

SAFETY EVALUATION BY THE OFFICE OF NEW REACTORS

RELATED TO AMMENDMENT NO. 61

TO THE COMBINED LICENSE NOS. NPF-93 AND NPF-94

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

VIRGIL C. SUMMER NUCLEAR STATION UNITS 2 AND 3

DOCKET NOS. 52-027 AND 52-028

1.0 INTRODUCTION

By letter dated November 18, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16323A335), South Carolina Electric & Gas Company (SCE&G), on behalf of itself and the South Carolina Public Service Authority (both, hereafter, called the licensee), submitted license amendment request (LAR) 16-06 requesting that the U.S. Nuclear Regulatory Commission (NRC) amend the combined licenses (COLs) for Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3, COL Numbers NPF-93 and NPF-94, respectively.

The proposed amendment requests to depart from approved AP1000 Design Control Document (DCD) Tier 2 information (text, tables, and figures) as incorporated into the Updated Final Safety Analysis Report (UFSAR) as plant-specific DCD information, as well as, departures from involved plant-specific Tier 1 information (and associated COL Appendix C information) and from involved plant-specific Technical Specifications (PS-TS) as incorporated in Appendix A of the COL. Specifically, the request included changes to Tier 2 information in Chapters 1, 3, 5, 6, 7, 9, 14, 15, and 19 of the UFSAR. With the exceptions discussed below in the technical evaluation section, the request submitted by the licensee is essentially identical from a technical perspective to the exemption and departure request submitted by Duke Energy Florida (DEF) for the Levy Nuclear Plant (LNP) COL application (COLA). The DEF departure request addressed changes to passive core cooling system (PXS) condensate return. The staff's review of the LNP exemption and departure request to address PXS condensate return is provided in Section 21.1 of the staff's final safety evaluation report (FSER) for the LNP COLA review (ADAMS Accession No. ML16068A418). In LAR 16-06, Enclosure 5, "Applicability and Endorsement of Prior Docketed Information," SCE&G included a list of Technical Reports and responses to staff's request for additional information (RAIs) that are applicable supporting documents for the changes proposed in this LAR and are hereby incorporated by reference. For the portions of the evaluation where no technical changes were made to the submittal, the technical evaluation below reproduces the findings made by the staff in the FSER for the LNP COLA.

SCE&G has also requested a permanent exemption from the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, Appendix D, Section III.B, "Design Certification Rule

for the AP1000 Design, Scope and Contents,” to allow a departure from the elements of the certification information in Tier 1 of the plant-specific AP1000 DCD.¹

In order to modify UFSAR (the plant-specific DCD) Tier 1 information, the NRC must find the licensee’s exemption request included in its submittal for the LAR acceptable. The staff’s review of both the exemption request and the license amendment request are included in this safety evaluation.

The staff’s proposed no significant hazards consideration determination was published in the *Federal Register* on December 15, 2016 (81 FR 90871).

2.0 REGULATORY EVALUATION

The proposed changes in LAR 16-06 are required to meet the following 10 CFR Part 50, Appendix A General Design Criteria (GDC), which apply to the COLs for Summer Units 2 and 3:

10 CFR Part 50, Appendix A, GDC 34, “Residual heat removal,” as it applies to the capability of the Passive Residual Heat Removal (PRHR) Heat Exchanger (HX) to perform safety related safe shutdown cooling of the reactor coolant system (RCS),
10 CFR Part 50, Appendix A, GDC 44, “Cooling Water,” as it applies to the ability of the containment systems to transfer heat from the PRHR HX to the ultimate heat sink via the passive containment cooling system.

10 CFR 52.63(b)(1) permits a licensee who references a design certification rule to request NRC approval for an exemption from one or more elements of the certification information. The Commission may only grant such a request if it determines that the exemption will comply with the requirements of 10 CFR 52.7, which, in turn, points to the requirements listed in 10 CFR 50.12 for specific exemptions, and if the special circumstances present outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. Therefore, any exemption from the Tier 1 information certified by Appendix D to 10 CFR Part 52 must meet the requirements of 10 CFR 50.12, 52.7 and 52.63(b)(1).

10 CFR 52.98(f) states that any modification to, addition to, or deletion from the terms and conditions of a COL, including inspections, tests, analyses, or related acceptance criteria, is a proposed amendment to the license. LAR 16-06 involves a departure from plant-specific Tier 1 information, and corresponding change to COL Appendix C, Inspections, Tests, Analyses, and Acceptance Criteria information, and also involves a change to the plant-specific Technical Specifications as provided in COL Appendix A. Therefore, the proposed change requires a license amendment.

10 CFR Part 52, Appendix D, Section VIII.A.4 states that exemptions from Tier 1 information are governed by the requirements of 10 CFR 52.63(b)(1) and 52.98(f). It also states that the Commission will deny such a request if it finds that the design change will result in a significant decrease in the level of safety otherwise provided by the design.

¹ While the licensee describes the requested exemption as being from Section III.B of 10 CFR Part 52, Appendix D, the entirety of the exemption pertains to proposed departures from Tier 1 information in the generic DCD. In the remainder of this evaluation, the NRC will refer to the exemption as an exemption from Tier 1 information to match the language of Section VIII.A.4 of 10 CFR Part 52, Appendix D, which specifically governs the granting of exemptions from Tier 1 information.

10 CFR Part 52, Appendix D, Section VIII.B.5.a provides, among other things, that an applicant or licensee who references 10 CFR Part 52, Appendix D 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 Technical Specifications, or requires a license amendment under paragraphs B.5.b or B.5.c of 10 CFR Part 52, Appendix D, Section VIII.

3.0 TECHNICAL EVALUATION

LAR 16-06 addresses a standard issue outside the scope of the AP1000 design certification. NRC staff has previously conducted a thorough review of this issue during its review of the LNP COLA. To ensure that the staff's findings on standard content, documented in Section 21.1 of the FSER for the LNP COL review, were applicable to LAR 16-06, the staff undertook the following reviews:

- The staff compared the LNP COL FSAR, Revision 9 to the UFSAR with proposed updates under LAR 16-06, for Summer Units 2 and 3. In performing this comparison, the staff evaluated any differences in content to determine the impact on previously established findings.
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.
- The staff verified that the site-specific differences were insignificant and did not impact the staff's findings on standard content.

3.1 Differences in Technical Content

With the exceptions discussed below, the proposed changes to the licensing basis documents in LAR 16-06 are identical to the changes submitted by Duke Energy Florida under LNP Departure (DEP) 3.2-1 and LNP DEP 6.3-1.

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

LAR 16-06 did not include a change to Figure 3.8.2-1 because this change was previously made (this change did not require prior NRC approval in accordance with 10 CFR 52, Appendix D, paragraph VIII.B.5.a). Because Figure 3.8.2-1 in the Summer UFSAR is consistent with the same numbered figure in the LNP FSAR, the fact that no change was requested to Figure 3.8.2-1 in LAR 16-06 does not impact the staff's findings on standard content.

UFSAR Subsection 5.4.14.1

LAR 16-06 retains the following language from the AP1000 DCD, "the passive residual heat removal heat exchanger will prevent water relief from the pressurizer..." This language is consistent with the evaluation criteria for applicable Chapter 15 events. Thus, NRC staff concludes that this difference does not impact the staff's findings on standard content.

UFSAR Subsection 6.3.1.1.1

LAR 16-06 retains the following language from the AP1000 DCD, "The passive residual heat removal heat exchanger automatically actuates to ... prevent water relief through the pressurizer safety valves." This language is consistent with the evaluation criteria for applicable

Chapter 15 events. Thus, NRC staff concludes that this difference does not impact the staff's findings on standard content.

UFSAR Subsection 6.3.1.1.4

LAR 16-06 modified language from the standard content to clarify (1) the capability of the PXS to maintain stable plant conditions depends on normal plant systems, and (2) the Class 1E batteries discussed in UFSAR Subsection 6.3.1.1.4 are the 24 hour Class 1E batteries. The licensee stated, and NRC staff agrees, that these modifications are for clarification and have no substantive impact. Thus, NRC staff concludes that the differences do not affect the staff's findings on standard content.

UFSAR Subsection 6.3.2.1.1

NRC staff reviewed the proposed changes in LAR 16-06 and determined the technical content is identical to the standard content. Thus, NRC staff concludes that the differences do not impact the staff's findings on standard content.

UFSAR Subsection 6.3.2.8

LAR 16-06 modified language from the standard content to clarify that the batteries discussed in UFSAR Subsection 6.3.2.8 are the 24 hour Class 1E batteries. The licensee stated, and NRC staff agrees, that these modifications are for clarification and have no substantive impact. Thus, NRC staff concludes that this difference does not affect the staff's findings on standard content.

UFSAR Subsection 6.3.3.2.1.1

LAR 16-06 modified language from the standard content to clarify that the batteries discussed in UFSAR Subsection 6.3.3.2.1.1 are the 24 hour Class 1E batteries. The licensee stated, and NRC staff agrees, that these modifications are for clarification and have no substantive impact. Thus, NRC staff concludes this difference does not affect the staff's findings on standard content.

UFSAR Section 7.4

LAR 16-06 modified language from standard content to clarify a long term safe shutdown requires the core average temperature be less than 420°F. The licensee stated, and NRC staff agrees, that these modifications are for consistency in references to the 420°F criterion. Thus, NRC staff concludes that this difference does not impact the staff's findings on standard content.

UFSAR Subsection 7.4.1.1

LAR 16-06 modified language from the standard content to clarify that the batteries discussed in UFSAR Subsection 7.4.1.1 are the 24 hour Class 1E batteries. The licensee stated, and NRC staff agrees, that these modifications are for clarification and have no substantive impact. Thus, NRC staff concludes that this difference does not affect the staff's findings on standard content.

UFSAR Subsection 9.5, Table 9.5.1-1, Item 73

LAR 16-06 modified language from standard content to clarify a long term safe shutdown requires the core average temperature be less than 420°F. The licensee stated, and NRC staff agrees, that these modifications are for consistency in references to the 420°F criterion. Thus, NRC staff concludes that this difference does not impact the staff's findings on standard content.

UFSAR Subsection 15.2.6.1

LAR 16-06 modified language to remove the approximate time at which the in-containment refueling water storage tank (IRWST) reaches saturation temperature. NRC staff agrees that the time at which the IRWST reaches saturation temperature is not a Chapter 15 acceptance criterion and that the time for the IRWST to reach saturation following a loss of alternating current (ac) power event is provided in the UFSAR update to Table 19E.4.10-1. Thus, NRC staff concludes that this difference does not impact the staff's findings on standard content.

UFSAR Appendix 19E, Subsection 19E.4.10.2

LAR 16-06 modified language from standard content to clarify a long term safe shutdown requires the core average temperature be less than 420°F. The licensee stated, and NRC staff agrees, that these modifications are for consistency in references to the 420°F criterion. Thus, NRC staff concludes that this difference does not impact the staff's findings on standard content.

