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Docket Nos.: 52-025
52-026

ND-14-0588
10 CFR 50.90
10 CFR 52.63

U.S. Nuclear Regulatory Commission
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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-13-024)

Ladies and Gentlemen:

In accordance with 10 CFR 50.90, Southern Nuclear Operating Company (SNC), the licensee for Vogtle Electrical Generating Plant (VEGP) Units 3 and 4, requests an amendment to Combined License (COL) Numbers NPF-91 and NPF-92, for VEGP Units 3 and 4, respectively. 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.

The proposed amendment would revise the plant-specific Tier 1 and associated Tier 2 material to increase 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 description, technical evaluation, regulatory evaluation (including the significant hazards consideration determination), and environmental considerations for the proposed changes in the License Amendment Request (LAR) are contained in Enclosure 1 to this letter. Enclosure 2 provides the background and supporting basis for the requested exemption. Enclosure 3 provides markups depicting the requested changes to the licensing basis documents.

This request is consistent in technical content with a combined license application (COLA) update as submitted April 18, 2013 (and supplemented) by Progress Energy on the Levy Nuclear Plant dockets (No. 52-029 and 52-030) to address containment condensate return design. SNC is aware that requests for additional information (RAIs) are being addressed on the Levy docket and acknowledges that similar additional information will need to be addressed on this Vogtle docket upon NRC acceptance of this LAR.

SNC requests staff approval of the license amendment and associated exemption by October 15, 2014. Delayed approval of this license amendment could result in a delay of construction of the containment vessel ring 2 and subsequent dependent construction activities.

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.

This letter contains no regulatory commitments.

Should you have any questions, please contact Mr. C. Brian Meadors at (205) 992-7331.

Mr. Brian H. Whitley states that: he is the Regulatory Affairs Director of Southern Nuclear Operating Company; he 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

Brian H. Whitley

B. H. Whitley

BHW/ERG/kms

Sworn to and subscribed before me this 11th day of April, 2014

Notary Public: Kristin Marie Seibert

My commission expires: August 16, 2016

KRISTIN MARIE SEIBERT NOTARY PUBLIC ALABAMA STATE AT LARGE

Enclosures: 1) Request for License Amendment Regarding Passive Core Cooling System (PXS) Condensate Return (LAR-13-024)

2) Request for Exemption Regarding Passive Core Cooling System (PXS) Condensate Return (LAR-13-024)

3) Proposed Changes to the Licensing Basis Documents (LAR-13-024)

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**Southern Nuclear Operating Company**

**Vogtle Electric Generating Plant Units 3 and 4**

**ND-14-0588**

**Enclosure 1**

**Request for License Amendment Regarding**

**Passive Core Cooling System (PXS) Condensate Return**

**(LAR-13-024)**

Table of Contents

## 1. Summary Description

In accordance with 10 CFR 50.90, Southern Nuclear Operating Company (SNC), the licensee for Vogtle Electrical Generating Plant (VEGP) Units 3 and 4, requests an amendment to Combined License (COL) Numbers NPF-91 and NPF-92, for 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) and includes 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 (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.

To improve the efficiency of the condensate return, changes to the PXS are proposed. The proposed changes would revise the Combined Licenses (COLs) in regard to the condensate return function supporting prolonged non-LOCA events. The changes would 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, a rigorous 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 safety-related design criteria of cooling the RCS to 420°F within 36 hours and maintaining a safe, stable condition.

The requested amendment requires changes to UFSAR information, which involves changes to plant-specific Tier 1, and corresponding changes to COL Appendix C, information. (See Section 2 for details.) This enclosure requests approval of the license amendment necessary to implement the UFSAR changes and their involved Tier 1 and COL changes.

## 2. Detailed Description

<u>Background</u>

NRC regulations (in General Design Criterion 34 of Appendix A to 10 CFR Part 50) require the plant design to include a system to remove residual heat from the reactor core such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. The Passive Core Cooling System (PXS) provides emergency

core cooling during transients, accidents, and whenever the normal, nonsafety-related heat removal paths are unavailable.

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 6.3-2. 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 steam generated discharges through a series of vents located near the steam generator compartments 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, which results in the steam generator wall vents opening 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, and 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; as 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 is an angle stiffener and also 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 will be 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 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. However, since the efficiency of the PXS condensate return function is so important, a decision was made to conduct testing and to analyze the plant performance with a series of calculations that included quantification of steaming from the IRWST and the portion of that steam that condenses and returns to the IRWST.

Testing results showed that the current design of the PCG, internal stiffener, and IRWST gutter could contribute to losses at each location larger than assumed. These identified losses due to the physical geometry of the containment indicated the constant condensate return assumed in the safety analyses may not be appropriate. Analytically, when the constant condensate return assumption is replaced with the experimental design return rates including losses, the resultant PRHR HX performance is degraded and affects the design basis accident (DBA) safety analyses described in Chapter 15 of the UFSAR. However, it was also determined that the PCG and internal stiffener can be modified to improve condensate return such that the Chapter 15 design basis analyses would not be impacted.

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 Polar Crane Girder and internal stiffener,
5) Losses at support plates attached to the containment vessel,
6) Losses at the Equipment Hatch and Personnel Air Lock, and
7) Losses at entry to 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 the development of two new calculations and the revision of two existing calculations as discussed below.

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 would 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 polar crane girder would join together into a common header which extends below the

internal stiffener. The two connections from the internal stiffener would join together into a common line, which would connect to the header below the internal stiffener. The header would be 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 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 are identified as PXS-L301A/B to PXS-L308A/B, and are Safety Class C, seismic Category I.

