Enclosure 2
Westinghouse APP-GW-GLR-607 Revision 3
(NON-PROPRIETARY VERSION)

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DOCUMENT COVER SHEET

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Open Items

This document has no open items.

Record of Revisions

Revision	Changes
0	Initial issue.
1	Added detail to discussion of how the change meets the ISG-11 criterion in response to NRC and APOG comments. Changed description of downspout symmetry. Relocated containment peak pressure response discussion to Chapter 6 heading to coincide with its place in the DCD. Clarified justification for why containment peak pressure response is not impacted. Added details to discussion for why Chapter 15 is not affected. Added text to describe Chapter 19 changes. Clarified description of compliance the GDCs and with 10 CFR 50.46 and 10 CFR 50, Appendix K. Clarified use of the words "DCD," "FSAR," "plant-specific DCD" and "licensing basis" throughout the document. Clarified status of proposed changes in the significant hazards determination and added further justification. Change "PXS gutter" to "IRWST gutter" to align with the terminology used in the licensing basis ("PXS gutter" is not used in the DCD or CLB). Addressed customer comments.
2	This revision updates the Condensate Return submittal to include the recent changes to the condensate return evaluation, including: - Updates to the Licensing Basis markups based on DCP APP-GW-GEE-5007. - Updates to descriptions of the supporting calculations based on changes in the condensate return calculation process. - Minor editorial changes
3	This revision updates the Condensate Return submittal to include the following changes: - The complete set of FSAR/DCD markups is included in Appendix B. - Minor editorial changes, highlighted with track changes.

Changes to Passive Core Cooling System Condensate Return

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~~April~~ July 2015

Changes to Passive Core Cooling System Condensate Return

Table of Contents

1.0 Introduction

According to the **AP1000**® Design Control Document Revision 19 (Reference 3), subsection 6.3.1.1.1, "Emergency Core Decay Heat Removal," among the safety-related design bases of the Passive Core Cooling System (PXS) is the capability for the Passive Residual Heat Removal Heat Exchanger (PRHR HX) to cool the Reactor Coolant System (RCS) to the safe shutdown condition of 420°F in 36 hours. The Nuclear Regulatory Commission Staff recommended, in SECY-94-084 (Reference 1), that reactor designs utilizing passive safety systems include a residual heat removal system capable of bringing the reactor to a safe shutdown condition of 420°F (215.6°C) or lower within 36 hours following non-loss of coolant accident (LOCA) events. To support the capability of the **AP1000** design to meet these safe shutdown conditions, a safe shutdown temperature evaluation was performed, which assumed a condensate return fraction for the PXS.

Through a series of design reviews, the efficiency of the condensate return to the In-Containment Refueling Water Storage Tank (IRWST) was questioned. These questions initiated an investigation to quantify the returned fraction of condensate to the IRWST. Supplementary testing of the **AP1000** design revealed opportunities to improve the design with regard to the condensate return fraction used to evaluate long-term plant cooldown. In addition, a rigorous analysis methodology was applied to characterize both the thermodynamic and the geometric phenomena involved in prolonged non-LOCA events. The Shutdown Temperature Evaluation in Chapter 19E of the **AP1000** Design Control Document (DCD) has been updated to analyze the PRHR HX performance with the design modifications to confirm it meets its licensing performance criterion of cooling the RCS to 420°F within 36 hours and maintaining a safe, stable condition. In addition, an extended design basis accident evaluation is performed out to 72 hours that shows compliance with the Chapter 15 non-LOCA acceptance criteria. These changes were evaluated against the NRC Interim Staff Guidance DC/COL-011; and were determined to meet the criteria for a change that should not be deferred while a license application is under review.

This report discusses expected PRHR HX operation, condensate loss mechanisms, condensate return test results and the design modifications made to enhance performance of the PXS. Most importantly, changes to the licensing basis supporting these modifications to the PXS are detailed and compliance to NRC regulations evaluated.

2.0 Background

Regulatory Basis

NRC regulations require that the **AP1000** design include a system to remove residual heat from the reactor core so that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded (Reference 2). PXS provides emergency core cooling during transients, accidents, and whenever the normal, nonsafety-related heat removal paths are unavailable. The PXS is capable of bringing the plant to and maintaining the plant in a safe, stable condition following shutdown.

Operation

The PRHR HX is safety-related and provides emergency core decay heat removal. It is located in the IRWST as shown on Tier 2 DCD Figure 6.3-2. The heat exchanger is used in non-LOCA transients and also in LOCA events until voiding begins in the RCS Hot Leg. For any non-LOCA event, the PRHR HX plays an integral role in decay heat removal, as opening one of the two outlet isolation valves initiates natural circulation of the heat exchanger, transferring heat from the RCS into the IRWST. This transfer of heat from the RCS to the IRWST causes the water in the tank to heat up, eventually become saturated, and initiate steaming of the tank.

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

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

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

Condensate is collected in the IRWST gutter, which extends around the circumference of containment and returns condensate to the IRWST.

Upon actuation of the PRHR HX, two air-operated valves in series are actuated to isolate the normal gutter drain path to the Liquid Radwaste System, and divert condensate to the IRWST. It

is important that sufficient condensate return is achieved during non-LOCA PRHR HX operation. The ability to maintain closed-loop PRHR HX cooling for long periods minimizes the probability that open-loop cooling will be needed. Although maintaining IRWST level above the top of the HX tubes is not a prerequisite for maintaining adequate decay heat removal, reduction of IRWST level to below the top of the tubes will begin to degrade the heat exchanger performance.

Analyses

The **AP1000** Design Control Document (DCD) Revision 19 Chapter 19E **Safe** Shutdown temperature evaluation analysis assumes a constant portion of steam discharged to the containment is returned back to the IRWST. However, there was not a strong basis justifying the efficiency of the PXS condensate return function. Therefore, the decision was made to conduct testing and to characterize condensate return with calculations that included quantification of steaming from the IRWST and the portion of that steam that condenses and returns to the IRWST.

