52.7, and 52.63 state that the NRC may grant exemptions from the requirements of the regulations provided six conditions are met: 1) the exemption is authorized by law [§50.12(a)(1)]; 2) the exemption will not present an undue risk to the health and safety of the public [§50.12(a)(1)]; 3) the exemption is consistent with the common defense and security [§50.12(a)(1)]; 4) special circumstances are present [§50.12(a)(2)]; 5) the special circumstances outweigh any decrease in safety that may result

from the reduction in standardization caused by the exemption [§52.63(b)(1)]; and 6) the design change will not result in a significant decrease in the level of safety [Part 52, App. D, VIII.A.4].

The requested exemption satisfies the criteria for granting specific exemptions, as described below.

1. **This exemption is authorized by law**

   The NRC has authority under 10 CFR §§ 50.12, 52.7, and 52.63 to grant exemptions from the requirements of NRC regulations. Specifically, 10 CFR §§50.12 and 52.7 state that the NRC may grant exemptions from the requirements of 10 CFR Part 52 upon a proper showing. No law exists that would preclude the changes covered by this exemption request. Additionally, granting of the proposed exemption does not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission's regulations.

   Accordingly, this requested exemption is "authorized by law," as required by 10 CFR §50.12(a)(1).

2. **This exemption will not present an undue risk to the health and safety of the public**

   The proposed exemption from the requirements of 10 CFR 52, Appendix D, Section III.B allows changes to elements of the plant-specific Tier 1 DCD to depart from the AP1000 certified (Tier 1) design information. The plant-specific Tier 1 DCD continues to reflect the approved licensing basis for the Licensee, and maintains a consistent level of detail with that which is currently provided elsewhere in Tier 1 of the plant-specific DCD. Because the change to the condensate return portion of the passive core cooling system description maintains its design functions, the changed design continues to provide the protection of the health and safety of the public. Therefore, no adverse safety impact that presents any additional risk to the health and safety is present. The affected Design Description in the plant-specific Tier 1 DCD also continues to provide the detail necessary to support the performance of the associated ITAAC.

   Therefore, the requested exemption from 10 CFR 52, Appendix D, Section III.B does not present an undue risk to the health and safety of the public.

3. **The exemption is consistent with the common defense and security**

   The exemption from the requirements of 10 CFR 52, Appendix D, Section III.B changes elements of the plant-specific Tier 1 DCD by departing from the AP1000 certified (Tier 1) design information. The exemption does not alter the design, function, or operation of any structures or plant equipment that are necessary to maintain a safe and secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures.

   Therefore, the requested exemption is consistent with the common defense and security.

4. Special circumstances are present

10 CFR 50.12(a)(2) list six “special circumstances” for which an exemption may be granted. Pursuant to the regulation, it is necessary for one of these special circumstances to be present in order for the NRC to consider granting an exemption request. The requested exemption meets the special circumstances of 10 CFR 50.12(a)(2)(ii). That subsection defines special circumstances as when “Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.”

The rule under consideration in this request for exemption is 10 CFR 52, Appendix D, Section III.B, which requires that a licensee referencing the AP1000 Design Certification Rule (10 CFR Part 52, Appendix D) shall incorporate by reference and comply with the requirements of Appendix D, including Tier 1 information. The VEGP Units 3 and 4 COLs reference the AP1000 Design Certification Rule and incorporate by reference the requirements of 10 CFR Part 52, Appendix D, including Tier 1 information. The underlying purpose of Appendix D, Section III.B is to describe and define the scope and contents of the AP1000 design certification, and to require compliance with the design certification information in Appendix D.

The proposed changes to the condensate return portion of the passive core cooling system maintain the design margins of the Passive Core Cooling System. This change does not impact the ability of any structures, systems, or components to perform their functions or negatively impact safety. Accordingly, this exemption from the certification information enables the applicant to safely construct and operate the AP1000 facility consistent with the design certified by the NRC in 10 CFR 52, Appendix D.

Therefore, special circumstances are present, because application of the current generic certified design information in Tier 1 as required by 10 CFR Part 52, Appendix D, Section III.B, in the particular circumstances discussed in this request is not necessary to achieve the underlying purpose of the rule.

5. The special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption

Based on the nature of the changes to the plant-specific Tier 1 information and the understanding that these changes support the design function of the Passive Core Cooling System, it is expected that other AP1000 applicants and licensees will also request this exemption. This exemption request and the associated marked-up tables demonstrate that there is a minimal change from the generic AP1000 DCD, minimizing the reduction in standardization and consequently the safety impact from the reduction.

Therefore, the special circumstances associated with the requested exemption outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. In fact, as described in item 6 below, the exemption results in no reduction in the level of safety.

6. **The design change will not result in a significant decrease in the level of safety.**

   The exemption revises the plant-specific DCD Tier 1 information by altering the description of the passive core cooling system condensate return design. The components added to the condensate return function design enable the passive core cooling system to meet its design functions. Because these functions continue to be met, there is no reduction in the level of safety.

5.0 Risk Assessment

A risk assessment was determined to be not applicable to address the acceptability of this request.

6.0 Precedent

A similar change was previously submitted by Duke Energy Florida, on the Levy Nuclear Plant, Units 1 and 2 docket, Docket Nos. 52-029 and 52-030, and approved as part of their combined license application. The Levy changes and the supporting documents for the Levy changes applicable to the changes proposed in this LAR and are addressed in Enclosures 6 and 5, respectively.

7.0 Environmental Consideration

A review has determined that the proposed exemption changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or changes an inspection or surveillance requirement. However, the proposed exemption does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Specific justification is provided in Section 5 of the corresponding License Amendment Request in Enclosure 1. Accordingly, the proposed exemption meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed exemption.

8.0 Conclusion

The proposed changes to Tier 1 are necessary to revise the passive core cooling system design description in the plant-specific DCD Tier 1. The exemption request meets the requirements of 10 CFR 52.63, *"Finality of design certifications,"* 10 CFR 52.7, *"Specific exemptions,"* 10 CFR 50.12, *"Specific exemptions,"* 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," and 10 CFR 52 Appendix D, *"Design Certification Rule for the AP1000 Design."* Specifically, the exemption request meets the criteria of 10 CFR 50.12(a)(1) in that the request is authorized by law, presents no undue risk to public health and safety, and is consistent with the common defense and security. Furthermore, approval of this request does not result in a significant decrease in the level of safety, presents special circumstances, does not present a significant decrease in safety as a result of a reduction in standardization, and meets the eligibility requirements for categorical exclusion.

Southern Nuclear Operating Company

Vogtle Electric Generating Plant Units 3 and 4

ND-16-1892

Enclosure 3

Proposed Changes to the Licensing Basis Documents

(LAR-16-026)

Additions identified by blue underlined text.

~~Deletions Identified by red strikethrough of text.~~

Relocated existing text shown in green.

… indicates omitted existing text that is not shown.

(This Enclosure consists of 31 pages, including this cover page.)

Plant-Specific Tier 1 and associated COL Appendix C Table 2.2.3-1 is revised to include new line items as shown below.

Table 2.2.3-1

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
…									
pH Adjustment Basket 4B	PXS-MY-Y04B	No	Yes		- / -		- / -		
Downspout Screen 1A	PXS-MY-Y81	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 1B	PXS-MY-Y82	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 1C	PXS-MY-Y83	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 1D	PXS-MY-Y84	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2A	PXS-MY-Y85	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2B	PXS-MY-Y86	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2C	PXS-MY-Y87	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2D	PXS-MY-Y88	No	Yes	-	- / -	-	- / -	-	-
CMT A Inlet Isolation Motor-operated Valve	PXS-PL-V002A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/No	None	As Is
…									

Plant-Specific Tier 1 and associated COL Appendix C Table 2.2.3-2 is revised to include new line items at the end of the table as shown below.

Table 2.2.3-2 (cont.)

Line Name	Line Number	ASME Code Section III	Leak Before Break	Functional Capability Required
…				
IRWST gutter drain line	PXS-L142A, PXS-L142B	Yes	No	Yes
	PXS-L141A, PXS-L141B	Yes	No	No
Downspout drain lines from polar crane girder and internal stiffener to collection box A	PXS-L301A, PXS-L302A, PXS-L303A, PXS-L304A, PXS-L305A, PXS-L306A, PXS-L307A, PXS-L308A, PXS-L309A, PXS-L310A	Yes	No	Yes
Downspout drain lines from polar crane girder and internal stiffener to collection box B	PXS-L301B, PXS-L302B, PXS-L303B, PXS-L304B, PXS-L305B, PXS-L306B, PXS-L307B, PXS-L308B, PXS-L309B, PXS-L310B	Yes	No	Yes

COL Appendix A, Technical Specification (TS) 3.5.4, PRHR HX – Operating, SR 3.5.4.7, is revised as shown below.

SURVEILLANCE		FREQUENCY
SR 3.5.4.7	Verify by visual inspection that the IRWST gutter~~s~~ and downspout screens are not restricted by debris.	24 months

The UFSAR Subsection 1.9.4.2.2, Task Action Plan Items, Item A-31, Residual Heat Removal Requirements, AP1000 Response, first and second paragraphs are revised as shown below.

The AP1000 employs safety-related core decay heat removal systems that establish and maintain the plant in a safe, stable ~~shutdown~~ condition following design basis events. It is not necessary that these passive systems achieve cold shutdown as defined by Regulatory Guide 1.139.

The AP1000 complies with General Design Criteria 34 by using a more reliable and simplified system design. The passive core cooling system is employed for both hot-standby and long-term cooling modes. Hot-standby conditions are achieved immediately and a temperature of 420°F is reached within 36 hours as discussed in Subsection 19E.4.10.2. Reactor pressure is controlled and can be reduced to about 250 psig. The passive residual heat removal system provides a closed cooling system to maintain long-term core cooling. Passive feed and bleed cooling, using the passive injection features for the feed and the automatic depressurization system for bleed, provides safety-related cooling capability. ~~This capability eliminates dependency on open-loop cooling systems, which have limited ability to remain in hot standby for long-term core cooling.~~ See Section 7.4 for a discussion of safe shutdown and Section 6.3 for a description of the passive core cooling system.

The UFSAR Subsection 1.9.5.1.5, Station Blackout, AP1000 Response, third paragraph is revised as shown below.

The AP1000 safety-related passive systems automatically establish and maintain safe, stable ~~shutdown~~ conditions for the plant following design basis events, including an extended loss of ac power sources. The passive systems can maintain these safe, stable ~~shutdown~~ conditions after design basis events for at least 72 hours, without operator action, following a loss of both onsite and offsite ac power sources. Subsection 1.9.5.4 provides additional information on long-term actions following an extended station blackout beyond 72 hours.

UFSAR Section 3.2, Table 3.2-3 is revised to include the new downspout screens as shown below.

Table 3.2-3 (cont.)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
…					
PXS-MY-Y81	Downspout Screen 1A	C	I	Manufacturer Std	
PXS-MY-Y82	Downspout Screen 1B	C	I	Manufacturer Std	
PXS-MY-Y83	Downspout Screen 1C	C	I	Manufacturer Std	
PXS-MY-Y84	Downspout Screen 1D	C	I	Manufacturer Std	
PXS-MY-Y85	Downspout Screen 2A	C	I	Manufacturer Std	
PXS-MY-Y86	Downspout Screen 2B	C	I	Manufacturer Std	
PXS-MY-Y87	Downspout Screen 2C	C	I	Manufacturer Std	
PXS-MY-Y88	Downspout Screen 2D	C	I	Manufacturer Std	
PXS-PL-V002A	CMT A CL Inlet Isolation	A	I	ASME III-1	
…					

The UFSAR Subsection 5.4.5.2.1, Pressurizer, is revised to include a new final paragraph as shown below.

The AP1000 pressurizer has metallic reflective insulation (MRI) installed on the external surfaces; the insulation is designed to minimize heat losses from the pressurizer, to reduce heat load on the containment cooling system, and to limit temperatures in nearby concrete or components. During normal operating conditions, the insulation has an average maximum heat transfer rate of 65 Btu/hr-ft^2 at a containment design temperature of 120°F.

The UFSAR Subsection 5.4.11.2, System Description, second paragraph, is revised as shown below.