Pipe sizes are selected to prevent pipes from running full of water. The pipe sizes are also 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 would be after a LOCA event, or the IRWST gutter might be during refueling activities. The sections of piping routed horizontally are sloped 1/8 inch per foot or greater downward toward each of the respective collection boxes. The PXS piping and instrumentation diagrams are changed 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 was 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 and identified as 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 screen is 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 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. The WGOTHIC results are combined with detailed calculations of geometric condensate losses that incorporated the activities described.

The geometric and thermodynamic losses were incorporated into a calculation to verify acceptable PRHR HX performance. The results of this calculation showed that, with the intermediate collection points established by the addition of downspouts and blocking of the PCG and internal stiffener fabrication holes, enough condensate is returned to the IRWST to maintain the IRWST water level above the top of the PRHR HX tubes until long after the success criteria of the design basis non-LOCA events described in UFSAR Chapter 15 are met. This analysis also demonstrates that closed-loop PRHR HX operation can provide long-term core decay heat removal.

The time-dependent condensate return fraction and the resultant IRWST level and PRHR HX response are input into the LOFTRAN code to demonstrate the ability of the PRHR HX to cool the RCS temperature to 420°F within 36 hours in a closed-loop mode of operation. 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. Although the PRHR HX tubes do begin to uncover before the plant reaches 420°F (which occurs long after the endpoint of the Chapter 15 events), the analysis shows the plant can successfully meet this safety-related design criterion.

Additionally, UFSAR Table 19E.4.10-1, which reports the time at which the hot leg temperature in the RCS loop with the PRHR HX in it reaches safe shutdown temperature, is revised. The time shown is actually the time at which the core average temperature reaches safe shutdown temperature, as described in the text of UFSAR Subsection 19E.4.10.2.

Licensing Basis Change Descriptions

Revisions to the Plant-Specific Tier 1 information:

1. Table 2.2.3-1 is revised to include new downspout screens.

2. Table 2.2.3-2 is revised to include new downspout piping.

Revisions to the Tier 2 information incorporated into the UFSAR:

3. UFSAR Table 3.2-3 is revised to include the new downspout screens.

4. UFSAR Subsection 5.4.11.2 is updated to cross reference to Figure 6.3-1 (rather than Figure 6.3-2 which will not be used) for consistency across the chapters.

5. UFSAR Subsection 6.3.1.1.1 is updated to describe the downspouts in the safety-related design criteria.

6. UFSAR Subsection 6.3.2.1.1 is updated to include the intermediate collection points of the safety- related gutter arrangement.

7. UFSAR 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.

8. UFSAR Subsection 6.3.2.2.7.2 is updated to clarify the condensate return gutter arrangement related to LOCA operation.

9. UFSAR Figure 6.3-1 is relabeled Figure 6.3-1 (Sheet 1 of 3). This is the only change made to this figure. No technical changes are made.

10. UFSAR Figure 6.3-2 is now labeled 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.

11. UFSAR Figure 6.3-1 (Sheet 3 of 3) is a new P&ID sheet 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 Polar Crane Girder and internal stiffener.

12. Reference to UFSAR Figure 6.3-2 in Subsection 6.3.2.1 is changed to Figure 6.3-1 for consistency.

13. In UFSAR Table 14.3-2, cross reference to Figure 6.3-2 is changed to Figure 6.3-1 for consistency.

14. The plant-specific Technical Specification Bases LCO for B 3.3.3 are updated to reflect the addition of downspouts.

15. The plant-specific Technical Specification Bases Surveillance Requirement for SR 3.5.4.7 is updated to encompass the entire gutter arrangement, including the downspout screens, in the surveillance.

16. The plant-specific Technical Specification Bases Background for B 3.5.4 is updated to reflect the addition of downspouts.

17. UFSAR Subsection 19E.4.10.2 is changed to include a description of the analysis methodology for the safe shutdown evaluation, to update the transient description and to reflect the updated results of the evaluation. The results of the updated shutdown temperature evaluation are also presented in Table 19E.4.10-1 and Figures 19E.4.10-1 through 19E.4.10-4.

As indicated by the above listing of changes, 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 or safe shutdown temperature analysis methodology, the licensing basis changes proposed herein provide additional piping, components and adjustments to optimize the condensate return provisions and provide descriptions of the analysis methodology in the UFSAR.

Chapter 3

The new PXS downspout screens to be added would be 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 UFSAR Table 3.2-3 to capture these requirements.

Chapter 5

In UFSAR Subsection 5.4.11.2, the cross reference to Figure 6.3-2 is changed to Figure 6.3-1 for consistency with changes made to the Chapter 6 figures.

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. The specific 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.
- Subsection 6.3.2.1.1 is updated to include the intermediate collection points of the safety-related gutter arrangement.
- 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 identify 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.
- 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 would be 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 added to the licensing basis. This new figure would show the relocated IRWST gutter and the screens and piping comprising the PXS downspouts originating from the Polar Crane Girder and internal stiffener.
- Figure 6.3-2 is not used.
- References to Figure 6.3-2 in UFSAR Subsection 6.3.2.1 are changed for consistency.

Chapter 14

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

Chapter 16

The plant-specific Technical Specification Bases would be updated to include the downspouts in the descriptions of the gutter arrangement.

