ENGINEERED SAFETY FEATURES

6.3 Passive Core Cooling System (Related to RG 1.206, Section C.III.1, Chapter 6, C.I.6.3, "Emergency Core Cooling System")

6.3.1 Introduction

The passive core cooling system is designed to provide emergency core cooling to mitigate design-basis events that involve a decrease in the reactor coolant system (RCS) inventory, such as a loss-of-coolant accident (LOCA), a decrease in heat removal by the secondary system, such as a feedwater system piping failure, or an increase in heat removal by the secondary system, such as a steam system piping failure. It also provides core cooling for shutdown events, such as a loss of the normal residual heat removal system during a shutdown operation. The passive core cooling system is designed to perform the following safety-related functions:

- emergency core decay heat removal
- RCS emergency makeup and boration
- safety injection
- containment sump pH control

During long-term operation, the AP1000 passive core cooling system must withstand the effects of debris loading on the containment recirculation screens, in-containment refueling water storage tank (IRWST) screens and the fuel assemblies. The concern that debris may lead to unacceptable head loss for the recirculating flow was raised in Generic Safety Issue (GSI)-191 and it is the topic of Bulletin (BL) 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors," and Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors." Section 6.3 of the AP1000 Design Control Document (DCD) includes an evaluation of this issue and Section 6.2.1.8 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," includes the staff's review, which was performed in accordance with the U.S. Nuclear Regulatory Commission (NRC)-approved evaluation methodology.

In order to support long term operation in a closed loop configuration, the AP1000 passive core cooling system must also achieve a sufficient condensate return rate such that inventory in the IRWST is maintained in order to retain the heat transfer capability of the passive residual heat removal (PRHR) heat exchanger (HX) (and return condensate to the sump during recirculation). Water is steamed from the IRWST during transients that require the PRHR-HX to remove decay heat from the RCS. The steam that reaches the containment shell condenses and returns to the IRWST through a gutter system. LNP DEP 3.2-1, a departure from the AP1000 DCD requested by the applicant reviewed below, proposes design changes to improve condensate return to the IRWST and quantifies the condensate losses associated with the pressurizing of the containment atmosphere, condensation on heat sinks within the containment, and from dripping or splashing from structures and components attached to the containment.

6.3.2 Summary of Application

Section 6.3 of the LNP combined license (COL) Final Safety Analysis Report (FSAR), Revision 6, incorporates by reference Section 6.3 of the AP1000 DCD, Revision 19. Section 6.3 of the DCD includes Section 6.3.2.2.7, "IRWST and Containment Recirculation Screens"; Section 6.3.8.1, "Containment Cleanliness Program"; and Section 6.3.8.2, "Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA."

In addition, in LNP COL FSAR Section 6.3.8.1, the applicant provided the following:

Tier 1 and Tier 2 Departures

The applicant proposed the following Tier 1 and Tier 2 departures (DEP) from the AP1000 DCD:

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

In LNP DEP 3.2-1, the applicant proposed a departure from Tier 1 and Tier 2 information related to design changes of the containment condensate return cooling system used during non-LOCA accidents. As described in a partial request for additional information (RAI) response dated June 27, 2014, the proposed Tier 2 departure includes changes to FSAR Chapters 3, 5, 6, 7, 14, 15, 16, and 19 as well as technical specification bases appearing in Part 4 of the combined license application (COLA) and cited in FSAR Chapter 16. In addition, the applicant requested an exemption from the incorporation by reference of AP1000 DCD Tier 1 information, specifically Tier 1 Subsection 2.2.3, Tables 2.2.3-1 and 2.2.3-2. The exemption request proposes to revise these tables by adding components to the containment condensate return cooling system of the Passive Core Cooling System to more effectively perform its design functions.

In LNP DEP 6.3-1, the applicant proposed changes to FSAR Chapters 5, 6, 7, 9, 15, and 19 to address a departure related to quantifying the duration that the PRHR-HX can maintain safe shutdown conditions, changing the indefinite duration to greater than 14 days.

AP1000 COL Information Items

- STD COL 6.3-1

The applicant provided additional information in STD COL 6.3-1 to address COL Information Item 6.3-1 identified in AP1000 DCD Table 1.8-2, "Summary of AP1000 Standard Plant Combined License Information Items." STD COL 6.3-1 requires the applicant to develop a containment cleanliness program to limit the amount of debris that might be left in the containment following refueling and maintenance outages.