3.2 Endorsement of RAIs in Corresponding Standard Content

NRC staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed by the licensee, with one exception. The standard content evaluation describes how DEF, the Levy licensee, endorsed two RAI responses from Southern Nuclear Operating Company (SNC) (RAIs 7741 and 7756, responses in ADAMS Accession No. ML14338A658 and Accession No. ML15057A590) related to mitigation strategies for beyond design basis external events. SCE&G did not explicitly endorse those RAI responses in the same fashion; instead, they elected to provide a pair of letters with identical technical content to respond to the RAIs sent to SNC, as both licensees based their responses on the same Westinghouse report. SCE&G's responses are dated July 22, 2015 (ADAMS Accession No. ML15203A946), and March 31, 2016 (ADAMS Accession No. ML16091A483). The content of these responses serve as the basis for the staff's conclusion documented in the safety evaluation finding that the safety injection function of the PXS is not impacted by the LAR.

3.3 Site-Specific Differences

NRC staff verified that site specific characteristics for maximum safety air temperatures, provided in UFSAR Table 2.0-201, are bounded by the AP1000 DCD site characteristics. Thus, NRC staff concludes that there are no site-specific differences that impact their findings on standard content.

3.4 Standard Content

The staff has completed its review and found the evaluation performed for the standard content in the LNP FSER to be directly applicable to LAR 16-06. The standard content material is

identified in this SER by use of italicized, double-indented formatting. The following portion of this technical evaluation section is reproduced from Section 21.1.4 of the LNP COL application FSER (ADAMS Accession No. ML16068A418):

- *LNP DEP 3.2-1 and LNP DEP 6.3-1*

LNP DEP 3.2-1 proposes to change the PXS to increase the fraction of condensate returning to the IRWST when there is steam in the containment building. This change creates intermediate gutters at the top and bottom of the polar crane girder and at the containment shell intermediate ring stiffener. It blocks drain holes that were in these structures and adds dams where needed to collect condensate. It adds downspouts from these gutters to the IRWST. It also modifies the gutter drip lip so that condensate is not lost between the containment wall and the gutter. Condensate that is “lost” does not return to the IRWST, and instead drips off of the shell into various containment holdup volumes, such as the loop compartments or reactor vessel cavity.

LNP DEP 6.3-1 proposes additional changes to the FSAR in conjunction with the design changes described in LNP DEP 3.2-1 to clarify the duration of operation of the PRHR HX and separate the description of the safety functions from the non-safety design function of the PXS.

The staff reviewed a request for an exemption submitted by the applicant. The request proposed changes to Tier 1 Tables 2.2.3-1 and 2.2.3-2 and generic TS Surveillance Requirement (SR) 3.5.4.7 in the AP1000 DCD. Additionally, the staff reviewed the Tier 2 changes for potential effects on safety functions of the PXS and the associated Chapter 15 safety analyses, the safe-shutdown temperature evaluation in Chapter 19E, the seismic classification in Chapter 3, and the TS and Bases in Chapter 16. The regulatory evaluation of the exemption request appears in Subsection A, below, and the technical evaluation of the exemption request and departure appears in Subsection B, below.

A. Regulatory Evaluation of Exemption Request

A.1 Summary of Exemption

The applicant requested an exemption from the provisions of 10 CFR Part 52, Appendix D, Section III.B, “Design Certification Rule for the AP1000 Design, Scope and Contents,” that require the applicant referencing a certified design to incorporate by reference Tier 1 information. Specifically, the applicant proposed to revise Tier 1 Tables 2.2.3-1 and 2.2.3-2 by adding components to the condensate return design to enable the PXS to more effectively perform its design functions and revised TS SR 3.5.4.7 to address downspout screens.²

² While the applicant describes the requested exemption as being from Section III.B of 10 CFR Part 52, Appendix D, the entirety of the exemption pertains to proposed departures from Tier 1 information and generic TS in the generic DCD. In the remainder of this evaluation, the NRC will refer to the exemption as an exemption from Tier 1 information and generic TS to match the language of Sections VIII.A.4 and VIII.C.4 of 10 CFR Part 52, Appendix D, which specifically govern the granting of exemptions from Tier 1 information and generic TS.

A.2 Regulations

- *10 CFR Part 52, Appendix D, Section VIII.A.4 states that exemptions from Tier 1 information are governed by the requirements of 10 CFR 52.63(b) and 10 CFR 52.98(f). It also states that the Commission may deny such a request if the design change causes a significant reduction in plant safety otherwise provided by the design. This subsection of Appendix D also provides that a design change requiring a Tier 1 change shall not result in a significant decrease in the level of safety otherwise provided by the design.*
- *10 CFR Part 52, Appendix D, Section VIII.C.4 states that an applicant may request an exemption from the generic TS or other operational requirements. The Commission may grant such a request only if it determines that the exemption will comply with the requirements of 10 CFR 52.7.*
- *10 CFR 52.63(b)(1) allows an applicant or licensee to request NRC approval for an exemption from one or more elements of the certification information. The Commission may only grant such a request if it complies with the requirements of 10 CFR 52.7 which in turn points to the requirements listed in 10 CFR 50.12 for specific exemptions, and if the special circumstances present outweigh the potential decrease in safety due to reduced standardization. Therefore, any exemption from the Tier 1 information certified by Appendix D to 10 CFR Part 52 must meet the requirements of 10 CFR 50.12, 52.7, and 52.63(b)(1).*

A.3 Evaluation of Exemption

As stated in Section VIII.A.4 of Appendix D to 10 CFR Part 52, an exemption from Tier 1 information is governed by the requirements of 10 CFR 52.63(b)(1) and 52.98(f). Additionally, the Commission will deny an exemption request if it finds that the requested change to Tier 1 information will result in a significant decrease in safety. Pursuant to 10 CFR 52.63(b)(1), the Commission may, upon application by an applicant or licensee referencing a certified design, grant exemptions from one or more elements of the certification information, so long as the criteria given in 10 CFR 50.12 are met and the special circumstances as defined by 10 CFR 50.12 outweigh any potential decrease in safety due to reduced standardization.

As stated in Section VIII.C.4 of Appendix D to 10 CFR Part 52, the Commission may grant an exemption from generic TS of the DCD only if it determines that the exemption will comply with the requirements of 10 CFR 52.7. As stated above, Section 52.7 points to 10 CFR 50.12 for specific exemptions.

Applicable criteria for when the Commission may grant the requested specific exemption are provided in 10 CFR 50.12(a)(1) and (a)(2). Section 50.12(a)(1) provides that the requested exemption must be authorized by law, not present an undue risk to the public health and safety, and be consistent with the common defense and security. The provisions of 10 CFR 50.12(a)(2) list six special circumstances for which an exemption may be granted. It is necessary for one of

these special circumstances to be present in order for NRC to consider granting an exemption request. The applicant stated that the requested exemption meets the special circumstances of 10 CFR 50.12(a)(2)(ii). That subsection defines special circumstances as when “[a]pplication 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 staff’s analysis of each of these findings is presented below.

A.3.1 Authorized by Law

This exemption would allow the applicant to implement approved changes to Tier 1 Tables 2.2.3-1 and 2.2.3-2 and generic TS SR 3.5.4.7. This is a permanent exemption limited in scope to particular Tier 1 information and generic TS, and subsequent changes to this information or any other Tier 1 information or generic TS would be subject to full compliance with the change processes specified in Sections VIII.A.4 and VIII.C.4 of Appendix D to 10 CFR Part 52. As stated above, 10 CFR 52.63(b)(1) allows the NRC to grant exemptions from one or more elements of the certification information, namely, as discussed in this exemption evaluation, the requirements of Tier 1. Moreover, Section VIII.C.4 allows the NRC to grant exemptions from generic TS if the exemption meets the requirements of 10 CFR 52.7 and 50.12. The NRC staff has determined that granting of the applicant’s proposed exemption will not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission’s regulations. Therefore, as required by 10 CFR 50.12(a)(1), the exemption is authorized by law.

A.3.2 No Undue Risk to Public Health and Safety

The underlying purpose of AP1000 Tier 1 Tables 2.2.3-1 and 2.2.3-2 and generic TS SR 3.5.4.7 is to ensure that the plant will be constructed and operated with a safe and reliable condensate return system in the event of an accident.

Additions to the condensate return portion of the passive core cooling system improve the reliability and effectiveness of the condensate return system; these additions to the system, therefore, support the system’s intended design functions. The plant-specific Tier 1 DCD and TS will continue to reflect the approved licensing basis for the applicant and will maintain a level of detail consistent with that which is provided elsewhere in Tier 1 of the plant-specific DCD. The affected design description in the plant-specific Tier 1 DCD provides the detail to support the performance of the associated ITAAC. The proposed changes to Tier 1 information and generic TS are evaluated and found to be acceptable in Section 6.3 of this safety evaluation. Therefore, the staff finds the exemption presents no undue risk to public health and safety as required by 10 CFR 50.12(a)(1).

A.3.3 Consistent with Common Defense and Security

The proposed exemption would allow the applicant to implement modifications to the Tier 1 information and generic TS requested in the applicant’s submittal. This is a permanent exemption limited in scope to particular Tier 1 information and a specific TS. Subsequent changes to this information or any other Tier 1 information or generic TS would be subject to full compliance with the change

processes specified in Sections VIII.A.4 and VIII.C.4 of Appendix D to 10 CFR Part 52. This change is not related to security issues. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the exemption is consistent with the common defense and security.

A.3.4 Special Circumstances

Special circumstances, in accordance with 10 CFR 50.12(a)(2)(ii), are present whenever application of the regulation in the particular circumstances would not serve the underlying purposes of the rule or is not necessary to achieve the underlying purpose of the rule. The underlying purpose of the specific Tier 1 Tables 2.2.3-1 and 2.2.3-2 and TS SR 3.5.4.7 being modified in the exemption request is to identify and conduct surveillances of the components that will be added to the design of the condensate return portion of the passive core cooling system. The additional components and new surveillance requirements for those components are needed so that the passive core cooling system can perform its intended function, that is, to bring the reactor coolant system to safe shutdown conditions during certain non-loss-of-coolant-accident events.

Application of the requirements in Tier 1 Tables 2.2.3-1 and 2.2.3-2 and generic TS SR 3.5.4.7 is not necessary to achieve the underlying purpose of those portions of the rule. The proposed additions to the condensate return portion of the passive core cooling system support the system's intended design functions, as does the addition of a generic TS to conduct surveillances of those additional components. The system and tables listing its components and surveillances, as modified in the requested exemption, will continue to perform their intended functions and will, therefore, meet the underlying purposes of the rule. Accordingly, because application of the requirements in Tier 1 Tables 2.2.3-1 and 2.2.3-2 and the generic TS SR 3.5.4.7 is not necessary to achieve the underlying purpose of the rule, special circumstances are present. Therefore, the staff finds that special circumstances exist as required by 10 CFR 50.12(a)(2)(ii) for the granting of an exemption from the Tier 1 information and generic TS described above.

A.3.5 Special Circumstances Outweigh Reduced Standardization

This exemption, if granted, would allow the applicant to change certain Tier 1 information incorporated by reference from the AP1000 DCD into the LNP COL application. An exemption from Tier 1 information may only be granted if the special circumstances of the exemption request, required to be present under 10 CFR 52.7 and 10 CFR 50.12, outweigh any reduction in standardization. The proposed exemption would modify the condensate return portion of the passive core cooling system to improve the reliability and effectiveness of the condensate return system. The proposed additions to the system support the system's intended design functions and the key design functions of the passive core cooling system will be maintained.³

³ Based on the nature of the proposed changes to the generic Tier 1 information in Tables 2.2.3-1 and 2.2.3-2 and TS SR 3.5.4.7, both of which maintain and support the design functions of the passive core cooling system, other AP1000 licensees and applicants may request the same exemption, preserving the intended level of standardization.