- The Bases LCO for B 3.3.3 are updated to reflect the addition of downspouts.
- The Bases Surveillance Requirement for SR 3.5.4.7 is updated to encompass the entire gutter arrangement, including the downspout screens, in the surveillance.
- The Bases Background for B 3.5.4 is updated to reflect the addition of downspouts.

Chapter 19

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 Subsection 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.

Tier 1 (COL Appendix C)

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, test, 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 will 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 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 will be met,

plant-specific Tier 1 Table and the corresponding COL Appendix C Table 2.2.3-2 are updated to include the new PXS pipe lines.

## 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 safety-related design 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 non-LOCA 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 described in Chapter 15 continue to 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 described response does not change because even if the time-dependent condensate return fraction were applied, the PRHR HX would remain submerged well beyond the duration of the relevant design basis analyses listed in Table 1.

**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..

To further analyze the effects of the variable condensate return fraction, the PRHR Performance (see Figure 1) calculation presents an independent analysis of PRHR HX performance under various operating conditions. One performance case (CD2) tracks IRWST level and RCS temperature with PRHR HX operation under design basis conditions comparable to the Chapter 15 events. Case CD2 considers the changing condensate return fraction as a function of time and predicts PRHR HX performance for durations much longer than the Chapter 15 analyses. Case CD2 confirms that the PRHR HX tubes remain submerged for longer than the recovery time of any of the design basis analyses listed in Table 1. When the PRHR HX tube sheet begins to uncover, the available heat transfer area of the heat exchanger begins to decrease. However, the heat exchanger is sufficiently sized to continue to match decay heat well after uncovery begins. Case CD2 demonstrates the effects of time-dependent condensate return are not important in relation to the time scales of the design basis accidents analyzed in Chapter 15 and confirms there is no cliff in PRHR HX performance that would warrant re-analysis of the Chapter 15 events that credit the PRHR HX.

In addition, the WGOTHIC peak containment pressure analysis was considered during the course of testing and analysis for this change; and was 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 guillotine cold leg 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 and 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 would result 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 to be constructed of only those materials appropriately suited for exposure to the reactor coolant environment as described in UFSAR Subsection 6.1.1.4. No aluminum is permitted to be used in the construction of these components so that they do not contribute to hydrogen production in containment. No system

or design function or equipment qualification is affected 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, per SECY-94-084 (reference 1), 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 at best estimate conditions, with a number of conservatisms maintained, and assumed a constant condensate return fraction. The plant response after shutdown following non-LOCA events was reanalyzed with a series of interdependent calculations. The information flow between these calculations is illustrated in Figure 1.