Testing results showed the current design of the PCG, internal stiffener, and IRWST gutter contributed to losses at each location, which were larger than assumed. In addition to the losses due to the physical geometry of containment, there were also losses due to pressurization and heat-up of containment structures. These losses proved that the constant condensate return fraction assumed in the safety analyses was incorrect. Analytically, when the constant condensate return assumption was replaced with the experimental design return rates including losses, the resultant PRHR HX performance was degraded and could have affected the temperature profiles and the event times of the non-LOCA design basis accident (DBA) safety analyses described in Chapter 15 of the DCD if left uncorrected. The PCG and internal stiffener can be modified to improve condensate return such that the Chapter 15 design basis analyses would not be impacted.

The Safe Shutdown temperature evaluation is an analysis that extends beyond the end of the analyses described in Chapter 15 of the DCD. Therefore, to provide long term confirmation of acceptability with regard to the Chapter 15 non-LOCA acceptance criteria, it is deemed appropriate to consider an extended design basis accident evaluation out to 72 hours.

3.0 Discussion

Condensation

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

1) Steam to pressurize the containment
2) Steam condensation on passive heat sinks
3) Raining from the containment roof, Containment ring misalignment
4) Losses at the PCG and Stiffener
5) Losses at support plates attached to the containment vessel
6) Losses at the Equipment Hatch and Personnel Airlock
7) Losses at entry to IRWST Gutter

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

A full scale section of the containment wall was constructed at the Westinghouse Waltz Mill facility [ ]^{a,c}.
The testing at Waltz Mill is discussed in Section 4.0.

The locations of the Polar Crane Girder, internal stiffener, and IRWST gutter are illustrated in Figure 1.

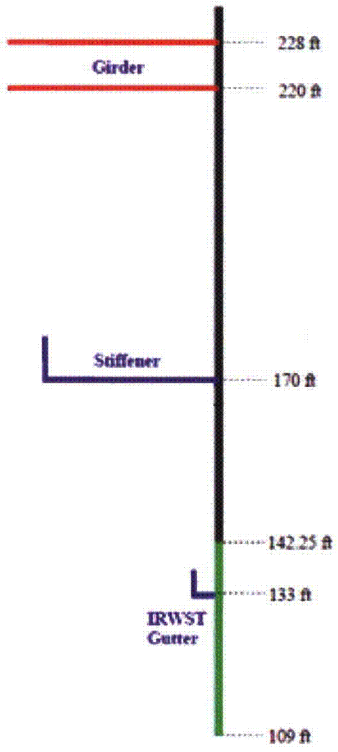


Figure 1: Example Containment Wall Schematic

4.0 Testing

A full scale section of the containment vessel (CV) wall, 55 feet tall, was constructed at the Westinghouse Waltz Mill Facility to accurately test and quantify the various loss mechanisms along the length of the containment wall. [

] a,c

[

] a,b,c

.] a,b,c

]a,b,c

]a,b,c

]a,c

Test Observations and Insights

Testing of the aforementioned configurations determined that a number of design optimizations were available to improve the performance of the condensate return system. Modifications were made at the half circle fabrication hole locations at the PCG and stiffener. A downspout piping system was deemed to be the most effective option for optimizing condensate return. The specific modifications needed at the PCG and stiffener are highlighted in Section 5.0.

[

]a,b,c

The details related to the condensate return testing conducted at Waltz Mill are available for inspection by the NRC.

5.0 Design Changes

As a result of the condensate return testing conducted at the Waltz Mill Test Facility, modifications to the PCG, internal stiffener, and IRWST gutter design 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. Each of these items is discussed in detail on the following pages.

Polar Crane Girder and Internal Stiffener Modifications

1) PXS Downspout Piping

A downspout piping network would be added to collect and transport condensation from the PCG and stiffener to the IRWST gutter collection boxes. The downspouts would consist of two downspout branches, each with two connections to the top of the PCG, two connections to the bottom of the PCG and two connections to the internal stiffener. Figure 3 illustrates the two downspout branches incorporated into the PXS system design. In each branch, the four connections from the PCG would join together into a common header which extends below the internal stiffener. The two connections from the stiffener would join together into a common line, which would connect to the header below the stiffener. The header would be routed to one of the two PXS collection boxes at either side of the IRWST. The downspouts would be situated with approximate symmetry around the circumference of containment. The common header for each branch would pass through the internal stiffener. These pass-through locations would include penetration sleeves to allow sufficient depth for collection at the stiffener downspout inlets.

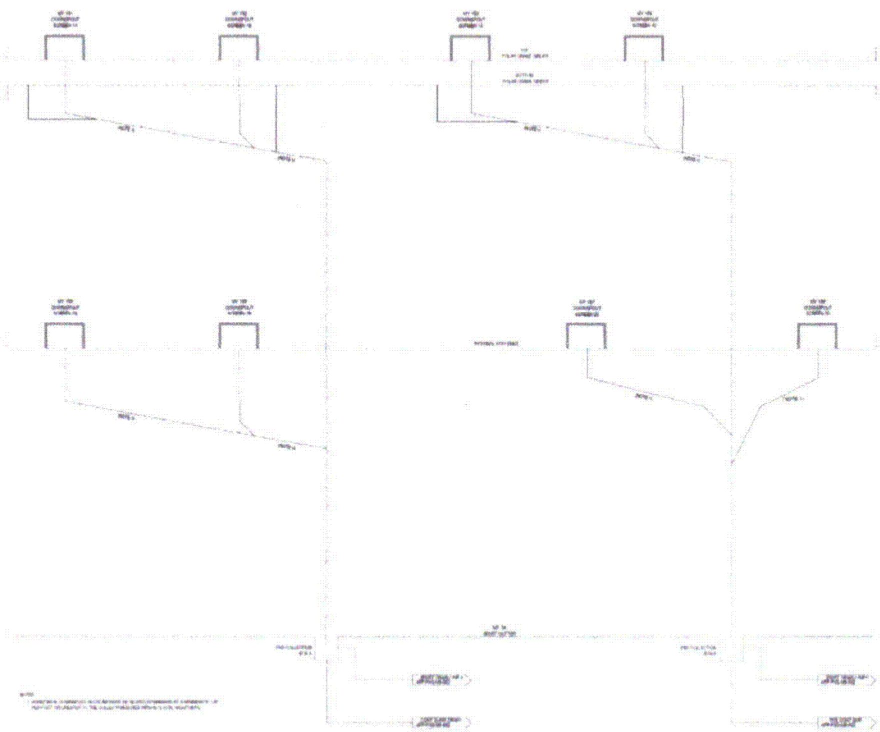


Figure 3: Simplified Downspout Configuration

The PCG boxes would be modified to allow condensate to drain from inside the PCG. The configuration of the collection boxes is modified to accommodate the additional downspouts. The piping is constructed of materials approved for use inside containment, consistent with DCD Tier 2, Section 6.1.1.4 (Reference 3). The downspouts will have tag numbers PXS-L301A/B to PXS-L310A/B. The downspouts are **AP1000** Safety Class C, Seismic Category I.