As described below in the technical evaluation, the changes to the condensate return system (1) ensure the capability of the PRHR HX to maintain the RCS in a safe, stable condition, as described in DCD Chapter 19E, "Shutdown Temperature Evaluation," and (2) demonstrate the existing non-loss-of-coolant accident (LOCA) analyses in Chapter 15 that credit the PRHR HX remain valid. Consequently, while there is a small possibility that standardization may be slightly reduced by the granting the exemption from the specified Tier 1 requirements, the proposed exemption modifying the condensate return portion of the passive core cooling system will improve the reliability and effectiveness of the condensate return system, to better allow the system to perform its intended function. For this reason, the staff determined that even if other AP1000 licensees and applicants do not request similar departures, the special circumstances supporting this exemption outweigh the potential decrease in safety due to reduced standardization of the AP1000 design, as required by 10 CFR 52.63(b)(1).

A.3.6 No Significant Reduction in Safety

The proposed exemption would modify the passive core cooling system from the design presented in the original application. As described below in the technical evaluation, the changes to the condensate return system (1) ensure the capability of the PRHR HX to maintain the RCS in a safe, stable condition, as described in DCD Chapter 19E, "Shutdown Temperature Evaluation," and (2) demonstrate the existing non-LOCA analyses in Chapter 15 that credit the PRHR HX remain valid. The proposed changes to the PXS design will increase the reliability of the system, maintain its key design functions, and will not adversely affect its function. Therefore, the staff finds that granting the exemption would not result in a significant decrease in the level of safety otherwise provided by the design, as required by 10 CFR Part 52, Appendix D, Section VIII.A.4.

A.4 Conclusion

The staff has determined that pursuant to Section VIII.A.4 of Appendix D to 10 CFR Part 52, the exemption: (1) is authorized by law, (2) presents no undue risk to the public health and safety, (3) is consistent with the common defense and security, (4) has special circumstances that outweigh the potential decrease in safety due to reduced standardization, and (5) does not significantly reduce the level of safety at the licensee's facility. The staff has also determined, pursuant to Section VIII.C.4 of Appendix D to 10 CFR Part 52, that the generic TS portion of the exemption request: (1) is authorized by law, (2) presents no undue risk to the public health and safety, (3) is consistent with the common defense and security, (4) demonstrates the existence of special circumstances. Therefore, the staff grants the applicant an exemption from the requirements of Tier 1 Tables 2.2.3-1 and 2.2.3-2 and generic TS SR 3.5.4.7 of the generic DCD associated with the LNP Units 1 and 2.

B. Technical Evaluation of Exemption Request and Departure

B.1 Passive Core Cooling System, Accident Analysis, and Shutdown Temperature Evaluation

Letter NPD-NRC-2014-005, submitted by the applicant and dated February 7, 2014, requested the previously described departures from 10 CFR Part 52, Appendix D, Section III.B. A revised submittal, letter NPD-NRC-2015-015, dated May 5, 2015, included two supporting reports as Enclosures 2 and 3: APP-GW-GLR-161, Revision 2 (proprietary) and APP-GW-GLR-607, Revision 2 (non-proprietary), respectively, both titled "Changes to Passive Core Cooling System Condensate Return." These reports describe the change and the basis for the change. In addition, APP-GW-GLR-161 and APP-GW-GLR-607 references three calculations and a test report further described below. Enclosure 6 provides the applicant's request for exemption related to this topic. Enclosures 7 and 8 present, respectively, changes to AP1000 DCD Revision 19 and the LNP COLA information that will be included in a future revision to the COLA. Letter NPD-NRC-2014-005 and its enclosures are the subject of the following review by the staff.

The applicant indicated that the changes described in LNP DEP 3.2-1 are necessary to (1) ensure the capability of the PRHR HX to maintain the RCS in a safe, stable condition, as described in DCD Chapter 19E, "Shutdown Temperature Evaluation," and (2) to demonstrate the existing non-LOCA analyses in Chapter 15 that credit the PRHR HX remain valid. The safe shutdown temperature evaluation, presented in DCD Chapter 19E Revision 19, assumes a constant condensate return fraction (the fraction of the water boiled off from the IRWST that will condense on the containment shell and return to the IRWST). Water that does not return to the IRWST can be referred to as condensate losses. The NRC staff understands that the applicant's analyses showed there are a number of mechanisms for condensate losses that vary with time including: steam to pressurize the containment atmosphere, condensation on passive heat sinks within the containment, and condensate splashing from the containment vessel and its attachments that does not reach to the PXS gutter system. The NRC staff's review of this departure request indicates some of these losses, such as the steam to pressurize the atmosphere, initially account for the majority of the condensation losses but decrease as the transient progresses, while other losses, such as the splashing from the attachments to the shell, are relatively time-independent and only a function of the amount of condensation on the shell. Condensate return is one of the primary factors influencing the performance of the PRHR HX.

Section 5.0, "Design Changes," of APP-GW-GLR-607 and APP-GW-GLR-161 detail the changes proposed by the applicant for increasing the condensate return rate. Subsection 1 describes the PXS downspout piping network added at the polar crane girder and stiffener, the routing for which is shown in the revised Figure 6.3-1 of the FSAR. Four collection points are located on both the upper portion and the lower flange of the polar crane girder and the stiffener ring that are routed to common lines that empty into two collection points already existing on either side of the IRWST. These downspouts, collection points and connecting piping serve to capture condensate that previously would have been lost, and are

sized such that any one line can accommodate the full flow anticipated during a transient to prevent a single failure from impacting the return flow to the IRWST. Subsection 2 describes the screens added to the downspouts and new guttering that is similar to screens existing on the IRWST gutter. These screens are designed to keep larger debris from blocking piping while still allowing condensate flow. The seismic qualifications of the downspouts and screens are further discussed later in this section. Subsection 3 explains how fabrication holes are blocked in the polar crane girder and the stiffener. Subsection 4 details the dam added to the polar crane girder to alleviate flow interactions between the containment shell and polar crane girder that contributed to losses. Furthermore, changes to the gutter drip lip and gutter routing were made to reduce losses from the gutter-wall interaction as much as possible. The effect of these changes on the transient analysis is described in detail below.

The design changes, which are intended to reduce the condensate losses, prompted review of the analyses associated with transients that rely on condensate return. The effectiveness of the condensate return to the IRWST is captured in a series of proprietary calculations supporting the submittal, which were audited by the staff (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML14219A200 and ML15187A248) and are described in Section A.2 of APP-GW-GL-161 and APP-GW-GLR-607. The containment response is analyzed in calculation APP-PXS-M3C-071, "Containment Response Analysis for the Long Term PRHR Operation," via modifying the NRC-approved AP1000 WGOTHIC model used for containment peak pressure calculation that is part of the licensing basis, and provides transient containment pressure, temperature, and condensate holdup volumes input to the other calculations. Condensate losses implemented in WGOTHIC are obtained from a second calculation, APP-PXS-M3C-072, "Condensate Return to IRWST for Long Term PRHR Operation," which uses the parameters from WGOTHIC in concert with test results to provide a bounding condensate loss fraction from the containment shell. The test data used to calculate the losses are summarized in Section 4 of APP-GW-GL-161 and APP-GW-GLR-607 and described in detail in report TR-SEE-III-12-01, "AP1000 Condensate Return Test Report." A further calculation, APP-SSAR-GSC-536, "AP1000 Safe Shutdown Temperature Evaluation," incorporated the containment parameters and condensate behavior from the WGOTHIC analysis into LOFTRAN to calculate the behavior of the RCS and PRHR heat exchanger. This calculation was performed both for a 72-hour design basis case to verify that the assertions in Chapter 6 of the FSAR remain valid for all FSAR Chapter 15 events reliant on the PRHR, and for the 36-hour cooldown case depicted in Chapter 19 of the FSAR. A further calculation, APP-SSAR-GSC-009, "AP1000 Plant Safe Shutdown Duration Evaluation," justifies the duration of extended operation to 14 days using a LOFTRAN analysis. Further discussion of the analyses is located below in the "Evaluation of Containment Response," "Safety Design Bases," and "Non-Safety Design Bases" subsections of this SER section.

B.1.1 Evaluation of Containment Response

Although the staff audited the calculations referenced in the February 7, 2014 submittal by the applicant (ADAMS Accession Nos. ML14219A200 and ML15187A248), the submittal did not contain sufficient information for the staff to

make a safety finding based on the docketed information, and thus the staff issued RAI 7439 in a letter dated March 6, 2014, asking the applicant to summarize the containment response calculation and its relationship with the other calculations. In its response dated May 5, 2014, the applicant provided a summary to address the impact of the cited calculation on the changes in LNP DEP 3.2-1. The staff requested in RAI 7439, Question 6.03-1, that the applicant provide additional detail on the results described in "Containment Response Analysis for the Long Term PRHR Operation" (ADAMS Accession Nos. ML14077A609 and ML14126A702), which describes the WGOTHIC model used to calculate the containment pressure and temperature as well as the steaming rate from the IRWST to the containment atmosphere, heat sinks and the containment shell, to address the technical merits of the changes in LNP DEP 3.2-1. The staff reviewed this response and finds it acceptable, as it provides an accurate summary of the analysis explaining how the containment response calculation relates to other calculations, inputs, and key results with sufficient information for the staff to make its finding.

Operation of the PRHR HX is affected by the amount of condensate returned to the IRWST. Therefore, in order to bound all events that credit the PRHR HX, the staff considered events requiring operation of the PRHR HX. The applicant identified the loss of normal feedwater coincident with a loss of alternating current (ac) power to the plant auxiliaries as the most limiting transient. The discussion below analyzes this scenario, and the justification for the loss of ac power as the most limiting transient is provided below in the "Safety Design Basis" subsection of this SER.

Using WGOTHIC, the applicant modeled the containment behavior during a transient involving the actuation of the PRHR by modifying the containment model used for the peak pressure calculation such that it conservatively captured the phenomena that would challenge the performance of the PRHR HX. This was accomplished by modifying the existing peak pressure calculation model in the following ways: increasing the area of the passive heat sinks as modeled by applying a multiplying factor, creating a volume to capture the condensate losses on the shell, adding a flow path to account for containment leakage, changing the IRWST (including a structure simulating PRHR heat exchanger using boundary conditions from LOFTRAN) to better represent the conditions during a non-LOCA transient, and adding a heat structure in the cavity to represent the vessel, among other minor changes. The net effect of these changes is to minimize the condensation rate on the containment inner shell, maximize the amount of steam and condensate that does not return to the IRWST—such as on passive heat sinks in containment and in the containment atmosphere—and maximize the amount of heat input to the IRWST, all of which are conservatisms for the non-LOCA transients that challenge the PRHR HX.