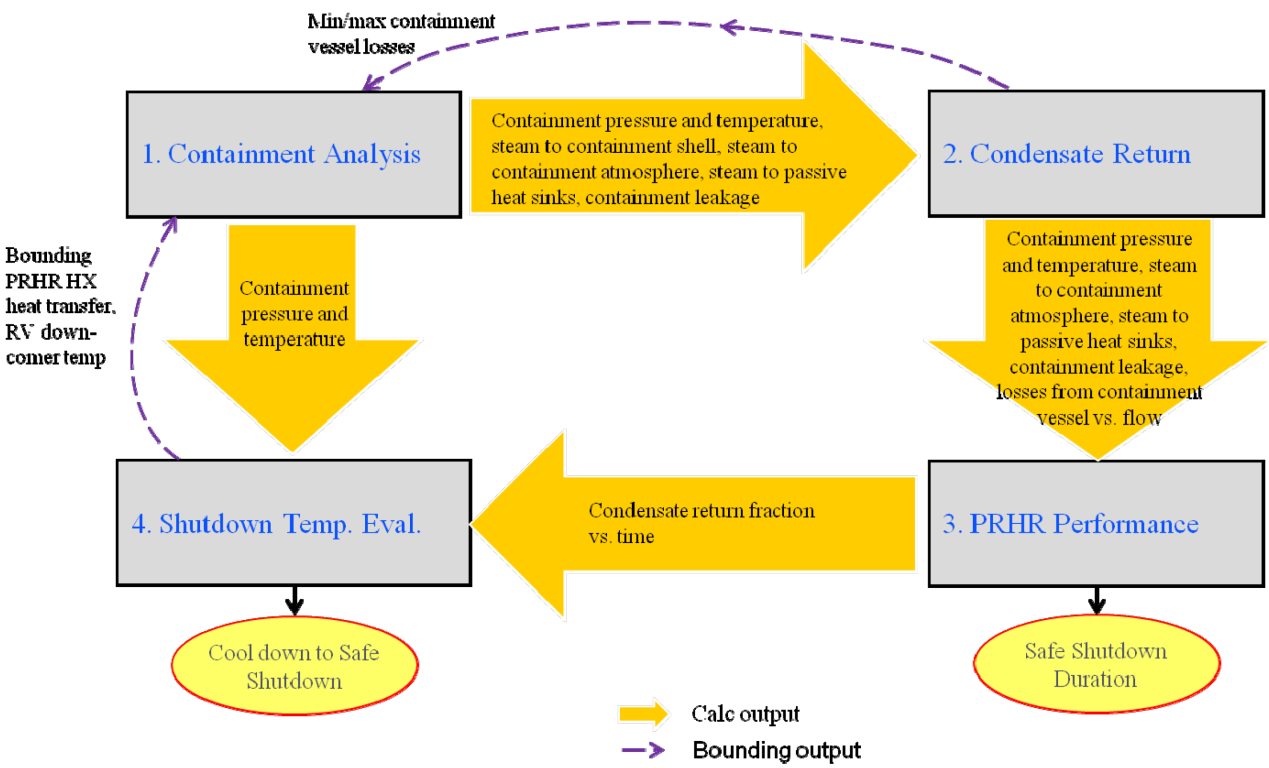



Figure 1: Calculation inter-relationships

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 were performed, which incorporate the lessons learned about condensate return from design review and testing. As described in the Chapter 19 Shutdown Temperature Evaluation, loss of normal feedwater coincident with loss of ac power event was identified to be the most limiting transient with regard to PRHR HX performance. The Shutdown Temperature Evaluation was updated 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 safe shutdown temperature evaluation, as the maximum pressure reached during the loss of normal feedwater coincident with loss of ac power event does not approach 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 were made to the approved WGOTHIC containment response model to increase condensation and produce conservative results. The Containment Analysis performed with the WGOTHIC model is used to determine the transient mass of condensate on passive heat sinks, steam in the containment atmosphere, and steam lost to containment leakage. In the Condensate Return calculation, the WGOTHIC results are combined with calculation of losses from the containment shell surfaces to determine a time-dependent fraction of condensate return. The variable condensate return fraction is used to develop detailed PRHR HX performance parameters (PRHR Performance case CD7), which are then incorporated into the modified LOFTRAN computer code (described in UFSAR Subsection 15.0.11.2) to produce the Shutdown Temperature Evaluation summarized in Appendix 19E. 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 RCS temperature to 420°F in less than 36 hours. Therefore, the plant continues to meet its safety-related design criterion and the provisions 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 Polar Crane Girder and internal stiffener to augment the IRWST gutter arrangement to capture condensate at the PCG and stiffener locations. Those changes affect material in the UFSAR (Tier 2) and involve plant-specific Tier 1 information and the corresponding COL Appendix C Table 2.2.3-1 and Table 2.2.3-2. As the updated Shutdown Temperature Evaluation demonstrates, 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 safety-related design criteria. 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 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 thus requires NRC approval for the Tier 1 and associated Tier 2 departures.

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

10 CFR 50, Appendix A, General Design Criterion 2, “Design bases for protection against natural phenomena,” requires 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 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 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 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 meets the requirements of General Design Criterion 17; and continues to support an exemption to the requirement of having two offsite power sources.

10 CFR 50, Appendix A, General Design Criterion 27, "Combined reactivity control systems capability," requires 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.

10 CFR 50, Appendix A, General Design Criterion 34, "Residual heat removal" requires 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 provide adequate PRHR HX performance. With the proposed changes, the updated safe shutdown analysis demonstrates the plant complies with its functional requirement of cooling the RCS to 420ºF within 36 hours.

10 CFR 50, Appendix A, General Design Criterion 35, "Emergency core cooling," requires 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 provide adequate PRHR HX 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.

10 CFR 50, Appendix A, General Design Criterion 36, "Inspection of emergency core cooling system," requires 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.

10 CFR 50, Appendix A, General Design Criterion 37, "Testing of emergency core cooling system," requires 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 capability.

10 CFR 52.98(f) 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; 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.

The Technical Specification (TS) Bases Control Program (incorporated into the COL Appendix A Technical Specifications as Technical Specification 5.5.6) requires prior NRC approval for any proposed change to the TS Bases that involves a change in the TS incorporated in the license; or a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59. While 50.59 does not explicitly address changes that would be in conflict with other portions of the operating license other than the Appendix A Technical Specifications, in practice, many licensees have understood that changes that conflict with any portion of the operating license (such as license conditions) should also receive prior NRC approval. These changes involve a change to the terms and conditions of a COL, in particular, the COL Appendix C, Inspections, Tests, Analyses and Acceptance Criteria information; therefore, these Bases changes are provided in this proposed amendment for NRC approval.

### 4.2 Precedent

No precedent is identified.

### 4.3 Significant Hazards Consideration Determination

The proposed changes would revise the Combined Licenses (COLs) in regard to modifications to the addition of downspouts to capture condensate at the polar crane girder (PCG) and internal stiffener locations.

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.

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, thus preventing PRHR HX performance degradation from that considered in the 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 do not alter design features available during anticipated operational occurrences or accidents. The proposed changes do not involve any accident initiating component/system failure or 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 a radioactive material barrier, and this activity does not involve the containment of radioactive material. Thus, the proposed changes do not affect any safety-related accident mitigating function. 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 meets its acceptance criterion, i.e., to cool the Reactor Coolant System (RCS) to below 420ºF in 36 hours. The affected 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 functional 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.

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

#### 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, diversity or performance of any safety-related function. 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 show the PRHR HX continues to meets its acceptance criterion. The UFSAR Chapters 6 and 15 analyses results are not affected, thus margins to their regulatory acceptance criteria are unchanged. The added components are classified as safety-related, seismically qualified, and comply with their applicable design codes. 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 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.

### 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 licensing 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 stiffener locations.

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

This review has determined the proposed change would require an amendment from the COL; however, a review of the anticipated construction and operational effects of the proposed amendment has determined 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 Determination, 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 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 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 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), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

**Southern Nuclear Operating Company**

**Vogtle Electric Generating Plant Units 3 and 4**

**ND-14-0588**

**Enclosure 2**

**Exemption Request**

**Regarding Passive Core Cooling System Condensate Return Changes**

**(LAR-13-024)**

## 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 will apply 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, test, 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 will be 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 will be 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) includes 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 (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

## 3.0 Technical Justification of Acceptability

General design criteria 34 and 35 require the PXS to 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 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 would allow 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 will continue to reflect the approved licensing basis for the Licensee, and will maintain 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 would present any additional risk to the health and safety is present. The affected Design Description in the plant-specific Tier 1 DCD will also continue 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 would 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 would change 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 will enable the applicant to safely construct and operate the AP1 000 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 will result 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**

No precedent for this request is identified.

