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10CFR50.12
10CFR52.63

September 29, 2014

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: Duke Energy Carolinas, LLC
Docket Nos. 52-018 and 52-019
AP1000 Combined License Application for the
William States Lee III Nuclear Station Units 1 and 2
Voluntary Submittal of Exemption Request and Design Change Description for Departure from AP1000 DCD Revision 19 to Address Containment Condensate Return Cooling Design
Ltr# WLG2014.09-01

References:

1. Letter from Christopher Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated April 18, 2013, "Submittal of Exemption Request and Design Change Description for Departure from AP1000 DCD Revision 19 to Address Containment Condensate Return Cooling Design", Serial: NPD-NRC-2013-010 [ML13109A533]

2. Letter from Christopher Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated June 3, 2013, "Supplement to Submittal of Exemption Request and Design Change Description for Departure from AP1000 DCD Revision 19 to Address Containment Condensate Return Cooling Design", Serial: NPD-NRC-2013-023 [ML13156A007]

3. Letter from Christopher Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated October 21, 2013, "Supplement 2 to Submittal of Exemption Request and Design Change Description for Departure from AP1000 DCD Revision 19 to Address Containment Condensate Return Cooling Design", Serial: NPD-NRC-2013-044 [ML13296A034]

4. Letter from Christopher Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated February 7, 2014, "Supplement 3 to Submittal of Exemption Request and Design Change Description for Departure from AP1000 DCD

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Revision 19 to Address Containment Condensate Return Cooling Design", Serial: NPD-NRC-2014-005 [ML14042A035-Non-Proprietary Portion]

5. Letter from Christopher Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated July 10, 2014, "Supplement 4 to Submittal of Exemption Request and Design Change Description for Departure from AP1000 DCD Revision 19 to Address Containment Condensate Return Cooling Design", Serial: NPD-NRC-2014-023 [ML14196A074]

Duke Energy Carolinas, LLC (DEC) hereby submits a request for exemption and associated design change description to address a departure from the AP1000 Design Control Document (DCD) Revision 19. The departure involves a design change that requires Nuclear Regulatory Commission (NRC) notification and review in accordance with Interim Staff Guidance DC/COL-ISG-011, "Finalizing Licensing-basis Information." The William States Lee III Nuclear Station Units 1 and 2 (WLS) Combined License Application (COLA) incorporates the AP1000 DCD by reference.

The design change modifies the condensate return portion of the Passive Core Cooling System (PXS) on the interior of the containment vessel to improve containment condensation returned to the In-Containment Refueling Water Storage Tank (IRWST). The change involves the addition of components to tables contained in Tier 1 of the DCD and associated changes to Tier 2 text, tables and figures.

By letters dated April 18, 2013 (Reference 1), February 7, 2014 (Reference 4), and July 10, 2014 (Reference 5), Duke Energy Florida (DEF) provided voluntary submittals for the Levy Nuclear Plant, Units 1 and 2, COLA. Pursuant to the provisions of 10 CFR 52.63(b)(1), DEF also requested an exemption from elements of the design as certified in the 10 CFR Part 52, Appendix D, design certification rule for the plant-specific DCD Tier 1 material departures.

The proposed changes in this submittal would revise the WLS plant-specific Tier 1 and associated Tier 2 material to increase the efficiency of the return of condensate to the IRWST that is utilized by the PXS to support the capability for long term cooling. This exemption and departure request is supported by the technical documents provided with the exemption and departure request submitted for the Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030 on February 7, 2014 (Reference 4) and by the technical content of Supplement 4 submittal (Reference 5), as well as the Levy RAI responses described in Enclosure 8.

Enclosures 1, 2, 3, and 4 identify specific information provided by Reference 4 supporting the WLS specific request for exemption.

A WLS specific request for exemption is contained in Enclosure 5, which includes a summary and detailed description of the change to the design of the condensate return portion of the PXS and associated changes to the licensing basis, the technical evaluation of the change, the regulatory evaluation (including the exemption, justification and the Significant Hazards Consideration determination) of the change, and the risk assessment.

Enclosure 6 provides the complete listing of the references for the proposed mark-ups depicting the requested changes to the AP1000 Tier 1 information tables and mark-ups of the associated departures from the Tier 2 information contained in the AP1000 DCD Revision 19.

Enclosure 7 contains revisions to the WLS COLA Part 2 (FSAR), Part 4 (Technical Specifications), Part 7 (Departures and Exemptions Requests), and Part 10 (Proposed License Conditions (Including ITAAC)) required to implement the requested changes in the WLS COLA. The revisions will be incorporated into a future revision of the WLS COLA. The proposed WLS COLA changes are identical in context to those approved by NRC for the Levy COLA except for site specific identification.

Following review of Levy Nuclear Plant submittals (References 1 and 4), the NRC staff issued several requests for additional information (RAIs) on the Levy Nuclear Plant, Units 1 and 2, dockets. For convenience, Enclosure 8 provides a complete listing of the NRC requests for additional information, along with the corresponding Levy Nuclear Plant response submittals. These RAIs and the Levy docketed responses have been reviewed and DEC has determined that the response information is also pertinent to the WLS COLA. Any updates to the COLA revisions resulting from the RAI responses have been included in the proposed COLA revisions in Enclosure 7.

If you have any further questions, or need additional information, please contact Bob Kitchen at (704) 382-4046, or me at (704) 382-9248.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on September 29, 2014.

Sincerely,

Christopher M Fallon

Christopher M. Fallon
Vice President
Nuclear Development

Enclosures:

1. Westinghouse APP-GW-GLR-161, Revision 1 (PROPRIETARY)
2. Westinghouse APP-GW-GLR-607, Revision 1 (NON-PROPRIETARY VERSION)
3. Westinghouse Application Letter CAW-14-3877 and Affidavit
4. Proprietary Information Notice and Copyright Notice
5. Request for Exemption Regarding Containment Condensate Return Cooling
6. Tier 1 and Tier 2 Licensing Basis Documents - Proposed Changes
7. Revisions to the William States Lee III COL Application
8. Levy Docketed Requests for Additional Information Responses Regarding Passive Core Cooling System (PXS) Condensate Return

xc (w/out enclosures):
Frederick Brown, Deputy Regional Administrator, Region II

xc (w/ enclosures):
Brian Hughes, Senior Project Manager, DNRL

Duke Energy Voluntary Submittal

Condensate Return

William States Lee III COL Application

Enclosure 1

Westinghouse APP-GW-GLR-161, Revision 1

The following Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, letter has been reviewed and found to be applicable to William States Lee III Nuclear Station Units 1 and 2 for the Duke Energy Carolinas voluntary submittal of exemption request and design change description for departure from AP1000 DCD Revision 19 to address Containment Condensate Return Cooling design.

1. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-005, dated February 7, 2014 (ADAMS Accession No. ML14042A035), Enclosure 1 is found to be applicable to William States Lee III Nuclear Station Units 1 and 2 (contains proprietary information).

Duke Energy Voluntary Submittal

Condensate Return

William States Lee III COL Application

Enclosure 2

Westinghouse APP-GW-GLR-607, Revision 1

(NON-PROPRIETARY VERSION)

The following Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, letter has been reviewed and found to be applicable to William States Lee III Nuclear Station Units 1 and 2 for the Duke Energy Carolinas voluntary submittal of exemption request and design change description for departure from AP1000 DCD Revision 19 to address Containment Condensate Return Cooling design.

1. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-005, dated February 7, 2014 (ADAMS Accession No. ML14042A035), Enclosure 2 is found to be applicable to William States Lee III Nuclear Station Units 1 and 2.

Duke Energy Voluntary Submittal

Condensate Return

William States Lee III COL Application

Enclosure 3

Westinghouse Application Letter CAW-14-3877 and Affidavit

The following Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, letter has been reviewed and found to be applicable to William States Lee III Nuclear Station Units 1 and 2 Duke Energy Carolinas voluntary submittal of exemption request and design change description for departure from AP1000 DCD Revision 19 to address Containment Condensate Return Cooling design.

1. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-005, dated February 7, 2014 (ADAMS Accession No. ML14042A035), Enclosure 3 is found to be applicable to William States Lee III Nuclear Station Units 1 and 2.

Duke Energy Voluntary Submittal

Condensate Return

William States Lee III COL Application

Enclosure 4

Proprietary Information Notice and Copyright Notice

The following Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, letter has been reviewed and found to be applicable to WLS Nuclear Station Units 1 and 2 for the Duke Energy Carolinas voluntary submittal of exemption request and design change description for departure from AP1000 DCD Revision 19 to address Containment Condensate Return Cooling design.

1. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-005, dated February 7, 2014 (ADAMS Accession No. ML14042A035), Enclosure 4 is found to be applicable to William States Lee III Nuclear Station Units 1 and 2.

Duke Energy Voluntary Submittal

Condensate Return

William States Lee III COL Application

Enclosure 5

Request for Exemption Regarding

Containment Condensate Return Cooling

1.0 Summary Description

The Passive Residual Heat Removal Heat Exchanger (PRHR HX) is safety-related and provides emergency core decay heat removal. It is located in the In-containment Refueling Water Storage Tank (IRWST) as shown on Tier 2 DCD Figure 6.3-2. The heat exchanger is used in non-loss of coolant accident (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 Reactor Coolant System to the IRWST causes the water in the tank to heat up, eventually become saturated, and initiate steaming of the tank.

The steam generated will discharge 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 ensures the steam generator wall vents will open first. The location of the steam generator wall vents (near the center of containment) contributes to mixing of the containment atmosphere. The steam released from the IRWST condenses on "passive heat sinks" within the containment, such as the containment vessel wall, Polar Crane Girder (PCG), concrete, piping, components, or any other subcooled surface until these passive heat sinks reach saturation temperature. Condensation on the inside of the containment vessel wall forms a thin fluid film and runs down the containment wall surface. Provisions are made to collect and channel condensate to the IRWST.

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

The PCG is a box girder consisting of 80 enclosed boxes; and is shown in Tier 2 DCD 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 collected 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 will begin to degrade the heat exchanger performance.

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, containment ring misalignment
4) Losses at the Polar Crane Girder and Stiffener
5) Losses at support plates attached to the containment vessel
6) Losses at the Equipment Hatch and Personnel Airlock
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. A full scale section of the containment wall was constructed to test condensate losses.

As a result of the condensate return testing, modifications to the Polar Crane Girder, internal stiffener, and IRWST gutter design were made. The fabrication holes at the top surface of the PCG and in the internal hoop stiffener are blocked, drainage holes in the bottom of the PCG boxes are blocked, and flow communication holes between PCG boxes are added. A downspout piping network was added to collect and transport condensation from the top and interior of the PCG and the internal hoop stiffener to the Passive Core Cooling System (PXS) Collection Boxes. Eight new PXS downspout screens were added at the entrance of each of the downspouts at the top of the PCG and the stiffener to prevent any larger debris from blocking the downspout piping. Extensions of the IRWST gutter were added above the Upper Personnel Airlock and Upper Equipment Hatch.

2.0 Description of Licensing Basis Impacts

Tier 1 Changes

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. To provide assurance that ITAAC design commitments will be met, the component numbers for the following downspout screens are added to Table 2.2.3-1:

PXS-MY-Y81	PXS-MY-Y83	PXS-MY-Y85	PXS-MY-Y87
PXS-MY-Y82	PXS-MY-Y84	PXS-MY-Y86	PXS-MY-Y88

To provide assurance that ITAAC design commitments will be met, the additional downspout piping is added to the PXS recirculation system as captured in Table 2.2.3-2:

PXS-L301A	PXS-L306A	PXS-L301B	PXS-L306B
PXS-L302A	PXS-L307A	PXS-L302B	PXS-L307B
PXS-L303A	PXS-L308A	PXS-L303B	PXS-L308B
PXS-L304A	PXS-L309A	PXS-L304B	PXS-L309B
PXS-L305A	PXS-L310A	PXS-L305B	PXS-L310B

Tier 2 Changes

The new PXS downspout screens are AP1000 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 one-third of the safe shutdown earthquake ground motion before resuming operations in accordance with 10 CFR Part 50, Appendix S. Component numbers for the following downspout screens are added to Table 3.2-3, AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment, to capture these requirements.

PXS-MY-Y81	PXS-MY-Y83	PXS-MY-Y85	PXS-MY-Y87
PXS-MY-Y82	PXS-MY-Y84	PXS-MY-Y86	PXS-MY-Y88

Pictorial detail of the Polar Crane Girder is shown in DCD Figure 3.8.2-1 (Sheet 3 of 3), Containment Vessel General Outline, and shows the fabrication holes in the top right figure. As the fabrication holes in the PCG would be blocked in the modified configuration, this detail would be removed from this figure.

To reflect the changes to the PXS system, the additional downspout piping is captured in the gutter discussion of DCD Subsection 6.3.2.1.1 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, Passive Core Cooling System Piping and Instrumentation Diagram will be expanded to include continuation flags for condensate returning to the IRWST originating from PXS Collection Boxes A and B in the IRWST Gutter (Sheet 2) and show the relocated IRWST Gutter and the screens and piping comprising the PXS downspouts originating from the Polar Crane Girder and internal stiffener (Sheet 3). Subsection 6.3.1.1.1 will be updated to describe the downspouts in the safety-related design criteria, subsections 6.3.2.2.7 and 6.3.2.2.7.1 will be updated to clarify the number of screen sets in the PXS and to which set of screens the criteria in this section apply, and subsection 6.3.2.2.7.2 will be updated to clarify the condensate return gutter arrangement related to LOCA operation.

The Technical Specification Bases in DCD Chapter 16 will be updated to include the downspouts in the descriptions of the IRWST gutter arrangement. The Bases LCO for B 3.3.3 will be updated to reflect the addition of downspouts, the Bases Surveillance Requirement for SR 3.5.4.7 will be updated to encompass the entire gutter arrangement, including the downspout screens, in the surveillance, and the Bases Background for B 3.5.4 will be updated to reflect the addition of downspouts.

The safe shutdown temperature evaluation was revised to address the effects of the design modifications and supporting analyses and calculations of condensate return to the IRWST on PXS performance. The resultant changes to the Chapter 19 shutdown temperature evaluation are shown in text revisions to subsection 19E.4.10.2, changes to Table 19E.4.10-1, and changes to Figures 19E.4.10-1 through 19E.4.10-4.