Section 1.9 of the LNP COL FSAR incorporates by reference Section 1.9, "Compliance with Regulatory Criteria," of the AP1000 DCD. Section 1.9 of the DCD includes Section 1.9.4.2.3, "New Generic Issues," and Section 1.9.5.5, "Operational Experience."

In addition, in LNP COL FSAR Section 1.9, the applicant provided the following information related to the effect of debris accumulation on long-term cooling:

- STD COL 1.9-3

The applicant provided additional information in STD COL 1.9-3 to address the review of GSI-191.

- STD COL 1.9-2

The applicant provided additional information in STD COL 1.9-2 to address the review of BL 03-01 and GL 04-02.

6.3.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In conducting its review of STD COL 6.3-1, the NRC staff used the guidance and staff positions of RG 1.82, Revision 3, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," and NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Revision 0, Volume 1, and in the "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02," in NEI 04-07, Revision 0, Volume 2.

The changes proposed in LNP DEP 3.2-1 and LNP DEP 6.3.1 are also required to meet the following general design criteria (GDC), which also apply to the AP1000 DCD:

> Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, GDC 34, "Residual heat removal," as it applies to the capability of the PRHR-HX to perform safety related safe shutdown cooling of the RCS. Additionally, LNP DEP 3.2-1 and LNP DEP 6.3.1 are required to meet 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.

6.3.4 Technical Evaluation

The NRC staff reviewed Section 6.3 of the LNP COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the passive core cooling system. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in

evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (Vogtle Electric Generating Plant (VEGP) Units 3 and 4) were equally applicable to the LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 6 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- 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 not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the LNP COL application. This standard content material is identified in this safety evaluation report (SER) by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application.

Tier 1 and Tier 2 Departures

- 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 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.

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 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 technical specification bases in Chapter 16.

Exemption Request

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.

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 52.63(b)(1) allows the 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).

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.

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.

Authorized by Law

This exemption would allow the applicant to implement approved changes to Tier 1 information. This is a permanent exemption limited in scope to particular Tier 1 information, and subsequent changes to this Tier 1 information or any other Tier 1 information would be subject to full compliance by the applicant as specified in Section III.B 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, the requirements of Section III.B of Appendix D to 10 CFR Part 52. 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 Commissions regulations. Therefore, as required by 10 CFR 50.12(a)(1), the exemption is authorized by law.

No Undue Risk to Public Health and Safety

The underlying purpose of Section III.B of Appendix D to 10 CFR Part 52 is to ensure that the plant will be constructed and operated based on the DCD that has been incorporated in the plant's licensing basis.

Additions to the condensate return portion of the passive core cooling system support the system's intended design functions. The plant-specific Tier 1 DCD will continue to reflect the approved licensing basis for the applicant and will maintain a level of detail consistent with that which is currently provided elsewhere in Tier 1 of the plant-specific DCD. The affected design description in the plant-specific Tier 1 DCD will continue to provide the detail necessary to support the performance of the associated ITAAC. These proposed changes 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).

Consistent with Common Defense and Security

The proposed exemption would allow the applicant to implement modifications to the Tier 1 information requested in the applicant's submittal. This is a permanent exemption limited in scope to particular Tier 1 information. Subsequent changes to this Tier 1 information or any other Tier 1 information would be subject to full compliance by the applicant as specified in Section III.B 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.

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 being modified in the exemption request is to identify the components of the design of the condensate return portion of the passive core cooling system such that it 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.

The condensate return portion of the passive core cooling system as modified in the requested exemption will continue to perform its intended function and will, therefore, meet the underlying purpose of the rule. Accordingly, special circumstances are present because application of the requirement to incorporate the certified design information in Tier 1 Tables 2.2.3-1 and 2.2.3-2 is not necessary to achieve the underlying purpose of the rule. Therefore, the staff finds that special circumstances required by 10 CFR 50.12(a)(2)(ii) for the granting of an exemption from Section III.B of Appendix D to 10 CFR Part 52 exist.

Special Circumstances Outweigh Reduced Standardization

This exemption would allow the applicant to change certain Tier 1 information proposed in the LNP COL application. The key design functions of the passive core cooling system will be maintained. 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 the understanding that these changes support the design function of the passive core cooling system, it is likely that all other AP1000 licensees and applicants will request the same exemption, preserving the intended level of standardization.