The discharge from each of two groups of automatic depressurization system valves is connected to a separate sparger below the water level in the in-containment refueling water storage tank. The piping and instrumentation diagram for the connection between the automatic depressurization system valves and the in-containment refueling water storage tank is shown in Figure ~~6.3-2~~ 6.3-1. The…

The UFSAR Subsection 5.4.14.1, Design Bases, first, second and third paragraphs, are revised as shown below.

The passive residual heat removal heat exchanger~~s~~ automatically actuates to remove~~s~~ core decay heat for 72 hours as discussed in Section 6.3 ~~an unlimited period of time~~, assuming the condensate from steam generated in the in-containment refueling water storage tank (IRWST) is returned to the tank. The passive residual heat removal heat exchanger is designed to withstand the design environment of 2500 psia and 650°F.

The passive residual heat removal heat exchanger and the in-containment refueling water storage tank are designed to delay significant steam release to the containment for at least one hour. The passive residual heat removal heat exchanger will ~~keep the reactor coolant subcooled and~~ prevent water relief from the pressurizer and remove sufficient decay heat from the reactor coolant system to satisfy the applicable post-accident safety evaluation criteria detailed in Chapter 15 for at least 72 hours. {combine paragraphs}

~~The passive residual heat removal heat exchanger in conjunction with the passive containment cooling system can remove heat for an indefinite time in a closed-loop (that is, no pipe break) mode of operation.~~ In addition, the passive residual heat removal heat exchanger will cool the reactor coolant system, with reactor coolant pumps operating or in the natural circulation mode, so that the reactor coolant system pressure can be lowered ~~depressurized~~ to reduce stress levels in the system if required. See Section 6.3 for a discussion of the capability of the passive core cooling system.

The UFSAR Subsection 6.3.1.1.1, Emergency Core Decay Heat Removal, is revised as shown below.

For postulated non-LOCA events, where a loss of capability to remove core decay heat via the steam generators occurs, the passive core cooling system is designed to perform the following functions for at least 72 hours:

- The passive residual heat removal heat exchanger automatically actuates to provide reactor coolant system cooling and to prevent water relief through the pressurizer safety valves.

- The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, the condensate collection features, and the passive containment cooling system, is designed to remove decay heat following a design basis event~~ for an indefinite time in a closed-loop mode of operation~~. Automatic depressurization actuation is not expected, but may occur depending on the amount of reactor coolant system leakage and when normal systems are recovered (refer to Subsection 6.3.1.1.4).

- The passive residual heat removal heat exchanger is designed to maintain acceptable reactor coolant system conditions following a non-LOCA event. The applicable post-accident safety evaluation criteria are discussed in Chapter 15.

- The passive residual heat removal heat exchanger is capable of performing its post-accident safety functions~~ automatically removing core decay heat following such an event~~, assuming the steam generated in the in-containment refueling water storage tank is condensed on the containment vessel and returned by gravity via the in-containment refueling water storage tank condensate return gutter and downspouts.

- ~~The passive residual heat removal heat exchanger, in conjunction with the passive containment cooling system, is designed to remove decay heat for an indefinite time in a closed-loop mode of operation. The passive residual heat removal heat exchanger is designed to cool the reactor coolant system to 420°F in 36 hours, with or without reactor coolant pumps operating. This allows the reactor coolant system to be depressurized and the stress in the reactor coolant system and connecting pipe to be reduced to low levels. This also allows plant conditions to be established for initiation of normal residual heat removal system operation.~~

- During a steam generator tube rupture event, the passive residual heat removal heat exchanger removes core decay heat and reduces reactor coolant system temperature and pressure, equalizing with steam generator pressure and terminating break flow, without overfilling the steam generator.

System operation beyond 72 hours is described in Subsection 6.3.1.2.1.

The UFSAR Subsection 6.3.1.1.4, Safe Shutdown, is revised as shown below.

The functional requirements for the passive core cooling system specify that the plant be brought to a safe, stable condition using the passive residual heat removal heat exchanger for events not involving a loss of coolant. As stated in Subsection 6.3.1.1.1, the passive residual heat removal heat exchanger in conjunction with the passive containment cooling system provides sufficient heat removal to satisfy the post-accident safety evaluation criteria for at least 72 hours. Additionally~~For these events~~, the passive core cooling system, in conjunction with the passive containment cooling system and the automatic depressurization system, has the capability to establish long-term safe shutdown conditions~~, cooling~~ in the reactor coolant system as identified in Subsection 7.4.1.1~~to about 420°F in 36 hours, with or without the reactor coolant pumps operating~~.

The core makeup tanks automatically provide injection to the reactor coolant system after they are actuated on low reactor coolant temperature or low pressurizer pressure or level~~as the temperature decreases and pressurizer level decreases, actuating the core makeup tanks~~. The passive core cooling system can maintain stable plant conditions for a long time in this mode of operation, depending on the reactor coolant leakage and the availability of normal systems~~ac power sources~~. For example, with a technical specification leak rate of 10 gpm, stable plant conditions can be maintained for at least 10 hours. With a smaller leak a longer time is available. ~~However in scenarios when ac power sources are unavailable for as long as 24 hours, the automatic depressurization system will automatically actuate.~~

In scenarios when ac power sources are unavailable for approximately 22 hours, the automatic depressurization system automatically actuates. However, after the initial plant cooldown following a non-LOCA event, operators assess plant conditions and have the option to perform recovery actions to further cool and depressurize the reactor coolant system in a closed-loop mode of operation, i.e., without actuation of the automatic depressurization system. After verifying the reactor coolant system is in an acceptable, stable condition such that automatic depressurization is not needed, the operators may take action to extend the passive residual heat removal heat exchanger operation by de-energizing the loads on the 24-hour Class 1E dc batteries powering the protection and monitoring system actuation cabinets. After operators have taken action to extend its operation, the passive residual heat removal heat exchanger, in conjunction with the passive containment cooling system, has the capability to maintain safe, stable conditions for at least 72 hours. The automatic depressurization system remains available to maintain safe shutdown conditions at a later time.

In most sequences, the operators would return the plant to normal system operations and terminate passive system operation within several hours in accordance with the plant emergency operating procedures. For loss of coolant accidents, when the core makeup tank level reaches the automatic depressurization system actuation setpoint and other postulated events where~~ac power sources are lost~~ the passive residual heat removal heat exchanger operation is not extended or is exhausted, ~~or when the core makeup tank levels reach the automatic depressurization system actuation setpoint,~~ the automatic depressurization system ~~initiates~~ may be initiated. This results in injection from the accumulators and subsequently from the in-containment refueling water storage tank, once the reactor coolant system is nearly depressurized. For these conditions, the reactor coolant system depressurizes to saturated conditions at about 250°F within 24 hours. The

passive core cooling system can maintain this safe shutdown condition indefinitely for the plant as identified in Subsection 7.4.1.1.

The ~~basis used to define the~~ passive core cooling system functional requirements ~~are derived from Section 7.4 of the Standard Review Plan. The functional requirements~~ are met over the range of anticipated events and single failure assumptions. The primary function of the passive core cooling system during a safe shutdown using only safety-related equipment is to provide a means for boration, injection, and core cooling. Details of the safe shutdown design bases are presented in Subsection 5.4.7 and Section 7.4. The performance of the passive residual heat removal heat exchanger to bring the plant to 420°F in 36 hours is summarized in Subsection 19E.4.10.2.

The UFSAR Subsection 6.3.1.1.6, Reliability Requirements, last sentence, is revised as shown below.

Subsection 6.3.1.3~~2~~ includes specific nonsafety-related design requirements that help to confirm satisfactory system reliability.

The UFSAR Subsection 6.3.1.2, Power Generation Design Basis, is moved to 6.3.1.3, and a new Subsection 6.3.1.2 is inserted as shown below.

6.3.1.2 Nonsafety Design Basis

6.3.1.2.1 Post Accident Core Decay Heat Removal

The passive residual heat removal heat exchanger is designed to cool the reactor coolant system to 420°F in 36 hours, with or without reactor coolant pumps operating. This allows the reactor coolant system to be depressurized and the stress in the reactor coolant system and connecting pipe to be reduced to low levels. This non-bounding, conservative evaluation is discussed in Subsection 19E.4.10.2.

The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, the condensate return features, and the passive containment cooling system, has the capability to maintain the reactor coolant system in the specified, long-term safe shutdown condition of 420°F for greater than 14 days in a closed-loop mode of operation. The automatic depressurization system can be manually actuated by the operators during the extended passive residual heat removal heat exchanger operation to initiate open-loop cooling. The operator actions necessary to achieve safe shutdown using the passive residual heat removal heat exchanger in a closed-loop mode of operation involve preventing unnecessary actuation of the automatic depressurization system as detailed in Subsection 7.4.1.

Eventually, if pressurizer heaters are not available, the pressurizer subcools due to ambient heat loss. When this happens, the steam void within the pressurizer is transferred to the reactor coolant system. It has been determined that this condition is safe as long as the passive residual heat removal performance is not affected.

If passive residual heat removal performance is affected by subcooling (or other plant conditions) and non-safety systems to control core cooling are not reestablished, then the final, long-term safe shutdown conditions may be achieved and maintained using the automatic depressurization system as discussed in Subsection 7.4.1.1.

6.3.1.3 Power Generation Design Basis

The UFSAR Subsection 6.3.2.1, Schematic Piping and Instrumentation Diagram, first sentence, is revised as shown below.

Figure~~s~~ 6.3-1 ~~and 6.3-2~~ shows the piping and instrumentation drawings of the passive core cooling system.

The UFSAR Subsection 6.3.2.1.1, Emergency Core Decay Heat Removal at High Pressure and Temperature Conditions, last three paragraphs, are revised as shown below and a 4th paragraph is added.

The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, the condensate return features, and the passive containment cooling system, can provide core cooling for at least 72 hours~~an indefinite period of time~~. After the in-containment refueling water storage tank water reaches its saturation temperature (in several ~~about 2~~ hours), the process of steaming to the containment initiates. Containment pressure increases as steam is released from the in-containment refueling water storage tank. As containment temperature increases, condensation begins to form on the subcooled metal and concrete surfaces inside containment. Condensation on these heat sink surfaces transfers energy to the bulk metal and concrete until they come into equilibrium with the containment atmosphere. Condensation that is not returned to the in-containment refueling water storage tank drains to the containment sump.

Condensation occurs on the steel containment vessel, which is cooled by the passive containment cooling system. ~~The~~ Most of the condensate formed on the containment vessel wall is collected in a safety-related gutter arrangement. A gutter is located near ~~at~~ the operating deck ~~level which returns the~~ elevation, and a downspout piping system is connected at the polar crane girder and internal stiffener, to collect steam condensate ~~to the~~ inside the containment during passive containment cooling system operation and return it to the in-containment refueling water storage tank. The gutter normally drains to the containment sump, but when the passive residual heat removal heat exchanger actuates, safety-related isolation valves in the gutter drain line shut and the gutter overflow returns directly to the in-containment refueling water storage tank. Recovery of the condensate maintains the passive residual heat removal heat exchanger heat sink for greater than 14 days~~an indefinite period of time~~.

The passive residual heat removal heat exchanger is used to maintain an acceptable, stable reactor coolant system ~~a safe shutdown~~ condition. It transfers ~~removes~~ decay heat and sensible heat from the reactor coolant system to the in-containment refueling water storage tank, the containment atmosphere, the containment vessel, and finally to the ultimate heat sink–the atmosphere outside of containment. This occurs after in-containment refueling water storage tank saturation is reached and steaming to containment initiates.

The duration the passive residual heat removal heat exchanger can continue to remove decay heat is affected by the efficiency of the return of condensate to the in-containment refueling water storage tank. The in-containment refueling water storage tank water level is affected by the amount of steam that leaves the tank and does not return. Resources are typically recovered within 72 hours, which allows the operators to place active, defense-in-depth systems into service and to terminate passive system operation. If resources are not recovered within this time frame, cooling can be extended as described in Subsection 7.4.1.1 to maintain a safe, stable condition after a design basis event.

The UFSAR Subsection 6.3.2.2.7, IRWST and Containment Recirculation Screens, first paragraph, is revised as shown below.