All proposed changes are shown in Enclosure 6.

3.0 Technical Evaluation

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 2.2.3.8.b. The changes described herein do not change the commitment to complete the performance test of the PRHR HX. This evaluation is based on information provided in Westinghouse report APP-GW-GLR-161, Revision 1, which is included as Enclosure 1 to this submittal.

4.0 Regulatory Evaluation

4.1 Exemption Justification

4.1.1 Pursuant to 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 requested for plant-specific Tier 1 material departures from the AP1000 DCD for Tier 1 information. These material departures are contained in Tier 1 Subsection 2.2.3, Tables 2.2.3-1 and 2.2.3-2, and involve the addition of components to the condensate return design to enable the Passive Core Cooling System to more effectively perform its design functions. This exemption request is in accordance with the provisions of 10 CFR §50.12, 10 CFR §52.7, and 10 CFR Part 52, Appendix D, as demonstrated below.

Applicable Regulation(s): 10 CFR Part 52, Appendix D, Section III.B
Specific wording from which exemption is requested:

"III. Scope and Contents

B. An applicant or licensee referencing this appendix, in accordance with Section IV of this appendix, shall incorporate by reference and comply with the requirements of this appendix, including Tier 1, Tier 2 (including the investment protection short-term availability controls in Section 16.3 of the DCD), and the generic TS except as otherwise provided in this appendix. Conceptual design information in the generic DCD and the evaluation of severe accident mitigation design alternatives in appendix 1B of the generic DCD are not part of this appendix."

4.1.2 DEC evaluated this exemption request in accordance with 10 CFR Part 52, Appendix D, Section VIII.A.4, 10 CFR §50.12, 10 CFR §52.7 and 10 CFR §52.63, which state that the NRC may grant exemptions from the requirements of the regulations provided the following 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, Appendix 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 with proper justification. 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 applicant, 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 will ensure the protection of the health and safety of the public. Therefore, no adverse safety impact which would present any additional risk to the health and safety is present. The affected Design Description in the plant-specific Tier 1 DCD will 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 relating to the condensate return portion of the passive core cooling system. The exemption does not alter the design, function, or operation of any structures or plant equipment that are necessary to maintain a 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) lists 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 from Tier 1 Subsection 2.2.3, Tables 2.2.3-1 and 2.2.3-2, is 10 CFR 52, Appendix D, Section III.B, which requires that an applicant 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 William States Lee III Units 1 and 2 COLA references the AP1000 Design Certification Rule and incorporates 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 to maintain the level of safety in the design.

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 in Tier 1 Subsection 2.2.3, Tables 2.2.3-1 and 2.2.3-2, will enable the applicant to safely construct and operate the AP1000 facility consistent with the design certified by the NRC in 10 CFR 52, Appendix D.

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

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

Based on the nature of the changes to the plant-specific Tier 1 information and the understanding that these changes support the design function of the Passive Core Cooling System, it is likely that other AP1000 applicants and licensees will request this exemption. However, if this is not the case, the special circumstances continue to outweigh any decrease in safety from the reduction in standardization because the key design functions of the Passive Core Cooling System associated with this request will continue to be maintained. This exemption request and the associated marked-up tables demonstrate that the Passive Core Cooling System function continues to be maintained following implementation of the change from the generic AP1000 DCD, thereby minimizing the safety impact resulting from any reduction in standardization.

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 Condition 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 adding components to Subsection 2.2.3, Tables 2.2.3-1 and 2.2.3-2, which were added to the condensate return design to enable the Passive Core Cooling System to more effectively perform its design functions. Because these functions are met, there is no reduction in the level of safety.

Therefore, the design change will not result in a significant decrease in the level of safety.

As demonstrated above, this exemption request satisfies NRC requirements for an exemption to the design certification rule for the AP1000.

4.2 Significant Hazards Consideration

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

No accident previously evaluated in the plant-specific DCD is attributed to the failure of the condensate return features of the design. The proposed changes add passive components that do not rely on instrumentation and control systems to move them to a safe position. The proposed changes also meet applicable NRC general design criteria requirements. As the proposed changes do not involve any components that could initiate an event by means of component or system failure, the changes do not increase the probability of a previously evaluated accident.

The added components are constructed of only those materials appropriately suited for exposure to the post-accident environment as described in DCD Subsection 6.1.1.4 of the plant-specific DCD. No aluminum is permitted to be used in the construction of these components to ensure they will not contribute to hydrogen production in containment. The changes do not alter design features available during normal operation or anticipated operational occurrences. Nonsafety-related features used for reactor coolant activity monitoring, or reactor coolant chemistry control remain unaffected. The changes do not adversely impact accident source term parameters or affect any release paths used in the safety analyses, which could increase radiological dose consequences. Thus the radiological releases associated with the Chapter 15 accident analyses are not affected.

As previously described, the proposed changes would not adversely affect the ability of the PRHR HX to meet the design requirements of GDCs 34 and 35. The proposed equipment does not adversely interact with or affect safety-related equipment or a radioactive material barrier. The components added by this change would not increase the consequences of an accident previously evaluated in the plant-specific DCD. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

An evaluation of the downspout and gutter return subsystem determined the components are capable of acceptably performing their safety-related function, even if one of the downspouts were blocked. The new equipment does not interface with components in other systems that provide safety-related or defense-in-depth support to the plant, thus precluding the possibility condensate could be diverted to another system before reaching the gutter. The affected equipment does not 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. Consequently, the proposed component changes do not introduce new failure modes, interactions or dependencies, the malfunction of which could lead to new accident scenarios. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

4.2.3 Does the proposed change involve a significant reduction in a margin of safety?

The proposed changes do not involve a significant reduction in the margin of safety. The proposed changes do not reduce the redundancy or diversity of any safety-related functions. The proposed changes increase the amount of condensate available in the IRWST for heat transfer after shutdown following a non-LOCA event with long-term loss of AC power. Though the fraction of condensate returned is smaller than originally assumed, the proposed changes provide sufficient condensate return flow to maintain adequate IRWST water level for those events using the PRHR HX cooling function. While lower condensate return rates result in an earlier transition to PRHR HX uncovery, the long-term shutdown temperature evaluation results show that the PRHR HX would continue to meet its acceptance criteria.

The DCD Chapters 6 and 15 analyses results are not affected, thus margins to the regulatory acceptance criteria are unchanged. No design basis safety analysis or acceptance criterion is challenged or exceeded by the proposed changes. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

4.3 Applicable Regulatory Requirements/Criteria

10 CFR 52, Appendix D, Section VIII.B.5.a requires that an applicant or licensee who references this appendix may 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 that section. When evaluating the proposed departure, an applicant or licensee shall consider all matters described in the plant-specific DCD. This exemption request involves a departure from Tier 1 Subsection 2.2.3, Tables 2.2.3-1 and 2.2.3-2, with Tier 2 involved departures.

4.4 Precedent

No precedent is cited.

4.5 Conclusions

Based on the considerations discussed above:

(1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and

(2) such activities will be conducted in compliance with the Commission's regulations, and

(3) the issuance of the exemption will not be inimical to the common defense and security or to the health and safety of the public.

The above evaluations demonstrate the requested changes can be accommodated without an increase in the probability or consequences of an accident previously evaluated, without creating the possibility of a new or different kind of accident from any accident previously evaluated, and without a significant reduction in a margin of safety. Having arrived at negative declarations with regard to the criteria of 10 CFR 50.92, this assessment determines the requested change does not involve a Significant Hazards Consideration.

5.0 Risk Assessment

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

6.0 References

1) Westinghouse Electric Company, AP1000 Design Control Document, Revision 19, June 2011

2) Westinghouse Electric Company, Topical Report, APP-GW-GLR-161, Revision 1, Changes to Passive Core Cooling System Condensate Return

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Enclosure 6

Tier 1 and Tier 2 Licensing Basis Documents – Proposed Changes

The following Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, letters have been reviewed and found to be applicable to William States Lee III (WLS) Nuclear Station Units 1 and 2 for the Duke Energy Carolinas voluntary submittal of exemption request and design change description for departure from AP1000 DCD Revision 19 to address Containment Condensate Return Cooling design.

1. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2013-010,dated April 18, 2013 (ADAMS Accession No. ML13109A533), Enclosure 6 is found to be applicable to WLS Nuclear Station Units 1 and 2.

2. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-005, dated February 7, 2014 (ADAMS Accession No. ML14042A035), Enclosure 6 is found to be applicable to WLS Nuclear Station Units 1 and 2.

3. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-021, dated June 27, 2014 (ADAMS Accession No. ML14182A106), Enclosure 4 is found to be applicable to WLS Nuclear Station Units 1 and 2.

4. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-028, dated July 24, 2014 (ADAMS Accession No. ML14206A953), Enclosure 4 is found to be applicable to WLS Nuclear Station Units 1 and 2.

Enclosure 7

Duke Energy Voluntary Submittal

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Revisions to the William States Lee III COL Application

Attachment 1	**Revisions to Part 2, Final Safety Analysis Report**
Attachment 2	**Revisions to Part 4, Technical Specifications**
Attachment 3	**Revisions to Part 7, Departures and Exemptions Requests**
Attachment 4	**Revisions to Part 10, Proposed License Conditions (Including ITAAC)**

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Attachment 1

Revisions to Part 2, Final Safety Analysis Report

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Attachment 1A

Revisions to Part 2, FSAR Chapter 1

1. COLA Part 2, FSAR Chapter 1, Table 1.8-201, Summary of FSAR Departures from the DCD, will be revised to add departures WLS DEP 3.2-1 and WLS DEP 6.3-1 as follows:

Departure Number	Departure Description Summary	FSAR Section or Subsection
WLS DEP 3.2-1	The condensate return portion of the Passive Core Cooling System has been upgraded to add downspouts and plug fabrication holes in the Polar Crane Girder in order to maximize the return of condensate to the In-Containment Refueling Water Storage Tank and ensure long-term operation of the Passive Residual Heat Removal Heat Exchanger to meet design requirements. The following are the departures from the DCD: Tier 1 Table 2.2.3-1 and Table 2.2.3-2, Tier 2 Table 3.2-3 (Sheet 16 of 75), Figure 3.8.2-1 (Sheet 3), Subsections 5.4.11.2 and 5.4.14.1, Chapter 6, Subsections 6.3.1.1.1, 6.3.1.1.4, 6.3.1.1.6, 6.3.1.2, 6.3.1.3, 6.3.2.1, 6.3.2.1.1, 6.3.2.2.7, 6.3.2.8, 6.3.3, and 6.3.3.2.1.1, Figure 6.3-1 (Sheets 1 through 3), Figure 6.3-2 (Not Used), Subsection 7.4.1.1, Table 14.3-2 (Sheets 7 and 8 of 17), Subsection 15.0.13, Chapter 16 (TS Bases B3.3.3 and B3.5.4), Subsections 19E.4.10.2, 19E.9, Table 19E.4.10-1, and Figures 19E.4.10-1 through 19E.4.10-4.	Table 3.2-201, Figure 3.8-205, 5.4.11.2, 5.4.14.1, 6, 6.3.1.1.1, 6.3.1.1.4, 6.3.1.1.6, 6.3.1.2, 6.3.1.3, 6.3.2.1, 6.3.2.1.1, 6.3.2.2.7, 6.3.2.8, 6.3.3, 6.3.3.2.1.1, Figure 6.3-201, 7.4.1.1, 14, Table 14.3-202, 15.0.13, 16 (TS Bases B3.3.3 and B3.5.4), 19, 19E.4.10.2, 19.E.9, Table 19E.4.10-201, Figures 19E.4.10-201 through 19E.4.10-204

Departure Number	Departure Description Summary	FSAR Section or Subsection
WLS DEP 6.3-1	The DCD states that the PRHR HX can maintain safe shutdown conditions for non-LOCA accidents "indefinitely." A quantitative duration of greater than 14 days has been adopted based on that time being long enough to minimize the need to switch to passive feed and bleed cooling except for very unlikely or extreme hazard events. The following are the departures from the DCD: Subsection 5.4.14.1, Subsections 6.3.1.1.1, 6.3.1.2, 6.3.1.3, 6.3.2.1.1, and 6.3.3.4.1, Subsection 7.4.1.1, Table 9.5.1-1 (Sheet 11), Subsection 15.2.6.1, Table 19.59-18 (Sheet 6), and Subsection 19E.4.10.2	5.4.14.1, 6.3.1.1.1, 6.3.1.2, 6.3.1.3, 6.3.2.1.1, 6.3.3.4.1, 7.4.1.1, Table 9.5.1-201, 15.2.6.1, Table 19.59-201, 19E.4.10.2

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Attachment 1B

Revisions to Part 2, FSAR Chapter 3

2. COLA Part 2, FSAR Chapter 3 is revised with new Table 3.2-201 with a left margin annotation of WLS DEP 3.2-1 as follows:

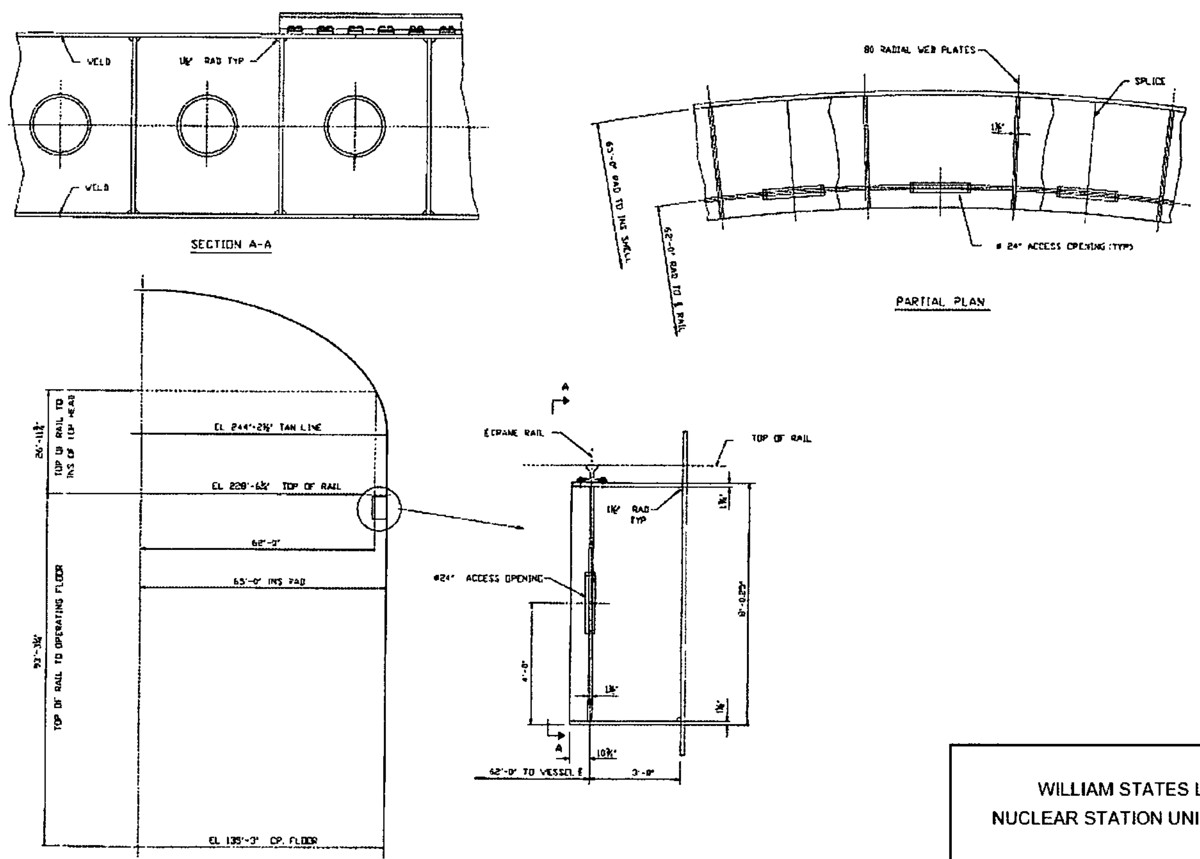
WLS DEP 3.2-1

TABLE 3.2-201
AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
...					
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.	
...					