However, this exemption request and the associated changes to LNP Tier 1 information demonstrate that there is a minimal change from the standard information provided in the generic AP1000 DCD. Consequently, the decrease in safety due to reduced standardization would also be minimal. For this reason, the staff determined that even if other AP1000 licensees and applicants do not request similar departures, the special circumstances outweigh the potential decrease in safety due to reduced standardization of the AP1000 design, as required by 10 CFR 52.63(b)(1).

No Significant Reduction in Safety

The proposed exemption would modify the passive core cooling system from the design presented the original application. The proposed changes to the PXS design 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.

Conclusion

The staff has determined that pursuant to Section VIII.A.4 of Appendix D to 10 CFR 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. Therefore, the staff grants the applicant an exemption from the requirements of 10 CFR Part 52, Appendix D, Section III.B, to allow a departure from elements of the certification information in Tier 1 of the generic DCD associated with the LNP Units 1 and 2 Tier 1 Tables 2.2.3-1 and 2.2.3-2, (collectively Tier 1 information).

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 these departures and included two supporting reports as Enclosures 1 and 2: APP-GW-GLR-161, Revision 1 (proprietary) and APP-GW-GLR-607, Revision 1 (non-proprietary), 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 references four calculations further described below. Enclosure 5 provides the applicant's request for exemption related to this topic. Enclosures 6 and 7 present, respectively, changes to AP1000 DCD Revision 19 and the LNP COLA information that will be included in a future revision to the COLA.

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 preserve the existing non-LOCA analyses in Chapter 15 that credit the PRHR-HX. The safe shutdown temperature evaluation, presented in DCD Chapter 19E Revision 19, assumes a constant condensate return fraction (i.e., 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 do 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 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 details 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 boxes are located on both the PCG and stiffener that are routed to common lines that empty into two collection boxes already existing on either side of the IRWST. These downspouts, collection boxes 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 preclude a single failure. 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 block 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 PCG and the stiffener. Subsection 4 details the dam added to the PCG to alleviate flow interactions between the containment shell and PCG 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 revision 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 No. ML14219A200). The containment response is analyzed in "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. A second calculation, "Condensate Return to IRWST for Long Term PRHR Operation," uses the parameters from WGOTHIC in concert with test results to provide a transient condensate loss fraction from the containment shell. Two further calculations – one using LOFTRAN, "AP1000 Safe Shutdown Temperature Evaluation," used for shorter duration calculations of the PRHR-HX performance leading up to safe shutdown, and another, "PRHR-HX Sizing/Performance," incorporating the condensate behavior from the WGOTHIC calculation and the shell behavior providing the transient return rate that is used to determine the duration of the cool down under various initial conditions – are used to provide input to the analysis depicted in the FSAR. 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 section.

Evaluation of Containment Response

Although the staff audited the calculations referenced in the submittal by the applicant, 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 their 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 No. ML14219A200), 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 a 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.

Using WGOTHIC, the applicant modeled the containment behavior during a transient involving the actuation of the PRHR-HX 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 leakage, changing the IRWST 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 changes serve to minimize the condensation rate on the outer 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.

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, staff requested 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 new approach to determine film thickness for condensation on surfaces utilizing a maximum contact angle for wetting in the design basis analyses and a realistic contact angle for "non-bounding, conservative" 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. After a non-LOCA transient, condensate return rates are significantly lower for the first few hours. In addition, there is no steaming from the reactor coolant system. Consequently, the effect of additional holdup resulting from the more conservative film holdup calculation is lessened and the level in the IRWST is 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 applicant's submittal shows a minor reduction in capability of the system beyond 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 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. The NRC staff's confirmatory analysis shows a peak of 3.9 pounds per square inch less than the applicant's WGOTHIC analysis, both of which are well below the design pressure. More importantly for this transient, the applicant's pressure 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 °F and outside containment is 115 °F. The analysis submitted used an in-containment initial temperature of 85 °F (capturing all the heat sinks as well as the IRWST) and an environment temperature of 115 °F. In RAI 7439, question 6.03-4, staff requested the applicant justify the assumption of 85 °F for the initial temperature of containment for the design-basis accident (DBA) analysis. In the response dated July 2, 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 for in-containment initial temperature was based on the use of an exterior temperature of 115 °F, the technical specification 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 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 for an ambient temperature of 115 °F. The influence of exterior temperatures is more dramatic on PRHR-HX performance: 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 presents an acceptably conservative value for a transient given a bounding environmental temperature of 115 °F, 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 staff's understanding of the system as discussed in this subsection, and is acceptable because water that does not return to the IRWST eventually reaches the lowest point in containment and fills the sump and associated floodup volumes.