The addition of the heat structure to represent the reactor vessel in the reactor cavity, although used appropriately to capture a physical phenomenon present in the problem, is not the most conservative modeling choice with respect to the calculation of condensate return. Most condensate that is lost from the containment shell eventually reaches the reactor cavity. This water fills the cavity to the point that it reaches the vessel and begins steaming. The vessel is surrounded by metallic insulation material designed to admit water through gaps

and release the resultant steam through larger gaps between the insulation and the vessel. Although steaming from the reactor vessel cavity has competing effects on the system performance, as it both cools the reactor vessel and results in additional mixing below the operating deck, it does result in a larger net condensate return fraction to the IRWST. The applicant explored mechanisms that stimulate mixing within containment, but the precise extent of the mixing beneath the operating deck is not fully defined. The applicant states that additional mixing below the operating deck results in more condensate holdup on passive heat sinks, but also that in the long term steaming from the reactor vessel results in additional inventory return to the IRWST.

The analysis in WGOTHIC accounts for the heat removal from the reactor vessel by subtracting it from heat that would be removed by the PRHR HX so that the energy balance is maintained. Temperature data from LOFTRAN is extracted and input into one boundary of the WGOTHIC vessel, while the other boundary exposed to the control volume uses a boiling correlation. The amount of heat removed by the boiling from the vessel is stored and subtracted from the PRHR HX heat input. Due to the nature of the modeling of the heat structure in the cavity in WGOTHIC, the entirety of the structure participates in heat transfer to the fluid in the reactor cavity. To mitigate against the effects of this, the applicant subtracted the volume in the cavity underneath the vessel and added it to the reactor coolant drain tank room so as to increase the holdup volume that must fill prior to condensate reaching the reactor vessel. This still results in additional boiling from the condensate that reaches the reactor vessel, as a larger area available (at least until the water would have reached the top of the bottom head) results in higher heat transfer. Conversely, in the very long term, the WGOTHIC model does not consider additional area that would participate as the water in the cavity rises above the lower head of the reactor vessel. In "Containment Response Analysis for the Long Term PRHR Operation," the applicant documents a sensitivity study that explores the effect on IRWST level of no condensate return resulting from reactor vessel steaming. The analysis shows that IRWST level is reduced by as much as 7 inches in the 72-hour period following the transient as a result of not accounting for reactor vessel steaming. This reduction in IRWST inventory does not appreciably impact system performance during the first 72 hours and would not challenge the operability of the system until much later in the transient. The staff performed a confirmatory analysis on the effect of the lower condensate return rate using LOFTRAN, which showed the lack of steaming from the reactor vessel would have less impact than was calculated by the applicant in their sensitivity study. In addition, the staff confirmatory calculation in MELCOR documented below tracks level along the reactor vessel heat structure and uses a conservatively high holdup volume such that steaming from the cavity is not established until almost one day into the transient. The applicant's design basis calculation bounds the confirmatory analysis performed by the staff. As a result, the staff finds the treatment of steaming from the vessel bottom head acceptable for this analysis.

The applicant made additional changes as compared to the approved WGOTHIC model used for peak pressure analyses in the most recent revisions of the calculations referenced in the May 5, 2015, submittal. The elevation of a modeled volume was changed, (resulting in changes to flow paths not representative of pipes but rather a function of the modeling divisions) in the analysis to prevent

condensate build up in the control volume from inhibiting air flow between the control volumes to prevent non-physical behavior and better represent real conditions. The condensate return fraction was further modified to be a flat value representative of the loss rate determined by testing at the highest flow rate (discussed further below) plus a margin of 0.7 percent. In addition, the heat structures representing the PRHR HX and reactor vessel receive temperature conditions from iterative runs of the LOFTRAN model discussed later in the “Safety Design Basis” section of this report, rather than bounding values.

In the applicant’s supporting analysis, condensation on most of the heat sinks is directly analyzed in WGOTHIC, while condensation holdup on surfaces such as the operating deck floor and other equipment was incorporated into a horizontal film holdup volume assumed proportional to the cross sectional area of containment multiplied by a factor with no provided justification. Therefore, in RAI 7439, Question 6.03-3, the staff requested that the applicant justify the multiplication factor used and the treatment of the horizontal film in the WGOTHIC model. In a response dated June 12, 2014, the applicant determined that the earlier treatment of film may not have been conservative. Thus, the applicant performed a sensitivity study to determine the effect of a different approach. The approach detailed in the response changed the representative area to a value incorporating the total surface area of the heat sinks modeled within containment in WGOTHIC, which are a conservative representation of the total passive heat sink area inside containment, incorporating the fixed components. For direct condensation in WGOTHIC, the applicant further increased this value to bound the total passive heat sink area within containment. Though this value does not directly represent the film holdup area as some heat sinks like the core makeup tanks (CMTs), polar crane girder and stiffener are excluded, the use of total surface area rather than horizontal surface area incorporates margin such that this treatment is conservative.

In addition, the applicant used a different approach to determine film thickness for condensation on surfaces utilizing a maximum contact angle for wetting in the design basis analyses and a more realistic contact angle for the “conservative, non-bounding” analyses to determine the thickness of the film. Although these changes increase the film holdup by a factor of more than three, there is a negligible effect on the performance of the PRHR HX during the first 72 hours. Initially following a non-LOCA transient, the significantly lower condensate return rates for the first few hours and lack of steaming from the reactor vessel cause the impact of additional holdup resulting from the more conservative film holdup calculation to be lessened and the level in the IRWST to be relatively unchanged. As condensate return increases to its long term value, and steaming from the reactor vessel begins to have a measurable impact on the transient, the submittal shows a minor reduction in the time before the RCS begins to reheat, well after the safety-related 72-hour period. The PRHR is required to remove decay heat following a design basis event for a minimum of 72 hours, in accordance with the revised FSAR Section 6.3.1.1.1, “Emergency Core Decay Heat Removal” in LNP DEP 6.3-1. The staff verified that this calculation was incorporated into “Containment Response Analysis for the Long Term PRHR Operation” calculation in a subsequent audit (ADAMS Accession No. ML15187A248).

The amount of condensation held up on surfaces within containment is also an important parameter during containment floodup following a LOCA or automatic depressurization system (ADS) actuation. Because the AP1000 relies on gravity for the driving force for recirculation in the long-term following an accident, the height of water in containment must be sufficient to force flow through the direct vessel injection lines for an opening in the RCS above the floodup level. The NRC staff's confirmatory analysis applying the revised film holdup to the floodup calculation shows a negligible impact on the containment water level following a LOCA or ADS actuation. Thus, the staff finds the treatment of film holdup on surfaces within containment acceptable because it conservatively accounts for condensation on surfaces using conditions for maximum condensate losses, and does not adversely affect current bounding analyses for other transients.

Containment response heavily depends on the initial conditions assumed for the transient of interest. Containment pressure and temperature, IRWST temperature, and the ambient outside temperature (equal to passive containment cooling system (PCS) water temperature) all have an impact. Pressure response can be divided into two phases for this transient, an initial spike up in pressure as the IRWST boils off, followed by a slow levelling off to a peak and decay as passive cooling occurs. Confirmatory analysis performed by the staff using MELCOR for design basis conditions follows a similar trend as the analysis performed by the applicant documented in "Containment Response Analysis for the Long Term PRHR Operation" (ADAMS Accession No. ML14219A200), although the pressure calculated by the applicant bounds the pressure in MELCOR at all points within an hour after steaming begins for the design basis. For best estimate conditions, the staff's confirmatory analysis shows a peak pressure of approximately 2 pounds per square inch greater than the applicant's WGOTHIC analysis, while design basis conditions result in confirmatory analysis yielding a pressure approximately 5 pounds per square inch less than the conservative value calculated by the applicant in WGOTHIC; these events, like all events involving PRHR actuation, do not challenge the design pressure. More importantly for this transient, the applicant's pressure used for the design basis analysis results in a higher saturation pressure for water in containment, which results in additional holdup in the containment atmosphere and higher IRWST temperatures and, therefore, reduced heat transfer through the PRHR. As such, the applicant's modeled pressure response in containment is conservative because it uses bounding inputs into an approved methodology and yields a more conservative value than staff models of the same conditions.

In each analysis performed by the applicant, calculations were performed for design basis conditions for Chapter 15 and "non-bounding, conservative" conditions for Chapter 19. Design basis conditions should represent the conservatively bounding set of values for any given transient, and the design basis values for the maximum temperature inside containment is 120 degrees Fahrenheit (°F) (48.9 degrees Celsius (°C)) and outside containment is 115 °F (46.1 °C). The analysis submitted used an in-containment initial temperature of 85 °F (29 °C) (capturing all the heat sinks as well as the IRWST) and an environment temperature of 115 °F (46.1 °C). In RAI 7439, Question 6.03-4, the staff requested the applicant justify the assumption of 85 °F (29 °C) for the initial temperature of containment for the design-basis accident (DBA) analysis. In the response dated July 1, 2014, the applicant explained that

the effect of the temperature of the heat sinks outweighed the effect of the IRWST temperature. That is, a lower heat sink temperature results in more condensation on heat sinks and, therefore, more losses when compared with the effect of a change in the initial enthalpy in the IRWST, which affects the time to begin boiling. The NRC staff reviewed analysis supporting this assertion (ADAMS Accession No. ML14219A200), and although the effect is slight, lower heat sink temperatures result in a lower IRWST level as the transient progresses.

The choice of 85 °F (29 °C) for in-containment initial temperature was based on the use of an exterior temperature of 115 °F (46.1 °C), the TS maximum for ambient air temperatures for the environment outside containment. The applicant performed a study for a plant located at a site where meteorological data indicates ambient temperatures could reach 115 °F (46.1 °C) and calculated in-containment temperatures for an operating facility with containment coolers running to show that containment temperatures (and therefore the temperatures of the heat sinks and the IRWST) would not reach below 88 °F (31 °C) for an ambient temperature of 115 °F (46.1 °C). The influence of exterior temperatures is more dramatic on PRHR HX performance: while lower temperatures inside containment would result in additional condensation on heat sinks, higher ambient temperatures result in higher initial PCS water temperatures, which result in less heat removal from containment during a transient and thus higher containment pressures and temperatures. The staff agrees that 85 °F (29 °C) for the in containment temperature presents an acceptably conservative value for a transient given a bounding environmental temperature of 115 °F (46.1 °C), due to the large thermal inertia of the heat sinks within containment and the sizable heat load for the operating plant under the steady state conditions leading up to the transient, in addition to the applicant's justification based on ambient temperatures.

Section 6.3.2.1.1 of the revised FSAR, "Emergency Core Decay Heat Removal at High Pressure and Temperature Conditions," in LNP DEP 6.3-1, addresses the impact of the revised analysis due to the design changes. The revised FSAR discusses the integrated system, including emphasis on the condensate return features, and explicitly describes the mechanics of in-containment condensation as the heat transfer mechanism. In addition, the FSAR now highlights that "[c]ondensation that is not returned to the in-containment refueling water storage tank drains to the containment sump." This is in accordance with the staff's understanding of the system as discussed in this subsection, and is acceptable because most water that does not return to the IRWST fills holdup volumes, which must fill to a certain level before overflowing and eventually reaching the lowest point in containment and filling the reactor coolant drain tank room and reactor cavity.

Section 6.3.2.1.1 also explains the impact of the condensate return rate on the duration of operation of the PRHR HX, and explains that if ac power is not recovered, the PRHR HX can continue to perform for a period of time beyond 72 hours. The plant also retains the ability to transition to open loop cooling via the automatic depressurization system if inventory in the IRWST is insufficient. This agrees with the staff analysis of the performance of the system and is an acceptable change to the FSAR, discussed further in the following section, "Safety Design Basis."