The passive core cooling system~~s~~ has two different sets of screens that are used ~~following a LOCA; IRWST screens and containment recirculation screens. These screens~~ to prevent debris from entering the reactor and blocking core cooling passages during a LOCA: IRWST screens and containment recirculation screens. The screens are AP1000 Equipment Class C and are designed to meet seismic Category I requirements. The structural frames, attachment to the building structure, and attachment of the screen modules use the criteria of ASME Code, Section III Subsection NF. The screen modules are fabricated of sheet metal and are designed and fabricated to a manufacturer's standard. The~~se~~ IRWST screens and containment recirculation screens are designed to comply with applicable licensing regulations including:

The UFSAR Subsection 6.3.2.2.7.1, General Screen Design Criteria, is revised to include a new introductory paragraph to be clear that the discussion is specific to the IRWST and containment recirculation screens as shown below.

The IRWST screens and containment recirculation screens are designed with the following criteria.

1. Screens are design to Regulatory Guide 1.82, including:

The UFSAR Subsection 6.3.2.2.7.2, IRWST Screens, third paragraph, is revised as shown below.

During a LOCA, steam vented from the reactor coolant system condenses on the containment shell~~,~~ and drains down the shell to the ~~operating deck elevation~~ polar crane girder or internal stiffener where it is drained via downspouts to the IRWST. Steam that condenses below the internal stiffener drains down the shell and is collected in a gutter near the operating deck elevation. It is very unlikely that debris generated by a LOCA can reach the downspouts or the gutter because of ~~its~~ their locations. Each downspout inlet is covered with a coarse screen that prevents larger debris from entering the downspout. The gutter is covered…

The UFSAR Subsection 6.3.2.8, Manual Actions, third paragraph, is revised as shown below.

The operator can take action to avoid actuation of the automatic depressurization system when it is not needed. For non-LOCA events during which ac power has been lost for more than 22 hours, the protection and safety monitoring system will automatically open the automatic depressurization system valves to begin a controlled depressurization of the reactor coolant system and, eventually, containment floodup and recirculation prior to depletion of the 24-hour Class 1E actuation batteries. However, the operators can take action to block actuation of the automatic depressurization system should actuation be deemed unnecessary based on reactor coolant system conditions. This action allows closed loop passive residual heat removal heat exchanger operation to continue as long as acceptable reactor coolant system conditions are maintained.

Section 7.4 describes the anticipated operator actions to block the unnecessary automatic depressurization system actuation and to achieve recovery using available systems to remove decay heat. Section 7.5 describes the post-accident monitoring instrumentation available to the operator in the main control room following an event.

The UFSAR Subsection 6.3.3, Performance Evaluation, seventh paragraph, is revised as shown below.

For non-LOCA events, the passive residual heat removal heat exchanger is actuated so that it can remove core decay heat. The passive residual heat removal heat exchanger can operate for at least 72 hours after initiation of a design basis event to satisfy Condition I, II, III, and IV safety evaluation criteria described in the relevant safety analyses. Subsection 6.3.3.2.1.1 provides an evaluation of the duration of the passive residual heat removal heat exchanger operation using the LOFTRAN code described in Subsection 15.0.11.2. In this evaluation, it is assumed that the operators power down the protection and safety monitoring actuation cabinets in the 22-hour time frame prior to the automatic timer actuating the automatic depressurization system.

In addition to mitigating the initiating events, the passive residual heat removal heat exchanger is capable of cooling the reactor coolant system to the specified safe shutdown condition of 420°F within 36 hours as described in Subsection 19E.4.10.2. A non-bounding, conservative analysis of the plant response during operator-initiated, extended operation of the passive residual heat removal heat exchanger is demonstrated in the shutdown temperature evaluation of Subsection 19E.4.10.2. The closed-loop cooling mode allows the reactor coolant system pressure to decrease and reduces the stress in the reactor coolant system and connecting pipe.

The UFSAR Subsection 6.3.3, Performance Evaluation, ninth paragraph, is revised as shown below.

As the core makeup tanks drain down, the automatic depressurization system valves are sequentially actuated. The depressurization sequence establishes reactor coolant pressure conditions that allow injection from the accumulators, and then from the in-containment refueling water storage tank and the containment recirculation path. Therefore, an injection source is continually available. If onsite or offsite ac power has not been restored after 72 hours, the post-72 hour support actions described in Subsection 1.9.5.4 maintain this mode of core cooling and provide adequate decay heat removal for an unlimited time.

The transient analyses summarized in Chapter 15 are extended long enough to demonstrate the applicable safety evaluation criteria are met. It is expected that normal systems would be available such that operators could terminate the passive safety systems and proceed with an orderly shutdown. However, as discussed in Subsection 6.3.1.1.4, the passive systems are capable of bringing the plant to a safe, stable condition for at least 72 hours in closed loop cooling mode and for longer in an open loop mode.

The UFSAR Subsection 6.3.3.2.1, Loss of Main Feedwater, is revised to add a new subsection as shown below.

6.3.3.2.1.1 Loss of AC Power to Plant Auxiliaries

The most severe conditions resulting from a loss of ac power to the plant auxiliaries are associated with loss of offsite power with a loss of main feedwater system flow at full power. A loss of main feedwater with a loss of ac power lasting longer than a few hours presents the highest demand on passive residual heat removal heat exchanger operation. Subsection 15.2.6 provides a description of this short-term event, including criteria and analytical results.

During most events, the passive systems would be terminated in hours. When an ac power source is restored and passive core cooling system termination criteria are satisfied, the operator terminates passive core cooling system operation and initiates normal plant shutdown operations (as discussed in Subsection 6.3.1.2.1).

However, if normal systems are not recovered as expected, the passive residual heat removal heat exchanger removes core decay heat and maintains acceptable reactor coolant system conditions for at least 72 hours. For a non-LOCA event where ac power is lost, the automatic depressurization system will actuate in approximately 22 hours if operators do not act to avoid actuation when it is not needed. For this long-term transient, it is assumed operators extend passive residual heat exchanger operation as described in the Subsection 7.4.1.1.

The loss of main feedwater with loss of ac power event is analyzed for a 72-hour period, assuming operators extend closed-loop cooling beyond the time the automatic depressurization system would be actuated by the protection and safety monitoring system. This event mirrors the loss of ac power to the plant auxiliaries as described in Subsection 15.2.6, but the loss of ac power extends to 72 hours. In this event, operation of the passive residual heat removal heat exchanger continues for 72 hours and maintains acceptable reactor coolant system conditions such that the applicable Condition II safety evaluation criteria are met. If non-safety systems capable of removing decay heat are not recovered, operator action to actuate automatic depressurization system is eventually required. This condition would then be bounded by the Condition III event of inadvertent automatic depressurization system actuation.

Reactor coolant system leakage could limit closed-loop capacity. A reactor coolant system leak could produce conditions that would preclude the operators from de-energizing the loads on the 24-hour Class 1E batteries, or could require the operators to re-energize the buses powered by the Class 1E batteries before 72 hours so that the automatic depressurization system valves could be actuated.

The UFSAR Subsection 6.3.3.4.1, Loss of Startup Feedwater During Hot Standby, Cooldowns, and Heatups, last sentence, is revised as shown below.

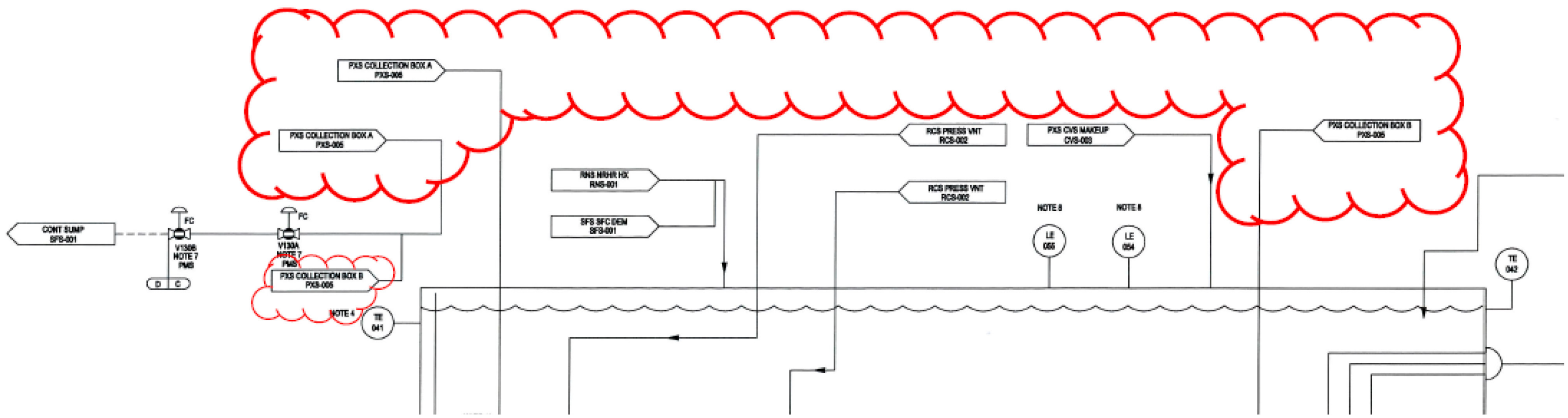
… This allows it to ~~indefinitely~~ function as a heat sink for greater than 14 days, as discussed in Subsection 6.3.1.2.1.

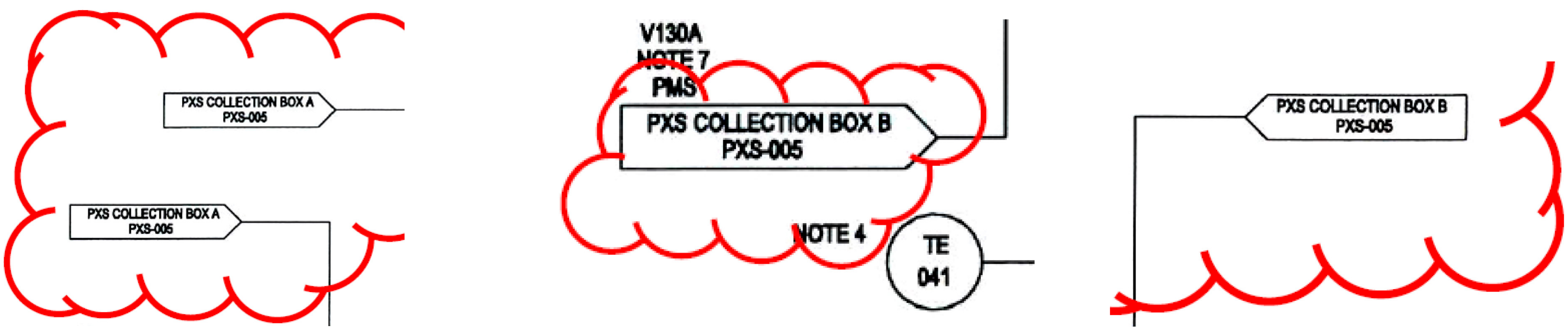
The UFSAR Section 6.3, Figure 6.3-1 Sheet 1 is revised to show that it is (Sheet 1 of 3) rather than (Sheet 1). No other changes are made to this sheet of the figure.

The UFSAR Section 6.3, Figure 6.3-2 is revised to become Figure 6.3-1 (Sheet 2 of 3) rather than Figure 6.3-2, and revised as identified below.