Pipe sizes were selected to prevent pipes from running full of water. The pipe sizes were also selected to accommodate a single failure (blockage) of one of the screens over the inlet to the downspouts. [

]a,c All sections of piping routed horizontally are sloped 1/8 inch per foot or greater downward toward each of the respective collection boxes. The PXS piping and instrumentation diagrams were changed accordingly.

2) Downspout Screen Design

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

Eight (8) new PXS downspout screens were added. The screens will have the tag numbers PXS-MY-Y81 to PXS-MY-Y88. The screen at each downspout entrance is **AP1000** Safety Class C, Seismic Category I.

.]

Figure 4 provides an illustration of a downspout screen.

[ ]a,c

The screens would be constructed of materials compatible with the post-accident environment, consistent with DCD Tier 2 subsection 6.1.1.4. Aluminum would not be used for these components. The screen would be designed to allow small debris to pass through; and provide sufficient flow area to accommodate design basis flow rates at the PCG and internal stiffener

locations. [.]^{a,c}

3) Blocking of PCG/Stiffener Fabrication Holes

The PCG is made up of sections, which are welded together around the circumference of containment. At each interface where the top and bottom plate sections are welded together, all four corners have openings to prevent multiple welds from joining at a common location. Therefore, the assembled PCG has open fabrication holes at the corners where the sections interface. The stiffener is similarly assembled in sections and contains fabrication holes at the interface where each section is welded together. DCD Tier 2 Figure 3.8.2-1 (Sheet 3 of 3) shows an example of the fabrication holes. The fabrication holes in the PCG and in the stiffener would be blocked.

4) Addition of Dam on the PCG

[ ]^{a,c} a dam is welded to the top of the PCG between the CV wall and the crane rail and to the bottom front edge of the PCG. [ ]^{a,c} The proposed top side dam is depicted in Figure 5.

[ ]^{a,c}

[

.]a,c

[

]a,c

Personnel Airlock and Equipment Hatch Gutter Routing

The original IRWST gutter was routed to the edges of the Upper Personnel Airlock and Upper Equipment hatch. [

] The IRWST gutter will be modified to have an upper gutter above each of the hatches that connects with the existing gutter. The extended gutter is of the same size as the sections which currently interface with the sides of the hatches. The gutter is sloped the same as the existing portions of the gutter. The extended portion of the gutter above the Upper Personnel Airlock is routed to interface with the lower portion of the gutter. Figure 8 shows a schematic of the proposed modification to the IRWST gutter to above the Upper Personnel Airlock. The Upper Equipment Hatch is similarly treated.

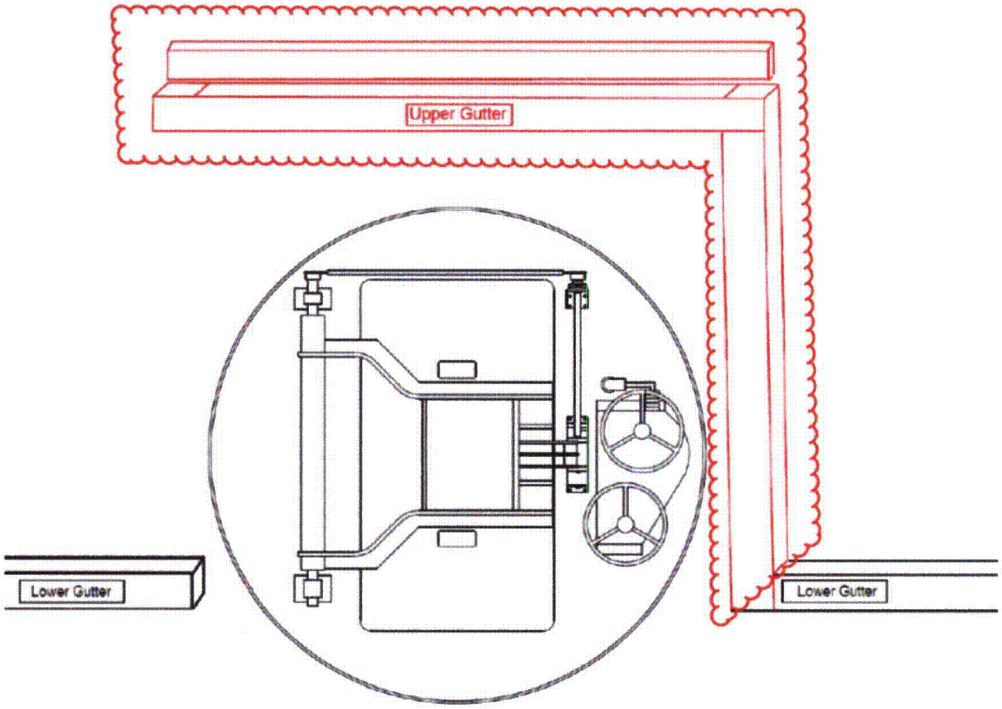


Figure 8: Personnel Airlock Gutter Modification Schematic

The Shutdown Temperature Evaluation summarized in DCD subsection 19E.4.10.2 was updated (Reference 7). The analysis is performed using the LOFTRAN computer code, as before, with a more detailed input for the condensate return fraction. Condensate return is affected by the containment pressure, which determines the PRHR HX heat sink (IRWST water) temperature. The WGOTHIC containment model described in DCD subsection 6.2.1.1.3 was used to model the peak containment pressure, with limited changes made to the model to maximize condensate losses (as opposed to maximizing peak pressure). The WGOTHIC model was used to calculate thermodynamic condensate losses due to containment pressurization, containment leakage and passive heat sink saturation (Reference 8). A bounding condensate loss from the containment shell (confirmed by Reference 9, which incorporates the changes described in section 5.0) is used as input to the WGOTHIC model, which in turn outputs a condensate return flow and a IRWST steaming rate. Subsequently, these two outputs are combined to develop a time dependent condensate return fraction, which is input to the downstream LOFTRAN calculation.