The changes made to Figures 6.3-1 and 6.3-2 in the FSAR appropriately capture the design changes as modeled in the analyses described in the submittal and are acceptable. The components in these figures added to Tier 1 are discussed in the "Classification of Structures, Components, and Systems" subsection below.

The applicant stated that the modifications referenced above to the WGOTHIC model, such as those incorporating condensate return to the IRWST, have no effect on the peak containment pressure calculation. Peak containment pressure is reached well before condensate return has a measurable impact on the transient, and any benefits from condensate return at later times are not credited. The addition of downspouts at the polar crane and stiffener have no impact on the current peak pressure analysis because the model already assumes that condensate reaching the polar crane and stiffener makes its way to the reactor coolant drain tank room, which overflows to the reactor cavity region. The assumptions used in these analyses for initial conditions for temperature, humidity, and heat sink area limiting the amount of condensate return are less bounding for the case of peak containment pressure and, therefore, would not be applicable to the peak pressure calculation. The staff finds the peak pressure analysis in the licensing basis is unaffected by the changes implemented in the current analyses.

For the analyses supporting LNP DEP 3.2-1, the treatment of the PCS water coverage of the outside of the containment shell is consistent with that used in the peak pressure calculation model previously approved by the staff. That is, an assumed film coverage below the weir of 90 percent (for design basis conditions) at nominal flow rates, decreasing as the level in the PCS water storage tank drops during the 72-hour period (discussed in Section 6.2.1 of NUREG-1273 and Table 6.2.2-1 of the AP1000 DCD). Thus, that treatment is conservative for this analysis, as minimizing shell coverage maximizes the energy within containment, which maximizes the containment pressure and saturation temperature.

The calculation, "Containment Response Analysis for the Long Term PRHR Operation," receives inputs from the "Condensate Return to IRWST for Long Term PRHR Operation" calculation (ADAMS Accession No. ML14219A200), which calculates the effective condensate losses on the inside surface of the containment shell. The NRC staff requested in RAI 7439, Question 6.03-2 that the applicant submit additional detail on the results described in "Condensate Return to IRWST for Long Term PRHR Operation," which describes the methodology used to calculate losses over the containment shell, including the tests used to determine losses over attachments to the shell. This request was to address deficiencies in the submittal related to insufficient justification of the applicability of the development of the condensate loss model. The applicant summarized the calculation in a response dated June 12, 2014. The NRC staff reviewed the response and found it acceptable because it provides a summary with sufficient information on the calculation for the staff to make its finding.

Tests for losses over attachments to the shell were performed at lower temperatures than the prototypic conditions on the containment shell during a non-LOCA transient, which could peak in excess of 220 °F (104 °C). Therefore, in RAI 7439, Question 6.03-5, the staff requested the applicant justify the extrapolation from the losses for tested values of condensate losses over

attachments to the wall to the values used in the analysis at containment pressure and temperature. In its response to the RAI dated June 27, 2014, the applicant explained that although the losses over wall attachments are extrapolated, the extrapolation is overly conservative and prior research indicates that film thickness should decrease at the same Reynolds number at higher temperatures and thus decrease the condensate losses. In addition, the applicant performed sensitivity studies on the effect of increasing the losses on the performance of the PRHR HX. Those sensitivities indicate that even for a case when losses over attachments are increased by a factor of 1.4 to 1.75, there is a negligible effect on the performance of the system in the first 72 hours and only a minor (approximately 5 percent) reduction in the long term capability of the system. The NRC staff remains unconvinced as to the validity of the applicant's temperature scaling argument, especially given the relative variance in the test results. However, on the basis of the large degree of conservatism inherent in the extrapolation and the fact that a further 40 percent increase in losses over wall attachments results in an insignificant impact to the system performance, the staff finds the treatment of film losses over attachments to the containment shell acceptable.

The analysis described above using WGOTHIC passes a set of inputs to analyses in LOFTRAN (discussed below). The applicant extracts a table including time, condensate return flow, condensate temperature, IRWST steaming rate, containment pressure, and CMT compartment temperature. The data for condensate return flow and condensate temperature are combined to create a recirculation ratio (the fraction of boil off from the IRWST returning as condensate). The recirculation ratio and containment pressure are then used in the LOFTRAN analysis; in the case of the LOFTRAN run using design basis conditions, the recirculation ratio is further reduced and the pressure is increased from the values calculated in WGOTHIC for additional conservatism.

On the bases that the modifications to the gutter system are appropriately incorporated into the analyses for events that actuate the PRHR, that the data from tests used to determine the losses on the containment shell conservatively bound realistic losses, and that condensate loss mechanisms have been quantified and captured in the analysis, the staff finds the treatment of containment conditions in calculations supporting LNP DEP 3.2-1 and LNP DEP 6.3-1 acceptable. Therefore, the staff finds the proposed LNP DEP 3.2-1 FSAR revisions related to containment response noted above to be acceptable pending the staff's confirmation that the proposed FSAR revisions are incorporated in the LNP Units 1 and 2 COL application. The staff is tracking these revisions as **LNP Confirmatory Item 21.1-1**.

Resolution of LNP Confirmatory Item 21.1-1

LNP Confirmatory Item 21.1-1 is a commitment by the applicant to revise the LNP COL FSAR to provide additional information related to containment response as indicated in the letter dated January 14, 2016. The staff confirmed that the LNP COL FSAR has been appropriately revised. As a result, LNP Confirmatory Item 21.1-1 is now closed.

B.1.2 Safety Design Bases

The PXS performs the following safety-related functions:

- 1. Emergency decay heat removal*
- 2. Emergency reactor makeup/boration*
- 3. Safety injection*
- 4. Containment pH control*

The following subsections evaluate the impact of LNP DEP 3.2-1 and LNP DEP 6.3-1 on each safety function of the PXS.

B.1.2.1 Emergency Decay Heat Removal

LNP DEP 3.2-1 impacts the condensate return rate to the IRWST and thus impacts the emergency decay heat removal function of the PRHR HX. Under LNP DEP 3.2-1 and LNP DEP 6.3-1, the revised FSAR Section 6.3 states that for non-LOCA events in which a loss of core decay heat removal capability via the steam generators (SGs) occurs, the PRHR HX is designed to perform the following functions:

- 1. Remove core decay heat following a design basis event.*
- 2. Maintain acceptable reactor coolant system conditions for a minimum of 72 hours following a non-LOCA event. Applicable post-accident evaluation criteria are specified in Chapter 15.*
- 3. Sufficiently reduce RCS temperature and pressure during an SG tube rupture (SGTR) event to terminate breakflow, without overflowing the SG.*

Emergency decay heat removal functions 1 and 3 are design criteria that have been evaluated in DCD Chapter 15, Revision 19 for the events identified in Table 21.1-1 and reviewed in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." Previous staff review of DCD Chapter 15 events did not consider the possibility of PRHR HX tube uncover. Therefore, calculations could be terminated once the acceptance criteria for the design basis events were initially met. LNP DEP 3.2-1 revealed that the PRHR HX can provide cooling for a finite period of time before performance degrades and transition to open-loop cooling, via ADS actuation, is required to maintain the reactor in a safe, stable shutdown condition. LNP DEP 3.2-1 states that the water level in the IRWST remains above the uppermost points of the PRHR HX for the duration of all DCD Chapter 15 analyses and, therefore, there is no impact to the calculated heat transfer through the heat exchanger. This caused the staff to question the mission time for the PRHR HX

and the termination criteria for DCD Chapter 15 analyses for events that credit the PRHR HX (Table 21.1-1).

Table 21.1-1. Chapter 15 Events that Credit the PRHR HX for Decay Heat Removal

DCD Section	Scenario	Calculation Duration
15.2.6	Loss of AC Power to Plant Auxiliaries	6.2 hrs
15.2.7	Loss of Normal Feedwater Flow	5.4 hrs
15.2.8	Feedwater System Pipe Break	3.1 hrs
15.5.1	Inadvertent Operation of CMTs During Power Operation	8.6 hrs
15.5.2	CVCS Malfunction that Increases RCS Inventory	5.6 hrs
15.6.3	Steam Generator Tube Rupture	6.7 hrs

Section 4.3.3.5 of the Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document (URD) and Section 2.3.2 of the staff's corresponding safety evaluation (NUREG-1242, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document, Evolutionary Plant Designs," Volume 3) both state that a design expectation for the passive decay heat removal system is to have sufficient water capacity in the passive decay heat water pools to permit 72 hours of operation after SCRAM without the need for refill. The 72-hour capacity of the passive residual heat removal system was approved by the Commission in their responses to SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," and SECY-95-132, "Policy and Technical Issues Associated with Regulatory Treatment of Non-Safety Systems in Passive Plant Designs (SECY-94-084)." Based upon the Commission position expressed in SECY-94-084 and SECY-95-132, the licensing guidance in the URD, NUREG-1242, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document, Evolutionary Plant Designs," and the Regulatory Treatment of Non-Safety Systems as discussed in Section 19.3 of the Standard Review Plan, in order for the PRHR HX to meet the requirements of GDC 34 and GDC 44, the IRWST should have sufficient capacity to permit a minimum of 72 hours of operation after SCRAM following an accident without the need for refill. In RAI-7475, Question 6.03-10, the staff requested clarification of the mission time for the PRHR HX. In a response dated June 27, 2014 (ADAMS Accession No. ML14182A106), the applicant stated that the PRHR HX operates to bring the RCS to an acceptable, stable condition and maintain this condition for at least 72 hours after a non-LOCA event to allow ample time for decision-making and initiation of recovery actions. During this 72-hour time period, applicable Chapter 15 design basis safety evaluation criteria are met. The 72-hour operational requirement for the PRHR HX following a non-LOCA event is consistent with the Commission position for compliance with GDC 34 and GDC 44.

DCD Chapter 15 analyses that credit the PRHR HX, shown in Table 21.1-1, terminate before the 72-hour operational requirement of the PRHR HX. This caused the staff to question the possibility of PRHR HX tube uncover during the 72-hour time period, and the resulting impact to Chapter 15 analyses. In RAI 7440, Question 15.02.06-2, the staff requested the applicant to (1) identify

the bounding Chapter 15 event in terms of PRHR HX performance, and (2) extend the calculation for the bounding event out to 72 hours in order to demonstrate the 72-hour operational requirement of the PRHR HX.

In their response dated June 27, 2014 (ADAMS Accession No. ML14182A106), to the first part of RAI 7440, Question 15.02.06-2, the applicant identified the Loss of AC Power to Plant Auxiliaries (LOAC) as the limiting event in terms of PRHR HX performance. The applicant explained that the LOAC event combines a relatively late reactor trip with a significant loss of secondary side inventory in both steam generators, and a loss of forced reactor coolant flow. It therefore, represents the largest mismatch between primary side energy and secondary side/PRHR HX heat removal capability. The applicant's response to RAI 7440, Question 15.02.06-2 included a sensitivity study, performed with the MAAP4.0.7 code, to evaluate the impact of different events on PRHR HX performance. The results demonstrated that the plant response to different events begins to converge after approximately 8 hours into the event with the LOAC event producing slightly bounding heat loads on the PRHR HX over the 72-hour calculation time. The NRC staff performed confirmatory calculations as part of the review, which include a sensitivity study to investigate the impact of the initiating event. The result of the staff's sensitivity study is consistent with the applicant's response to RAI 7440, Question 15.02.06-2. Based upon considerations discussed in this paragraph, the staff finds the selection of LOAC as the limiting event in terms of PRHR HX performance to be acceptable.