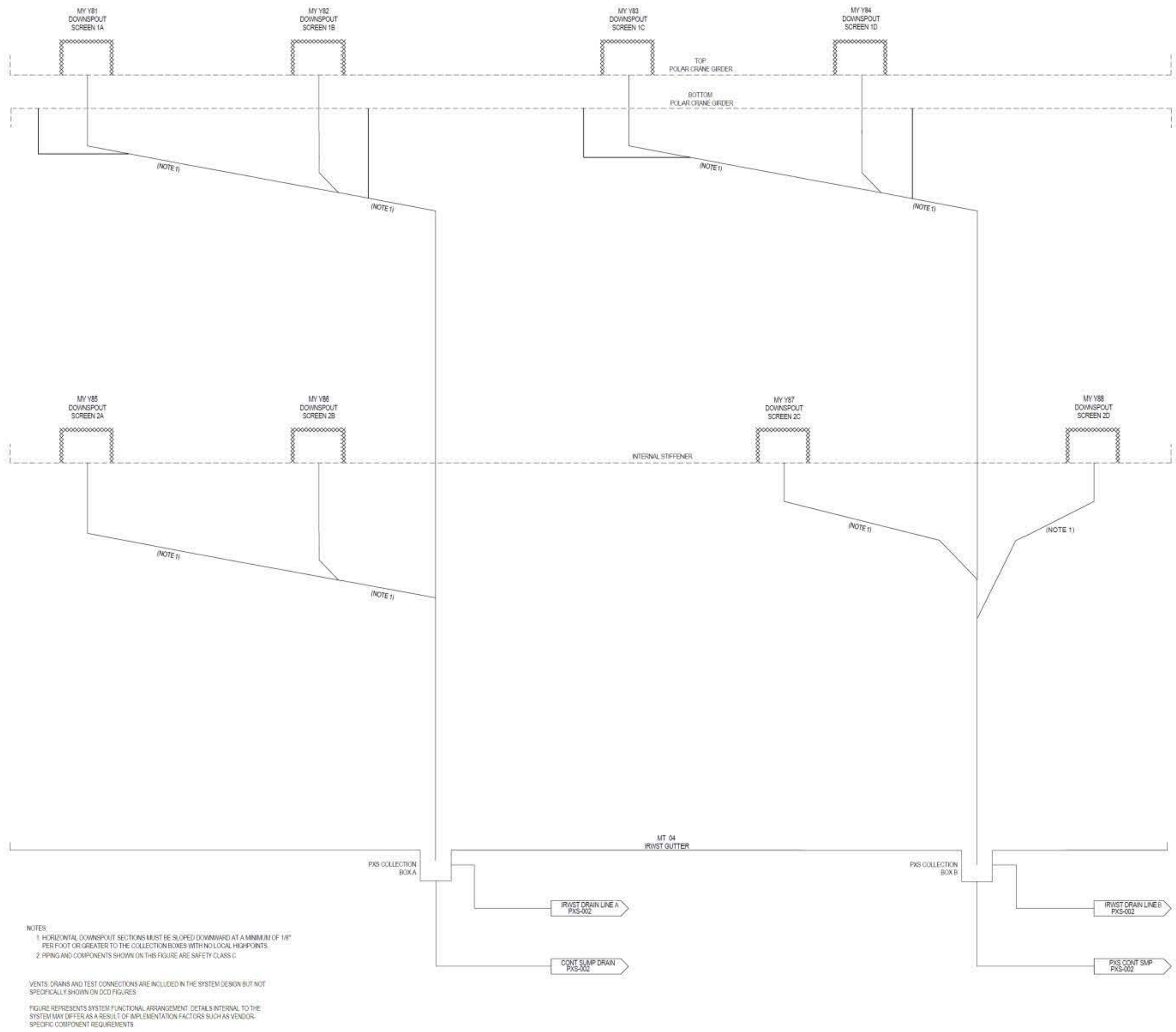
The UFSAR Section 6.3, is revised to include a new Figure 6.3-1 (Sheet 3 of 3), as identified below.

The UFSAR Section 6.3, Figure 6.3-1 (Sheet 2 of 3) [that was Figure 6.3-2] is revised in the upper portion of the figure to relocate the IRWST gutter to a new sheet 3, and to include continuation flags for condensate returning to the IRWST originating from PXS Collection Boxes A and B in the IRWST gutter as shown below. Individual flags are shown larger for clarity below.





New UFSAR Section 6.3, Figure 6.3-1 (Sheet 3 of 3) is included to show the downspout collection system as provided below.



The Notes on this inserted new Figure 6.3-1 (Sheet 3 of 3) are enlarged for ease of review as follows:

1. HORIZONTAL DOWNSPOUT SECTIONS MUST BE SLOPED DOWNWARD AT A MINIMUM OF 1/8” PER FOOT OR GREATER TO THE COLLECTION BOXES WITH NO LOCAL HIGHPOINTS.
2. PIPING AND COMPONENTS SHOWN ON THIS FIGURE ARE SAFETY CLASS C.

VENTS, DRAINS AND TEST CONNECTIONS ARE INCLUDED IN THE SYSTEM DESIGN BUT NOT SPECIFICALLY SHOWN ON DCD FIGURES.

FIGURE REPRESENTS SYSTEMS FUNCTIONAL ARRANGEMENT. DETAILS INTERNAL TO THE SYSTEM MAY DIFFER AS A RESULT OF IMPLEMENTATION FACTORS SUCH AS VENDOR SPECIFIC COMPONENT REQUIREMENTS.

The UFSAR Section 6.3, Figure 6.3-2 location is revised to indicate that Figure 6.3-2 is not used.

The UFSAR Section 7.4, Systems Required for Safe Shutdown, fifth paragraph, is revised as shown below.

The long-term safe shutdown conditions are the same as the short-term conditions except that the core average ~~coolant~~ temperature shall be less than 420°F. This long-term condition must be achieved within 36 hours ~~and~~ following a non-LOCA event using the passive residual heat removal heat exchanger as shown in Appendix 19E. These safe shutdown conditions can be maintained by the passive residual heat removal heat exchanger for greater than 14 days based on a non-bounding, conservative analysis that only credits using safety-related equipment. In addition, these safe shutdown conditions can be maintained indefinitely using the automatic depressurization system and passive injection/recirculation as discussed in Subsection 7.4.1.1~~safety-related equipment~~.

The UFSAR Subsection 7.4.1.1, Safe Shutdown Using Safety-Related Systems, first, sixth, eighth, ninth and eleventh paragraphs, are revised as shown below.

The following describes the process that establishes safe shutdown conditions for the plant, based on a conservative, non-bounding analysis using the safety-related systems, and no operator action. The reactor coolant system is assumed to be intact for this discussion of safe shutdown.

…

The engineered safety system actuation signal generated on low pressurizer pressure also actuates containment isolation. This prevents loss of water inventory from containment and permits ~~indefinite~~ operation of the passive residual heat removal heat exchanger and the in-containment refueling water storage tank for greater than 14 days.

…

A gutter located at the operating deck elevation collects condensate from the inside of the containment shell. Valves located in drain lines from the gutter to the containment waste sump close on a passive residual heat removal heat exchanger actuation signal. This action diverts the condensate to the in-containment refueling water storage tank. The system ~~indefinitely~~ provides core decay heat removal in this configuration for greater than 14 days with~~out~~ a limited ~~significant~~ increase in the containment water level.

Once the reactor coolant system and the safety systems are in this configuration, the plant is in a safe, stable shutdown condition. The reactor coolant system temperatures and pressures continue to slowly decrease. The passive residual heat removal heat exchanger has the capacity to maintain a safe, stable reactor coolant system condition during a design basis event for at least 72 hours in a closed-loop mode of operation. A non-bounding, conservative analysis of extended operation in this mode shows the ~~The~~ passive residual heat removal heat exchanger cools the reactor coolant system to 420°F in 36 hours.

…

The 24-hour Class 1E dc batteries that power the automatic depressurization system valves provide power for at least 24 hours. There is a timer that measures the time that ac power sources are unavailable. This timer provides for automatic actuation of the automatic depressurization system before the 24-hour Class 1E dc batteries are discharged. The emergency response guidelines direct the operator to assess the need for automatic depressurization before the timer completes its count (approximately 22 hours). The operator assessment considers ~~includes consideration for a visible refueling water storage tank level, full~~ core makeup tank~~s~~ levels, ~~and a high and stable in-containment refueling water storage tank level~~ reactor coolant system hot leg level, temperature, and pressure. If automatic depressurization is not needed, the operator is directed to de-energize all loads on the 24-hour Class 1E dc batteries. This action preserves the capability for the operator to initiate automatic depressurization at a later time based on assessment of the same parameters.

The UFSAR Subsection 9.5, Table 9.5.1-1, Sheet 11, item 73, is revised as shown below.

Table 9.5.1-1 (Sheet 11 of 29)
AP1000 Fire Protection Program Compliance with BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[1]	Remarks
…			
73. Fire damage should be limited so that systems necessary to achieve and maintain cold shutdown from either the control room or emergency control station can be repaired within 72 hours.	C.5.b(1)	AC	Safe shutdown following a fire is defined for the AP1000 plant as the ability to achieve and maintain the reactor coolant system (RCS) core average temperature below 215.6°C (420°F) without uncontrolled venting of the primary coolant from the RCS. This is a departure from the criteria applied to the evolutionary plant designs, and the existing plants where safe shutdown for fires applies to both hot and cold shutdown capability. With expected RCS leakage, the AP1000 plant can maintain safe shutdown conditions for greater than 14 days. Therefore, repairs to systems necessary to reach cold shutdown need not be completed within 72 hours.
…			

The UFSAR Subsection 14.3, Table 14.2-3 (Sheets 7 and 8) references to Figure 6.3-2 are revised as shown below.

Table 14.3-2 (Sheet 7 of 17) Design Basis Accident Analysis		
Reference	**Design Feature**	**Value**
Figure 6.3-1~~2~~	The PRHR inlet line (hot leg to high point) has no downward sloping sections.	
Figure 6.3-1~~2~~	The maximum elevation of the IRWST injection lines (from the connection to the IRWST to the reactor vessel) and the containment recirculation lines (from the containment to the IRWST injection lines) is less than the bottom inside surface of the IRWST.	
Figure 6.3-1~~2~~	The maximum elevation of the PRHR outlet line from the PRHR to the SG) is less than the PRHR lower channel head top inside surface.	

The UFSAR Subsection 15.0.13, Operator Action, is revised as shown below.

For events where the PRHR heat exchanger is actuated, the plant automatically cools down to ~~the~~ a safe, stable ~~shutdown~~ condition. Where a stabilized condition is reached automatically following a reactor trip, it is expected that the operator may, following event recognition, take manual control and proceed with orderly shutdown of the reactor in accordance with the normal, abnormal, or emergency operating procedures. The exact actions taken and the time at which these actions occur depend on what systems are available and the plans for further plant operation.

However, for these events, operator actions are not required to maintain the plant in a safe and stable condition for at least 72 hours. Operator actions typical of normal operation are credited for the inadvertent actuations of equipment in response to a Condition II event.

The UFSAR Section 15.2, Decrease in Heat Removal by the Secondary System, is revised to include a new third paragraph as shown below.

For events in this section where PRHR heat exchanger actuation occurs, transients are presented until the PRHR heat exchanger heat removal matches decay heat generation. After that point in time, PRHR heat exchanger performance is driven by the performance of the passive containment cooling systems to control containment pressure and the ability of the condensate collection features to return condensate to the in-containment refueling water storage tank. The performance of these systems, for extended decay heat removal, is described in Subsection 6.3.1.1.1.

The UFSAR Subsection 15.2.6.1, Identification of Causes and Accident Descriptions, fourth paragraph, is revised as shown below.

... The PRHR heat exchanger, in conjunction with the passive containment cooling system, ~~keeps the~~ provides core cooling and maintains reactor coolant ~~subcooled indefinitely~~ system conditions to satisfy the evaluation criteria. After the IRWST water reaches saturation ~~(in about two and half hours)~~, steam starts to vent to the containment atmosphere. …

The UFSAR Section 19.59, Table 19.59-18 (Sheet 6), item 1e, is revised as shown below.

Table 19.59-18 (Sheet 6 of 25) AP1000 PRA-Based Insights	
Design Feature	**Disposition**
1e. (cont.) … The PRHR HX, in conjunction with the IRWST, the condensate return features, and the PCS, can provide core cooling for ~~an indefinite period of time~~ greater than 14 days. After the IRWST water reaches its saturation temperature, the process of steaming to the containment initiates. … …	6.3.2.1.1 & 6.3.7.6

The UFSAR Appendix 19E, Subsection 19E.2.3.2.6, Discussion of Safe Shutdown for AP1000, is revised as shown below.

The functional requirements for the PXS specify that the plant be brought to a safe, stable condition using the PRHR HX for events not involving a loss of coolant. As stated in Subsection 6.3.1.1.1, the PRHR HX, in conjunction with the passive containment cooling system (PCS), provides sufficient heat removal to satisfy the post-accident safety evaluation criteria for at least 72 hours. ~~For these events~~ Additionally, the PXS, in conjunction with the ~~passive containment cooling system (~~PCS~~)~~, and the ADS, has the capability to establish long-term safe shutdown conditions in the RCS as identified in Subsection 7.4.1.1~~, cooling the RCS to less than 420°F within 36 hours, with or without the RCPs operating~~.