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 some 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 becomes insufficient. This agrees with staff analysis of the performance of the system and is an acceptable change to the FSAR, discussed further in the following section.

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 sump. 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 of 90 percent 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.

"Containment Response Analysis for the Long Term PRHR Operation" provides inputs into "Condensate Return to IRWST for Long Term PRHR Operation" (ADAMS Accession No. ML14219A200), which calculates the total condensate losses on the containment shell as a function of time. 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. Therefore, in RAI 7439, question 6.03-5, 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.

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. Steaming from the reactor vessel cavity has competing effects on the system performance, cooling the reactor vessel, but also resulting in additional mixing below the operating deck, increasing the available passive heat sink area that must be accounted for in a conservative analysis. 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. Confirmatory analysis performed by NRC staff indicates that heat removal from vessel bottom head cooling is at least two orders of magnitude lower than that removed by the PRHR-HX. Within the safety-related 72-hour period of operation, however, steaming from the vessel bottom head has a negligible effect on system performance because very little steaming occurs up to this point in the transient. As a result, the staff finds the treatment of vessel bottom head cooling acceptable for this analysis.

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 3.2-1.**

Safety Design Bases

The PXS is comprised of six major components (CMTs, accumulators, IRWST, pH adjustment baskets, PRHR-HX, and ADS) that function together to perform 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.

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, 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 meet the following safety-related criteria:

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 overfilling the SG

Emergency decay heat removal design criteria 1 and 3 are safety-related design criteria that have been evaluated in DCD Chapter 15, Revision 19 for the events identified in Table 6.3-1 and reviewed in NUREG-1793. Previous staff review of DCD Chapter 15 events did not consider the possibility of PRHR-HX tube uncovery. 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 Chapter 15 is not affected. This caused 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 6.3-1).

Table 6.3-1 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 Aux.	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 Utility Requirements Document (URD) and Section 2.3.2 of the staff's corresponding safety evaluation (NUREG-1242, Volume 3) both state that a design requirement 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, staff requested clarification of the mission time for the PRHR-HX. In a response dated June 27, 2014, 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 6.3-1, terminate before the 72-hour operational requirement of the PRHR-HX. This caused staff to question the possibility of PRHR-HX tube uncovery during the 72-hour time period, and the resulting impact to Chapter 15 analyses. In RAI 7440, question 15.02.06-2, 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, 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, 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, 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, staff concluded that the applicant's use of LOFTRAN as described in WCAP-15644 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 uncovery of the PRHR-HX tubes. Modeling of tube uncovery in LOFTRAN uses a collapsed liquid level within the IRWST, where surface area of the PRHR-HX above the collapsed liquid level is conservatively not credited for heat removal, and transfer area below the liquid level is calculated as described in WCAP-14234 and approved in the staff's safety evaluation of the DCD in NUREG-1793. In RAI 7475, question 6.03-10, 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." Based upon considerations discussed in this paragraph, staff finds the use of LOFTRAN and the associated modeling of the PRHR-HX for the 72-hour analysis of LOAC to be acceptable.

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. 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, staff finds the submitted analysis of the 72-hour LOAC event acceptable.

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 staff to question whether containment backpressure effects on PRHR-HX performance were accounted for in Chapter 15. During the staff audit of applicant documents related to LNP DEP 3.2-1 and LNP DEP 6.3-1 (ADAMS Accession No. ML14219A200), 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, staff is interpreting extended operation to be at least 72 hours. Based upon the considerations discussed within this subsection, the staff found 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 6.3-1**.

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.

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. 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 staff to question whether ADS actuation can be inhibited by a low IRWST level. In RAI 7440, question 15.02.06-1, 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.

In addition, 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 conservatisms inherent in the film holdup analysis in in RAI 7439, question 6.03-3, staff finds no significant impact to containment floodup level as a result of LNP DEP 3.2-1. Based upon the considerations regarding ADS actuation and containment floodup level, staff finds that the safety injection function of the PXS is not impacted by LNP DEP 3.2-1.

Containment pH Control

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

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-related functions, 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 safety design basis for the PRHR-HX.