Reference 7 confirms that sufficient condensate is returned to the IRWST to maintain the IRWST water level above the top of the PRHR HX tubes meeting the success criteria of the design basis non-LOCA events described in DCD Chapter 15, even when extended out to 72 hours. The time-dependent condensate return fraction was input into the LOFTRAN code to demonstrate the ability of the PRHR HX to cool the RCS temperature to 420°F within 36 hours in a closed-loop mode of operation. This analysis demonstrated that the proposed changes to channel condensate that reaches the PCG and internal stiffener back to the IRWST maintains sufficient IRWST inventory. The analysis shows the plant can successfully meet this licensing performance criterion using conservative, non-bounding methods (Reference 7). A description of the

differences and conservatisms applied in these analyses is presented in Appendix A.

Based on the updates to the safe shutdown evaluation, the supporting licensing basis tables and figures need to be updated to reflect the current design. Section 19E.4.10.2, Table 19E.4.10-1, and the supporting figures, Figure 19E.4.10-1, -2, -3, and -4 will be updated by this change package.

6.0 Impacts to the Licensing Basis

Condensate return to the IRWST is discussed widely throughout DCD Revision 19 in conjunction with PRHR HX operation. Though the changes described in Section 5.0 do not change the condensate return concept or the safe shutdown temperature analysis methodology, the licensing basis changes proposed herein provide additional piping, components and adjustments to optimize the descriptions of the condensate return provisions and provide descriptions of the analysis methodology in the plant-specific DCD.

Tier 1

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

The downspout screens would support the capability of the PRHR HX to maintain the reactor in a safe shutdown condition by preventing large objects from entering the downspout piping. As required by General Design Criterion 2 of Appendix A to 10 CFR Part 50, the PXS is designed to withstand the effects of natural phenomena and normal and accident conditions without loss of capability to perform its safety functions. The PXS downspout screens would be safety-related, located on the Nuclear Island; and required to withstand design basis seismic and post-accident operating loads without losing the capability to perform their safety function. The component numbers for the following downspout screens are added to Table 2.2.3-1 to provide assurance that ITAAC design commitments will be met. The resultant change to Tier 1 is shown in Appendix B.

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

The downspout piping would support the capability of the PRHR HX to maintain the reactor in a safe shutdown condition by inhibiting containment floodup during PRHR HX operation and delaying the need for containment recirculation following RCS depressurization. As required by general design criterion 4 of Appendix A to 10 CFR Part 50, the PXS containment downspout piping would be safety-related and required to withstand normal and seismic design basis loads without losing functional capability. The additional downspout piping added to the PXS is captured in Table 2.2.3-2 to provide assurance that ITAAC design commitments will be met. The resultant change to Tier 1 is shown in Appendix B.

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

General design criteria 34 and 35 require that the PXS be capable of removing core decay and residual heat and provide an abundance of core cooling such that fuel design limits and the RCS

design conditions are not exceeded. As the PXS provides core decay heat removal during design basis events, performance of this safety-related function is confirmed through ITAAC design commitment 8.b. The changes described herein do not change the commitment to complete the performance test of the PRHR HX. No further changes to Tier 1 are required to assure the desired PXS performance is confirmed in this performance test.

Tier 2

Chapter 3: Impacted

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. The screens added to Tier 1 Table 2.2.3-1 are also added to Table 3.2-3 to capture these requirements. The markup to Table 3.2-3 is shown in Appendix B.

Pictorial detail of the PCG is shown in DCD Chapter 3. Figure 3.8.2-1 (Sheet 3 of 3) shows the fabrication holes in the top right figure. As the fabrication holes in the PCG would be blocked in the modified configuration, this detail ~~would~~ will be removed from this figure. The changes are shown in Appendix B.

Chapter 5: Impacted

In subsection 5.4.11.2, cross reference to Figure 6.3-2 is changed to Figure 6.3-1 for consistency across the chapters. ~~The changes to chapter 5 are shown in Appendix B.~~

Changes were also made to subsection 5.4. ~~1.~~14.1 ~~in Reference 11.~~ to add discussion about the design bases of the PRHR HX.

These changes are shown in Appendix B, note that this portion of the markups was originally made by References 10 and 11.

Chapter 6: Impacted

To reflect the changes to the PXS system, the additional downspout piping is captured in the gutter discussions of UFSAR subsection 6.3 and on a new sheet of the PXS piping and instrumentation diagrams (P&IDs). In order to add the new P&ID sheet to the licensing basis, Figure 6.3-1 will be expanded to include all sheets of the PXS P&IDs and Figure 6.3-2 will not be used.

- Subsection 6.3.1.1.1 ~~would~~ will be updated to describe the downspouts in the safety-related design criteria and provide clarification regarding the safety related design basis of the PRHR HX during closed-loop passive core cooling.
- Subsection 6.3.1.1.4 will be updated to provide discussion about the safety related cooldown capabilities of the PRHR HX. Additional discussion has also been added regarding operator action to block ADS if the RCS is in a safe, stable condition.
- Section 6.3.1.2 was added to discuss the nonsafety design basis of the PRHR HX.
- The original section 6.3.1.2 was renumbered as section 6.3.1.3.
- Subsection 6.3.2.1.1 will be updated to include the intermediate collection points of the safety-related gutter arrangement and additional discussion about the operation of the

condensate return features of passive core cooling system.