In their response to the second part of RAI 7440, Question 15.02.06-2, the applicant performed a 72-hour calculation of the LOAC event. The analysis utilized the LOFTRAN code to model the response of the reactor coolant system. In evaluating the applicant's response, the staff evaluated the analytical procedure (i.e., use of LOFTRAN) and the results of the calculation. In the NRC staff's safety evaluation for the AP1000 DCD, NUREG-1793, the staff concluded that the applicant's use of LOFTRAN as described in WCAP-15644 (ADAMS Accession No. ML040890663) is acceptable for licensing calculations of the AP1000 subject to the following limitation:

- LOFTRAN is approved to analyze the transients listed in Table 21-2 of NUREG-1793. Use of the code for other analytical purposes will require additional justification.*

Previous licensing calculations that utilized LOFTRAN extended less than 10 hours and did not experience uncovering of the PRHR HX tubes. Thus, the staff investigated the applicability of the code to the analyses referenced in the departure. Modeling of tube uncovering in LOFTRAN uses a collapsed liquid level within the IRWST, where surface area of the PRHR HX above the collapsed liquid level is not credited for heat removal. The surface area below the liquid level is calculated as described in WCAP-14235 (ADAMS Accession No. 9709290174) and approved in the staff's safety evaluation of the AP1000 DCD in NUREG-1793. During pool boiling, the secondary side heat transfer is modeled using a modified Rosenhow correlation. This modified Rosenhow correlation was developed from experimental data obtained from the AP600 PRHR HX test program described in WCAP-13573 (ADAMS Accession No. 9705280203). The AP600 PRHR HX test program included a series of tests where PRHR HX tubes

were uncovered to different levels (75 percent, 50 percent, and 25 percent) which demonstrated insignificant heat transfer for the uncovered tubes and heat transfer consistent with nucleate boiling for the covered tubes. Details of the staff review of the PRHR HX test program are available in Section 21.5.3 of NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design." Of specific concern were the flow distribution and behavior in the tubes and two-phase flow behavior in the IRWST, especially within the tube bundle. High heat transfer rates could cause violent boiling on the outer surface of the tube, resulting in vapor blanketing of some portion of the heat exchanger surface and drastic reduction in heat transfer. Westinghouse analyzed the PRHR HX performance and concluded that it is unlikely that vapor blanketing could occur, and that if it did occur, such behavior would be limited to a very short length near the inlet of the tube bundle, leaving sufficient heat transfer area to meet its design performance requirements. Based upon the Westinghouse analysis and that vapor blanketing was not observed at any of the integral test facilities (OSU/APEX, SPES-2, or ROSA/LSTF), the staff concluded in NUREG-1512 that Westinghouse resolved the concern of vapor blanketing. The potential for the vapor generated by the lower tubes to impede the heat transfer of the upper (covered) tubes is reduced as the PRHR HX begins to uncover. Based upon considerations discussed in this paragraph, the staff finds the previous resolution of the vapor blanketing issue to remain valid for the case of tube uncover and the heat transfer modeling of the PRHR HX to be acceptable.

In order to understand the limits of the analysis, the staff explored additional input considerations. In RAI 7475, Question 6.03-10, the staff requested the tube plugging assumption used for DBA analyses. In the response, dated June 27, 2014, the applicant stated that a design change was implemented to reduce the allowable number of plugged tubes for the PRHR-HX from the number of tubes making up 8 percent of the heat transfer area to the number of tubes making up 5 percent of the heat transfer area. However, the original 8 percent assumption is utilized for the DBA analysis presented in the response to RAI 7440, Question 15.02.06-2. Existing Chapter 15 analyses assume 8 percent tube plugging in the PRHR-HX (in terms of heat transfer area) for scenarios where minimizing heat removal is bounding and 0 percent tube plugging in the PRHR-HX where maximizing heat removal is bounding (e.g., steam line break). Boundary conditions for the containment response (i.e., containment pressure and condensate return ratio) were input as functions of time and have been evaluated above in subsection "Evaluation of Containment Response" of this SER. During an audit, the NRC staff identified that the initial power utilized in the 72 hour analysis accounted for a 1 percent uncertainty. Section 15.0.3.2 of the AP1000 DCD, Revision 19, states that a 1 percent uncertainty is supported by the main feedwater flow measurement instrumentation, but that a bounding value of 2 percent is used in the analysis. The Levy COL FSAR contains COL Information Item STD COL 15.0-1, which identifies the plant operating instrumentation which when properly calibrated will support 1 percent uncertainty in the core power based on flow measurement uncertainty. Additionally, the NRC staff performed a sensitivity study investigating the impact of the reduced core power uncertainty on the 72-hour LOAC event. The results of this study demonstrated that the reduction in core power uncertainty has an insignificant impact on the RCS response and Chapter 15 acceptance criteria.

The analysis of the LOAC event submitted by the applicant demonstrates that during the 72-hour period the top horizontal portion of the PRHR HX becomes uncovered. However, the PRHR HX capacity remains sufficient to prevent RCS heatup for a time period greater than 72 hours. The submitted analysis demonstrates that once the Chapter 15 acceptance criteria are satisfied, at approximately 6.2 hours, they remain satisfied for a time period exceeding 72 hours. The NRC staff performed confirmatory calculations as part of the review, which include a 72-hour analysis of the LOAC event. The staff's confirmatory calculation for the LOAC event is consistent with the applicant's submitted analysis. Based upon the identification of the LOAC event being the bounding event in terms of PRHR HX operation, the acceptable modeling of the LOAC event, and the result demonstrating the 72-hour operational requirement for the PRHR HX, the staff finds the submitted analysis of the 72-hour LOAC event acceptable.

In a letter dated January 14, 2016 (ADAMS Accession No. ML16020A250), the applicant updated their submittal, which included the consideration of ambient heat losses from the RCS during Chapter 15 non-LOCA events. Previous analyses had assumed the RCS to be adiabatic, which would result in the highest required heat removal from the PRHR HX; due to ambient heat losses from the RCS, from the pressurizer in particular, and in the absence of positive pressure control associated with pressurizer heaters, the applicant was concerned that pressure in the RCS could be reduced to the point that subcooled margin is lost. A loss of subcooling was thought to have the potential to inhibit the performance of the PRHR HX. Additional analyses were conducted by the applicant to investigate the impact of ambient heat loss from the RCS. A description of these analyses is provided in APP-GW-GLR-607, Revision 4 "Changes to Passive Core Cooling System Condensate Return," which is included as an enclosure to the letter of January 14, 2016. The NRC staff audited the supporting calculations (documented in the audit report, ADAMS Accession No. ML16034A034). The audit resulted in a supplemental RAI response, provided in letter dated January 14, 2016 (ADAMS Accession No. ML16020A105), to establish the basis for the ambient heat losses associated with the pressurizer. The RAI response included (1) a description of the ambient heat loss flow paths from the pressurizer and their treatment in transient analyses, and (2) a FSAR update to Section 5.4.5.2.1 to include the average maximum heat transfer rate specification for the metallic reflective insulation installed on the external surfaces of the RCS. The NRC staff found the RAI response identified the applicable heat loss mechanisms from the pressurizer during a DBA. NRC reviewed the details of the heat loss calculation during their audit of the supporting calculations and observed that additional conservatism was included in pressurizer heat loss calculations. Additionally, the NRC staff performed confirmatory calculations for the heat losses from the pressurizer which resulted in values that were consistent with the applicant's analyses. The conservative modeling of the heat losses from the pressurizer is further supported by data from applicable literature identified in the NRC staff's audit report. Based upon the information discussed above, the NRC staff finds the treatment of ambient heat losses in the analysis of DBAs to be suitably conservative. The applicant performed a DBA analysis that considers ambient heat losses, performed with LOFTRAN, showing that the RCS remains

subcooled for a time period exceeding 72 hours. Therefore, the only impact on the DBA analysis was a lower temperature in the RCS due to the increased heat removal. The NRC staff performed confirmatory calculations as part of this review and obtained results that were consistent with the applicant's analysis. Based on the information in this paragraph, the NRC staff finds that ambient heat losses do not adversely impact DBA analyses for the AP1000.

The staff performed confirmatory calculations, which included the Chapter 15 LOAC event, to assist in evaluating the impacts of LNP DEP 3.2-1 to Chapter 15. The calculations caused the staff to question whether containment backpressure effects on PRHR HX performance were accounted for in Chapter 15. During the staff audit of the applicant's documents related to LNP DEP 3.2-1 and LNP DEP 6.3-1 (ADAMS Accession No. ML14219A200), the staff verified that in Revision 19 of the DCD, Chapter 15 analyses that credit the PRHR HX for decay heat removal do not account for containment backpressure effects on the PRHR HX. Not accounting for containment backpressure on PRHR HX performance introduces a slightly non-conservative boundary condition that affects PRHR HX performance late in the transient. However, the staff verified that this effect does not alter the conclusions of Chapter 15 analyses and thus produces no consequential impact.

The change from indefinite operation of the PRHR HX to the 72-hour operational requirement, and subsequent analysis demonstrating the 72-hour operational requirement, are reflected in the applicant's proposed changes under FSAR Sections 5.4, 6.3, 7.4, and Table 19.59-18 in letter dated June 27, 2014. In the proposed FSAR changes noted above, indefinite operation is changed to extended operation at several locations. For consistency among the proposed changes, the staff is interpreting extended operation to be at least 72 hours. Based upon the considerations discussed within this subsection, the staff finds the proposed FSAR revisions noted above to be acceptable pending the staff's confirmation that the proposed revisions are incorporated in the LNP Units 1 and 2 COL application. The staff is tracking these revisions as **LNP Confirmatory Item 21.1-1**.

Resolution of LNP Confirmatory Item 21.1-1

LNP Confirmatory Item 21.1-1 is a commitment by the applicant to revise the LNP COL FSAR to provide additional information related to ambient heat losses as indicated in the letter dated January 14, 2016. The staff confirmed that the LNP COL FSAR has been appropriately revised. As a result, LNP Confirmatory Item 21.1-1 is now closed.

Indefinite is still used in the revised FSAR (in Sections 6.3.1.1.4, 6.3.3.3.3, 6.3.3.4.3, and 7.4) when considering the entirety of the passive core cooling system; that is, when ADS is actuated and the system transitions to open-loop cooling with gravity driven injection. At that point, the system is nominally limited by normal containment leakage. This treatment remains unchanged from the system as reviewed by the staff in Revision 19 of the DCD.

B.1.2.2 Emergency Makeup and Boration

Emergency makeup and boration for non-LOCA events are functions performed by the CMTs and are not impacted by LNP DEP 3.2-1.

B.1.2.3 Safety Injection

LNP DEP 3.2-1 is evaluated to ensure ADS actuation and transition to open loop cooling is retained as a defense-in-depth means of providing emergency core cooling during non-LOCA events. The evaluation includes investigating the impact of IRWST level on the performance of the ADS spargers, the impact of LNP DEP 3.2-1 on the containment floodup level, and the availability of the ADS, IRWST injection, and containment recirculation valves during an extended station blackout.