Transient analyses demonstrating safety-related design requirements must include methods to estimate the uncertainty in the calculation or ensure that the results of the analysis are demonstrably conservative to meet the acceptance criteria established in Section 15.0.2 of the Standard Review Plan. The shutdown temperature evaluation, provided in Section 19E.4.10.2, utilizes several conservative assumptions, but significant contributors to uncertainty attributed to core power, initial power and decay heat, are taken at nominal and mean values, respectively. Sections 6.3.3 and 7.4.1.1 of the revised FSAR refer to this analysis as "non-bounding, conservative." The use of nominal and mean values caused staff to question the basis for the

safe shutdown temperature requirement. In RAI 7475, question 6.03-11, staff requested the applicant to clarify the basis of the safe shutdown condition. In its response dated June, 27, 2014, the applicant stated the capability to bring the plant to the specified long term safe shutdown condition of 420 °F within 36 hours after an event is not a Chapter 15 success criteria, but is still considered a safety-related design requirement. The applicant stated the reasons for not utilizing bounding assumptions in the shutdown temperature evaluation as:

1. The PRHR-HX is an extremely reliable, safety-related component
2. The PRHR-HX is backed up by a separate, diverse, safety-related residual heat removal system (i.e., open loop cooling)
3. The probability that the PRHR-HX would be unable to adequately perform the specified safe shutdown function, and open loop cooling would be required to bring the plant to safe shutdown after a non-LOCA event is remote.
4. Establishing an RCS temperature of 420 °F is not a prerequisite for maintaining safe, stable RCS conditions. Maintaining a stable, post-accident condition with an RCS temperature higher than 420 °F would not result in exceeding any of the safety evaluation criteria evaluated in the bounding, conservative Chapter 15 analyses.

Although staff identifies the points raised by the applicant as valid, staff finds that the shutdown temperature analysis provided in Chapter 19E does not adequately account for uncertainty to the level that is required to demonstrate a safety-related design requirement. However, the Commission position identifying 420 °F as the safe shutdown temperature, stated in SECY-94-084, does not provide guidance on how achieving the safe shutdown temperature should be demonstrated. In Section 5.3.3 of the staff's safety evaluation of the URD, NUREG-1242, it is stated that passive system capabilities must be demonstrated by appropriate evaluations during detailed design analyses including a safety analysis, assuming the limiting single failure, to demonstrate that the capabilities of the passive systems are such that they can bring the plant to and maintain a safe stable condition, and that no transients will result in the specified acceptable fuel design limits or pressure boundary design limit being violated. The NRC staff finds that the bounding safety analyses presented in Chapter 15 of the DCD, and the DBA analysis performed in response to RAI 7440, question 15.02.06-2, satisfy the safety analysis requirement identified by the staff in NUREG-1242.

The staff identified in NUREG-1242 and SECY-94-084 that there are other plant conditions besides cold shutdown that constitute a safe shutdown state as long as reactor subcriticality, decay heat removal, and radioactivity containment are properly maintained on a long term basis. As highlighted in the applicant's response to RAI 7475, question 6.03-11, the same argument holds for the safe shutdown temperature of 420 °F. Based on the safe, stable shutdown condition demonstrated at a temperature above 420 °F in the DBA analyses provided in response to RAI 7440, question 15.02.06-2, staff finds that cooling the RCS below 420 °F within 36 hours following an event is not a safety-related requirement, but is a design goal that supports the safety-related requirement of obtaining and maintaining a safe, stable condition following an event. Based upon considerations discussed in this paragraph, staff finds the approach taken in Chapter 19E to demonstrate the safe shutdown requirement of cooling the RCS below 420 °F within 36 hours is consistent with SECY-94-084. The applicant's proposed changes under FSAR Section 6.3.1.1 in letters dated June 27, 2014, and July 24, 2014, remove the language that identifies the shutdown requirement of cooling the RCS below 420 °F within

36 hours as safety-related. Therefore, 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 3.2-1**.

The shutdown temperature evaluation, provided in Chapter 19E, is updated with the containment response evaluated by staff in the subsection "Evaluation of Containment Response." The boundary conditions that impact the LOFTRAN calculation consist of the containment pressure and condensate return rate to the IRWST as functions of time. Additionally, in RAI 7475, question 6.03-10, staff requested the tube plugging assumption used for the shutdown temperature analyses. In a letter 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 a number of tubes making up 5 percent of the heat transfer area. This reduction in allowable tube plugging is implemented in the revised shutdown temperature analysis. The results of the updated analysis demonstrate the RCS average temperature decreases below 420 °F within 36 hours. Staff performed confirmatory calculations as part of the review, which include a shutdown temperature evaluation. The result of staff's confirmatory calculation for the shutdown temperature evaluation is consistent with the applicant's submittal. Based upon the considerations within this subsection, staff finds the plant is consistent with SECY-94-084. The shutdown temperature evaluation discussed in this section is reflected in the applicant's proposed changes to FSAR Section 7.4 and Chapter 19E in the letter dated June 27, 2014. The staff finds the proposed FSAR revisions related to safe shutdown 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 3.2-1**.