- Subsections 6.3.2.2.7 and 6.3.2.2.7.1 ~~would~~ will be updated to clarify the number of screen sets in the PXS and to which set of screens the criteria in this section apply.
- Subsection 6.3.2.2.7.2 ~~would~~ will be updated to clarify the condensate return gutter arrangement related to LOCA operation.
- Section 6.3.2.8 will be updated to add discussion regarding the manual actions that can be taken by the operator to block ADS if the RCS is in a safe, stable condition.
- Section 6.3.3 will be updated to provide additional information regarding the operation of the PRHR HX.
- Subsection 6.3.3.2.1.1 "Loss of AC Power to the Plant Auxiliaries" will be added to describe how the plant will respond during a loss of ac power combined with a loss of main feedwater event.
- Subsection 6.3.3.4.1 will be updated to remove the term "indefinitely" from the length of time the heat sinks need to function.
- Figure 6.3-1 will be relabeled Figure 6.3-1 (Sheet 1 of 3). This editorial change is the only change made to this figure. No technical changes are made.
- Figure 6.3-2 will be relabeled Figure 6.3-1 (Sheet 2 of 3). On relabeled Sheet 2, the IRWST gutter has been relocated to a new sheet 3 of the PXS P&IDs. Sheet 2 has been modified to include continuation flags for condensate returning to the IRWST originating from PXS Collection Boxes A and B in the IRWST gutter.
- Figure 6.3-1 (Sheet 3 of 3) is a new P&ID sheet and will be added to the licensing basis. This new figure shows the relocated IRWST gutter and the screens and piping comprising the PXS downspouts originating from the Polar Crane Girder and internal stiffener.
- Figure 6.3-2 will not be used.
- The Chapter 6 List of Figures will be updated to reflect the PXS figure relabeling and the additional PXS P&ID sheet.
- Reference to Figure 6.3-2 in subsection 6.3.2.1 is changed for consistency.

The changes Chapter 6 are shown in Appendix B. Note that some of these markups were originally made by References 10 and 11.

The WGOTHIC peak containment pressure analysis was considered during the course of testing and analysis for this change; and was determined not to be affected by this change for the following reasons. With regard to peak containment pressure, the limiting design basis event is a double-ended guillotine cold leg break (DECLG) LOCA. The containment peak pressure for the DECLG LOCA case is not sensitive to the time-dependent condensate return, as the peak pressure is reached well before condensate return plays a factor in the event. Additionally, in the later stages of the transient (24 and 72 hours) the beneficial effects of condensate return are not considered in the containment peak pressure and temperature analysis. The current WGOTHIC containment response model assumes condensate that reaches the polar crane girder and internal stiffener is deposited in the containment sump and no longer contributes to the film thickness at lower elevations of the containment wall. Therefore, the containment analysis methodology remains bounding and is consistent with the modified design as described in Section 5.0.

~~In addition to the impacts noted above, additional impacts are documented in References 10 and 11.~~

Chapter 7: Impacted

~~The impacts to Chapter 7 are documented in References 10 and 11.~~

The changes ~~made to~~that will be made to section 7.4 and 7.4.1.1 are shown in Appendix B and are considered clarifications to the original words about safe shutdown. The changes that will be made to subsection 7.4.1.1 are additions to describe the operation of the PRHR HX during a design basis event. Note that the 7.4.1.1 markups were originally made by References 10 and 11.

Chapter 9: Impacted

~~The impacts to Chapter 9 are documented in References 10 and 11.~~Table 9.5.1-1 will be updated to add clarifications about the safe shutdown operation of the AP1000 plant. The changes to Chapter 9 are shown in Appendix B, note that this portion of the markups was originally made by References 10 and 11.

Chapter 14: Impacted

In Table 14.3-2, cross reference to Figure 6.3-2 ~~is changed~~will be changed to Figure 6.3-1 for consistency across the chapters. The changes to Chapter 14 are shown in Appendix B.

Chapter 15: ~~No impact~~Impacted

Chapter 15 design basis transients that credit PRHR HX operation, along with the analysis run time are listed in Table 1. In these analyses, a constant condensate return fraction was used for the safety analysis models supporting Chapter 15. However, though the condensate return fraction has changed, the transient analyses in Chapter 15 bound the plant response expected as a result of the proposed design changes. During the transients which credit PRHR HX operation, there is no impact to the heat transfer rate of the heat exchanger until the point that the water level in the IRWST drops below the top of the tube sheet, reducing the available heat transfer area. For the transient analyses in Chapter 15, the response will not change because even if the time-dependent condensate return fraction were applied, the PRHR HX would remain submerged well beyond the duration of the relevant design basis analyses listed in Table 1. In order to demonstrate that there is no impact to Chapter 15, an extended 72 hour case of the limiting event was completed (Appendix D of Reference 7). The analysis shows all Chapter 15 acceptance criteria are met through the duration of the analysis. To add additional conservatism to the WGOTHIC calculated variable condensate return fraction, the analysis assumes a 5% reduction throughout the transient.

Table 1

DCD Chapter 15 Design Basis Accidents Crediting PRHR HX Operation

DCD subsection	Transient Name	Run Time
15.2.2	Loss of external electrical load$^{(1)}$	(2)
15.2.3	Turbine trip$^{(1)}$	<1 minute
15.2.6	Loss of ac power to the plant auxiliaries	<6.2 hours

Table 1

DCD Chapter 15 Design Basis Accidents Crediting PRHR HX Operation

DCD subsection	Transient Name	Run Time
15.2.7	Loss of normal feedwater flow	<5.5 hours
15.2.8	Feedwater system pipe break	<3.2 hours
15.5.1	Inadvertent operation of the core makeup tanks during power operation	<8.6 hours
15.5.2	Chemical and volume control system malfunction that increases reactor coolant inventory	<5.7 hours
15.6.3	Steam generator tube rupture	<6.7 hours

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

~~Additional changes to Chapter 15 are noted in References 10 and 11.~~

While the discussion above demonstrates the changes to the gutter systems and condensate return features of the PXS do not impact the licensing basis, minor changes were made to the surrounding Chapter 15 text. These changes are shown in the markups in Appendix B, note that this portion of the markups was originally made by References 10 and 11.

- Section 15.0.13 will be updated to make a minor editorial change.
- Subsection 15.2.6.1 will be updated to add description to the operation of the PRHR HX.

Chapter 16: Impacted

The Technical Specification Bases ~~would~~ will be updated to include the downspouts in the descriptions of the gutter arrangement.

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

The changes to Chapter 16 are shown in Appendix B.

Chapter 19: Impacted

Two portions of Chapter 19 were impacted. Table 19.59-18, will be updated to add clarification about the PRHR HX. This markup is shown in Appendix B. Note that this portion of the markups was originally made by References 10 and 11.