3. COLA Part 2, FSAR Chapter 3 will be revised to add Figure 3.8-205 with a left margin annotation of WLS DEP 3.2-1 as follows:

WLS DEP 3.2-1



WILLIAM STATES LEE III
NUCLEAR STATION UNITS 1 AND 2

Containment Vessel General Outline

Figure 3.8-205

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Attachment 1C

Revisions to Part 2, FSAR Chapter 5

4. COLA Part 2, FSAR Chapter 5, will be revised to add new Subsection 5.4.11.2, with an LMA of WLS DEP 3.2-1, as follows:

5.4.11.2 System Description

Replace the second sentence of the second paragraph of DCD Subsection 5.4.11.2 with the following.

WLS DEP 3.2-1 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 DCD Figure 6.3-1.

5. COLA Part 2, FSAR Chapter 5, will be revised to add new Subsection 5.4.14.1 with left margin annotations of WLS DEP 3.2-1 and WLS DEP 6.3-1 as follows:

5.4.14.1 Design Bases

Replace the first sentence of the first paragraph of DCD Subsection 5.4.14.1 with the following information.

WLS DEP 6.3-1 The passive residual heat removal heat exchanger automatically actuates to remove core decay heat for an extended period of time as discussed in Section 6.3, assuming the condensate from steam generated in the in-containment refueling water storage tank (IRWST) is returned to the tank.

Combine the second and third paragraphs of DCD Subsection 5.4.14.1 and revise to read as follows:

WLS DEP 3.2-1
WLS DEP 6.3-1 The passive residual heat removal heat exchanger and the in-containment refueling water storage tank are designed to delay significant steam release to the containment for at least one hour. The passive residual heat removal heat exchanger will remove sufficient decay heat from the reactor coolant system to satisfy the applicable post-accident safety evaluation criteria detailed in DCD Chapter 15. In addition, the passive residual heat removal heat exchanger will cool the reactor coolant system, with reactor coolant pumps operating or in the natural circulation mode, so that the reactor coolant system pressure can be lowered to reduce stress levels in the system if required. See Section 6.3 for a discussion of the capability of the passive core cooling system.

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Attachment 1D

Revisions to Part 2, FSAR Chapter 6

6. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.1.1.1, with an LMA of WLS DEP 3.2-1 as follows:

6.3.1.1.1 Emergency Core Decay Heat Removal

Replace the bulleted list following the first paragraph of DCD Subsection 6.3.1.1.1 with the following information.

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

WLS DEP 3.2-1
- The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, condensate collection features and the passive containment cooling system, are designed to remove decay heat following a design basis event. Automatic depressurization actuation is not expected; but may occur depending on the amount of reactor coolant system leakage and when normal systems are recovered (refer to Subsection 6.3.1.1.4).
- The passive residual heat removal heat exchanger is designed to maintain acceptable reactor coolant system conditions for at least 72 hours following a non-LOCA event. The applicable post-accident safety evaluation criteria are discussed in Chapter 15. Operator action may be taken in accordance with emergency procedures to de-energize the loads on the Class 1E batteries to avoid unnecessary automatic actuation of the automatic depressurization system. Specific safe shutdown criteria are described in Subsection 6.3.1.1.4.
- The passive residual heat removal heat exchanger is capable of performing its post-accident safety functions, 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.

WLS DEP 3.2-1
WLS DEP 6.3-1
- Deleted

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

7. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.1.1.4, with an LMA of WLS DEP 3.2-1 as follows:

6.3.1.1.4 Safe Shutdown

Replace DCD Subsection 6.3.1.1.4 with the following information.

WLS DEP 3.2-1 The functional requirements for the passive core cooling system specify that the plant be brought to a stable condition using the passive residual heat removal heat exchanger for events not involving a loss of coolant. As stated in Subsection 6.3.1.1.1, the passive residual heat removal heat exchanger in conjunction with the passive containment cooling system provides sufficient heat removal to satisfy the post-accident safety evaluation criteria for at least 72 hours. Additionally, the passive core cooling system, in conjunction with the passive containment cooling system and the automatic depressurization system, has the capability to establish long-term safe shutdown conditions in the reactor coolant system.

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

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

For loss of coolant accidents and other postulated events where ac power sources are lost, or when the core makeup tank levels reach the automatic depressurization system actuation setpoint, the automatic depressurization system initiates. This results in injection from the accumulators and subsequently from the in-containment refueling water storage tank, once the reactor coolant system is nearly depressurized. For these conditions, the reactor coolant system depressurizes to saturated conditions at about 250°F within 24 hours. The passive core cooling system can maintain this safe shutdown condition indefinitely for the plant.

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

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8. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.1.1.6, with an LMA of WLS DEP 3.2-1 as follows:

6.3.1.1.6 Reliability Requirements

Replace the last sentence of DCD Subsection 6.3.1.1.6 with the following:

WLS DEP 3.2-1

DCD Subsection 6.3.1.3 includes specific non-safety related design requirements that help to confirm satisfactory system reliability.

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9. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.1.2 (new DCD Subsection 6.3.1.2), with LMAs of WLS DEP 3.2-1 and WLS DEP 6.3-1 as follows:

Add the following subsection after DCD Subsection 6.3.1.1.6.

DCD Subsection 6.3.1.2 is renumbered as Subsection 6.3.1.3.

6.3.1.2 Non-safety Design Basis

6.3.1.2.1 Long-Term Core Decay Heat Removal

WLS DEP 3.2-1
WLS DEP 6.3-1

The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, the condensate return features and the passive containment cooling system, has the capability to maintain the reactor coolant system in the specified, long-term safe shutdown condition of 420°F for 14 days in a closed-loop mode of operation. The automatic depressurization system can be manually actuated by the operators at any time during extended passive residual heat removal heat exchanger operation to initiate open-loop cooling. The operator actions necessary to achieve safe shutdown using the passive residual heat removal heat exchanger in a closed-loop mode of operation involve preventing unnecessary actuation of the automatic depressurization system as detailed in DCD Subsection 7.4.1.1.

10. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.1.3, title only, to reflect the numbering change of DCD Subsection 6.3.1.2 to 6.3.1.3, with left margin annotations of WLS DEP 3.2-1 and WLS DEP 6.3-1 as follows:

WLS DEP 3.2-1
WLS DEP 6.3-1

6.3.1.3 Power Generation Design Basis

11. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.1, with an LMA of WLS DEP 3.2-1 as follows:

6.3.2.1 Schematic Piping and Instrumentation Diagrams

Replace the first sentence of the first paragraph of DCD Subsection 6.3.2.1 with the following:

WLS DEP 3.2-1

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

12. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.1.1 with left margin annotations WLS DEP 3.2-1 and WLS DEP 6.3-1 as follows:

6.3.2.1.1 Emergency Core Decay Heat Removal at High Pressure and Temperature Conditions

Replace the seventh and eighth paragraphs of DCD Subsection 6.3.2.1.1 with the following:

WLS DEP 3.2-1
WLS DEP 6.3-1

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

Condensation occurs on the steel containment vessel, which is cooled by the passive containment cooling system. Most of the condensate formed on the containment vessel wall is collected in a safety-related gutter arrangement. A gutter is located near the operating deck 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 gutter and downspouts normally drain to the containment sump, but when the passive residual heat removal heat exchanger actuates, safety-related isolation valves in the gutter drain line shut and the gutter overflow returns directly to the in-containment refueling water storage tank. Recovery of the condensate maintains the passive residual heat removal heat exchanger heat sink for an extended period of time.

Revise the first and second sentences of the ninth paragraph of DCD Subsection 6.3.2.1.1 as follows:

WLS DEP 3.2-1 The passive residual heat removal heat exchanger is used to maintain an acceptable, stable reactor coolant system condition. It transfers decay heat and sensible heat from the reactor coolant system to the in-containment refueling water storage tank, the containment atmosphere, the containment vessel, and finally to the ultimate heat sink–the atmosphere outside of containment.

Add a new tenth paragraph to DCD Subsection 6.3.2.1.1 to read as follows:

WLS DEP 3.2-1 The duration the passive residual heat removal heat exchanger can continue to remove decay heat is affected by the efficiency of the return of condensate to the in-containment refueling water storage tank. The in-containment refueling water storage tank water level is affected by the amount of steam that leaves the tank and does not return. Offsite or onsite ac power sources are typically recovered within a day, which would allow the operators to place active, defense-in-depth systems into service and to terminate passive system operation. If ac power is not recovered within this time frame, closed-loop cooling using the passive residual heat removal heat exchanger can be extended as described in DCD Subsection 7.4.1.1 to maintain a safe, stable condition after a design basis event.

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13. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.2.7, with an LMA of WLS DEP 3.2-1 as follows:

6.3.2.2.7 IRWST and Containment Recirculation Screens

Replace the first paragraph of DCD Subsection 6.3.2.2.7 with the following:

WLS DEP 3.2-1 The passive core cooling system has two different sets of screens that are used 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 frame's attachment to the building structure, and attachment of the screen modules use the criteria of ASME Code, Section III Subsection NF. The screen modules are fabricated of sheet metal and are designed and fabricated to a manufacturer's standard. The IRWST screens and containment recirculation screens are designed to comply with applicable licensing regulations including:

- GDC 35 of 10 CFR 50 Appendix A
- Regulatory Guide 1.82
- NUREG-0897

14. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.2.7.1, with an LMA of WLS DEP 3.2-1 as follows:

6.3.2.2.7.1 General Screen Design Criteria

Insert the following at the beginning of DCD Subsection 6.3.2.2.7.1.

WLS DEP 3.2-1 The IRWST screens and containment recirculation screens are designed to comply with the following criteria.

15. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.2.7.2, with an LMA of WLS DEP 3.2-1 as follows:

6.3.2.2.7.2 IRWST Screens

Replace the third paragraph of DCD Subsection 6.3.2.2.7.2 with the following:

WLS DEP 3.2-1 During a LOCA, steam vented from the reactor coolant system condenses on the containment shell and drains down the shell to 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 their locations. Each downspout inlet is covered with a coarse screen that prevents larger debris from entering the downspout. The gutter is covered with a trash rack which prevents larger debris from clogging the gutter or entering the IRWST through the two 4-inch drain pipes. The inorganic zinc coating applied to the inside surface of the containment shell is safety – Service Level I, and will stay in place and will not detach.

16. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.8, with an LMA of WLS DEP 3.2-1 as follows:

6.3.2.8 Manual Actions

Replace the third paragraph of DCD Subsection 6.3.2.8 with the following information:

WLS DEP 3.2-1 The operator can take action to avoid actuation of the automatic depressurization system when it is not needed. For non-LOCA events during which ac power has been lost for more than 22 hours, the protection and safety monitoring system will automatically open the automatic depressurization system valves to begin a controlled depressurization of the reactor coolant system and, eventually, containment floodup and recirculation prior to depletion of the actuation batteries. However, the operators can take action to block actuation of the automatic depressurization system should actuation be deemed unnecessary based on reactor coolant system conditions. This action allows closed loop passive residual heat removal heat exchanger operation to continue as long as acceptable reactor coolant system conditions are maintained.

Section 7.4 describes the anticipated operator actions to block unnecessary automatic depressurization system actuation. DCD Section 7.5 describes the post-accident monitoring instrumentation available to the operator in the main control room following an event.

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17. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.3, with an LMA of WLS DEP 3.2-1 as follows:

6.3.3 Performance Evaluation

Replace the seventh, eighth, and ninth paragraphs of DCD Subsection 6.3.3 with the following information.

WLS DEP 3.2-1 For non-LOCA events, the passive residual heat removal heat exchanger is actuated so that it can remove core decay heat. The passive residual heat removal heat exchanger can operate for at least 72 hours after initiation of a design basis event to satisfy Condition I, II, III and IV safety evaluation criteria described in the relevant safety analyses. Subsection 6.3.3.2.1.1 provides an evaluation of the duration of the passive residual heat removal heat exchanger operation using the LOFTRAN code described in DCD Subsection 15.0.11.2. In this evaluation it is assumed that the operators power down the protection and monitoring actuation cabinets in the 22 hour time frame prior to the automatic timer actuating ADS.