In the event that operator action is taken to prolong closed loop mode of PXS operation for an extended period of time, the level in the IRWST can drop below the ADS spargers, causing the staff to question whether ADS actuation can be inhibited by a low IRWST level. In RAI 7440, Question 15.02.06-1, the staff requested information regarding the minimum IRWST level required for ADS actuation. In a letter dated June 19, 2014, the applicant stated that no minimum IRWST level is required for ADS actuation because:

- 1. ADS spargers do not limit the containment pressure increase for the bounding mass and energy release. The associated mass and energy release attributed to ADS actuation is bounded by the large break LOCA accident or a large main steam line break inside containment.*
- 2. IRWST vents are more than sufficient to vent the amount of steam released if ADS Stages 1-3 are actuated after the spargers are uncovered. The IRWST vents are sized to vent steam relief from ADS stages 1-3 at high system pressures following several hours of PRHR HX operation during which the IRWST has reached saturation pressure.*
- 3. During a long-term non-LOCA event, during which the IRWST level has fallen below the elevation of the ADS spargers, RCS pressure at the time of ADS actuation will be relatively low.*
- 4. Steam relief from uncovered ADS spargers actually improves ADS Stages 1-3 performance due to the lower backpressure provided by the IRWST water. Limitations are imposed on the maximum sparger submergence depth to limit sparger discharge backpressure.*
- 5. No damage is done to spargers, IRWST, or surrounding structures.*

The NRC staff identifies the reasons as valid, but requested further justification for the argument that no damage is done to the ADS spargers, IRWST, or surrounding structures. In a supplemental letter dated July 24, 2014, the applicant stated that the ADS spargers are designed to withstand spurious actuation of ADS Stages 1-3 at normal operating conditions. Spurious actuation of ADS Stages 1-3 is bounding in terms of stress on the spargers because it results in bounding mass flows and temperatures experienced by the spargers. Additionally, with the IRWST water level below the spargers, the hydrodynamic loads associated with the initial discharge of air (trapped in the ADS valve discharge lines) or of the subsequent discharge of steam into the water are

eliminated. Forces encountered by the IRWST and surrounding structures due to ADS actuation would not be large because the spargers contain a large number of small jets that would interact and dissipate over a relatively short distance. Based upon the considerations mentioned above and the equipment classification of the associated structures and components, the staff finds that ADS actuation is not inhibited by low IRWST level.

The NRC staff reviewed the potential changes to containment holdup during floodup following a LOCA or ADS actuation as a result of the changes in LNP DEP 3.2-1. The NRC staff audited the "Containment Floodup Level" calculation (ADAMS Accession No. ML14219A200), and found that steam in the containment atmosphere and film on surfaces was accounted for. Applying the calculation for film condensing on surfaces used in RAI 7439, Question 6.03-3, results in a higher holdup than calculated in the supporting analysis in the form of film, which would reduce the containment level following depressurization of the RCS by less than 2 inches. Given the conservatism inherent in the film holdup analysis in RAI 7439, Question 6.03-3, the staff finds no significant impact to containment floodup level as a result of LNP DEP 3.2-1.

An additional consideration is the availability of the ADS, IRWST injection, and containment recirculation valves during an extended station blackout event. The operator action to establish open loop cooling, if required, may occur at a time that exceeds the operating times for the ADS, IRWST injection, and containment recirculation valves specified in Table 3.11-1 of the FSAR. As part of the staff review of submittals from Southern Nuclear Operating Company (SNC) in response to "Order to Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events, Order EA-12-049," issued on March 12, 2012, for Vogtle Electric Generating Plant Units 3 and 4, which is licensed based on the same AP1000 certified design as the LNP Units 1 and 2 applicant, the NRC staff issued RAI 7741 and RAI 7756 to SNC seeking further justification that the AP1000 can transition to open loop cooling during an extended station blackout. SNC's response in letters dated December 4, 2014 (ADAMS Accession No. ML14338A658), and February 26, 2015 (ADAMS Accession No. ML15057A590), provided justification regarding (1) equipment qualification of the ADS, IRWST injection, and containment recirculation valves, and (2) diverse actuation capability for the squib valves.

SNC demonstrated the equipment qualification envelope for the ADS, IRWST injection, and containment recirculation valves is bounding for an event that utilized the PRHR HX long term. This was done by performing a best estimate calculation for the containment response to an event that utilized the PRHR HX over a 30-day duration. The pressure profile for the qualification envelope was shown to bound the results of the containment response calculation. The temperature profile from the containment response calculation was converted into an equivalent time at 150 °F (65.6 °C) using the Arrhenius method. This equivalent time is bounded by the qualification time specified for the ADS, IRWST injection, and containment recirculation valves. The Arrhenius methodology has been previously reviewed and approved by the NRC staff for modeling the temperature effects in a post-LOCA environment (ADAMS Accession No. ML003701987). Based on the discussion in this paragraph, the NRC staff finds the equipment qualification envelope for the ADS, IRWST

injection, and containment recirculation valves bounds the expected containment environment during an extended station blackout for at least 30 days.

Additionally, SNC discussed the diverse capability for establishing open loop cooling. The primary means of establishing open loop cooling utilizes the Class 1E dc and uninterruptible power supply system (IDS). SNC's response included an analysis of the capacity of the IDS batteries. This analysis considered temperature de-rating of the batteries and self-discharge over a month and showed that sufficient margin is available for the batteries to perform their intended function during an extended station blackout. Should the battery supplies become completely exhausted, the ADS Stage 4, IRWST injection, and containment recirculation valves can be actuated via a diverse actuation system power independent device located at the secondary diverse actuation system station. Based upon the considerations in this paragraph, the NRC staff finds reasonable assurance that open loop cooling can be actuated during an extended station blackout event.

In a letter dated July 16, 2015 (ADAMS Accession No. ML15201A129), the applicant endorsed the RAI responses of SNC discussed above. Based upon the considerations of the environmental qualification of the ADS, IRWST injection, and containment recirculation valves, the containment floodup level, and the diverse actuation for establishing open loop cooling, the NRC staff finds that the safety injection function of the PXS is not impacted by LNP DEP 3.2-1.

B.1.2.4 Containment pH Control

Control of the pH in the containment sump post-accident is achieved through the use of pH adjustment baskets containing granulated trisodium phosphate (TSP) and is not impacted by LNP DEP 3.2-1.

B.1.2.5 Safe Shutdown

Short term safe shutdown conditions, defined in Section 7.4 of the DCD, include:

- Maintaining the reactor in a subcritical condition*
- Maintaining RCS average temperature less than or equal to no load temperature*
- Retaining adequate coolant inventory*
- Providing adequate core cooling*

Establishing short term safe shutdown conditions after an event has been demonstrated through DCD Chapter 15 analyses and reviewed by the staff in NUREG-1793. Through the evaluation of the PXS safety functions, the staff finds that short term safe shutdown is not impacted by LNP DEP 3.2-1.

Long term safe shutdown conditions, defined in Section 7.4 of the DCD, are the same as the short term conditions except that the RCS average temperature shall be less than 420 °F. The design requirement of entering a long term safe shutdown condition within 36 hours (i.e., reaching an average RCS temperature less than 420 °F in 36 hours) following an event is established in the URD and

SECY-94-084. In Section 6.3 of the DCD, Revision 19, cooling the RCS to 420 °F in 36 hours is identified as part of the design basis for the PRHR HX. The ability of the PRHR HX to satisfy this design requirement is demonstrated in the shutdown temperature evaluation provided in DCD Section 19E.4.10.2.

The shutdown temperature evaluation utilizes the same model and evaluates the same event as discussed in subsection "Emergency Decay Heat Removal" of this SER. The analysis in Section 19E.4.10.2 differs in that several model inputs (e.g., containment response pressure, condensate return rate, initial power, and core decay heat) utilize more realistic values. Sections 6.3.3 and 7.4.1.1 of the revised FSAR refer to this analysis as "non-bounding, conservative." In order to better understand the sources of conservatism in the calculation, the NRC staff issued RAI 7475, Question 6.03-11. The response, provided in letter from the applicant dated June 27, 2014, identified conservatism inherent in the condensate return rate and several modeling choices that were taken to increase the heat load on the PRHR HX and limit the heat removal capability of the PRHR HX. The use of nominal and best-estimate values for reactor power and decay heat remains consistent with the shutdown temperature evaluation supporting the design certification as verified by the staff during an audit of the original calculation (ADAMS Accession No. ML14219A200). The results of the updated analysis demonstrate the RCS average temperature decreases below 420 °F within 36 hours. The staff performed confirmatory calculations as part of the review, which include a shutdown temperature evaluation. The result of the staff's confirmatory calculation for the shutdown temperature evaluation is consistent with the applicant's submittal. Based upon the considerations within this subsection, and the results of the bounding calculation discussed in subsection "Emergency Decay Heat Removal" of this SER, the staff finds the plant is consistent with SECY-94-084. The updated analysis is reflected in the applicant's proposed changes to FSAR Section 19E described in a letter from the applicant dated May 5, 2015.

In Revision 19 of the AP1000 DCD, the cooldown requirement of reaching an RCS temperature of 420 °F in 36 hours is the only performance criteria listed in Section 6.3.1.1.1 that is not demonstrated by a Chapter 15 analysis. In reading the original DCD, it would be possible to incorrectly conclude that this performance requirement was demonstrated by a Chapter 15 analysis. The applicant's proposed changes under FSAR Sections 6.3.1.1 in letters dated June 27, 2014, and July 24, 2014, clarify how this design requirement is demonstrated. Based upon considerations within this subsection, the staff finds the proposed FSAR revisions in Sections 6.3.1.1 and 19E, noted above, to be acceptable.

B.1.3 Non-Safety Design Basis

In the proposed FSAR revision under Section 6.3.1.2 the applicant states that the PRHR HX, in conjunction with the IRWST and the condensate return features of the PXS, has the capability to maintain the reactor coolant system in the specified, long-term shutdown condition for 14 days in a closed loop mode of operation. The 14-day operation is also reflected in the applicant's proposed changes under FSAR Section 19E. The basis for this duration is provided by extending the duration of the non-bounding conservative LOFTRAN calculation

that was discussed in subsection "Safe Shutdown" of this SER. The staff verified the results of the analysis in an audit (see ADAMS Accession No. ML15187A248). In an update to the departure provided in a letter dated January 14, 2016 (ADAMS Accession No. ML16020A250), the applicant identified calculations incorporating ambient heat losses performed using RELAP 5, a transient analysis code, as LOFTRAN was not suited for demonstrating two-phase flow through the RCS. The RELAP calculations showed a loss of subcooling in the RCS occurring after 72 hours, but prior to 14 days. The calculations showed that the PRHR HX was capable of performing its function out to 14 days even with the loss of subcooling. The applicant provided test results from the APEX facility to demonstrate the ability of the PRHR HX to perform its function with a saturated RCS. The staff verified the results of the calculation and test results in an audit (ADAMS Accession No. ML16034A034). Operation of the PXS for 14 days in closed loop mode is not required to satisfy Commission regulations. The operational requirements of the PRHR HX have been evaluated in subsection "Safety Design Basis" of this SER. The staff finds the changes made to the operational duration and safety classification of the PRHR HX in LNP DEP 6.3-1 acceptable.