The second portion is in Chapter 19E. Per SECY-94-084, the NRC recommends the requirement of 420°F or below as a safe, stable shutdown condition. The results of the shutdown temperature evaluation are represented in DCD Revision 19 subsection 19E.4.10.2, Table 19E.4.10-1 and Figures 19E.4.10-1 through 19E.4.10-4. The original evaluation was performed at best estimate conditions, with a number of conservatisms maintained, and assumed a constant condensate return rate. The plant response after shutdown following non-LOCA events was reanalyzed with a series of interdependent calculations, made using non-bounding, conservative assumptions. The updated information flow between these calculations is illustrated in Figure 9.

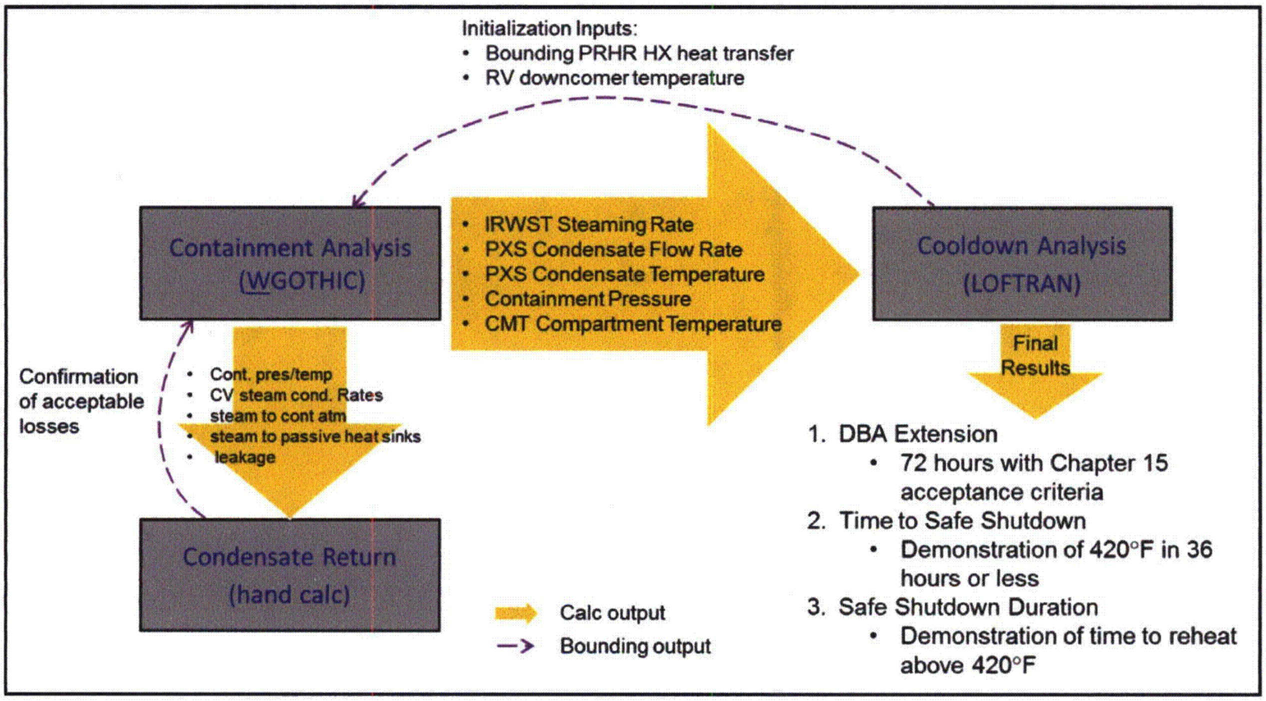


Figure 9: Calculation inter-relationships

The design changes described in Section 5.0 ensure sufficient condensate is returned to the IRWST to preserve PRHR HX performance after shutdown following a non-LOCA event. To verify the effectiveness of the proposed changes to the PXS system, several analyses (discussed in Appendix A) were performed, which incorporate the lessons learned about condensate return from design review and testing. As described in subsection 4.2 of the Shutdown Temperature Evaluation (Reference 7), loss of normal feedwater coincident with loss of ac power event was identified to be the most limiting transient with regard to PRHR HX performance. The Shutdown Temperature Evaluation is performed to demonstrate the adequacy of the PRHR HX to reduce the core average temperature to 420°F within 36 hours after shutdown following a loss of normal feedwater coincident with loss of ac power event. The containment peak pressure and temperature design limits are not challenged by the long-term loss of normal feedwater with loss of ac power event that forms the limiting basis for the safe shutdown temperature evaluation, as the maximum pressure reached during the loss of normal feedwater coincident with loss of ac power event does not approach the containment design pressure.

Following a loss of ac power event, reactor coolant system energy is slowly transferred to the IRWST following actuation of the PRHR HX. The water in the IRWST will begin to heat up, eventually coming to a boil. The steam released by boiling of the IRWST will cause the containment temperature and pressure to increase. To evaluate the containment response to

IRWST steaming and time-dependent condensate return on PRHR HX performance, minor modifications were made to the approved WGOTHIC containment response model to increase condensation and produce conservative results. The Containment Analysis (APP-PXS-M3C-071, Reference 8) performed with the WGOTHIC model calculates several key inputs to both the Condensate Return calculation (Reference 9) and the Safe Shutdown Temperature Evaluation (Reference 7). The Containment Analysis provides the containment pressure and temperature, containment vessel steam condensate rates, steam to the containment atmosphere, transient mass of condensate on the passive heat sinks, and steam lost to containment leakage. APP-PXS-M3C-072 (Reference 9) determines the percentage of losses as a function of flow from the containment vessel shell and this percent loss is bounded as an input to the WGOTHIC model. The Containment Analysis also provides the IRWST steaming rate, the PXS condensate flow rate and temperature, containment pressure, and CMT compartment temperature to the Safe Shutdown Temperature Evaluation (APP-SSAR-GSC-536, Reference 7). This analysis performs the Safe Shutdown Temperature Evaluation using the modified LOFTRAN computer code (described in DCD subsection 15.0.11.2).

The Shutdown Temperature Evaluation (Reference 7) and its analytical inputs implementing the variable condensate return fraction produced by the proposed changes demonstrate the efficacy of the proposed changes in helping to bring the RCS temperature to 420°F in less than 36 hours. Therefore, the plant continues to meet the licensing performance criterion established in SECY-94-084.