**7.0 Environmental Consideration**

A review has determined that the proposed exemption would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change 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, and 10 CFR 52 Appendix D, *"Design Certification Rule for the AP1000."* 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-14-0588**

**Enclosure 3**

**Proposed Changes to the Updated Final Safety Analysis Report**

**(LAR-13-024)**

**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 |
| 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 | - | - / - | - | - / - | - | - |

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

| Table 2.2.3-2 (cont.) | | | | |
|---|---|---|---|---|
| **Line Name** | **Line Number** | **ASME Code Section III** | **Leak Before Break** | **Functional Capability Required** |
| 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 |

**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 | |

**The UFSAR Subsection 6.3.1.1.1, Emergency Core Decay Heat Removal, second bullet of the first paragraph, is revised to include the downspouts as shown below.**

- The passive residual heat removal heat exchanger is capable of 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 UFSAR Subsection 6.3.2.1.1, Emergency Core Decay Heat Removal at High Pressure and Temperature Conditions, eighth paragraph, is revised to update the description of the condensate return gutter arrangement as shown below.**

Condensation occurs on the steel containment vessel, which is cooled by the passive containment cooling system. The condensate is collected in a safety-related gutter arrangement. A gutter is located at near 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 inside the containment during passive containment cooling system operation and return it to the in-containment refueling water storage tank…

**The UFSAR Subsection 6.3.2.2.7, IRWST and Containment Recirculation Screens, first paragraph, is revised to be clear that the discussion is specific to the IRWST and containment recirculation screens as shown below.**

The passive core cooling systems 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. These 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:

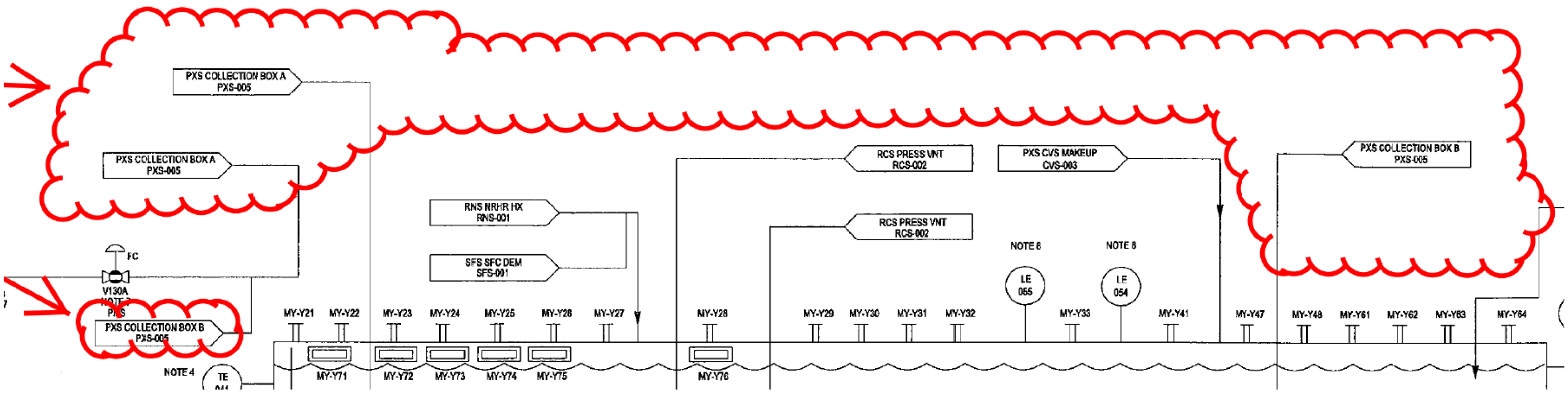
**The UFSAR Subsection 6.3.2.2.7.2, IRWST Screens, is revised to update the description of the condensate return gutter arrangement as shown below.**

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 the 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…