The CMTs automatically provide injection to the RCS after they are actuated on low reactor coolant temperature or low pressurizer pressure or level ~~as the temperature decreases and the pressurizer level decreases, actuating the CMTs~~. The PXS can maintain stable plant conditions for a long time in this mode of operation, depending on the reactor coolant leakage and the availability of ac power sources. For example, with a technical specification leak rate of 10 gpm, stable plant conditions can be maintained for at least 10 hours. With a smaller leak, a longer time is available. ~~However, in scenarios when ac power sources are unavailable for as long as 24 hours, the ADS will automatically actuate.~~

In scenarios when ac power sources are unavailable for approximately 22 hours, the ADS automatically actuates. However, after the initial plant cooldown following a non-LOCA event, operators assess plant conditions and have the option to perform recovery actions to further cool and depressurize the RCS in a closed-loop mode of operation, i.e., without actuation of the ADS. After verifying the RCS is in an acceptable, stable condition, such that automatic depressurization is not needed, the operators may take action to extend PRHR HX operation by de-energizing the loads on the Class 1E dc batteries powering the protection and safety monitoring system actuation cabinets. After operators have taken action to extend its operation, the PRHR HX, in conjunction with the PCS, has the capability to maintain safe, stable conditions. The ADS remains available to maintain safe shutdown conditions at a later time.

In most sequences, the operators would return the plant to normal system operations and terminate passive system operation within several hours in accordance with the plant emergency operating procedures. For LOCAs and other postulated events, when the core makeup tank level reaches the automatic depressurization actuation setpoint, and other postulated events where ~~ac power sources are lost, or when the CMT levels reach the ADS actuation setpoint,~~ the PRHR HX operation is not extended or exhausted, ADS ~~initiates~~ may be initiated. This results in injection from the accumulators and subsequently from the in-containment refueling water storage tank, once the RCS is nearly depressurized. For these conditions, the RCS depressurizes to saturated conditions at about ~~240~~ 250°F within 24 hours. The PXS can maintain this safe shutdown condition ~~indefinitely~~ as identified in Subsection 7.4.1.1.

The primary function of the PXS during a safe shutdown using only safety-related equipment is to provide a means for boration, injection, and core cooling. Analysis is provided in subsection 19E.4.10.2 of this appendix that verifies the ability of the AP1000 passive safety systems to meet the safe shutdown requirements.

The UFSAR Subsection 19E.4.10.2, Shutdown Temperature Evaluation, is revised to reflect the results of the design changes as shown below.

~~In SECY-94-084, Item C, Safe Shutdown (Reference 14), the NRC staff recommended the Commission's approval of 420°F or below, rather than cold shutdown condition as a safe stable condition, which the PRHR HX must be capable of achieving and maintaining following non-LOCA events, predicated on acceptable passive safety system performance and an acceptable resolution of the regulatory treatment of nonsafety systems (RTNSS) issue. The NRC requested a safety~~ As discussed in Subsection 6.3.1.1.4, the PRHR HX is required to be able to cool the RCS to a safe, stable condition after shutdown following a non-LOCA event. The following summarizes a non-bounding, conservative analysis, which demonstrates the PRHR HX can meet this criterion and cool the RCS to the specified, safe shutdown condition of 420°F within 36 hours. This analysis ~~to~~ demonstrates that the passive systems can bring the plant to a ~~stable~~ safe, stable condition and maintain this condition so that no transients will result in the specified acceptable fuel design limit and pressure boundary design limit being violated and that no high-energy piping failure being initiated from this condition results in 10 CFR 50.46 (Reference 15) criteria.

As discussed in Subsections 6.3.3 and 7.4.1.1, the PRHR HX operates to reduce the RCS core average temperature to the safe shutdown condition following ~~an~~ a non-LOCA event. An analysis of the loss of main feedwater with a loss of ac power event demonstrates that the passive systems can bring the plant to a ~~stable~~ safe, stable condition following postulated transients. ~~The results of this~~ A non-bounding, conservative analysis ~~are~~ is represented in Figures 19E.4.10-1 through 19E.4.10-4. The progression of this event is outlined in Table 19E.4.10-1. Though some of the assumptions of this evaluation are based on nominal conditions, many of the analysis assumptions are bounding.

The performance of the PRHR HX is affected by the containment pressure. Containment pressure determines the PRHR HX heat sink (the IRWST water) temperature. The WGOTHIC containment response model described in Subsection 6.2.1.1.3 was used to determine the containment pressure response to this transient, which was used as an input to the plant cooldown analysis performed with LOFTRAN. Some changes were made to the WGOTHIC model to provide conservative results for the long-term safe shutdown analysis.

The PRHR HX performance is also affected by the IRWST water level when the level drops below the top of the PRHR HX tubes. The IRWST water level is affected by the heat input from the PRHR HX and by the amount of steam that leaves the IRWST and does not return to the IRWST through the IRWST gutter arrangement. The principal steam condensate losses include steam that stays in the containment atmosphere, steam that condenses on heat sinks inside containment other than the containment vessel, and dripping or splashing losses due to obstructions on the inner containment vessel wall. The WGOTHIC containment response model also provided the mass balance with respect to the steam lost to the containment atmosphere and to condensation on passive heat sinks other than the containment vessel. The WGOTHIC analysis inputs (including the mass of the heat sinks and heat transfer rates) were biased to increase steam condensate losses. The WGOTHIC model provides the time-dependent condensate return rate, which was incorporated into the LOFTRAN computer code described in Subsection 15.0.11.2 to demonstrate that the RCS core average temperature could be cooled to 420°F within 36 hours.

Summarizing this transient, the loss of normal ac power occurs (offsite and onsite), followed by the reactor trip. The PRHR HX ~~heat exchanger~~ is actuated on the low steam generator narrow range level coincident with low startup feed water flow rate signal. Eventually a safeguards actuation signal is actuated on Low cold leg temperature and the CMTs are actuated.

Once actuated, at about ~~600~~ 2,700 seconds, the CMTs operate in recirculation mode, injecting cold borated water into the RCS. In the first part of their operation, due to the injection of cold ~~flow rate~~ water, the CMTs operate in conjunction with the PRHR HX to reduce RCS temperature. Due to the primary system cooldown, the PRHR heat transfer capability drops below the decay heat and the RCS cooldown is essentially driven by the CMT cold injection flow. However, at about ~~3,500~~ 6,000 seconds, the CMT cooling effect decreases and the RCS starts heating up again (Figure 19.E.4.10-1). The RCS temperature increases until the PRHR HX can match decay heat. At about ~~31,000~~ 46,700 seconds, the PRHR heat transfer matches decay heat and it continues to operate to reduce the RCS temperature to below 420°F within 36 hours. As seen from Figure 19E.4.10-1, the cold leg temperature in the loop with the PRHR is reduced to 420°F at ~~82,600~~ about 52,900 seconds, while the core average temperature reaches 420°F ~~in 123,600~~ at about 120,900 seconds (approximately 34 hours).

As discussed in Subsection 7.4.1.1, ~~this mode of operation can last for up to 72 hours. However, in about 22 hours after the event, if no ac power is available, or if condensate return is not available, then the operator is instructed to actuate the ADS.~~ a timer is used to automatically actuate the ADS if offsite and onsite power are lost for about 24 hours. This timer automates putting the open loop cooling features into service prior to draining the Class 1E dc 24-hour batteries that operate the ADS valves. Before 22 hours, if the plant conditions indicate that the ADS would not be needed until well after 24 hours, the operators are directed to de-energize all loads on the 24-hour batteries. This action will block actuation of the ADS and preserves the ability to align open loop cooling at a later time. Operation of the ADS in conjunction with the CMTs, accumulators, and IRWST reduces the RCS pressure and temperature to below 420°F. The ability to actuate ADS and IRWST injection provides a safety-related, backup mode of decay heat removal that is diverse to extended PRHR HX operation.

As discussed in Subsection 6.3.3.2.1.1, the PRHR HX can operate in this mode for at least 72 hours to maintain RCS conditions within the applicable Chapter 15 safety evaluation criteria. In addition, the analysis supporting this section shows the PRHR HX is expected to maintain safe shutdown conditions for greater than 14 days. One important consideration with regard to the duration closed-loop cooling can be maintained is the RCS leak rate. This duration of closed-loop cooling can be achieved with expected RCS leak rates. For abnormal leak rates, it may become necessary to initiate open-loop cooling earlier than 14 days.

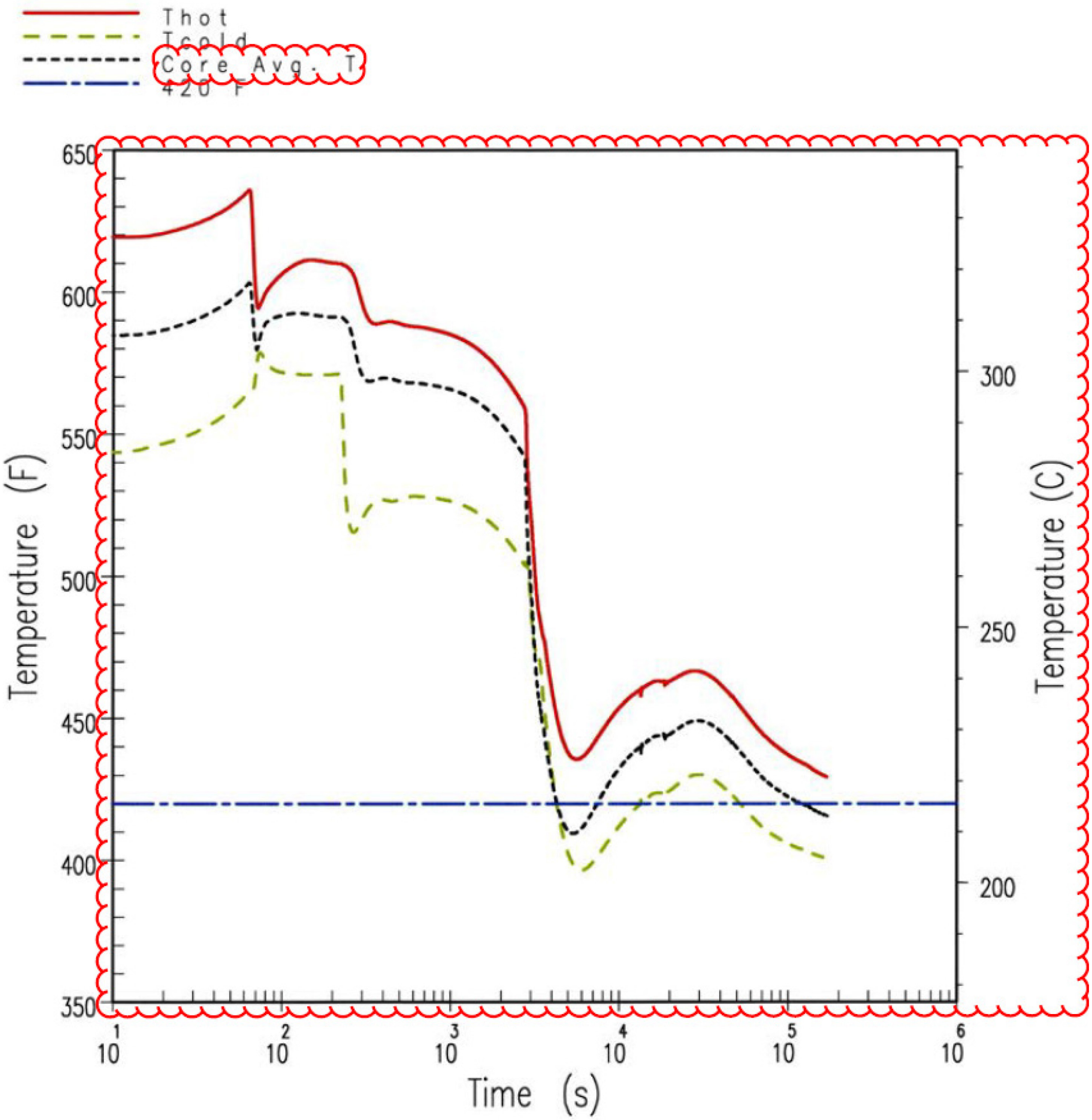
The UFSAR Subsection 19E.9, References, is revised as shown below.