Changes to Chapter 19 ~~would~~ will include changes to subsection 19E.4.10.2, Shutdown Temperature Evaluation, to describe the analysis methodology for a non-LOCA shutdown event, the time for the cold leg and core average temperatures to reach the specified safe, stable condition after shutdown following a loss of ac power event, and updates to corresponding tables and figures, which further detail the sequence of events. The changes to Chapter 19 are shown in Appendix B.

7.0 Regulatory Evaluation

The design changes and the changes to the licensing basis described in Sections 5.0 and 6.0 were evaluated against the NRC Interim Staff Guidance DC/COL-ISG-011 (Reference 4). That evaluation determined the changes are necessary to reflect a "significant technical correction associated with the design described in the licensing document that, if not changed, would preclude operation within the bounds of the licensing basis" (Reference 4). Specifically, without the changes described in Sections 5.0 and 6.0, the capability of the PRHR HX to maintain the RCS in a safe, stable condition as described in DCD Chapter 19E, "Shutdown Temperature Evaluation," would be challenged. Without the proposed changes, less condensate would be returned to the IRWST and the PRHR HX tubes would uncover sooner than anticipated. PRHR HX performance degrades as the heat exchanger tubes become uncovered. Without the proposed changes, the conclusions of the LOFTRAN shutdown temperature evaluation described in Chapter 19E.4.10.2 of DCD Revision 19 would have been inaccurate. Without the proposed changes, the descriptions of the Chapter 15 non-LOCA analyses would have required revision as well. Therefore, the changes meet the criteria for a change that should not be deferred during review of an application for a combined license.

This section provides an evaluation of the updated PXS condensate return design against the regulations satisfied by the PXS and analyses supporting the PXS design. In addition, a discussion of significant hazard considerations is included for informative purposes.

<u>Applicable Regulatory Requirements and Criteria</u>

Title 10 Code of Federal Regulations, Part 52, Appendix D, Section VIII applies to changes to Tier 1 and Tier 2 changes which involve changes to Tier 1 . The Tier 2 changes to the licensing basis described in Section 6.0 also require departures from Tier 1 information. Therefore, NRC approval is required prior to implementing the changes addressed in this departure.

Appendix A to 10 CFR Part 50:

1) General Design Criterion (GDC) 2, "Design bases for protection against natural phenomena," requires that the PXS be designed to withstand the effects of natural phenomena and normal and accident conditions without loss of capability to perform its safety functions.

   The PXS, including the additional PXS components added for the condensate return function, is designed to meet Seismic Category I design requirements; and is protected from the effects of external events such as earthquakes, tornadoes, and floods.

2) GDC 4, "Environmental and dynamic effects design bases," requires that the PXS be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.

   The PXS is designed to accommodate the environmental conditions associated with all modes of operation, and to prevent excessive dynamic events. Additionally, piping line sizes are selected to prevent steam flashing in the downspout piping. The additional piping and screens are constructed of materials compatible with the post-accident environment, consistent with DCD subsection 6.1.1.4.

3) GDC 5, "Sharing of structures, systems, and components," specifies that the PXS is prohibited from being shared among nuclear power units unless it can be demonstrated that sharing will not impair their ability to perform their safety function.

   The PXS contains no components that are shared between nuclear power units. Thus the PXS design to meets the requirements of GDC 5.

4) GDC 17, "Electric power systems," specifies that an onsite electric power system and an offsite electric power system be provided to provide sufficient capacity to ensure that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded and that the core is cooled during anticipated operational occurrences and accident conditions.

   The **AP1000** design does not require ac power sources to mitigate design-basis events. Likewise, the PXS condensate return design relies on natural forces; and does not require power sources to perform its safety-related functions. The components added are passive components maintained in their safety-related configuration for the duration of operation. Thus the **AP1000** design meets the requirements of GDC 17; and continues to support an exemption to the requirement of having two offsite power sources.

5) GDC 27, "Combined reactivity control systems capability," requires the PXS be designed to have a combined capability, in conjunction with poison addition, of reliably controlling reactivity changes to assure that, under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

   The proposed changes do not affect the capability of the PXS to control core reactivity with poison addition. The proposed changes do affect the ability of the PXS to provide adequate core cooling by increasing the fraction of condensate returned to the IRWST during an event where steaming from the IRWST to containment occurs.

6) GDC 34, "Residual heat removal," requires the plant be designed with a residual heat removal system to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the RCPB are not exceeded.

   The PRHR HX is capable of cooling the RCS in accordance with the provisions of SECY-94-084. The changes proposed in this departure assure the percentage of condensate returned to the IRWST over time exceeds the return fraction necessary to ensure adequate PRHR HX performance. With the proposed changes, the updated safe shutdown evaluation continues to demonstrate that the plant complies with its functional requirement of cooling the RCS to 420°F within 36 hours. In addition, the extended design basis accident evaluation confirms compliance with the Chapter 15 non-LOCA acceptance criteria for 72 hours during the most limiting event.

7) GDC 35, "Emergency core cooling," requires the PXS be able to provide an abundance of core cooling to transfer heat from the core at a rate so that fuel and clad damage will not interfere with continued effective core cooling.

   The functionality of components of the PXS providing direct injection to the RCS for emergency core cooling is not affected by the changes in this departure. The changes described herein provide assurance the PRHR HX can provide adequate core cooling during non-LOCA events, in conjunction with core makeup tank and accumulator

operation. Thus the PXS continues to satisfy GDC 35.

8) GDC 36, "Inspection of emergency core cooling system," requires the PXS be designed to permit appropriate periodic inspection of important components.

   The proposed modifications are accessible to periodic inspections. The proposed piping and downspout screens are accessible for inspection and maintenance as necessary. The PXS continues to comply with GDC 36.

9) GDC 37, "Testing of emergency core cooling system," requires the PXS be designed to permit appropriate periodic pressure and functional testing.

   The proposed modifications do not affect the ability to periodically test the emergency core cooling capability of the PXS. The periodic inspection and testing program for the PXS does not include requirements specifically for testing condensate return to the IRWST since steaming the containment is not practical. However, the added components are accessible for periodic inspection to confirm structural integrity and may be flow tested to confirm overall operability.

10) 10 CFR 50.46 and Appendix K to 10 CFR Part 50, as they relate to analysis of PXS performance, ensure the evaluation is accomplished in accordance with an acceptable evaluation model.