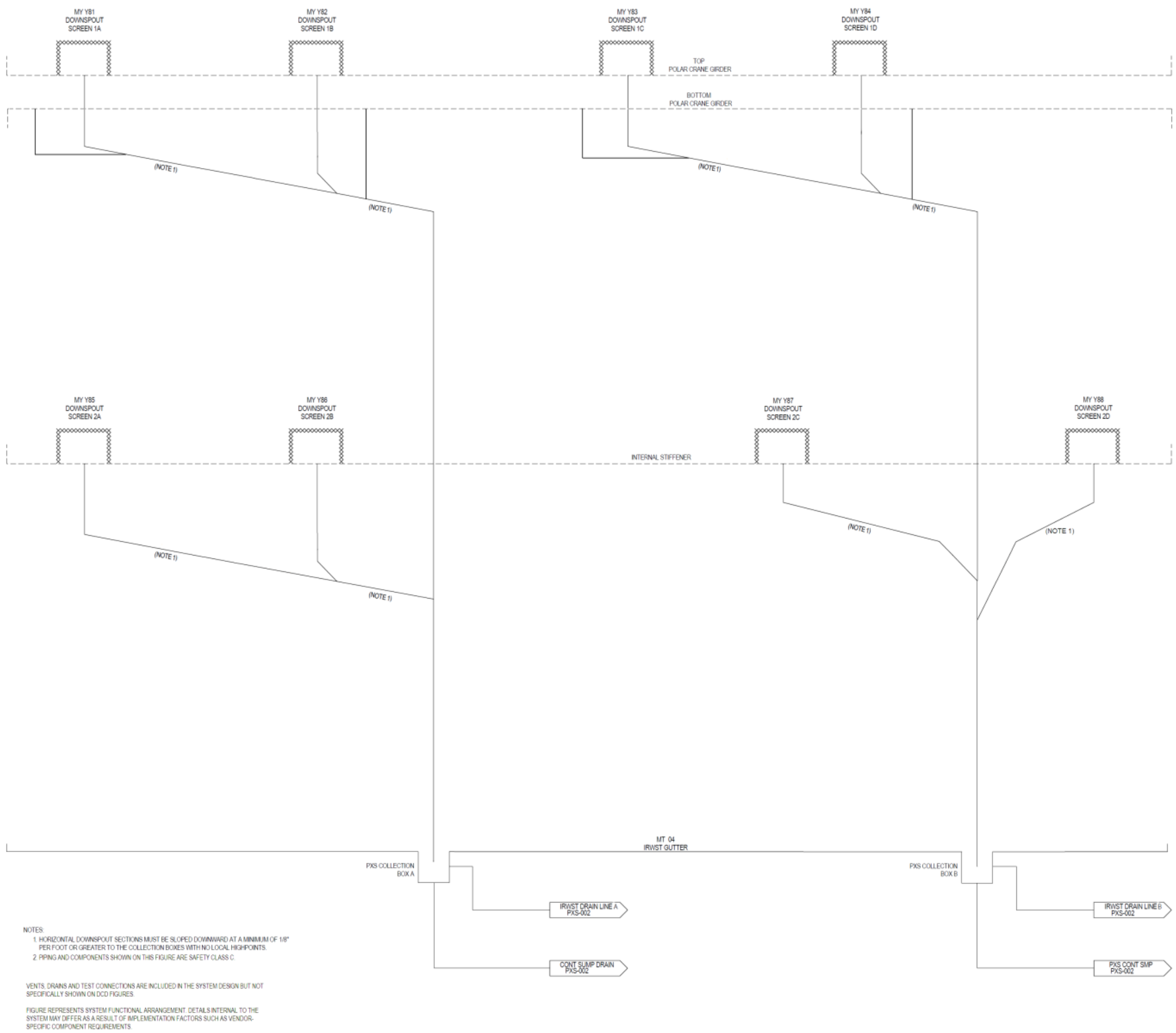
**The UFSAR Section 6.3, Figure 6.3-1 Sheet 1 is revised to show that it is (Sheet 1 of 3) rather than (Sheet 1 of 2).**

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

**The UFSAR Section 6.3, Figure 6.3-1 (Sheet 2 of 3) [that was Figure 6.3-2] is revised 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.**



**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 figure 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 ARRANGEMETN 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 renumbering of Figure 6.3-2 to become Figure 6.3-1 (Sheet 2 of 3) requires revision of several existing UFSAR references to Figure 6.3-2. These are addressed together here.

**The UFSAR Subsection 5.4.11.2, System Description, second paragraph, reference to Figure 6.3-2 is revised to reference Figure 6.3-1 as shown below.**

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~~1.

**The UFSAR Subsection 6.3.2.1, Schematic Piping and Instrumentation Diagram, first sentence, is revised to remove the reference to Figure 6.3-2 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 14.3, Table 14.2-2 Sheets 7 and 8 references to Figure 6.3-2 are revised to reference Figure 6.3-1 as shown below.**

**Table 14.3-2 (Sheet # 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 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 analysis to demonstrate that the passive systems can bring the plant to a stable safe 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 Subsection 7.4.1.1, the PRHR HX operates to reduce the RCS temperature to the safe shutdown condition following an event. An analysis of the loss of ac power event demonstrates that the passive systems can bring the plant to a stable safe condition following postulated transients. ~~The results of this~~ A bounding 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.

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.

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 efficiency of the gutter collection system was determined separate from the WGOTHIC analysis. The resulting time-dependent condensate return rate was incorporated into the LOFTRAN computer code described in Subsection 15.0.11.2 to demonstrate that the RCS could be cooled to 420°F within 36 hours.

Summarizing this transient, the loss of normal ac power occurs, followed by the reactor trip. The PRHR 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.