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 the PRHR-HX Sizing/Performance Calculation which incorporates the changes discussed in the "Evaluation of Containment Response" subsection of this review. The PRHR-HX Sizing/Performance Calculation was audited by the staff (ADAMS Accession No. ML14219A200). Staff is not accepting the calculation methodology employed in the PRHR-HX Sizing/Performance Calculation to support the conclusion of 14-day operation of the PRHR-HX. However, 14-day operation of the PXS 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.

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 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, staff requested clarification on post-72-hour actions following non-LOCA events. In a response dated June 19, 2014, 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. The impact of post-72-hour actions has been reviewed by staff in subsection "Safety Design Basis" of this SER.

Classification of Structures, Components, and Systems

Section 6.0, "Impacts to the Licensing Basis," of APP-GW-GLR-607, Revision 1 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, state that these eight additional downspout screens are not ASME Code Section III components and the principal construction code is manufacturer standard.

Sub-subsection "Tier 2," "Chapter 3: Impacted," 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." Subsection "Tier 1," further states that ITAAC design requirements will be met for these eight added downspout screens.

On the bases that these eight added downspout screens are classified as AP1000 Safety Class C and seismic Category I; will be subjected to the quality assurance requirements of

10 CFR 50, Appendix B; and that SRP 3.2.1, "System Quality Group Classification," and RG 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 standard 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 to be acceptable, pending the staff's confirmation that the proposed FSAR revisions noted above are incorporated in the LNP Units 1 and 2 COL application. The staff is tracking these revisions as **LNP Confirmatory Item 3.2-1**.

Section 6.0 of APP-GW-GLR-607, 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:

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

Section 5.0, "Design Changes," Subsection "Polar Crane Girder and Internal Stiffener Modifications," Sub-subsection "1) PXS Downspout Piping," of APP-GW-GLR-607 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, 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, 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; 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 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 3.2-1**.

Technical Specifications

In a letter dated February 7, 2014, Duke Energy submitted an exemption request titled "Supplement 3 to Submittal of Exemption Request and Design Change Description for Departure from AP1000 DCD Revision19 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 PCG and stiffener locations. These modifications result in minor editorial changes in the following three sections of the Technical Specifications Bases (Chapter 16B): LCO for B3.3.3 (1 change), the Background section for B3.5.4 (2 changes), and the Surveillance Requirements for SR B3.5.4.7 (2 changes). These changes are necessary to ensure that the Technical Specifications Bases accurately reflect the updated design. These changes in the Bases are described below.

LCO section of B3.3.3 (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…"

Surveillance Requirement B3.5.4.7

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 gutters or downspout screens could become restricted."

The staff finds these proposed changes acceptable because the changes make the Bases consistent with the revised design. Therefore, the staff finds the proposed revisions noted above to be acceptable pending the staff's confirmation that the proposed revisions are incorporated in Part 4 of the LNP Units 1 and 2 COL application. The staff is tracking these revisions as **LNP Confirmatory Item 3.2-1**.

AP1000 COL Information Items

The following portion of this technical evaluation section is reproduced from Section 6.3.4 of the VEGP SER:

> *AP1000 COL Information Items*
>
> - *STD COL 6.3-1*
>
> *The applicant provided additional information in STD COL 6.3-1 to address COL Action Item 6.2.1.8.1-1 identified in NUREG-1793 and COL Information Item 6.3-1 identified in Table 1.8-2 of the AP1000 DCD. The applicant added information to BLN COL FSAR Section 6.3.8.1, "Containment Cleanliness Program," providing details of the program and procedures to minimize the amount of debris that might be left in containment following refueling and maintenance outages, including requirements for cleanliness inspections and limits on materials introduced into containment. TVA states that the cleanliness program will be consistent with the evaluation discussed in the AP1000 DCD.*
>
> *In its June 9, 2009, response to RAI 6.2.2-1, the applicant addressed the changes made to Revision 17 of the AP1000 DCD in APP-GW-GLE-002 and staff questions on cleanliness measurements with a modification to STD COL 6.3-1. This included adding that the cleanliness program will meet the DCD limits on latent debris, that housekeeping procedures will be implemented to return work areas to original conditions upon completion of work, and that a sampling program will be used to quantify the amount of latent debris. The sampling program is stated to be consistent with NEI 04-07 Volumes 1 (guidance report) and 2 (NRC safety evaluation). The sampling will be done after containment exit cleanliness inspections, prior to start up, and the results will be evaluated post-start up. Any non-conforming results will be addressed in the Corrective Action Program.*
>
> *The resulting cleanliness program is consistent with the RG 1.82 recommendation that procedures be in place to regularly clean the containment and to control and remove foreign materials from containment. The sampling program included in STD COL 6.3-1 is required to demonstrate that the latent debris found in containment is within the AP1000 DCD specified limits of 130 pounds, of which, up to 6.6 pounds may be fibrous material. The DCD specified limits were demonstrated to be acceptable through scale testing and analysis. Thus, STD COL 6.3-1 is consistent with the RG 1.82 recommendation that the cleanliness program be correlated to the amount of debris used in the*

long term cooling analysis. It is appropriate that the sampling program be in accordance with NEI 04-07, Volumes 1 and 2, because these documents contain the most recent NRC-approved evaluation methodology for cleanliness programs. The response to RAI 6.2.2-1 is acceptable and incorporation of the changes to STD COL 6.3-1 in the BLN FSAR will be tracked as ***Confirmatory Item 6.3-1****.*

The staff reviewed the following information in the BLN COL FSAR as it relates to the effect of debris accumulation on long term cooling:

- *STD COL 1.9-3*

The applicant added information to Section 1.9.4.2.3, "New Generic Issues," regarding Issue 191. The applicant states that the design aspects are addressed by the AP1000 DCD and the COL applicant portions are the protective coatings program discussed in BLN COL FSAR Section 6.1.2.1.6 and the containment cleanliness program discussed in BLN COL FSAR Section 6.3.8.1. The staff agrees that these are the only two COL items identified in the staff's review of GSI-191 from Section 6.2.1.8 of NUREG-1793.

- *STD COL 1.9-2*

The applicant added line items for Bulletin 03-01 and GL 04-02 in Table 1.9-204, "Generic Communications Assessment." The new information states that the design aspects are addressed in the AP1000 DCD and that the COL applicant aspects are addressed in BLN COL FSAR Section 6.3 for Bulletin 03-01 and BLN COL FSAR Section 6.3.8.1 for GL 04-02. The staff agrees that the design aspects of these generic communications are addressed in the staff's review of GSI-191 from Section 6.2.1.8 of NUREG-1793. The COL applicant aspects are addressed in the staff's review of BLN COL FSAR Section 6.1.2.1.6 and BLN COL FSAR Section 6.3.8.1.

Resolution of Standard Content Confirmatory Item 6.3-1

Confirmatory Item 6.3-1 required the applicant to update its FSAR to include the information related to the cleanliness program provided in the BLN applicant's above-mentioned June 9, 2009, response to RAI 6.2.2-1 (which was endorsed by the VEGP applicant). The NRC staff verified that the VEGP COL FSAR was appropriately updated with this information. As a result, Confirmatory Item 6.3-1 is resolved.

6.3.5 Post Combined License Activities

There are no post-COL activities related to this section.

6.3.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the passive containment cleanliness program, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable and meets the regulatory requirements and guidance discussed in Section 6.3.3 of this SER. The staff based its conclusion on the following:

- LNP DEP 3.2-1 and LNP DEP 6.3-1 are acceptable because the described changes permit the applicant to meet the licensing basis within the bounds of the updated licensing document.

- STD COL 6.3-1 is acceptable because the containment cleanliness program complies with the guidance in RG 1.82.

- STD COL 1.9-3, related to GSI-191, is acceptable because the only two items that need to be addressed by the COL applicant have been resolved. The protective coatings program is evaluated in SER Section 6.1.2, and the containment cleanliness program is evaluated under STD COL 6.3-1.

- STD COL 1.9-2, related to BL 03-01 and GL 04-02, is acceptable because the only two items that need to be addressed by the COL applicant have been resolved. The protective coatings program is evaluated in SER Section 6.1.2, and the containment cleanliness program is evaluated under STD COL 6.3-1.