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

For loss of coolant accidents, the core makeup tanks deliver borated water to the reactor coolant system via the direct vessel injection nozzles. The accumulators deliver flow to the direct vessel injection line whenever reactor coolant system pressure drops below the tank static pressure. The in-containment refueling water storage tank provides gravity injection once the reactor coolant system pressure is reduced to below the injection head from the in-containment refueling water storage tank. The passive core cooling system flow rates vary depending upon the type of event and its characteristic pressure transient.

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

The transient analyses summarized in DCD Chapter 15 are extended long enough to demonstrate the applicable safety evaluation criteria are met. It is expected that normal systems would be available such that operators could terminate the passive safety systems and proceed with an orderly shutdown. However, as discussed in Subsection 6.3.1.1.4, the passive systems are capable of bringing the plant to a safe shutdown condition and maintaining that condition.

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18. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.3.2.1.1 with an LMA of WLS DEP 3.2-1 as follows:

6.3.3.2.1.1 Loss of AC Power to the Plant Auxiliaries

WLS DEP 3.2-1 The most severe conditions resulting from a loss of ac power to the plant auxiliaries are associated with loss of offsite power with a loss of main feedwater system flow at full power. A loss of main feedwater with a loss of ac power lasting longer than a few hours presents the highest demand on passive residual heat removal heat exchanger operation. DCD Subsection 15.2.6 provides a description of this short-term event, including criteria and analytical results.

During most events, the passive systems would be terminated in hours. However, if normal systems are not recovered as expected, the passive residual heat removal heat exchanger removes core decay heat and maintains acceptable reactor coolant system conditions for at least 72 hours. For a non-loss of coolant accident event lasting as long as 24 hours, the automatic depressurization system will actuate if operators do not act to avoid actuation when it is not needed. For this long-term transient, it is assumed operators extend passive residual heat removal heat exchanger operation as described in DCD Subsection 7.4.1.1, such that the automatic depressurization system does not actuate.

The loss of main feedwater with loss of ac power event is analyzed for a 72 hour period, assuming operators extend closed-loop cooling beyond the time the automatic depressurization system would be actuated by the protection and safety monitoring system. This event mirrors the loss of ac power to the plant auxiliaries event described in DCD Subsection 15.2.6, but the loss of ac power extends to 72 hours. In this event, operation of the passive residual heat removal heat exchanger continues for 72 hours and maintains acceptable reactor coolant system conditions such that the applicable Condition II safety evaluation criteria are met.

Reactor coolant system leakage could limit closed-loop capacity. A reactor coolant system leak could produce conditions that would preclude the operators from de-energizing the loads on the Class 1E batteries, or could require the operators to re-energize the buses powered by the Class 1E batteries before 72 hours so that the automatic depressurization system valves could be actuated. When an ac power source is restored and passive core cooling system termination criteria are satisfied, the operator terminates passive core cooling system operation and initiates normal plant shutdown operations.

---

19. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.3.4.1, with an LMA of WLS DEP 6.3-1 as follows:

6.3.3.4.1 Loss of Startup Feedwater During Hot Standby, Cooldowns, and Heat-ups

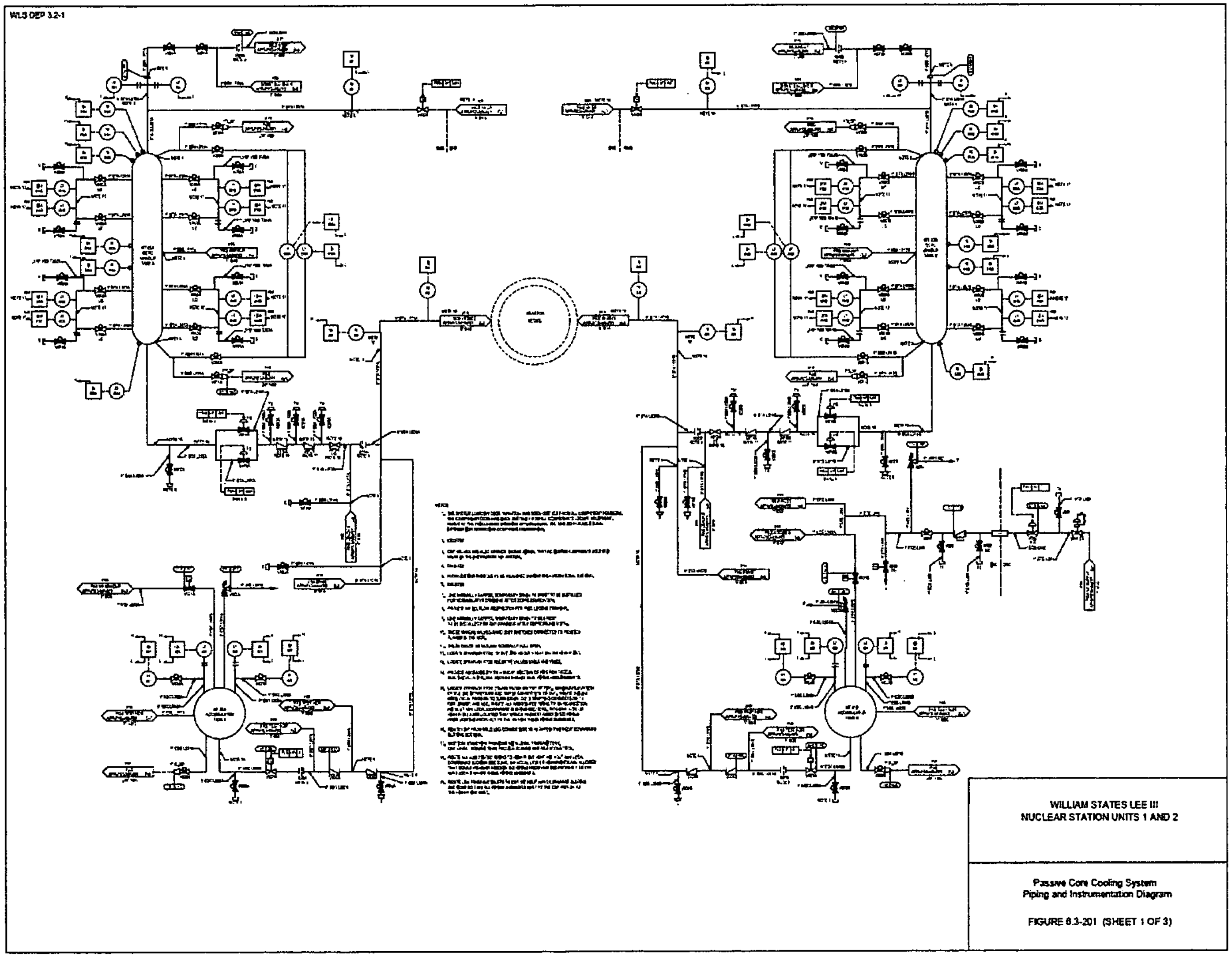
---

Revise the last sentence of the fourth paragraph of DCD Subsection 6.3.3.4.1 to read as follows:

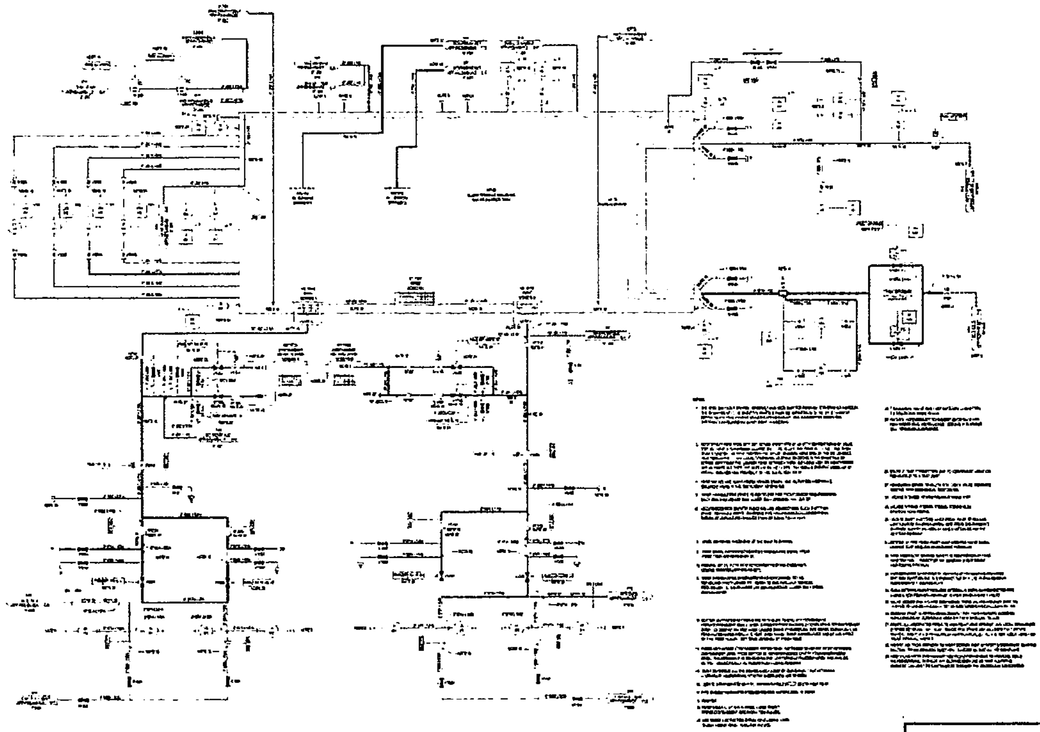
WLS DEP 6.3-1 This allows it to function as a heat sink.

---

20. COLA Part 2, FSAR Section 6.3 will be revised to add Figure 6.3-201 with an LMA of WLS DEP 3.2-1 as follows:



WLS DEP 3.2-1

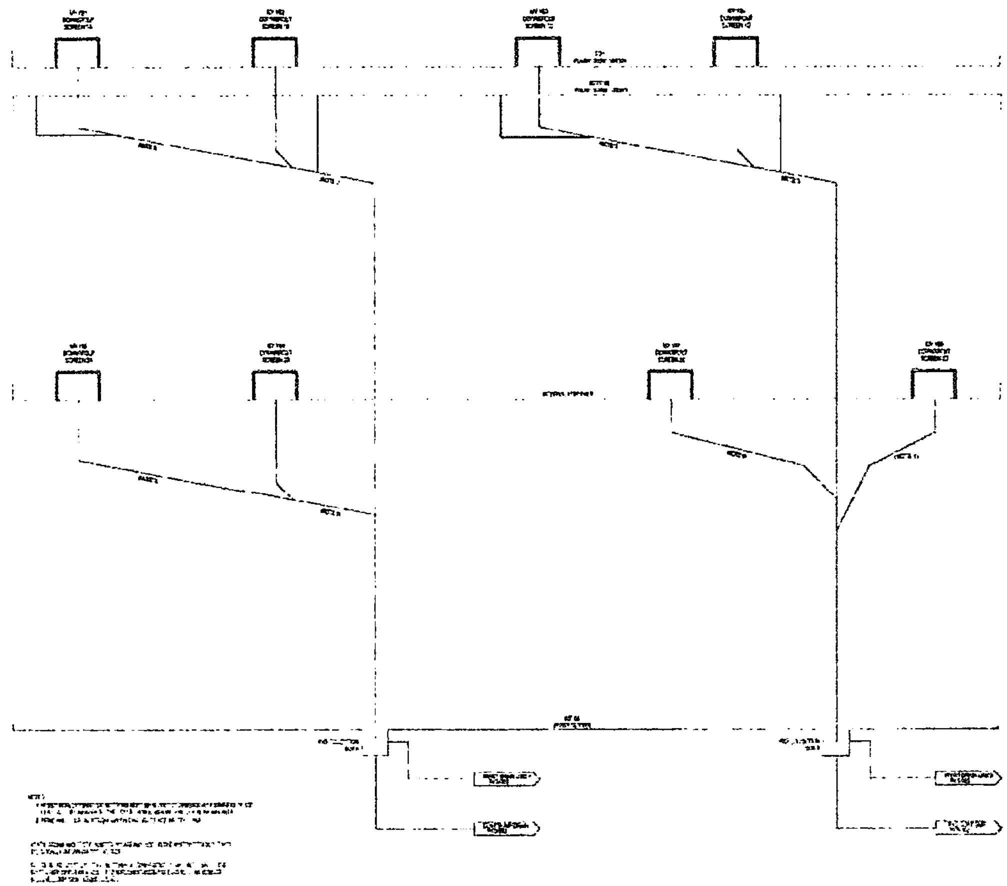


WILLIAM STATES LEE III
NUCLEAR STATION UNITS 1 AND 2

Passive Core Cooling System
Piping and Instrumentation Diagram

FIGURE 6.3-201 (Sheet 2 of 3)

WLS DEP 3.2-1



WILLIAM STATES LEE III
NUCLEAR STATION UNITS 1 AND 2

Passive Core Cooling System
Piping and Instrumentation Diagram

FIGURE 6.3-201 (Sheet 3 of 3)

Duke Energy Voluntary Submittal

Condensate Return

Attachment 1E

Revisions to Part 2, FSAR Chapter 7

21. COLA Part 2, FSAR Chapter 7, will be revised to add new Subsection 7.4.1.1 with left margin annotations WLS 3.2-1 and WLS 6.3-1 as follows:

7.4 SYSTEMS REQUIRED FOR SAFE SHUTDOWN

This Section of the referenced DCD is incorporated by reference with ~~no~~ the following departures and/or supplements.

7.4.1.1 Safe Shutdown Using Safety-Related Systems

Revise the second sentence of the sixth paragraph of DCD Subsection 7.4.1.1 as follows:

WLS DEP 6.3-1 This prevents loss of water inventory from containment and permits extended operation of the passive residual heat removal heat exchanger and the in-containment refueling water storage tank.

Revise the last sentence of the eighth paragraph of DCD Subsection 7.4.1.1 as follows:

WLS DEP 3.2-1
WLS DEP 6.3-1 The system provides core decay heat removal in this configuration with a limited increase in the containment water level.