14. Not used. ~~SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," March 28, 1994.~~

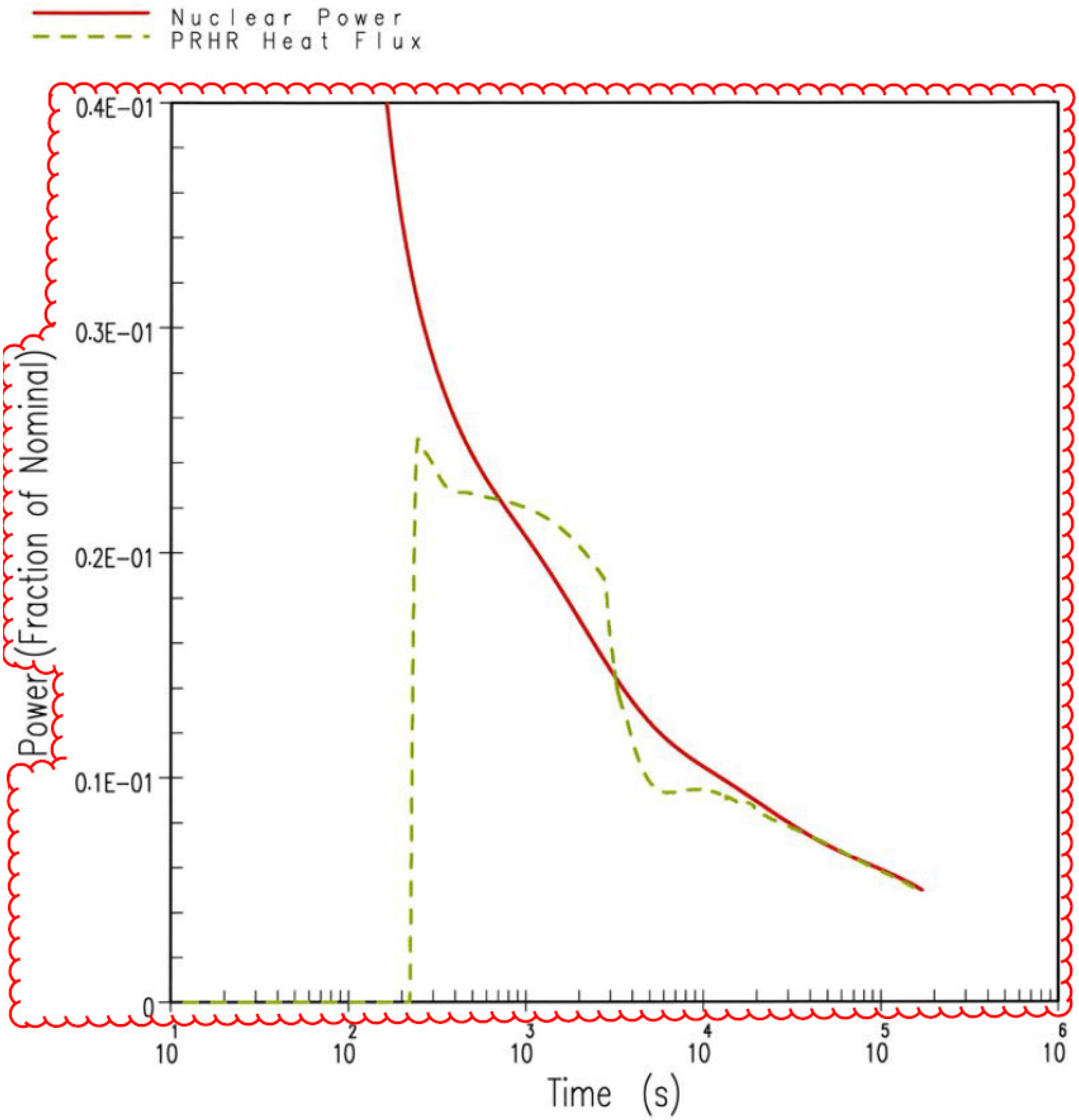
The UFSAR Appendix 19E, Table 19E.4.10-1, Sequence of Events Following a Loss of ac Power Flow with Condensate from the Containment Shell Being Returned to the IRWST, is revised as shown below.

Event	Time (seconds)
Feedwater is Lost	10.0
Low Steam Generator Water Level (Narrow-Range) Reactor Trip Setpoint Reached	60.6 ~~72.4~~
Rods Begin to Drop	62.6 ~~74.4~~
Low Steam Generator Water Level (Wide-Range) Reached	209.5
PRHR HX Actuation on Low Steam Generator Water Level (~~Wide-~~Narrow-Range Coincident with Low Startup Feedwater Flow)	221.5 ~~129.4~~
Low T_{cold} Setpoint Reached	2,752 ~~599.0~~
Steam Line Isolation on Low T_{cold} Signal	2,764 ~~611.0~~
CMTs Actuated on Low T_{cold} Signal	2,764 ~~617.0~~
IRWST Reaches Saturation Temperature	15,900 ~~17,600~~
Heat Extracted by PRHR HX Matches Core Decay Heat	46,700 ~~31,000~~
~~CMTs Stop Recirculating~~	~~43,500~~
Cold Leg Temperature Reaches 420°F (loop with PRHR)	52,900 ~~82,600~~
~~Hot Leg~~ Core Average Temperature Reaches 420°F (loop with PRHR)	120,900 ~~123,600~~

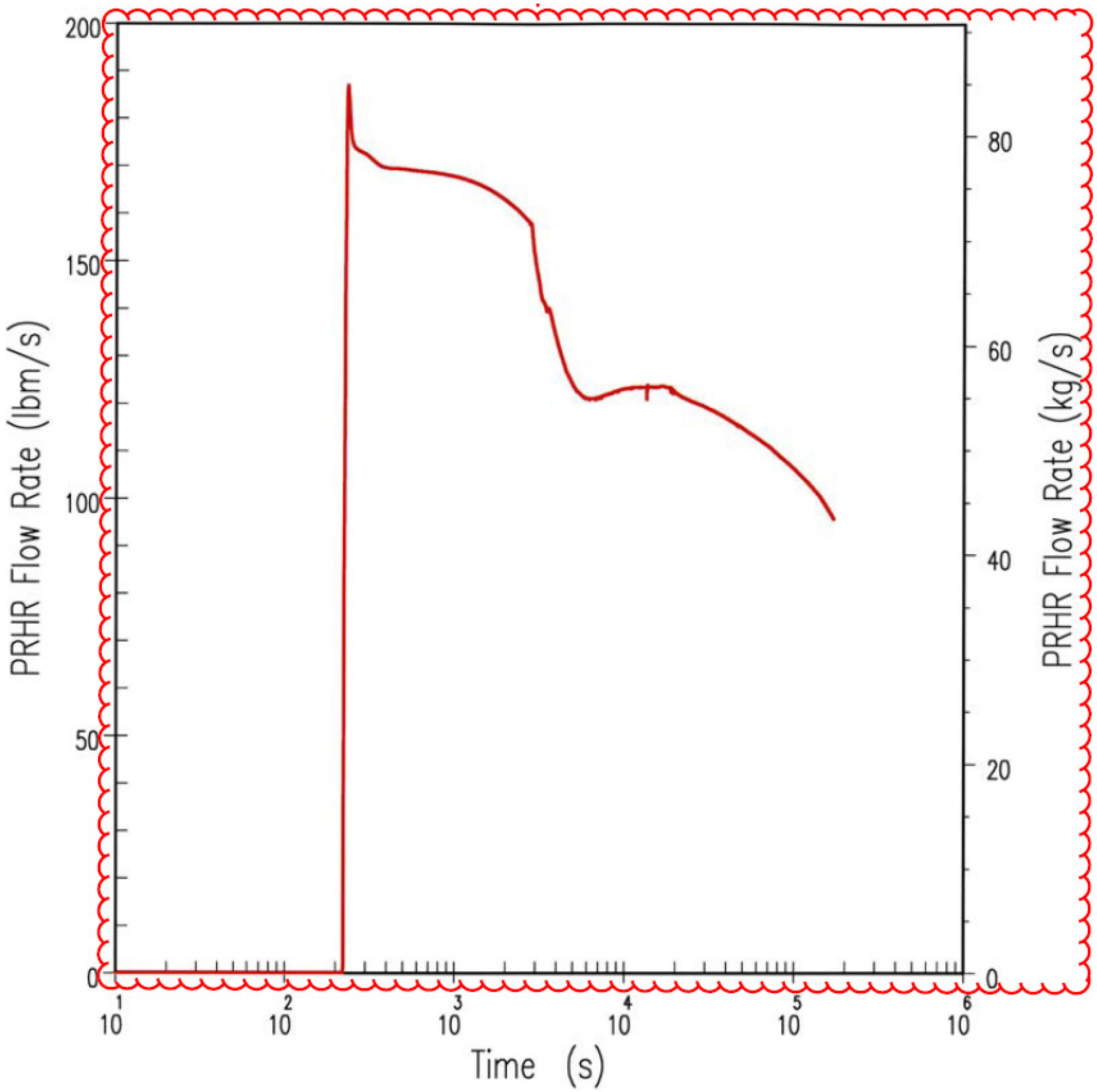
The UFSAR Appendix 19E, Figure 19E.4.10-1, Shutdown Temperature Evaluation, RCS Temperature, is revised to reflect the results of the design changes as shown below.



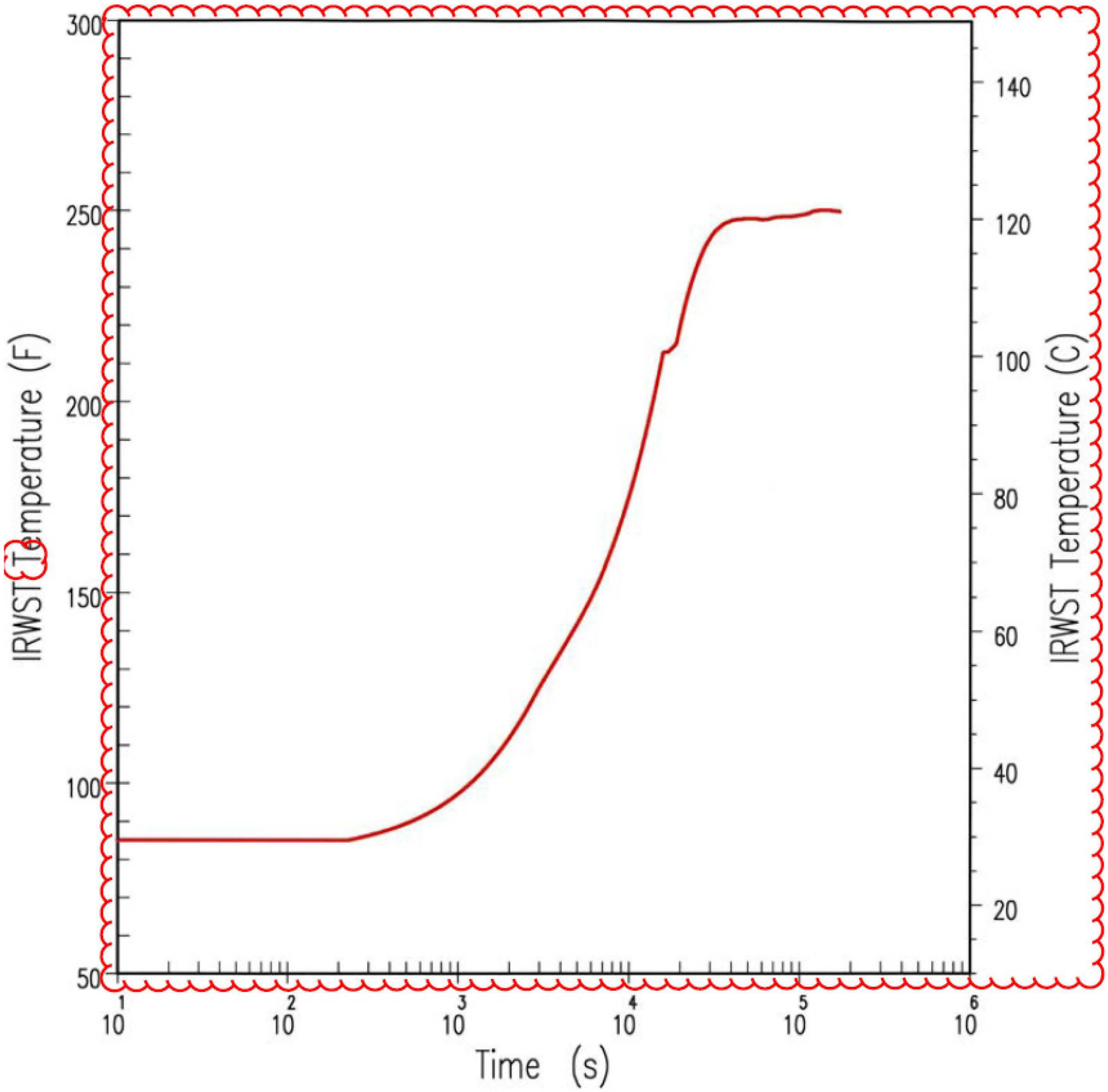
The UFSAR Appendix 19E, Figure 19E.4.10-2, Shutdown Temperature Evaluation, PRHR Heat Transfer, is revised to reflect the results of the design changes as shown below.