B.1.4 Post-72-Hour Actions

In DCD Section 6.3.4, it is stated that the only post-72-hour action required is a potential need for containment inventory makeup. This caused the staff to question the post-72-hour actions in the event that closed loop mode of PXS operation is extended following a non-LOCA event. In RAI-7440, Question 15.02.06-3, the staff requested clarification on post-72-hour actions following non-LOCA events. In a response dated June 19, 2014 (ADAMS Accession No. ML14171A453), the applicant stated that containment makeup would be necessary if containment leakage reduces the containment flood-up level, but there is no requirement to provide makeup to the IRWST to maintain PRHR HX operability. The primary post-72-hour actions are to provide water makeup to continue passive containment cooling and spent fuel cooling and, in the event that operators extend the closed loop mode of PXS operation, to provide power to the post-accident monitoring cabinets when transition to open loop cooling is required. In RAI 7440, Question 15.06.01, the NRC staff sought clarification on the criteria for operators to actuate ADS and transition to open loop cooling. The applicant's response provided in letter dated January 15, 2016 (ADAMS Accession No. ML16021A188), stated four criteria associated with reliable indication of core cooling which included (1) power availability to IDS divisions B and C, (2) hot leg and CMT level, (3) core exit thermocouple temperature, and (4) RCS pressure. The NRC staff finds this answer acceptable because it requires operators to check for diverse and reliable indication of adequate core cooling. The impact of post-72-hour actions has been reviewed by the staff in subsection "Safety Design Bases" of this SER.

B.2 Classification of Structures, Components, and Systems

Section 6.0, "Impacts to the Licensing Basis," of APP-GW-GLR-607 and APP-GW-GLR-161, Revision 2 describes the changes impacted to the COL application and provides the additional piping and components to the PXS. Subsection "Tier 1," states that "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." It further states that "As required by general design criterion 2 of Appendix A to 10 CFR Part 50, the PXS is designed to withstand the effects of natural phenomena and normal and accident conditions without loss of capability to perform its safety functions." The PXS containment recirculation downspout screens are identified as follows:

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

These component numbers will be added to the LNP Units 1 and 2 FSAR to supplement Table 2.2.3-1 of the AP1000 DCD, Revision 19, Tier 1. Mark-ups to Table 2.2.3-1 of the AP1000 DCD, Revision 19, Tier 1 and Table 3.2-3 of the AP1000 DCD, Revision 19, Tier 2, provided in Appendix B of APP-GW-GLR-607 and APP-GW-GLR-161, state that these eight additional downspout screens are not American Society of Mechanical Engineers (ASME) Code Section III components and the principal construction code is manufacturer standard.

In Section 6.0 of APP-GW-GLR-607 and APP-GW-GLR-161, under the subheadings "Tier 2," "Chapter 3: Impacted," the applicant states that, "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." Under the subheading "Tier 1," the applicant further states that ITAAC design requirements will be met for these eight added downspout screens.

On the basis of the safety and seismic classifications of these eight added downspout screens, their quality assurance requirements, and the fact that SRP 3.2.1, "System Quality Group Classification," and Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," do not provide specific guidance for the code of construction for non-ASME, non-pressure retaining components that belong to Quality Group C, the staff agrees that the use of manufacturer standards for the design of these downspout screens and the classification of AP1000 Safety Class C and seismic Category I is acceptable. Therefore, the staff finds the proposed FSAR revisions concerning these eight added downspout screens to be acceptable.

Section 6.0 of APP-GW-GLR-607 and APP-GW-GLR-161, Subsection "Tier 1," states that "As required by general design criterion 4 of Appendix A to 10 CFR Part 50, the PXS containment downspout piping would be safety-related and required to withstand normal and seismic design basis loads without losing functional capability." The following PXS containment downspout piping are the

proposed piping to be added to the LNP Units 1 and 2 FSAR to supplement Table 2.2.3-2 of AP1000 DCD, Revision 19, Tier 1:

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

Section 5.0, "Design Changes," Subsection "Polar Crane Girder and Internal Stiffener Modifications," Sub-subsection "1) PXS Downspout Piping," of APP-GW-GLR-607 and APP-GW-GLR-161 states that these added downspout piping are classified as AP1000 Safety Class C, seismic Category I. Mark-up of Table 2.2.3-2 to AP1000 DCD, Revision 19, Tier 1, provided in Appendix B of APP-GW-GLR-607 and APP-GW-GLR-161, further states that these added downspout piping are ASME Code Section III piping. According to the AP1000 DCD, Revision 19, Tier 2, Section 3.2.2, "AP1000 Classification System," Subsection 3.2.2.5, "Equipment Class C," Class C structures, systems and components are designed to codes and standards consistent with the guidelines for NRC Quality Group C. In addition, 10 CFR 50, Appendix B and ASME Code, Section III, Class 3 apply to pressure retaining components.

Section 6.0 of APP-GW-GLR-607 and APP-GW-GLR-161, Subsection "Tier 1," states that ITAAC design commitments will be met for these added downspout piping. In addition, Table 2.2.3-4 of the AP1000 DCD, Revision 19, Tier 1, provides ITAAC that 1) ensure the piping identified in Table 2.2.3-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements; 2) pressure boundary welds in piping identified in Table 2.2.3-2 as ASME Code Section III meet ASME Code Section III requirements; and 3) piping identified in Table 2.2.3-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.

On the bases that these downspout piping are designed to ASME Code Section III, Class 3 and the quality assurance requirements of 10 CFR 50, Appendix B, and that the ITAAC related to piping listed in Table 2.2.3-4 of the AP1000 DCD, Revision 19, Tier 1 apply, the staff finds the classification of this added downspout piping acceptable. Therefore, the staff finds the proposed FSAR revisions noted above to be acceptable.

B.3 Technical Specifications

In a letter dated February 7, 2014, the applicant submitted an exemption request titled "Supplement 3 to Submittal of Exemption Request and Design Change Description for Departure from AP1000 DCD Revision 19 to Address Containment Condensate Return Cooling Design," for LNP Units 1 and 2. As a result of the condensate return testing conducted at the Waltz Mill Test Facility, modifications to the polar crane girder, internal stiffener, and IRWST gutter designs were made. In addition, extensions of the gutter were added above the upper personnel airlock and upper equipment hatch. A downspout system was also added to capture condensation at the polar crane girder and stiffener locations.

These modifications result in minor editorial changes in a few sections of the TS and Bases (Chapter 16) in the COL application.

In a letter dated November 17, 2014, and titled “Supplement 5 to Submittal of Exemption Request and Design Change Description for Departure from AP1000 DCD Revision19 to Address Containment Condensate Return Cooling Design,” the applicant provided further details on the condensate return issue including other editorial modifications to the TS and Bases.

These changes are necessary to ensure that the TS and Bases accurately reflect the updated design and are described below.

LCO Section of B3.3.3 (Postaccident Monitoring (PAM) Instrumentation)

On page B3.3.3-4, in the last line of the first paragraph in Section 11, “In-Containment Refueling Water Storage Tank (IRWST) Water Level,” the text “...via a gutter.” is updated to “...via a gutter and downspouts.”

Background Section of B3.5.4 (Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Operating)

On page B3.5.4-1, in the first and second lines of the third paragraph of the Background section, the text “...PRHR HX operation, a gutter is provided...” is updated to “...PRHR HX operation, downspouts and a gutter are provided...”

Also in that paragraph, the text in the fourth and fifth line is updated from “...collected by the gutter is directed...” to “...collected by the downspouts or gutter is directed...”

TS and SR Sections for B3.5.4.7

On page 3.5.4-3 of the TS, the text in SR 3.5.4.7 is updated from “...gutter is...” to “...gutter and downspout screens are...”

On page B3.5.4-7, the text in the first and second lines of the only paragraph in SR 3.5.4.7 is updated from “...IRWST gutters to verify...” to “...IRWST gutters and downspout screens to verify...”

Also in that paragraph, the text in the fourth and fifth lines is updated from “...the gutters could become restricted.” to “...the gutter or downspout screens could become restricted.”

The staff finds the proposed changes in both Supplement 3 and 5 acceptable because the changes make the TS and Bases consistent with the revised design. Therefore, the staff finds the proposed revisions noted above to be acceptable.

B.4 Risk Results and Insights

The proposed departure did not entail any change to the models used for plant-specific PRA. However, FSAR Table 19.59-202, “AP1000 PRA-Based Insights”

item 1.e. was clarified to reflect how long the PRHR HX, IRWST, PCS, and condensate return features can now be relied on for core cooling.

The plant-specific PRA results and insights have been updated to account for this design change and departure. This is consistent with 10 CFR 52.79(d)(1) and is, therefore, acceptable to the staff.

Conclusion

The NRC staff reviewed LAR 16-06 and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the design change of the passive core cooling system, and there is no outstanding information expected to be addressed in the UFSAR for Summer Units 2 and 3 related to this amendment.

In addition, the staff concludes that the relevant information presented in LAR 16-06 is acceptable and meets the regulatory requirements discussed in Section 2.0 of this SER. The staff based its conclusion on the following:

- LAR 16-06 is acceptable because the described changes permit the licensee to meet the licensing basis within the bounds of the UFSAR.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations in 10 CFR 50.91(b), the designated South Carolina State official was notified of the proposed issuance of the amendment. The state official has no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, Standards for Protection Against Radiation. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite. Also, there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on December 15, 2016 (81 FR 90871). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

Because the exemption is necessary to allow the changes proposed in the license amendment, and because the exemption does not authorize any activities other than those proposed in the license amendment, the environmental consideration for the exemption is identical to that of the license amendment. Accordingly, the exemption meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the exemption.

5.0 CONCLUSIONS

The NRC staff has determined that pursuant to Section VIII.A.4 of Appendix D to 10 CFR Part 52, the exemption (1) is authorized by law, (2) presents no undue risk to the public health and safety, (3) is consistent with the common defense and security, (4) presents special circumstances, (5) the special circumstances outweigh the potential decrease in safety due to reduced standardization, and (6) does not significantly reduce the level of safety at the licensee's facility. Therefore, the staff grants the licensee an exemption from the Tier 1 information specified by the licensee in LAR 16-06 and evaluated in Section 3.0 of this safety evaluation.

Based on the considerations discussed in Section 3.0, and confirming that these changes do not change an analysis methodology, assumptions, or the design itself; the staff concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) there is reasonable assurance that the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. Therefore, the staff finds the changes proposed in this license amendment acceptable.

6.0 REFERENCES

1. Request for License Amendment and Exemption 16-06: Passive Core Cooling System (PXS) Condensate Return, letter from South Carolina Electric & Gas Company, dated November 18, 2016 (ADAMS Accession No. ML16323A335).
2. Levy Nuclear Plant final safety evaluation report (FSER), dated May 31, 2016 (ADAMS Accession No. ML16068A418).
3. VCSNS Units 2 and 3 UFSAR, Revision 4, dated July 1, 2016 (ADAMS Accession No. ML16193A229).
4. Final Safety Evaluation Report Related to the Combined Licenses for VCSNS Units 2 and 3, dated August 17, 2011 (ADAMS Accession No. ML110450305).
5. AP1000 DCD, Revision 19, dated June 13, 2012 (ADAMS Accession No. ML11171A500).
6. FSER Related to Certification of the AP1000 Standard Plant Design, NUREG-1793, Supplement 2, dated August 5, 2011 (ADAMS Accession No. ML112061231).