Revise the ninth paragraph of DCD Subsection 7.4.1.1 as follows:

WLS DEP 3.2-1 Once the reactor coolant system and the safety systems are in this configuration, the plant is in a stable shutdown condition. The reactor coolant system temperatures and pressures continue to slowly decrease. The passive residual heat removal heat exchanger has the capacity to maintain a safe, stable reactor coolant system condition during a design basis event for at least 72 hours in a closed-loop mode of operation. A non-bounding, conservative analysis of extended operation in this mode shows the passive residual heat removal heat exchanger cools the reactor coolant system to 420°F in 36 hours.

Revise the last three sentences of the eleventh paragraph of DCD Subsection 7.4.1.1 as follows:

WLS DEP 3.2-1 The operator assessment includes consideration for a visible refueling water storage tank level, full core makeup tanks, a high and stable pressurizer level, and decreasing or stable reactor coolant system temperature. If automatic depressurization is not needed, the operator is directed to de-energize all loads on the Class 1E dc batteries. This action preserves the capability for the operator to initiate automatic depressurization at a later time based on assessment of the same parameters.

Duke Energy Voluntary Submittal

Condensate Return

Attachment 1F

Revisions to Part 2, FSAR Chapter 9

22. COLA Part 2, FSAR Chapter 9 is revised to add new Table 9.5.1-201 with an LMA of WLS DEP 6.3-1 as follows:

WLS DEP 6.3-1

TABLE 9.5.1-201

AP1000 FIRE PROTECTION PROGRAM COMPLIANCE WITH BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[1]	Remarks
Safe Shutdown Capability			
72. Fire damage should be limited so that one train of systems necessary to achieve and maintain hot shutdown conditions from either the main control room or emergency control station is free of fire damage.	C.5.b(1)	C	
73. Fire damage should be limited so that systems necessary to achieve and maintain cold shutdown from either the control room or emergency control station can be repaired within 72 hours.	C.5.b (1)	AC	Safe shutdown following a fire is defined for the AP1000 plant as the ability to achieve and maintain the reactor coolant system (RCS) temperature below 215.6°C (420°F) without uncontrolled venting of the primary coolant from the RCS. This is a departure from the criteria applied to the evolutionary plant designs, and the existing plants where safe shutdown for fires applies to both hot and cold shutdown capability. With expected RCS leakage, the AP1000 plant can maintain safe shutdown conditions for at least 14 days. Therefore, repairs to systems necessary to reach cold shutdown need not be completed within 72 hours.
74. Separation requirements for verifying that one train of systems necessary to achieve and maintain hot shutdown is free of fire damage.	C.5.b (2)	C	

Notes:

1. Compliance with NUREG-0800 Section 9.5.1, Branch Technical Position CMEB 9.5-1 is indicated by the following codes:
   C - Compliance: AP1000 is committed to compliance with the guideline.
   AC - Alternate Compliance: compliance with the guideline by alternate means or intent. Alternative means or design are provided in the remarks column.

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Condensate Return

Attachment 1G

Revisions to Part 2, FSAR Chapter 14

23. COLA Part 2, FSAR Chapter 14 is revised to add new Table 14.3-202, with an LMA of WLS DEP 3.2-1 as follows:

WLS DEP 3.2-1

TABLE 14.3-202 (SHEET 1 of 2)

DESIGN BASIS ACCIDENT ANALYSIS

DCD Reference	Design Feature	Value
Subsection 6.3.6.1.3	The bottom of the in-containment refueling water storage tank is located above the direct vessel injection nozzle centerline (ft).	≥ 3.4
Subsection 6.3.6.1.3	The pH baskets are located below plant elevation 107' 2"	
Figure 6.3-1	The passive core cooling system has two direct vessel injection lines.	
Table 6.3-2	The passive core cooling system has two core makeup tanks, each with a minimum required volume (ft^3).	2500
Table 6.3-2	The passive core cooling system has two accumulators, each with a minimum required volume (ft^3)	2000
Table 6.3-2	The passive core cooling system has an in-containment refueling water storage tank with a minimum required water volume (ft^3)	73,900
Subsection 6.3.2.2.3	The containment floodup volume for a LOCA in PXS room B has a maximum volume (ft^3) (excluding the IRWST) below a containment elevation of 108 feet.	73,500
Table 6.3-2	Each sparger has a minimum discharge flow area (in^2).	≥ 274
Table 6.3-2	The passive core cooling system has two pH adjustment baskets each with a minimum required volume (ft^3).	280
Subsection 14.2.9.1.3f	The passive residual heat removal heat exchanger minimum natural circulation heat transfer rate (Btu/hr) - With 520°F hot leg and 80°F IRWST - With 420°F hot leg and 80°F IRWST	 ≥ 1.78 E+08 ≥ 1.11 E+08
Subsection 6.3.6.1.3	The centerline of the HX's upper channel head is located above the HL centerline (ft).	≥ 26.3
Figure 6.3-1	The CMT level sensors (PXS-11A/B/C/D, -12A/B/C/D, -13A/B/C/D, and -14A/B/C/D) upper level tap centerlines are located below the centerline of the upper level tap connection to the CMTs (in).	1" ± 1"
Figure 6.3-1	The CMT inlet lines (cold leg to high point) have no downward sloping sections.	
Figure 6.3-1	The maximum elevation of the CMT injection lines between the connection to the CMT and the reactor vessel is the connection to the CMTs.	
Figure 6.3-1	The PRHR inlet line (hot leg to high point) has no downward sloping sections.	

WLS DEP 3.2-1

TABLE 14.3-202 (SHEET 2 of 2)

DESIGN BASIS ACCIDENT ANALYSIS

DCD Reference	Design Feature	Value
Figure 6.3-1	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	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.	
Subsection 7.1.2.10	Isolation devices are used to maintain the electrical independence of divisions and to see that no interaction occurs between nonsafety-related systems and the safety-related system. Isolation devices serve to prevent credible faults in circuit from propagating to another circuit.	
Subsection 7.1.4.2	The ability of the protection and safety monitoring system to initiate and accomplish protective functions is maintained despite degraded conditions caused by internal events such as fire, flooding, explosions, missiles, electrical faults and pipe whip.	
Subsection 7.1.2	The flexibility of the protection and safety monitoring system enables physical separation of redundant divisions.	
Subsection 7.2.2.2.1	The protection and safety monitoring system initiates a reactor trip whenever a condition monitored by the system reaches a preset level.	
Subsection 7.2.2.2.8	The reactor is tripped by actuating one of two manual reactor trip controls from the main control room.	
Subsection 7.3.1.2.2	The in-containment refueling water storage tank is aligned for injection upon actuation of the fourth stage automatic depressurization system via the protection and safety monitoring system.	
Subsection 7.3.1.2.3	The core makeup tanks are aligned for operation on a safeguards actuation signal or on a low-2 pressurizer level signal via the protection and safety monitoring system.	
Subsection 7.3.1.2.4	The fourth stage valves of the automatic depressurization system receive a signal to open upon the coincidence of a low-2 core makeup tank water level in either core makeup tank and low reactor coolant system pressure following a preset time delay after the third stage depressurization valves receive a signal to open via the protection and safety monitoring system.	

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Condensate Return

Attachment 1H

Revisions to Part 2, FSAR Chapter 15

24. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.0.13, with an LMA of WLS DEP 3.2-1 as follows:

15.0.13 Operator Actions

Revise the first sentence of the first paragraph of DCD Subsection 15.0.13 as follows:

WLS DEP 3.2-1 For events where the PRHR heat exchanger is actuated, the plant automatically cools down to a safe, stable condition.

---

25. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.2.6.1, with an LMA of WLS DEP 6.3-1 as follows:

15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

This section of the referenced DCD is incorporated by reference with ~~no~~ the following departures or supplements.

15.2.6.1 Identification of Causes and Accident Description

WLS DEP 6.3-1 Revise the seventh sentence of the fourth paragraph of DCD Subsection 15.2.6.1 as follows:

The PRHR heat exchanger, in conjunction with the passive containment cooling system, provides core cooling and maintains reactor coolant system conditions to satisfy the evaluation criteria.

Duke Energy Voluntary Submittal

Condensate Return

Attachment 1I

Revisions to Part 2, FSAR Chapter 19

26. COLA Part 2, FSAR Chapter 19, Section 19.59 is revised to add departure from DCD Table 19.59-18, PRA Based Insights, Sheet 6 of 25, with new Table 19.59-201 with an LMA of WLS DEP 6.3-1 as follows:

WLS DEP 6.3-1

TABLE 19.59-201

AP1000 PRA-BASED INSIGHTS

Insight	Disposition
1e. (cont.)	
Long-term cooling of PRHR will result in steaming to the containment. The steam will normally condense on the containment shell and return to the IRWST by safety-related features. Connections are provided to IRWST from the spent fuel system (SFS) and chemical and volume control system (CVS) to extend PRHR operation. A safety-related makeup connection is also provided from outside the containment through the normal residual heat removal system (RNS) to the IRWST.	6.3.1 & system drawings
Capability exists and guidance is provided for the control room operator to identify a leak in the PRHR HX of 500 gpd. This limit is based on the assumption that a single crack leaking this amount would not lead to a PRHR HX tube rupture under the stress conditions involving the pressure and temperature gradients expected during design basis accidents, which the PRHR HX is designed to mitigate.	6.3.3 & 16.1
The positions of the inlet and outlet PRHR valves are indicated and alarmed in the control room.	6.3.7
PRHR air-operated valves are stroke-tested quarterly. The PRHR HX is tested to detect system performance degradation every 10 years.	3.9.6
PRHR is required by Technical Specifications to be available from Modes 1 through 5 with RCS pressure boundary intact.	16.1
The PRHR HX, in conjunction with the IRWST, condensate return features and the PCS, can provide core cooling for at least 72 hours. After the IRWST water reaches its saturation temperature, the process of steaming to the containment initiates. Condensation occurs on the steel containment vessel, and the condensate is collected in a safety-related gutter arrangement, which returns the condensate to the IRWST. The gutter normally drains to the containment sump, but when the PRHR HX actuates, safety-related isolation valves in the gutter drain line shut and the gutter overflow returns directly to the IRWST. The following design features provide proper re-alignment for the gutter system valves to direct water to the IRWST:	6.3.2.1.1 & 6.3.7.6
- IRWST gutter and its drain isolation valves are safety-related	
- These isolation valves are designed to fail closed on loss of compressed air, loss of Class 1E dc power, or loss of the PMS signal	
- These isolation valves are actuated automatically by PMS and DAS.	7.3.1.2.7
The PRHR subsystem provides a safety-related means of removing decay heat following loss of RNS cooling during shutdown conditions with the RCS intact.	16.1

27. COLA Part 2, FSAR Chapter 19, Appendix 19E Shutdown Evaluation, is revised as follows, with an LMA of WLS DEP 3.2-1 and WLS DEP 6.3-1, where noted, as follows:

APPENDIX 19E
SHUTDOWN EVALUAITON

This section of the referenced DCD is incorporated by reference with ~~no~~ the following departures and/or supplements.

19E.4.10.2 Shutdown Temperature Evaluation

Replace DCD Subsection 19E.4.10.2 with the following information.

WLS DEP 3.2-1 As discussed in Subsection 6.3.1.1.4, the passive residual heat removal heat exchanger is required to be able to cool the reactor coolant system to a safe, stable condition after shutdown following a non-LOCA event. The following summarizes a non-bounding, conservative analysis, which demonstrates the passive residual heat removal heat exchanger can meet this criterion and cool the RCS to the specified, safe shutdown condition of 420°F within 36 hours. This analysis demonstrates that the passive systems can bring the plant to a safe, stable condition and maintain this condition so that no transients will result in the specified acceptable fuel design limit and pressure boundary design limit being violated and that no high-energy piping failure being initiated from this condition results in 10 CFR 50.46 (DCD Reference 15) criteria.

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

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

The PRHR HX performance is also affected by the IRWST water level when the level drops below the top of the PRHR HX tubes. The IRWST water level is affected by the heat input from the PRHR HX and by the amount of steam that leaves the IRWST and does not return to the IRWST through the IRWST gutter arrangement. The principal steam condensate losses include steam that stays in the containment atmosphere, steam that condenses on heat sinks inside containment other than the containment vessel, and dripping or splashing losses due to obstructions on the inner containment vessel wall. The WGOTHIC containment response model also provided the mass balance with respect to the steam lost to the containment atmosphere and to condensation on passive heat sinks other than the containment vessel. The WGOTHIC analysis inputs (including the mass of the heat sinks and heat transfer rates) were biased to increase steam condensate losses. The 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 DCD 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 (offsite and onsite) 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 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 injection of cold water, the CMTs operate in conjunction with the PRHR HX to reduce RCS temperature. Due to the primary system cooldown, the PRHR heat transfer capability drops below the decay heat and the RCS cooldown is essentially driven by the CMT cold injection flow. However, at about 5,000 seconds, the CMT cooling effect decreases and the RCS starts heating up again (DCD Figure 19.E.4.10-1). The RCS temperature increases until the PRHR HX can match decay heat. At about 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 DCD Figure 19E.4.10-1, the cold leg temperature in the loop with the PRHR is reduced to 420°F within 48,600 seconds, while the core average temperature reaches 420°F within 124,400 seconds (approximately 34.6 hours).

As discussed in DCD 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. Operation of the ADS in conjunction with the CMTs, accumulators, and IRWST reduces the RCS pressure and temperature to below 420°F. The ability to actuate ADS and IRWST injection provides a safety-related, backup mode of decay heat removal that is diverse to extended PRHR HX operation.

WLS DEP 6.3-1 As discussed in DCD Subsection 6.3.3.2.1.1, the PRHR HX can operate in this mode for at least 72 hours to maintain RCS conditions within the applicable Chapter 15 safety evaluation criteria. In addition, the analysis supporting this Section shows the PRHR HX is expected to maintain safe shutdown conditions for more than 14 days. One important consideration with regard to the duration closed-loop cooling can be maintained is the RCS leak rate. This duration of closed-loop cooling can be achieved with expected RCS leak rates. For abnormal leak rates, it may become necessary to initiate open-loop cooling earlier than 14 days.

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28. COLA Part 2, FSAR Chapter 19 is revised to add a new Subsection 19E.9, with an LMA of WLS DEP 3.2-1 as follows:

19E.9 REFERENCES

WLS DEP 3.2-1 14. Not used.