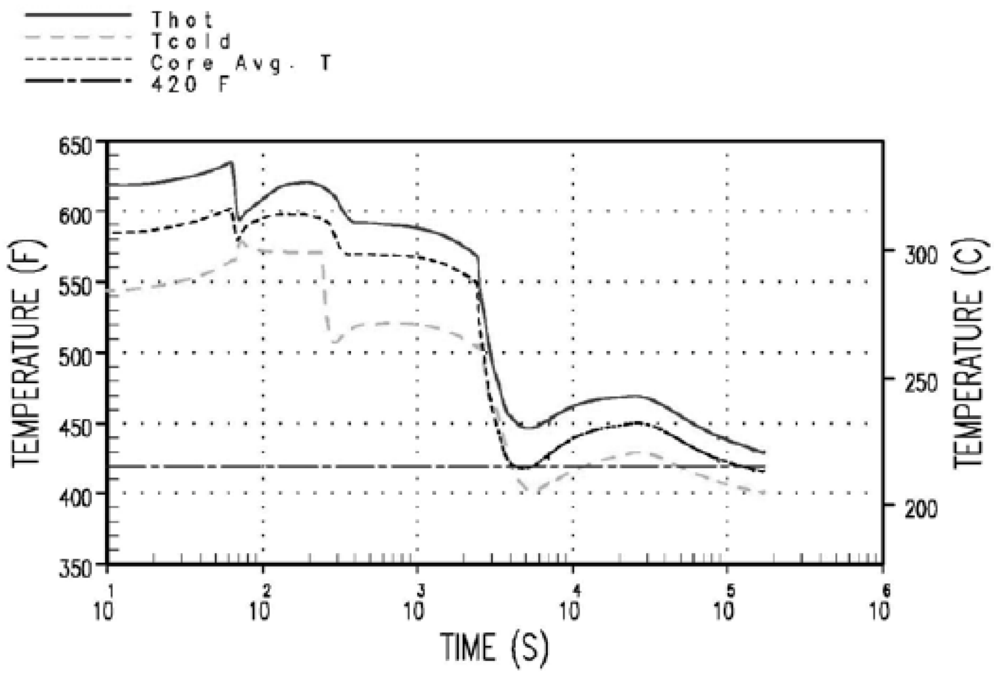
Once actuated, at about 600 2,400 seconds, the CMTs operate in recirculation mode, injecting cold borated water into the RCS. In the first part of their operation, due to the cold flow rate, the CMTs operate in conjunction with the PRHR to reduce RCS temperature. Due to the primary system cooldown, the 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 5,000 seconds, the CMT cooling effect decreases and the RCS starts heating up again (Figure 19E.4.10-1). The RCS temperature increases until the PRHR HX can match decay heat. At about 31,000 34,500 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 within 48,600 at 82,600 seconds, while the core average temperature reaches 420°F within 124,400 123,600 seconds (approximately 34.6 hours).

As discussed in Subsection 7.4.1.1, a timer is used to automatically actuate the automatic depressurization system 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. At approximately 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. 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. Operation of the ADS in conjunction with the CMTs, accumulators, and IRWST reduces the RCS pressure and temperature to below 420°F.

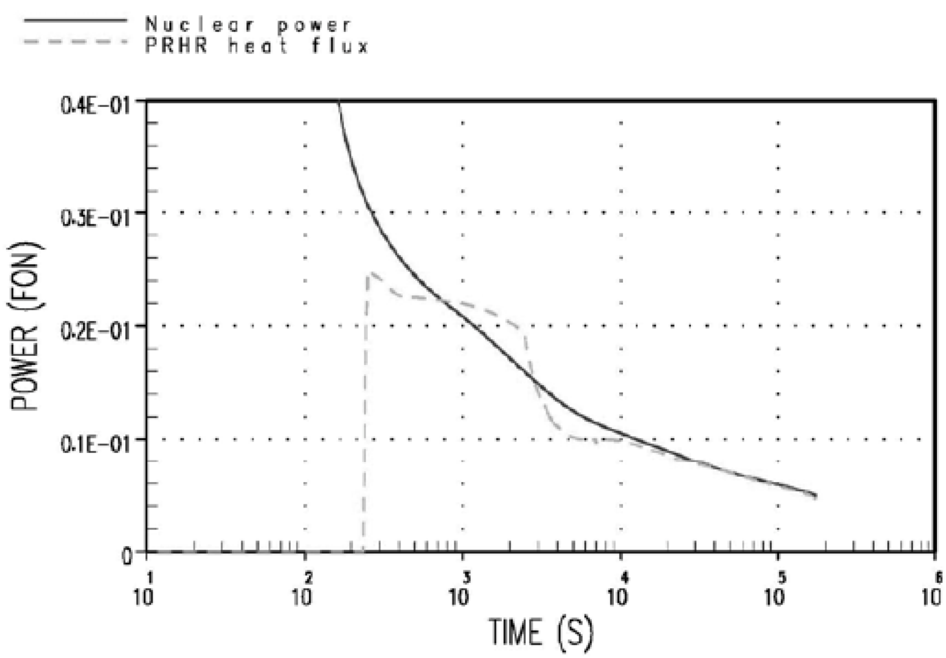
**The UFSAR Appendix 19E, Table 19E.4.10-1, Sequence of Events Following a Loss of ac Power Flow with Condensate from the Containment Shell Being Returned to the IRWST, is revised to reflect the results of the design changes as shown below.**

| Event | Time (seconds) |
|---|---|
| Feedwater is Lost | 10.0 |
| Low Steam Generator Water Level (Narrow-Range) Reactor Trip Setpoint Reached | ≤ 60 ~~72.4~~ |
| Rods Begin to Drop | ≤ 61 ~~74.4~~ |
| Low Steam Generator Water Level (Wide-Range) Reached | ≤ 230 |
| PRHR HX Actuation on Low Steam Generator Water Level (~~Wide~~Narrow-Range Coincident with Low Startup Feedwater Flow) | ≤ 240 ~~129.4~~ |
| Low $T_{cold}$ Setpoint Reached | ≤ 2,400 ~~599.0~~ |
| Steam Line Isolation on Low $T_{cold}$ Signal | ≤ 2,400 ~~611.0~~ |
| CMTs Actuated on Low $T_{cold}$ Signal | ≤ 2,400 ~~617.0~~ |
| IRWST Reaches Saturation Temperature | ≤ 15,500 ~~17,600~~ |
| Heat Extracted by PRHR HX Matches Core Decay Heat | ≤ 34,500 ~~31,000~~ |
| CMTs Stop Recirculating | -- ~~43,500~~ |
| Cold Leg Temperature Reaches 420°F (loop with PRHR) | ≤ 48,600 ~~82,600~~ |
| ~~Hot Leg~~ Core Average Temperature Reaches 420°F (loop with PRHR) | ≤ 124,400 ~~123,600~~ |