The UFSAR Appendix 19E, Figure 19E.4.10-3, Shutdown Temperature Evaluation, PRHR Flow Rate, is revised to reflect the results of the design changes as shown below.



The UFSAR Appendix 19E, Figure 19E.4.10-4, Shutdown Temperature Evaluation, IRWST Heatup, is revised to reflect the results of the design changes as shown below.



Southern Nuclear Operating Company

Vogtle Electric Generating Plant Units 3 and 4

ND-16-1892

Enclosure 4

Conforming Changes to the Technical Specifications Bases for Information

(LAR-16-026)

Additions identified by red underlined text.

~~Deletions Identified by red strikethrough of text.~~

… indicates omitted existing text that is not shown.

(This Enclosure consists of 2 pages, including this cover page.)

The Bases for Technical Specification 3.3.17, Post Accident Monitoring (PAM) Instrumentation, is revised in the last sentence of the first paragraph of the LCO discussion of Function 11, In-Containment Refueling Water Storage Tank (IRWST) Water Level, as shown below.

… The condensate is returned to the IRWST via a gutter and downspouts.

The Bases for Technical Specification 3.5.4, Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Operating, is revised in the third paragraph of the BACKGROUND discussion as shown below.

In order to preserve the IRWST water for long-term PRHR HX operation, downspouts and a gutter ~~is~~ are provided to collect and return water to the IRWST that has condensed on the inside surface of the containment shell. During normal plant operation, any water collected by the downspouts or gutter is directed to the normal containment sump. During…

The Bases for Technical Specification 3.5.4, Surveillance Requirement (SR) 3.5.4.7, is revised as shown below.

This surveillance requires visual inspection of the IRWST gutter~~s~~ and downspout screens to verify that the return flow to the IRWST will not be restricted by debris. A Frequency of 24 months is adequate~~,~~ since there are no known sources of debris with which the gutter~~s~~ or downspout screens could become restricted.

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Enclosure 5

Applicability and Endorsement of Prior Docketed Information

(LAR-16-026)

(This Enclosure consists of 7 pages, including this cover page.)

Pursuant to 10 CFR 2.390, the below incorporated documents submitted by Duke Energy and identified as proprietary should not be disclosed to the public. Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference the corresponding Westinghouse Application letter (“CAW” letter number), and should be addressed to the point of contact listed therein. Correspondence with respect to proprietary aspects of the information incorporated by reference should also be addressed to Brian H. Whitley at the contact information within this letter. SNC also makes a 10 CFR 2.390(b)(4) request for withholding based on: the reasons listed in corresponding Westinghouse Application letter; SNC's contractual obligation to seek proprietary treatment of the information; and the information is the type that has historically been held in confidence by SNC.

The appropriate information from each of these applicable, endorsed documents has been incorporated into the LAR. The technical information provided in the incorporated Duke Energy documents is supporting information and neither changes nor affects the scope of the Technical Evaluation nor the conclusions of the Significant Hazards Consideration determination in this license amendment request

Technical Reports

APP-GW-GLR-161-P, Revision 4, and its non-proprietary version, **APP-GW-GLR-607-NP, Revision 4**, were previously submitted by Duke Energy Florida, on the Levy Nuclear Plant, Units 1 and 2 docket, Docket Nos. 52-029 and 52-030, via Duke letter NPD-NRC-2016-002, dated January 14, 2016 [ML16020A105]. These documents, and their corresponding affidavits and notices (Enclosures 1, 2, 3 & 4 of the above referenced letter) are applicable supporting documents for the changes proposed in this LAR and are hereby incorporated by reference.

Design Changes

It is noted that the technical documents incorporated by reference above address several design changes, most of which are addressed in the requested license amendment. However, there were some associated design changes addressed in the enclosures which did not meet the regulation criteria for requiring prior NRC approval for implementation. The purpose of this discussion is to specifically identify these associated changes and confirm that these associated design changes have been incorporated into the plant design and licensing basis.

Specifically, Section 5.0 of **APP-GW-GLR-161-P, Revision 4**, and its non-proprietary version, **APP-GW-GLR-607-NP, Revision 4**, identifies the following design changes related to improvement of the PXS condensate return:

1) PXS downspout piping;
2) Downspout screen design (includes adding new screens);
3) Blocking of the polar crane girder / stiffener fabrication holes;
4) Addition of dam on the polar crane girder;
5) Gutter drip lip modification;
6) Personnel airlock and equipment hatch gutter routing; and
7) Update of the Shutdown Temperature Evaluation.

The changes to the PXS downspout piping (item 1 above), the addition of new screens (part of item 2 above), and the update of the shutdown temperature evaluation (item 7 above) are addressed in this License Amendment Request.

The change to block the polar crane girder and stiffener fabrication holes (item 3 above) affected UFSAR Figure 3.8.2-1, Sheet 3 of 3, in that the fabrication holes were shown on the figure. This change (which did not meet the criteria for requiring prior NRC approval) and its UFSAR impact on the associated figure have already been incorporated into the Vogtle Units 3&4 design and licensing basis. This departure was reported in the Vogtle 3 and 4 semi-annual departure report as LDCR 2014-028 via SNC letter ND-14-0768, dated June 13, 2014.

The remaining changes for the downspout screen design (remaining part of item 2 above), the addition of a dam on the polar crane girder (item 4 above), the gutter drip modification (item 5 above), and the personnel airlock and equipment hatch gutter routing (item 6 above) did not directly impact the current licensing basis documents, nor meet any of the criteria for requiring prior NRC approval. As such, these changes have also been implemented in the design to support improvement of the PXS condensate return.

Requests for Additional Information (RAIs)

During the course of the NRC technical review of the changes proposed on the Duke Energy Florida (DEF) Levy and Duke Energy Carolinas (DEC) Lee dockets, five Requests for Additional Information were issued by the NRC Staff. These are listed here.

1. Letter from Donald Habib (NRC) to Christopher M. Fallon (DEF), dated March 6, 2014, "Request for Additional Information Letter No. 116 Related to SRP Sections 6.3 and 15.2.6." [ML14065A362]
2. Letter from Donald Habib (NRC) to Christopher M. Fallon (DEF), dated April 10, 2014, "Request for Additional Information Letter No. 117 Related to SRP Section 6.3." [ML14100A040]
3. Letter from Donald Habib (NRC) to Christopher M. Fallon (DEF), dated April 24, 2014, "Request for Additional Information Letter No. 118 Related to SRP Section 6.3." [ML14114A050]
4. Letter from Donald Habib (NRC) to Christopher M. Fallon (DEF), dated December 5, 2014, "Request for Additional Information Letter No. 124 Related to SRP Section 6.3." [ML14341A003]
5. Letter from Donald Habib (NRC) to Christopher M. Fallon (DEF), dated January 13, 2015, "Request for Additional Information Letter No. 125 Related to SRP Section 6.3." [ML15013A500]

The Duke Levy docketed responses to these Requests for Additional Information are listed here. Reference information for these responses is provided following the listing.

NRC RAI #	Duke Energy RAI #	Duke Energy Response on the Levy Docket
15.02.06-1	L-1081	NPD-NRC-2014-017, dated June 19, 2014
15.02.06-1	L-1106	NPD-NRC-2014-024, dated July 24, 2014
15.02.06-1	L-1173	NPD-NRC-2016-006, dated January 15, 2016
15.02.06-2	L-1082	NPD-NRC-2014-021, dated June 27, 2014
15.02.06-3	L-1085	NPD-NRC-2014-017, dated June 19, 2014
06.03-1	L-1086	NPD-NRC-2014-014, dated May 5, 2014
06.03-2	L-1087	NPD-NRC-2014-016, dated June 12, 2014
06.03-3	L-1088	NPD-NRC-2014-016, dated June 12, 2014
06.03-4	L-1089	NPD-NRC-2014-022, dated July 1, 2014
06.03-5	L-1090	NPD-NRC-2014-021, dated June 27, 2014
06.03-6	L-1091	NPD-NRC-2014-014, dated May 5, 2014
06.03-6	L-1127	NPD-NRC-2015-023, dated June 11, 2015
06.03-7	L-1092	NPD-NRC-2014-012, dated April 17, 2014
06.03-8	L-1093	NPD-NRC-2014-012, dated April 17, 2014
06.03-9	L-1094	NPD-NRC-2014-015, dated May 19, 2014
06.03-9	L-1134	NPD-NRC-2015-023, dated June 11, 2015
06.03-10	L-1096	NPD-NRC-2014-021, dated June 27, 2014
06.03-11	L-1097	NPD-NRC-2014-021, dated June 27, 2014
06.03-12	L-1099	NPD-NRC-2014-021, dated June 27, 2014
06.03-13	L-1120	NPD-NRC-2015-015, dated May 5, 2015
06.03-14	L-1124	NPD-NRC-2015-004, dated January 21, 2015

The Duke Energy Florida (DEF), Levy Nuclear Plant (LNP), Units 1 and 2, Docket Nos. 52-029 and 52-030, letters responding to the RAIs listed above (with referenced information provided below) have been reviewed and found to be applicable as supporting documents for the changes proposed in this LAR and are hereby incorporated by reference.

1. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-012, dated April 17, 2014 [ML14112A371] - Enclosures 1, 2, 3, and 4. The referenced enclosures address RAIs 06.03-7 and 06.03-8. This letter contains proprietary information.

2. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-014, dated May 5, 2014 [ML14126A699] - Enclosures 1, 2, and 3. The referenced enclosures address RAIs 06.03-1 and 06.03-6. This letter contains proprietary information.
3. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-015, dated May 19, 2014 [ML14141A015] - Enclosure 1. The referenced enclosure addresses RAI 06.03-9.
4. Duke Energy, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-016, dated June 12, 2014 [ML14164A444] - Enclosures 1, 2, and 3. The referenced enclosures address RAIs 06.03-2 and 06.03-3. This letter contains proprietary information.
5. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-017, dated June 19, 2014 (ADAMS Accession No. ML14171A453) - Enclosure 1. The referenced enclosure addresses RAIs 15.02.06-1 and 15.02.06-3.
6. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-021, dated June 27, 2014 (ADAMS Accession No. ML14182A106) - Enclosures 1, 2, 3, and 4. The referenced enclosures address RAIs 15.02.06-2, 06.03-5, 06.03-10, 06.03-11 and 06.03-12. This letter contains proprietary information.
7. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-022 (note that page 1 of the cover letter incorrectly identifies this letter as NPD-NRC-2014-021}, dated July 1, 2014 (ADAMS Accession No. ML14183B342) - Enclosure 1. The referenced enclosure addresses RAI 06.03-4.
8. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-024, dated July 24, 2014 (ADAMS Accession No. ML14206A951) - Enclosures 1, 2, and 3. The referenced enclosures address RAI 15.02.06-1. This letter contains proprietary information.
9. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-028, dated July 24, 2014 (ADAMS Accession No. ML14206A953) - Enclosures 1, 2, 3, and 4. The referenced letter contains supplemental information addressing RAIs 06.03-10 through 06.03-12. This letter contains proprietary information.
10. Letter from Christopher Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated November 17, 2014, "Supplement 5 to Submittal of Exemption Request and Design Change Description for Departure from AP1000 DCD Revision 19 to Address Containment Condensate Return Cooling Design," Serial: NPD-NRC-2014-038 [ML14323A286].
11. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2015-004, dated January 21, 2015 (ADAMS Accession No. ML15023A036) - Enclosure 1. The referenced letter contains a response to RAI 06.03-14. This response was supplemented by NPD-NRC-2015-015, dated May 5, 2015.
12. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2015-015, dated May 5, 2015. [ML15128A604] - Enclosures 1, 2, 3, 4, and 5. The referenced letter contains a response to RAI 06.03-13. This letter contains proprietary information.

13. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2015-023, dated June 11, 2015 [ML15166A020] - Enclosure 1. The referenced enclosure contains revised responses addressing RAIs 06.03-06 and 06.03-09.
14. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2015-026, dated June 30, 2015 [ML15187A051]. This letter confirms that the long term analysis for PRHR HX operation is complete, and the results are available for NRC review and audit. These results confirm the licensing basis, of greater than 14 days for PRHR HX operation, is met.
15. Letter from Christopher Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated January 15, 2016, "Supplement 2 Response To NRC RAI Letter 116 – SRP Sections 6.3 and 15.2.6," Serial: NPD-NRC-2016-006 [ML16021A188]. This letter contains proprietary information.

Southern Nuclear Operating Company

Vogtle Electric Generating Plant Units 3 and 4

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Enclosure 6

Proposed Changes that Differ from the Duke Submittals for Information

(LAR-16-026)

(This Enclosure consists of 8 pages, including this cover page.)

The following comparison is based on the final Duke Energy Carolinas, Lee Nuclear Station, Units 1 and 2, Docket Nos. 52-018 and 52-019, Letter WLG2016.03-03, Enclosure 9, dated March 24, 2016. [ML16088A022]. Enclosure 9 of that letter provided a composite set of proposed changes to the Lee Nuclear Station Licensing Basis Documents.

Changes which are not "technically consistent" are identified and discussed following the listings of the "technically consistent" changes. "Technically consistent" allows for minor editorial differences that do not impact the technical details of the discussion, e.g., grammar, verb tense, capitalization, use of acronyms or initialisms, etc.

The following LAR change to COL Appendix A (Plant-Specific Technical Specifications (TS)) is technically consistent with the corresponding departure to COLA Part 4, provided in the Duke Energy Carolinas submittal for the Lee Nuclear Station (WLS), dated March 24, 2016.

- COL Appendix A (Plant-Specific TS) Surveillance Requirement (SR) 3.5.4.7

The following LAR changes to COL Appendix C (and Plant-Specific Tier 1) are technically consistent with the corresponding departure to COLA Part 10, Appendix B, provided in the Duke Energy Carolinas submittal for the Lee Nuclear Station (WLS), dated March 24, 2016.

- COL Appendix C (and Plant-Specific Tier 1) Table 2.2.3-1
- COL Appendix C (and Plant-Specific Tier 1) Table 2.2.3-2

The LAR changes to the UFSAR listed below are technically consistent with the corresponding departure to COLA Part 2, provided in the Duke Energy Carolinas submittal for the Lee Nuclear Station (WLS), dated March 24, 2016, or where the changes are not "technically consistent," additional information is provided and the differences are discussed following this listing.

- UFSAR Subsection 1.9.4.2.2, item A-31
- UFSAR Subsection 1.9.5.1.5
- UFSAR Section 3.2, Table 3.2-3
- UFSAR Section 3.8, Figure 3.8.2-1 (This change not included in this LAR; see basis below.)
- UFSAR Subsection 5.4.5.2.1
- UFSAR Subsection 5.4.11.2
- UFSAR Subsection 5.4.14.1 (generally consistent; see discussion of differences below)
- UFSAR Subsection 6.3.1.1.1 (generally consistent; see discussion of differences below)
- UFSAR Subsection 6.3.1.1.4 (generally consistent; see discussion of differences below)
- UFSAR Subsection 6.3.1.1.6
- UFSAR Subsection 6.3.2.1

- UFSAR Subsection 6.3.2.1.1 (generally consistent; see discussion of differences below)
- UFSAR Subsection 6.3.2.2.7
- UFSAR Subsection 6.3.2.2.7.1
- UFSAR Subsection 6.3.2.2.7.2
- UFSAR Subsection 6.3.2.8 (generally consistent; see discussion of differences below)
- UFSAR Subsection 6.3.3
- UFSAR Subsection 6.3.3.2.1.1 (generally consistent; see discussion of differences below)
- UFSAR Subsection 6.3.3.4.1
- UFSAR Subsection 6.3, Figure 6.3-1 (now Figure 6.3-1 (Sheet 1 of 3)
- UFSAR Subsection 6.3, Figure 6.3-2 (now Figure 6.3-1 (Sheet 2 of 3)
- UFSAR Subsection 6.3, Figure 6.3-1 (new Sheet 3 of 3)
- UFSAR Subsection 7.4 (generally consistent; see discussion of differences below)
- UFSAR Subsection 7.4.1.1 (generally consistent; see discussion of differences below)
- UFSAR Subsection 9.5, Table 9.5.1-1, item 73 (generally consistent; see discussion of differences below)
- UFSAR Section 14.3, Table 14.3-2
- UFSAR Subsection 15.0.13
- UFSAR Subsection 15.2
- UFSAR Subsection 15.2.6.1
- UFSAR Subsection 19.59, Table 19.59-18, item 1e
- UFSAR Appendix 19E, Subsection 19E.2.3.2.6
- UFSAR Appendix 19E, Subsection 19E.4.10.2 (generally consistent; see discussion of differences below)
- UFSAR Appendix 19E, Section 19E.4.10, Table 19E.4.10-1
- UFSAR Appendix 19E, Section 19E.4.10, Figure 19E.4.10-1
- UFSAR Appendix 19E, Section 19E.4.10, Figure 19E.4.10-2
- UFSAR Appendix 19E, Section 19E.4.10, Figure 19E.4.10-3
- UFSAR Appendix 19E, Section 19E.4.10, Figure 19E.4.10-4

The following LAR changes to the UFSAR are NOT fully consistent with the corresponding departure to COLA Part 2, provided in the Duke Energy Carolinas submittal for the Lee Nuclear Station (WLS), dated March 24, 2016. See discussion for each item below.

UFSAR Section 3.8, Figure 3.8.2-1, Sheet 3 of 3:

Duke/WLS: Revises the figure to remove weep holes (block the polar crane girder and stiffener fabrication holes).

SNC/VEGP: This change previously made, and thus, is not included in the requested changes for this LAR.

Basis for Difference:

The change to block the polar crane girder and stiffener fabrication holes did not meet the criteria for requiring prior NRC approval, and the UFSAR impact on the associated figure has already been incorporated into the design and licensing basis. This departure was reported in the Vogtle 3 and 4 semi-annual departure report as LDCR 2014-028 via SNC letter ND-14-0768, dated June 13, 2014.

UFSAR Subsection 5.4.14.1:

Duke/WLS: Revises the discussion of Passive Residual Heat Removal Heat Exchanger Design Basis to remove a phrase indicating that the PRHR Hx will "prevent water relief from the pressurizer."

SNC/VEGP: This phrase is not removed. Other changes in UFSAR Subsection 5.4.14.1 are consistent with the corresponding departure to COLA Part 2, provided in the Duke Energy Carolinas submittal for the Lee Nuclear Station (WLS), dated March 24, 2016.

Basis for Difference:

This change is currently being considered for processing in association with a separate change as this change is not directly a result of the revised calculations.

UFSAR Subsection 6.3.1.1.1:

Duke/WLS: Revises the discussion of Emergency Core Decay Heat Removal. The first bullet is revised to remove a phrase indicating that the PRHR Hx will "prevent water relief through the pressurizer safety valves."

SNC/VEGP: No change is made to this bullet. Other changes in UFSAR Subsection 6.3.1.1.1 are consistent with the corresponding departure to COLA Part 2, provided in the Duke Energy Carolinas submittal for the Lee Nuclear Station (WLS), dated March 24, 2016.

Basis for Difference:

This change is currently being considered for processing in association with a separate change as this change is not directly a result of the revised calculations.

UFSAR Subsection 6.3.1.1.4:

Duke/WLS: Revises the discussion of Safe Shutdown and includes a discussion of the capability of the PXS to maintain stable plant conditions "depending on the reactor coolant leakage and the availability of ac power sources."

The revised discussion also indicates that "de-energizing the loads on the Class 1E dc batteries powering the protection and monitoring system actuation cabinets."

SNC/VEGP: The discussion of the capability of the PXS to maintain stable plant conditions is revised to indicate that it is "depending on the reactor coolant leakage and the availability of normal systems ~~ac power sources~~."

The changes are made with the addition of clarifying the Class 1E dc batteries are the "24-hour" Class 1E dc batteries.

Other changes in UFSAR Subsection 6.3.1.1.4 are consistent with the corresponding departure to COLA Part 2, provided in the Duke Energy Carolinas submittal for the Lee Nuclear Station (WLS), dated March 24, 2016.

Basis for Difference:

These changes recognize and clarify that the capability of the PXS to maintain stable plant conditions is dependent not only on the power sources but also on the availability of the normal systems powered by the ac power sources.

UFSAR Subsection 6.3.2.1.1:

Duke/WLS: Revises the discussion of Emergency Core Decay Heat Removal at High Pressure and Temperature Conditions and indicates "The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, condensate return features and passive containment cooling system, can provide core cooling for greater than 14 days."

The discussion also indicates "If resources are not recovered within this time frame, closed-loop cooling using the passive residual heat removal heat exchanger can be extended as described in DCD Subsection 7.4.1.1 to maintain a safe, stable condition after a design basis event."

SNC/VEGP: The discussion of the Emergency Core Decay Heat Removal at High Pressure and Temperature Conditions indicates the capability is "for at least 72 hours" rather than "for greater than 14 days."

The discussion of extended cooling is simplified to include less detail of what is described in UFSAR Subsection 7.4.1.1; to read "If resources are not recovered within this time frame, cooling can be extended as described in Subsection 7.4.1.1 to maintain a safe, stable condition after a design basis event."

Other changes in UFSAR Subsection 6.3.1.1.4 are consistent with the corresponding departure to COLA Part 2, provided in the Duke Energy Carolinas submittal for the Lee Nuclear Station (WLS), dated March 24, 2016.

Basis for Difference:

These changes reflect revised capabilities of the emergency systems, consistent with expectations for these systems, and to recognize the variations in methods to extend the cooling beyond 72 hours.

UFSAR Subsection 6.3.2.8:

Duke/WLS: Revises the discussion of Manual Actions to address blocking of the automatic depressurization system when conditions do not require actuation.

SNC/VEGP: The discussion is supplemented to identify "the actuation batteries" more specifically as "the 24-hour Class 1E actuation batteries."

Basis for Difference:

This change is simply a clarification of which batteries are "the actuation batteries."

UFSAR Subsection 6.3.3.2.1.1:

Duke/WLS: Adds new discussion of Loss of AC Power to the Plant Auxiliaries.

SNC/VEGP: The discussion is supplemented to identify the ADS actuation occurs "in approximately 22 hours" if operator action is not taken.

The discussion is supplemented to identify "the actuation batteries" more specifically as "the 24-hour Class 1E actuation batteries."

Basis for Difference:

This change is simply clarification to reflect when the ADS actuation occurs and to identify which batteries are "the actuation batteries."

UFSAR Subsection 7.4:

Duke/WLS: Revises the discussion of Systems Required for Safe Shutdown for long-term conditions.

SNC/VEGP: The discussion is revised to more accurately identify the "core average temperature" conditions rather than "coolant temperature" conditions.

Basis for Difference:

This change made for consistency in the references to the 420 F temperature.

UFSAR Subsection 7.4.1.1:

Duke/WLS: Revises the discussion of Safe Shutdown Using Safety-Related Systems to address the capabilities of the systems.

SNC/VEGP: The discussion is supplemented to identify the "Class 1E dc batteries" more specifically as the "24-hour Class 1E dc batteries."

Basis for Difference:

This change is simply a clarification of which batteries are "the actuation batteries."

UFSAR Subsection 9.5, Table 9.5.1-1, item 73:

Duke/WLS: Revises the discussion of BTP CMEB 9.5-1 Guideline for fire damage repair.

SNC/VEGP: The discussion is revised to more accurately identify the "core average temperature" conditions rather than "RCS temperature" conditions.

Basis for Difference:

This change made for consistency in the references to the 420 F temperature.

UFSAR Appendix 19E, Subsection 19E.4.10.2:

Duke/WLS: Revises the discussion of the Shutdown Temperature Evaluation to reflect the non-bounding, conservative analysis results.

SNC/VEGP: The discussion is revised to more accurately identify the "core average temperature" conditions rather than RCS temperature conditions.

Basis for Difference:

This change made for consistency in the references to the 420 F temperature.