29. COLA Part 2, FSAR Chapter 19, Appendix 19E is revised to add Table 19E.4.10-201, with an LMA of WLS DEP 3.2-1 as follows:

WLS DEP 3.2-1

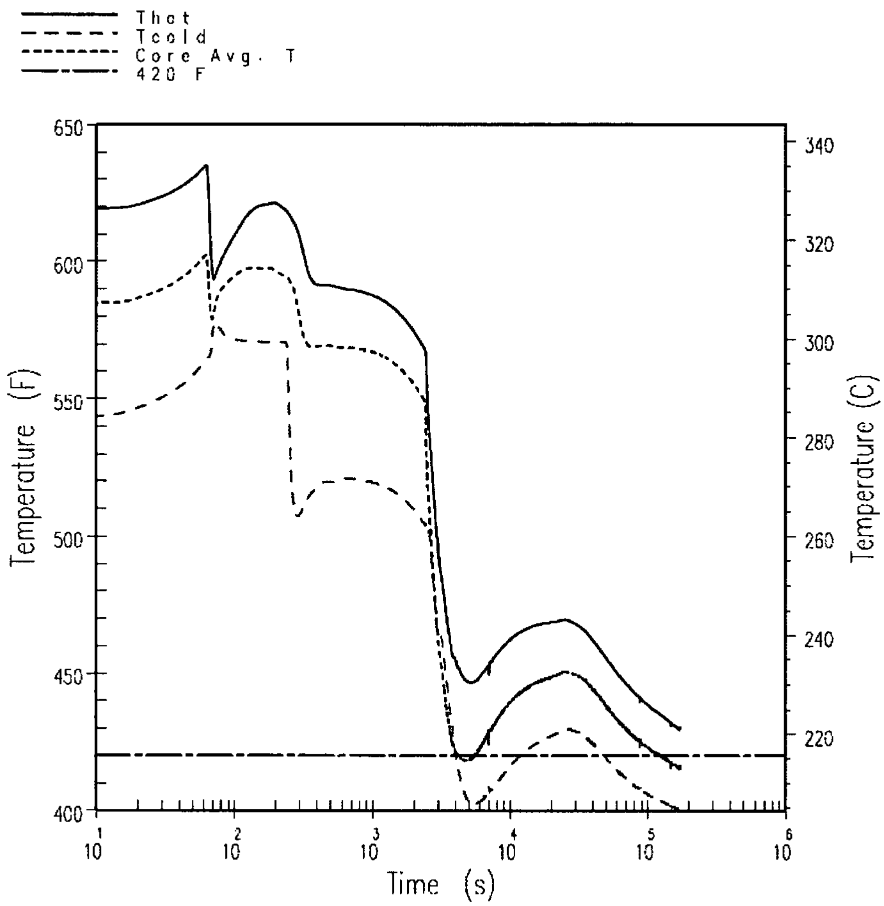
TABLE 19E.4.10-201

SEQUENCE OF EVENTS FOLLOWING A LOSS OF AC POWER FLOW WITH CONDENSATE FROM THE CONTAINMENT SHELL BEING RETURNED TO THE IRWST

Event	Time (seconds)
Feedwater is Lost	10.0
Low Steam Generator Water Level (Narrow-Range) Reactor Trip Setpoint Reached	≤ 60
Rods Begin to Drop	≤ 61
Low Steam Generator Water Level (Wide-Range) Reached	≤ 230
PRHR HX Actuation on Low Steam Generator Water Level (Narrow-Range Coincident with Low Startup Feedwater Flow)	≤ 240
Low T_{cold} Setpoint Reached	≤ 2400
Steam Line Isolation on Low T_{cold} Signal	≤ 2400
CMTs Actuated on Low T_{cold} Signal	≤ 2400
IRWST Reaches Saturation Temperature	≤ 15,500
Heat Extracted by PRHR HX Matches Core Decay Heat	≤ 34,500
CMTs Stop Recirculating	--
Cold Leg Temperature Reaches 420°F (loop with PRHR)	≤ 48,600
Core Average Temperature Reaches 420°F	≤ 124,400

30. COLA Part 2, FSAR Appendix 19E is revised to add Figures 19E.4.10-201 through 19E.4.10-204, with left margin annotations of WLS DEP 3.2-1 as follows:

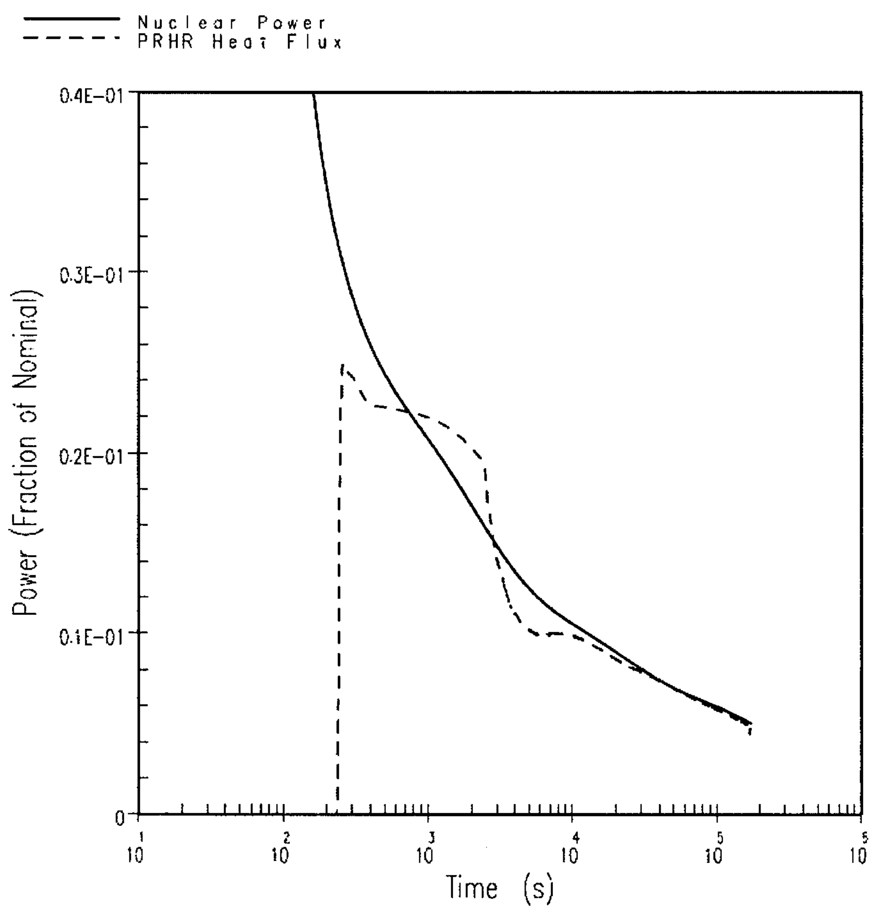
WLS DEP 3.2-1



WILLIAM STATES LEE III
NUCLEAR STATION UNITS 1 AND 2

Shutdown Temperature Evaluation,
RCS Temperature
FIGURE 19E.4.10-201

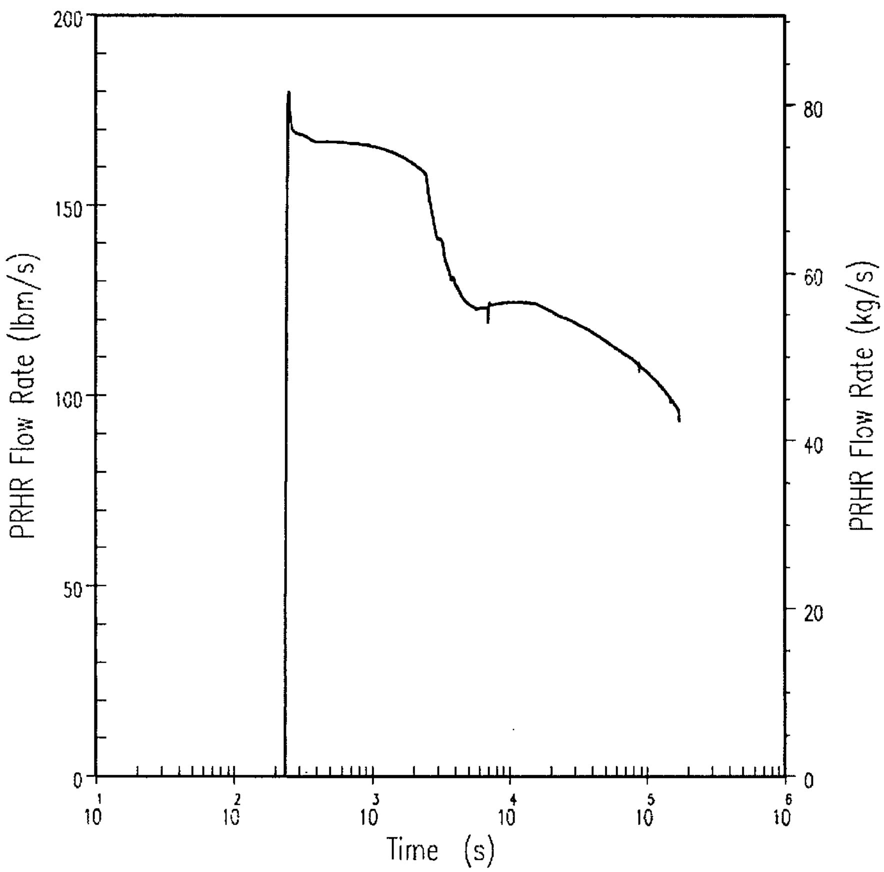
WLS DEP 3.2-1



WILLIAM STATES LEE III
NUCLEAR STATION UNITS 1 AND 2

Shutdown Temperature Evaluation,
PRHR Heat Transfer
FIGURE 19E.4.10-202

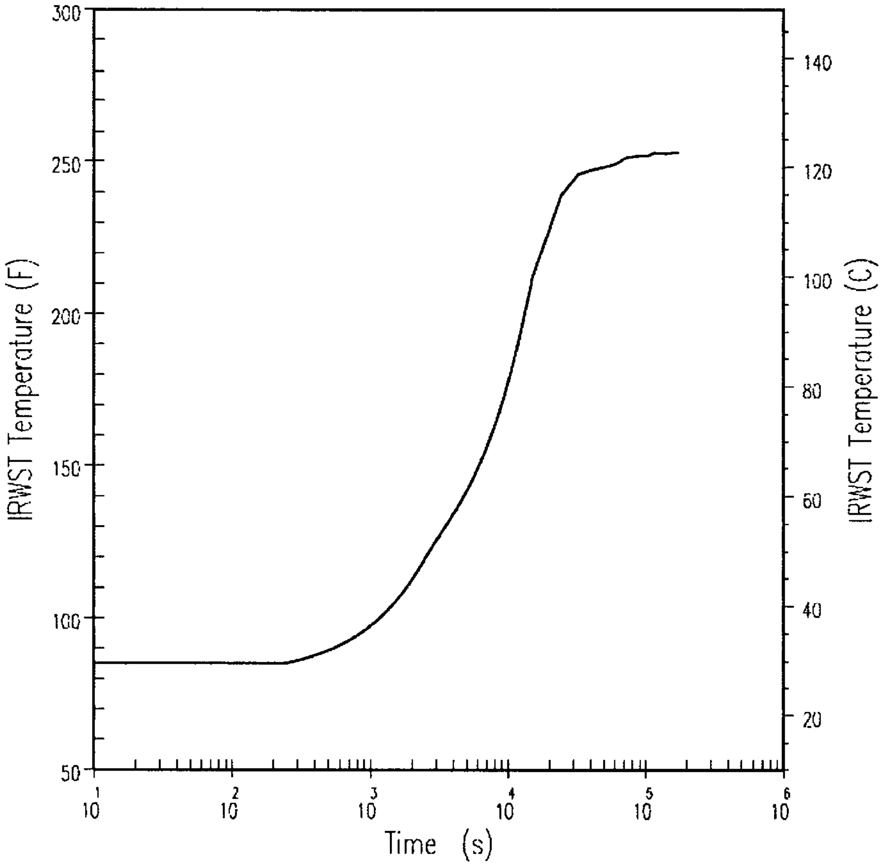
WLS DEP 3.2-1



WILLIAM STATES LEE III
NUCLEAR STATION UNITS 1 AND 2

Shutdown Temperature Evaluation,
PRHR Flow Rate
FIGURE 19E.4.10-203

WLS DEP 3.2-1



WILLIAM STATES LEE III
NUCLEAR STATION UNITS 1 AND 2

Shutdown Temperature Evaluation,
IRWST Heatup
FIGURE 19E.4.10-204

Duke Energy Voluntary Submittal

Condensate Return

Attachment 2

Revisions to Part 4, Technical Specifications

COLA Part 4, Technical Specifications

1. Revise LCO 11 for Part 4, TS Bases B 3.3.3, last sentence of the first paragraph with an LMA of WLS DEP 3.2-1 as follows:

WLS DEP 3.2-1 The condensate is returned to the IRWST via a gutter and downspouts.

---

2. Revise the first two sentences of the third paragraph for Part 4, TS Bases B 3.5.4, Background with an LMA of WLS DEP 3.2-1 as follows:

WLS DEP 3.2-1 In order to preserve the IRWST water for long-term PRHR HX operation, downspouts and a gutter 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.

---

3. Revise SR 3.5.4.7 of Part 4, TS Bases B 3.5.4, Surveillance Requirements with an LMA of WLS DEP 3.2-1 as follows:

WLS DEP 3.2-1 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.

Duke Energy Voluntary Submittal

Condensate Return

Attachment 3

Revisions to Part 7, Departures and Exemptions Requests

1. COLA Part 7, Departures and Exemption Requests, is revised to add the following departures to the table presented is Section A as follows:

Departure Number	Description
WLS DEP 3.2-1	Addition of downspouts to the condensate return portion of the Passive Core Cooling System
WLS DEP 6.3-1	Quantification of the term "indefinitely" as used in the DCD for maintenance of safe shutdown conditions using the PRHR HX during non-LOCA accidents.