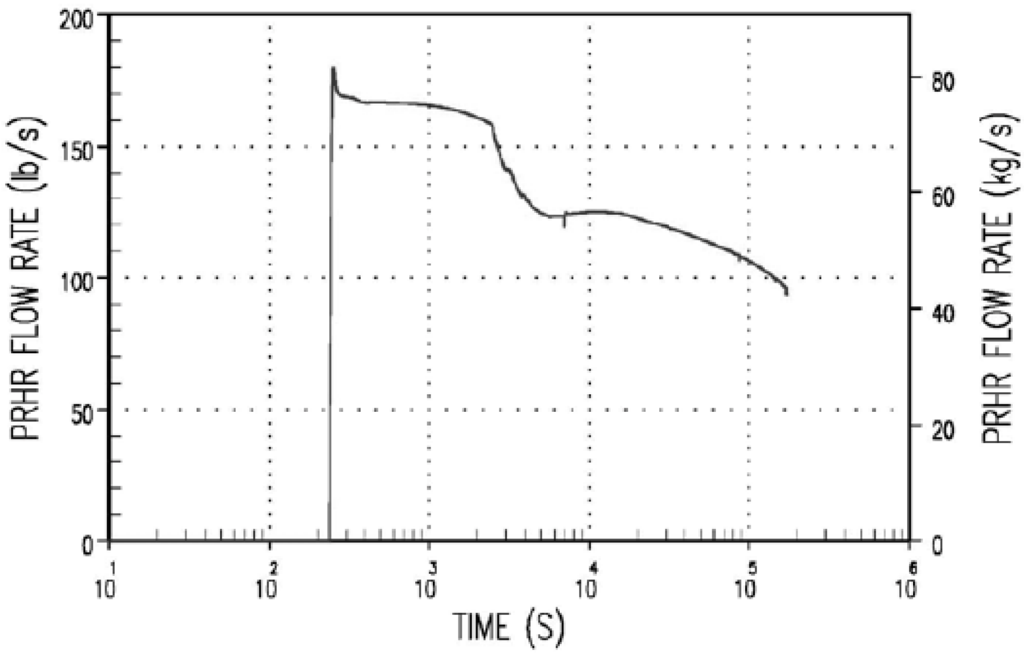
**The UFSAR Appendix 19E, Figure 19E.4.10-1, Shutdown Temperature Evaluation, RCS Temperature, is revised to reflect the results of the design changes as shown below.**



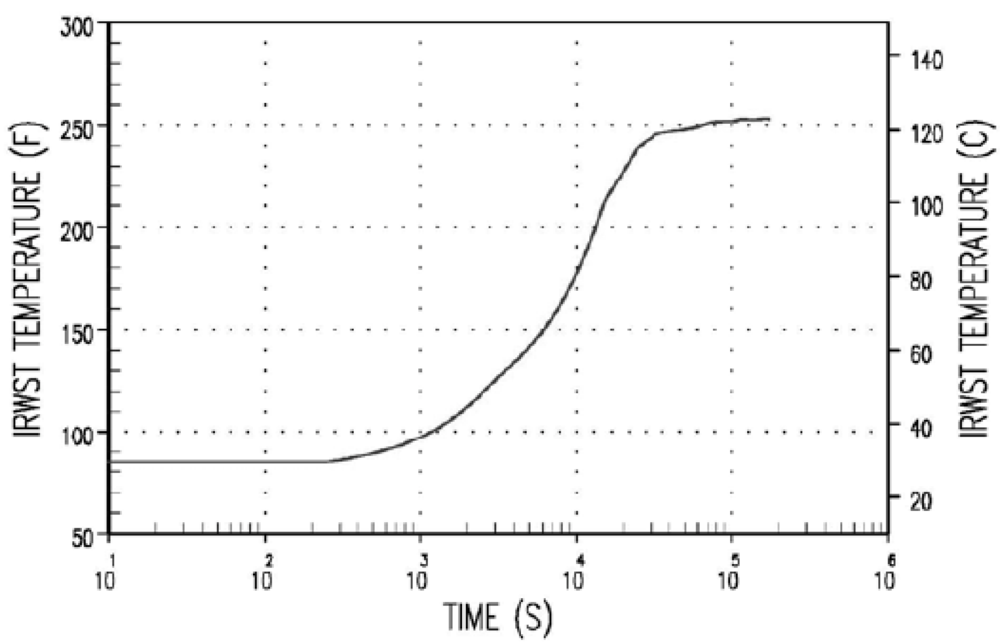

**The UFSAR Appendix 19E, Figure 19E.4.10-2, Shutdown Temperature Evaluation, PRHR Heat Transfer, is revised to reflect the results of the design changes as shown below.**



**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.**



**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, to reflect the design changes 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 to reflect the design changes 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 to reflect the design changes as shown below.**

This surveillance requires visual inspection of the IRWST gutters 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 gutters or downspout screens could become restricted.