2. COLA Part 7, Departures and Exemption Requests, is revised to add the following departures to the table presented in Section A.1, Departures That Can be Implemented Without Prior NRC Approval as follows:

Departure Number	Description
WLS DEP 6.3-1	Quantification of the term "indefinitely" as used in the DCD for maintenance of safe shutdown conditions using the PRHR HX during non-LOCA accidents.

3. COLA Part 7, Departures and Exemption Requests, is revised to add the following departure to Section A.1 as follows:

Departure Number WLS DEP 6.3-1

Affected DCD/FSAR Sections: Subsection 5.4.14.1, Subsections 6.3.1.1.1, 6.3.1.2, 6.3.1.3, 6.3.2.1.1, and 6.3.3.4.1, Subsection 7.4.1.1, Table 9.5.1-1 (Sheet 11), Subsection 15.2.6.1, Table 19.59-18 (Sheet 6), and Subsection 19E.4.10.2

Summary of Departure:

The Passive Residual Heat Removal Heat Exchanger (PRHR HX) has a functional requirement to be able to bring the AP1000 plant to a stable condition for events not involving a loss of coolant (i.e., non-LOCA event), DCD 6.3.1.1.4. The DCD in Subsection 6.3.1.1.1 further states "The PRHR HX in conjunction with the passive containment cooling system, is designed to remove decay heat for an indefinite time in a closed-loop mode of operation." Additional evaluations have been subsequently performed that have identified that the use of the term "indefinite" does not describe the predicted PRHR HX long term operation properly. The word "indefinite" can be defined as an "unknown" or "unidentified" length of time; "indefinite" does not mean "infinite" which means having no boundaries or limits in time. The word "indefinite" in regards to PRHR HX long term operation needs to be changed with a definitive time period.

Scope/Extent of Departure:

There are additional areas in the DCD that use the term "indefinite" in reference to long term PRHR HX operation that need to be changed in a departure to the DCD to more accurately reflect the PRHR HX long term operation during a non-LOCA event. The changes needed for the DCD departure WLS DEP 6.3-1 to incorporate this information include the following FSAR Sections or Tables:

Section 5.4.14.1
Section 6.3.1.1.1
Section 6.3.1.2
Section 6.3.1.3
Section 6.3.2.1.1
Section 6.3.3.4.1
Section 7.4.1.1
Table 9.5.1-201
Section 15.2.6.1,
Table 19.59-201
Section 19E.4.10.2

Departure Justification:

Recent PRHR HX evaluations performed under a variety of operating scenarios identified 14 days would be a conservative replacement time period for "indefinite". The Westinghouse evaluation of the PRHR HX operation under non-bounding, conservative conditions demonstrates the ability to keep the average RCS temperature in safe shutdown conditions for greater than 14 days under passive conditions (no operator action). The evaluation does indicate that if no action is taken, the average RCS temperature will increase at some point after 15 days but the PRHR HX operation would still keep the average RCS temperature below 420°F for a longer period of time of approximately 20 days (420°F is identified as the RCS temperature objective for safe shutdown). If no action is able to be taken after this period of time and there is adverse trending of RCS conditions that might be indicative of leading to an unstable condition, the operators do still have the ability to initiate Automatic Depressurization System (ADS), go to open loop cooling and retain the plant in a stable condition.

Departure Evaluation:

This Tier 2 departure is associated with defining the term "indefinite" as a conservative but specific duration (greater than 14 days). The departure results in a change to the DCD that does not impact the required design function (i.e., containment cooling condensate return). Accordingly, it does not:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the plant-specific DCD.
2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety and previously evaluated in the plant-specific DCD.
3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the plant-specific DCD.

4. Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the plant-specific DCD.
5. Create a possibility for an accident of a different type than any evaluated previously in the plant-specific DCD.
6. Create a possibility for a malfunction of an SSC important to safety with a different result than any evaluated previously in the plant-specific DCD.
7. Result in a design basis limit for a fission product barrier as described in the plant-specific DCD being exceeded or altered.
8. Result in a departure from a method of evaluation described in the plant-specific DCD used in establishing the design bases or in the safety analyses.

This departure does not affect resolution of a severe accident issue identified in the plant-specific DCD. Therefore, this departure has no safety significance.

NRC Approval Requirement:

This departure does not require NRC approval pursuant to 10 CFR Part 52, Appendix D, Section VIII.B.5.

4. COLA Part 7, Departures and Exemption Requests, is revised to add the following departure to the table presented in Section A.2, Departures That Require NRC Approval Prior to Implementation as follows:

Departure Number	**Description**
WLS DEP 3.2-1	Addition of downspouts to the condensate return portion of the Passive Core Cooling System

5. COLA Part 7, Departures and Exemption Requests, is revised to add the following departure to Section A.2 as follows:

Departure WLS DEP 3.2-1 is a departure from AP1000 Tier 1 information, in addition to Tier 2 information in the DCD; an exemption request and NRC approval is required prior to implementation.

Departure Number WLS DEP 3.2-1:

Affected DCD/FSAR Sections: Tier 1 Table 2.2.3-1 and Table 2.2.3-2, Tier 2 Table 3.2-3 (Sheet 16 of 75), Figure 3.8.2-1 (Sheet 3), Subsections 5.4.11.2 and 5.4.14.1, Subsections 6.3.1.1.1, 6.3.1.1.4, 6.3.1.1.6, 6.3.1.2, 6.3.1.3, 6.3.2.1, 6.3.2.1.1, 6.3.2.2.7, 6.3.2.8, 6.3.3, and 6.3.3.2.1.1, Chapter 6, Figure 6.3-1 (Sheets 1 through 3), Figure 6.3-2 (Not Used), Subsection 7.4.1.1, Table 14.3-2 (Sheets 7 and 8 of 17), Subsection 15.0.13, Chapter 16 (TS Bases B 3.3.3 and B 3.5.4), Subsections 19E.4.10.2 and 19E.9, Table 19E.4.10-1, and Figures 19E.4.10-1 through 19E.4.10-4.

Summary of Departure:

Modifications to the Polar Crane Girder (PCG), Internal Stiffener, and Passive Core Cooling System (PXS) gutters were made. The fabrication holes at the top surface of the

PCG and in the stiffener are blocked, drainage holes in the bottom of the PCG boxes are blocked, and flow communication holes between PCG boxes are added. A downspout piping network is added to collect and transport condensation from the top and interior of the PCG and the stiffener to the PXS Collection Boxes. Eight new PXS downspout screens are added at the entrance of each of the downspouts at the top of the PCG and the stiffener to prevent any larger debris from blocking the downspout piping. Visual inspection requirements to verify that the return flow to the IRWST will not be restricted by debris have been added to Technical Specification Bases.

Scope/Extent of Departure:

Upon actuation of the Passive Residual Heat Removal Heat Exchanger (PRHR HX), a series of air-operated valves are actuated to isolate the normal gutter drain path to the Liquid Radwaste System, and divert condensation to the In-containment Refueling Water Storage Tank (IRWST). It is important that sufficient condensate return is achieved during non-loss of coolant accident (LOCA) PRHR HX operation, since reduction of IRWST level to below the top of the tubes will begin to degrade the heat exchanger performance to the point where safe shutdown (<420 deg F in <36 hours) may not be achieved.

As steaming in the containment begins, following initiation of 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:

a. Steam to pressurize the containment
b. Steam condensation on Passive Heat Sinks
c. Raining from the containment roof, Containment ring misalignment
d. Losses at the Polar Crane Girder and Stiffener
e. Losses at support plates attached to the containment vessel
f. Losses at the Equipment Hatch and Personnel Airlock
g. Losses at entry to IRWST gutter

Losses due to pressurization and condensation on heat sinks are quantified with development of two new calculations. Two additional existing calculations have been revised based on the results of the new calculations in order to quantify the PRHR HX performance with the revised value of the condensate return and to ensure that the safe shutdown requirements are met. A full scale section of the containment wall was constructed to test condensate losses.

As a result of the condensate return testing, modifications to the Polar Crane Girder (PCG), Internal Stiffener, and Passive Core Cooling System (PXS) gutter designs are made. The fabrication holes at the top surface of the PCG and in the stiffener are blocked, drainage holes in the bottom of the PCG boxes are blocked, and flow communication holes between PCG boxes are added. A downspout piping network is added to collect and transport condensation from the top and interior of the PCG and the stiffener to the PXS Collection Boxes. Eight new PXS downspout screens are added at the entrance of each of the downspouts at the top of the PCG and the stiffener to prevent any larger debris from blocking the downspout piping. Visual inspection requirements to verify that the return flow to the IRWST will not be restricted by debris have been added to Technical Specification Bases.

Departure Justification:

The proposed change does not involve a significant reduction in the margin of safety. The proposed change does not reduce the redundancy or diversity of any safety-related SSCs. The proposed changes increase the amount of condensate available in the IRWST after the initiation of a design basis event compared to the design described in the AP1000 DCD Revision 19. Though the fraction of condensate returned is smaller than originally assumed, the proposed changes provide sufficient condensate return flow to maintain adequate IRWST water level for those events using the PRHR HX cooling function. While lower condensate return rates result in an earlier transition to PRHR HX uncovery, the long-term shutdown temperature evaluation results show that the PRHR HX would continue to meet its acceptance criteria.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that 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) approval of the change will not be inimical to the common defense and security or to the health and safety of the public.

Departure Evaluation:

This Tier 2 departure performs modifications to the PCG, Internal Stiffener, and PXS gutter designs. The fabrication holes at the top surface of the PCG and in the stiffener are blocked, drainage holes in the bottom of the PCG boxes are blocked, and flow communication holes between PCG boxes are added. A downspout piping network is added to collect and transport condensation from the top and interior of the PCG and the stiffener to the PXS Collection Boxes. Eight new PXS downspout screens are added at the entrance of each of the downspouts at the top of the PCG and the stiffener to prevent any larger debris from blocking the downspout piping. Visual inspection requirements to verify that the return flow to the IRWST will not be restricted by debris have been added to Technical Specification Bases. The proposed change does not involve a significant reduction in the margin of safety. The proposed change does not reduce the redundancy or diversity of any safety-related SSCs. The proposed changes increase the amount of condensate available in the IRWST after the initiation of a design basis event compared to the original design. Though the fraction of condensate returned is less than assumed in the original design, the proposed design does not result in significantly degraded overall PXS performance, in that the ability to achieve safe shutdown within the required time frame is accomplished. Therefore, this departure does not:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the plant-specific DCD.
2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety and previously evaluated in the plant-specific DCD.
3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the plant-specific DCD.
4. Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the plant-specific DCD.

5. Create a possibility for an accident of a different type than any evaluated previously in the plant-specific DCD.
6. Create a possibility for a malfunction of an SSC important to safety with a different result than any evaluated previously in the plant-specific DCD.
7. Result in a design basis limit for a fission product barrier as described in the plant-specific DCD being exceeded or altered.
8. Result in a departure from a method of evaluation described in the plant-specific DCD used in establishing the design bases or in the safety analyses. This departure does not affect resolution of a severe accident issue identified in the plant-specific DCD. Therefore, this departure has no safety significance.

NRC Approval Requirement:

This departure requires an exemption from the requirements of 10 CFR Part 52, Appendix D, Section III.B, which requires compliance with Tier 1 requirements of the AP1000 DCD. Therefore, an exemption is requested in Part B of this COL Application Part.

6. COLA Part 7, Departures and Exemption Requests, will be revised to add the following Exemption to Section B as follows:

B. Lee Nuclear Station Exemption Requests

Duke Energy Carolinas, Inc (DEC) requests the following exemptions related to:

3) Containment Cooling Changes in regard to Passive Core Cooling System Condensate Return

7. COLA Part 7, Departures and Exemption Requests, will be revised to add the following Exemption to Section B, under the discussion and justifications as follows:

3) Containment Cooling Changes in regard to Passive Core Cooling System Condensate Return

Applicable Regulation(s): 10 CFR Part 52 Appendix D, Section III.B
Specific wording from which exemption is requested:

"III. Scope and Contents

B. An applicant or licensee referencing this appendix, in accordance with Section IV of this appendix, shall incorporate by reference and comply with the requirements of this appendix, including Tier 1, Tier 2 (including the investment protection short-term availability controls in Section 16.3 of the DCD), and the generic TS except as otherwise provided in this appendix. Conceptual design information in the generic DCD and the evaluation of severe accident mitigation design alternatives in appendix 1B of the generic DCD are not part of this appendix."

Pursuant to 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 requested for plant-specific Tier 1 material departures from the AP1000 DCD for Tier 1 information. These material departures are contained in Tier 1 Subsection 2.2.3, Tables 2.2.3-1 and 2.2.3-2, and involve the addition of components to the condensate return design to enable the Passive Core Cooling System to more effectively perform its design functions. This exemption request is in accordance with the provisions of 10 CFR §50.12, 10 CFR §52.7, and 10 CFR Part 52, Appendix D.

Discussion:

The changes requested to Tier 1 Table 2.2.3-1 and Table 2.2.3-2 and associated Tier 2 changes to Table 3.2-3, Figure 3.8.2-1, Subsections 5.4.11.2 and 5.4.14.1, Subsections 6.3.1.1.1, 6.3.1.1.4, 6.3.1.1.6, 6.3.1.2, 6.3.1.3, 6.3.2.1, 6.3.2.1.1, 6.3.2.2.7, 6.3.2.8, 6.3.3, and 6.3.3.2.1.1, Figures 6.3-1 and 6.3-2, Subsection 7.4.1.1, Table 14.3-2, Subsection 15.0.3, Technical Specification Bases B 3.3.3 and B 3.5.4, Subsections 19E.4.10.2 and 19E.9, Table 19E.4.10-1, and Figures 19E.4.10-1 through 19E.4.10-4 provide additional equipment and surveillance requirements, provide reasonable assurance that the facility has been constructed and will be operated in conformity with the applicable design criteria, codes and standards, and demonstrate acceptable Passive Core Cooling System (PXS) system performance during design basis scenarios.

Conclusion:

This exemption request is evaluated in accordance with 10 CFR Part 52, Appendix D, Section VIII.A.4, 10 CFR §50.12, 10 CFR §52.7 and 10 CFR §52.63, which state that the NRC may grant exemptions from the requirements of the regulations provided the following 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, Appendix D, VIII.A.1]. 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 with proper justification. 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 applicant, 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 will ensure the protection of the health and safety of the public. Therefore, no adverse safety impact which would present any additional risk to the health and safety is present. The affected Design Description in the plant-specific Tier 1 DCD will 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 relating to the condensate return portion of the passive core cooling system. 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) lists 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 from Tier 1 Subsection 2.2.3, Tables 2.2.3-1 and 2.2.3-2, is 10 CFR 52, Appendix D, Section III.B, which requires that an applicant 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 WLS Units 1 and 2 COLA references the AP1000 Design Certification Rule and incorporates 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 to maintain the level of safety in the design.

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 in Tier 1 Subsection 2.2.3, Tables 2.2.3-1 and 2.2.3-2, will enable the applicant to safely construct and operate the AP1000 facility consistent with the design certified by the NRC in 10 CFR 52, Appendix D.

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

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

Based on the nature of the changes to the plant-specific Tier 1 information and the understanding that these changes support the design function of the Passive Core Cooling System, it is likely that other AP1000 applicants and licensees will request this exemption. However, if this is not the case, the special circumstances continue to outweigh any decrease in safety from the reduction in standardization because the key design functions of the Passive Core Cooling System associated with this request will continue to be maintained. This exemption request and the associated marked-up tables demonstrate that the Passive Core Cooling System function continues to be maintained following implementation of the change from the generic AP1000 DCD, thereby minimizing the safety impact resulting from any reduction in standardization.

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 Condition 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 adding components to Subsection 2.2.3, Tables 2.2.3-1 and 2.2.3-2, which were added to the condensate return design to enable the Passive Core Cooling System to more effectively perform its design functions. Because these functions are met, there is no reduction in the level of safety.

Therefore, the design change will not result in a significant decrease in the level of safety.

As demonstrated above, this exemption request satisfies NRC requirements for an exemption to the design certification rule for the AP1000.

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Condensate Return

Attachment 4

Revisions to Part 10, Proposed License Conditions (Including ITAAC)

1. COLA Part 10, Appendix B, Inspections, Tests, Analyses and Acceptance Criteria, is revised prior to the information on Physical Security ITAAC as follows:

Passive Containment Cooling System ITAAC

Passive Containment Cooling system components are added to support the capability of the Passive Residual Heat Removal Heat Exchanger (PRHR HX) to enable the reactor to achieve a safe shutdown condition of 420° F within 36 hours. Component numbers for downspout screens are added to DCD Tier 1 Table 2.2.3-1 and component numbers for downspout piping are added to DCD Tier 1 Table 2.2.3-2 to provide assurance that ITAAC design commitments will be met. These tables, with the subject component numbers added, are provided in the attached Tables 2.2.3-1 and 2.2.3-2, with an LMA of WLS DEP 3.2-1.

2. Part 10, Appendix B, Inspections, Tests, Analyses and Acceptance Criteria, insert the attached three pages from DCD Tier 1 Tables 2.2.3-1 and 2.2.3-2 containing the component numbers for downspout screens and downspout piping associated with WLS DEP 3.2-1 prior to the Table 2.6.9-2 Physical Security ITAAC as follows:

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**
Passive Residual Heat Removal Heat Exchanger (PRHR HX)	PXS-ME-01	Yes	Yes	-	- / -	-	- / -	-	-
Accumulator Tank A	PXS-MT-01A	Yes	Yes	-	- / -	-	- / -	-	-
Accumulator Tank B	PXS-MT-01B	Yes	Yes	-	- / -	-	- / -	-	-
Core Makeup Tank (CMT) A	PXS-MT-02A	Yes	Yes	-	- / -	-	- / -	-	-
CMT B	PXS-MT-02B	Yes	Yes	-	- / -	-	- / -	-	-
IRWST	PXS-MT-03	No	Yes	-	- / -	-	- / -	-	-
IRWST Screen A	PXS-MY-Y01A	No	Yes	-	- / -	-	- / -	-	-
IRWST Screen B	PXS-MY-Y01B	No	Yes	-	- / -	-	- / -	-	-
IRWST Screen C	PXS-MY-Y01C	No	Yes	-	- / -	-	- / -	-	-
Containment Recirculation Screen A	PXS-MY-Y02A	No	Yes	-	- / -	-	- / -	-	-
Containment Recirculation Screen B	PXS-MY-Y02B	No	Yes	-	- / -	-	- / -	-	-
pH Adjustment Basket 3A	PXS-MY-Y03A	No	Yes	-	- / -	-	- / -	-	-
pH Adjustment Basket 3B	PXS-MY-Y03B	No	Yes	-	- / -	-	- / -	-	-
pH Adjustment Basket 4A	PXS-MY-Y04A	No	Yes		- / -		- / -		
pH Adjustment Basket 4B	PXS-MY-Y04B	No	Yes		- / -		- / -		
Downspout Screen 1A	PXS-MY-Y81	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 1B	PXS-MY-Y82	No	Yes	-	- / -	-	- / -	-	-

WLS DEP 3.2-1

Note: Dash (-) indicates not applicable.

WLS DEP 3.2-1

TABLE 2.2.3-1 (cont.)

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 1C	PXS-MY-Y83	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 1D	PXS-MY-Y84	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2A	PXS-MY-Y85	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2B	PXS-MY-Y86	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2C	PXS-MY-Y87	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2D	PXS-MY-Y88	No	Yes	-	- / -	-	- / -	-	-
CMT A Inlet Isolation Motor-operated Valve	PXS-PL-V002A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/No	None	As Is
CMT B Inlet Isolation Motor-operated Valve	PXS-PL-V002B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/No	None	As Is
CMT A Discharge Isolation Valve	PXS-PL-V014A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
CMT B Discharge Isolation Valve	PXS-PL-V014B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
CMT A Discharge Isolation Valve	PXS-PL-V015A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
CMT B Discharge Isolation Valve	PXS-PL-V015B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
CMT A Discharge Check Valve	PXS-PL-V016A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-

Note: Dash (-) indicates not applicable.

TABLE 2.2.3-2 (cont.)				
Line Name	Line Number	ASME Code Section III	Leak Before Break	Functional Capability Required
IRWST screen cross-connect line	PXS-L180A, PXS-L180B	Yes	No	Yes
Containment recirculation line A	PXS-L113A, PXS-L131A, PXS-L132A	Yes	No	Yes
Containment recirculation line B	PXS-L113B, PXS-L131B, PXS-L132B	Yes	No	Yes
IRWST gutter drain line	PXS-L142A, PXS-L142B	Yes	No	Yes
	PXS-L141A, PXS-L141B	Yes	No	No
WLS DEP 3.2-1 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

Duke Energy Voluntary Submittal

Condensate Return

William States Lee III COL Application

Enclosure 8

Levy Docketed Requests for Additional Information Responses Regarding Passive Core Cooling System (PXS) Condensate Return

Levy Nuclear Plant, Units 1 and 2
Responses to NRC Requests for Additional Information
Letter No. 116 Related to SRP Sections 15.02.06 and 06.03, dated March 6, 2014,
Letter No. 117 Related to SRP Section 06.03, dated April 10, 2014, and
Letter No. 118 Related to SRP Section 06.03, dated April 24, 2014

NRC RAI No.	Duke Energy RAI No.	Levy Nuclear Plant Response
15.02.06-1	L-1081	NPD-NRC-2014-017, dated June 19, 2014
	L-1106	NPD-NRC-2014-024, dated July 24, 2014
15.02.06-2	L-1082	NPD-NRC-2014-021, dated June 27, 2014
15.02.06-3	L-1085	NPD-NRC-2014-017, dated June 19, 2014
06.03-1	L-1086	NPD-NRC-2014-014, dated May 5, 2014
06.03-2	L-1087	NPD-NRC-2014-016, dated June 12, 2014
06.03-3	L-1088	NPD-NRC-2014-016, dated June 12, 2014
06.03-4	L-1089	NPD-NRC-2014-022, dated July 1, 2014
06.03-5	L-1090	NPD-NRC-2014-021, dated June 27, 2014
06.03-6	L-1091	NPD-NRC-2014-014, dated May 5, 2014
06.03-7	L-1092	NPD-NRC-2014-012, dated April 17, 2014
06.03-8	L-1093	NPD-NRC-2014-012, dated April 17, 2014
06.03-9	L-1094	NPD-NRC-2014-015, dated May 19, 2014
06.03-10	L-1096	NPD-NRC-2014-021, dated June 27, 2014
06.03-11	L-1097	NPD-NRC-2014-021, dated June 27, 2014
06.03-12	L-1099	NPD-NRC-2014-021, dated June 27, 2014

NPD-NRC-2014-028, dated July 24, 2014, (Reference 12 on page 5 of 5) contains supplemental information addressing RAIs 06.03-10 through 06.03-12.

The following Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, letters have been reviewed and found to be applicable to William States Lee III Nuclear Station Units 1 and 2 for the Duke Energy Carolinas voluntary submittal of exemption request and design change description for departure from AP1000 DCD Revision 19 to address Containment Condensate Return Cooling design.

1. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-012, dated April 17, 2014 (ADAMS Accession No. ML14112A371), Enclosures 1, 2, 3, and 4 are found to be applicable to William States Lee III Nuclear Station Units 1 and 2. The referenced enclosures address RAIs 06.03-7 and 06.03-8 (contains proprietary information).

2. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-014, dated May 5, 2014 (ADAMS Accession No. ML14126A699), Enclosures 1, 2, and 3 are found to be applicable to William States Lee III Nuclear Station Units 1 and 2. The referenced enclosures address RAIs 06.03-1 and 06.03-6 (contains proprietary information).

3. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-015, dated May 19, 2014 (ADAMS Accession No. ML14141A015), Enclosure 1 is found to be applicable to William States Lee III Nuclear Station Units 1 and 2. The referenced enclosure addresses RAI 06.03-9.

4. Duke Energy, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-016, dated June 12, 2014 (ADAMS Accession No. ML14164A444), Enclosures 1, 2, and 3 are found to be applicable to William States Lee III Nuclear Station Units 1 and 2. The referenced enclosures address RAIs 06.03-2 and 06.03-3 (contains proprietary information).

5. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-017, dated June 19, 2014 (ADAMS Accession No. ML14171A453), Enclosure 1 is found to be applicable to William States Lee III Nuclear Station Units 1 and 2. The referenced enclosure addresses RAIs 15.02.06-1 and 15.02.06-3.

6. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-021, dated June 27, 2014 (ADAMS Accession No. ML14182A106), Enclosures 1, 2, 3, and 4 are found to be applicable to William States Lee III Nuclear Station Units 1 and 2. The referenced enclosures address RAIs 15.02.06-2, 06.03-5, 06.03-10, 06.03-11 and 06.03-12 (contains proprietary information).

7. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-022 (note that page 1 of the cover letter incorrectly identifies this letter as NPD-NRC-2014-021), dated July 1, 2014 (ADAMS Accession No.

ML14183B342), Enclosure 1 is found to be applicable to William States Lee III Nuclear Station Units 1 and 2. The referenced enclosure addresses RAI 06.03-4.

8. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-024, dated July 24, 2014 (ADAMS Accession No. ML14206A951), Enclosures 1, 2, and 3 are found to be applicable to William States Lee III Nuclear Station Units 1 and 2. The referenced enclosures address RAI 15.02.06-1 (contains proprietary information).

9. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, Docket Nos. 52-029 and 52-030, Letter NPD-NRC-2014-028, dated July 24, 2014 (ADAMS Accession No. ML14206A953), Enclosures 1, 2, 3, and 4 are found to be applicable to WLS Nuclear Station Units 1 and 2. The referenced letter contains supplemental information addressing RAIs 06.03-10 through 06.03-12 (contains proprietary information).

References:

1. Letter from Donald Habib (NRC) to Christopher M. Fallon (DEF), dated March 6, 2014, "Request for Additional Information Letter No. 116 Related to SRP Sections 6.3 and 15.2.6." [ML14065A362]

2. Letter from Donald Habib (NRC) to Christopher M. Fallon (DEF), dated April 10, 2014, "Request for Additional Information Letter No. 117 Related to SRP Section 6.3." [ML14100A040]

3. Letter from Donald Habib (NRC) to Christopher M. Fallon (DEF), dated April 24, 2014, "Request for Additional Information Letter No. 118 Related to SRP Section 6.3." [ML14114A050]

4. Letter from Christopher Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated April 17, 2014, "Partial Response to NRC RAI Letter 116—SRP Sections 6.3 and 15.2.6", Serial: NPD-NRC-2014-012 [ML14112A371]

5. Letter from Christopher Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated May 5, 2014, "Partial Response to NRC RAI Letter 116 – SRP Sections 6.3 and 15.2.6", Serial: NPD-NRC-2014-014 [ML14126A699]

6. Letter from Christopher Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated May 19, 2014, "Partial Response to NRC RAI Letter 116 – SRP Sections 6.3 and 15.2.6", Serial: NPD-NRC-2014-015 [ML14141A015]

7. Letter from Christopher Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated June 12, 2014, "Partial Response to NRC RAI Letter 116 – SRP Sections 6.3 and 15.2.6", Serial: NPD-NRC-2014-016 [ML14164A444]

8. Letter from Christopher Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated June 19, 2014, "Partial Response to NRC RAI Letter 116 – SRP Sections 6.3 and 15.2.6", Serial: NPD-NRC-2014-017 [ML14171A453]

9. Letter from Christopher Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated June 27, 2014, "Supplemental Response to NRC RAI Letter 116" - SRP Sections 6.3 and 15.2.6, Serial: NPD-NRC-2014-021 [ML14182A106]

10. Letter from Christopher Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated July 1, 2014, "Supplemental Response to NRC RAI Letter 116" - SRP Sections 6.3 and 15.2.6, Serial: NPD-NRC-2014-022 [ML14183B342]

11. Letter from Christopher Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated July 24, 2014, "Supplemental Response to NRC RAI Letter 116" - SRP Sections 6.3 and 15.2.6, Serial: NPD-NRC-2014-024 [ML14206A951]

12. Letter from Christopher Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated July 24, 2014, "Supplement to Partial Response to NRC RAI Letters 116, 117 and 118 – SRP Sections 6.3 AND 15.2.6", Serial: NPD-NRC-2014-028 [ML14206A953]