    The proposed design and licensing basis changes ensure the Chapter 6 and Chapter 15 safety analyses are not affected and remain bounding. The design basis analysis methods used to evaluate performance of the PXS include only methods approved for use by the Commission. The changes proposed do not include a new method of analysis.

<u>No Significant Hazards Consideration Determination</u> (provided for informative purposes only)

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

   Response: No.

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

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

could increase radiological dose consequences. Thus the radiological releases associated with the Chapter 15 accident analyses are not affected.

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

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident.

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

   Response: No.

   An evaluation of the downspout and gutter return subsystem determined the components are capable of acceptably performing their safety-related function, even if one of the downspouts were blocked. The new equipment does not interface with components in other systems that provide safety-related or defense-in-depth support to the plant, thus precluding the possibility condensate could be diverted to another system before reaching the gutter. The affected equipment does not interface with any component whose failure could initiate an accident, or any component that contains radioactive material. The modified components do not incorporate any active features relied upon to support normal operation. The downspout and gutter return components are seismically qualified to remain in place and functional during seismic and dynamic events. Consequently, the proposed component changes do not introduce new failure modes, interactions or dependencies, the malfunction of which could lead to new accident scenarios. Therefore, the proposed changes do not create the possibility for a new or different kind of accident.

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

   Response: No.

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

   The DCD Chapters 6 and 15 analyses results are not affected, thus margins to their regulatory acceptance criteria are unchanged. No design basis safety analysis or acceptance criterion is challenged or exceeded by the proposed changes, thus no margin of safety is reduced.

References

1) SECY-94-084, "Policy and Technical Issues Associated With the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," March 28, 1994
2) Appendix A to Part 50, Title 10 Code of Federal Regulations, General Design Criterion 34 - Residual heat removal
3) APP-GW-GL-700 Revision 19, "**AP1000** Design Control Document"
4) DC/COL-ISG-011, Final Interim Staff Guidance, Finalizing Licensing-basis Information, ML092890623, (Notice of Availability, ML092890577, November 2009).
5) WCAP-15846 (proprietary) and WCAP-15862 (nonproprietary), revision 1, "WGOTHIC Application to AP600 and **AP1000**."
6) [Not used]
7) APP-SSAR-GSC-536 Revision 3, "**AP1000** Safe Shutdown Temperature Evaluation," (proprietary).
8) APP-PXS-M3C-071 Revision 2, "Containment Response Analysis for Long-Term PRHR Operation," (proprietary).
9) APP-PXS-M3C-072 Revision 2, "Condensate Return to IRWST for Long Term PRHR Operation," (proprietary).
10) APP-FSAR-GF-006 Revision 0, "Levy County Units 1 & 2 Request for Additional Information Question 06.03-10, 06.03-11, and 06.03-12: Clarification of Passive Residual Heat Removal Heat Exchanger Design Requirements – Supplemental Response"
11) APP-FSAR-GF-007 Revision 0, "Levy County Units 1 & 2 Request for Additional Information Question 06.03-10, 06.03-11, and 06.03-12: Clarification of Passive Residual Heat Removal Heat Exchanger Design Requirements – Supplemental Response"

Note that the information contained in References 10 and 11 have been submitted to the NRC via DEF letter NPD-NRC-2014-021.

Appendix A

Summary of Changes to the Technical Basis and Transient Analyses

A.1 Non-LOCA Event Considerations

A station blackout event is postulated to occur less than once during the 60 year lifetime of the **AP1000** plant. This event is postulated to occur as a result of a loss of offsite power, failure of the rapid power reduction system such that the turbine trips and then failure of the Onsite Standby Power System to initiate. This sequence of events results in a complete loss of alternating current (AC), or station blackout, at the site. When this occurs, the reactor will trip on low RCS flow. The wide range level in the steam generators will decrease below the Low setpoint. When this occurs, the protection and safety monitoring system (PMS) will automatically open the normally closed air-operated valves isolating the PRHR HX and initiate flow through the heat exchanger.

Within 2 to 4 hours after transient initiation (the exact time depends on assumptions of the transient analysis), the water in the IRWST will reach saturation due to energy deposition from the PRHR HX. Boiling and steam release from the IRWST will increase the containment pressure and temperature. Note that there is a good chance that AC power (either offsite or onsite) would be recovered in this time frame, such that long term PRHR HX operation is very unlikely. Passive Containment Cooling System (PCS) flow will be delivered through the PCS valves and initiate gravity driven sub-cooled liquid flow to the outside of the containment vessel from the passive containment cooling water storage tank (PCCWST) after the High containment pressure signal is reached. The PCS water flow provides evaporative cooling for the containment shell. Steam released in the containment will condense on the inside surface of the containment shell, and the condensate will be collected in the gutters and returned back to the IRWST.

During this transient, some of the steam that is generated by the PRHR HX will be returned to the IRWST. During the first phase of the transient, the steam which is released into containment from the IRWST will contact surfaces and equipment in containment that are initially at cooler temperatures. As a result, condensation will occur on these surfaces and result in loss of IRWST inventory. This condensation will continue until these structures heat up to the containment temperature. Another way steam will be lost is due to the pressurization of the containment atmosphere. Finally, steam will condense on the containment vessel surface. Most of this steam condensate is expected to drain back into the IRWST through the IRWST gutter. However, some of this condensate will be lost from droplets dripping from the central dome of the containment vessel where the slope of the containment vessel wall is not sufficient to allow the condensate to adhere and flow over to the vertical wall. As the condensate film flows down the vertical containment wall it will flow onto the polar crane girder, the internal stiffener, and the IRWST gutter.

In order to identify all of the important phenomena that result in condensate losses, a phenomena identification and ranking table was developed. The results show that the overall IRWST condensate return rate study should address the following.:

1. Dripping from containment dome surface
    a. Near-horizontal surface dripping
    b. Dripping from interferences (weld plate misalignment, support plates)
2. Losses from the containment sidewalls due to obstacles and entrance to gutter
3. Condensation on passive heat sinks with no drain to IRWST or gutters
4. Steam stored in containment atmosphere
5. Leakage from containment
6. Water entrainment from the IRWST that is not returned to the IRWST

A.2 Supporting Design Documentation

To address the previous items, the following testing program and calculation notes are used. Each of these documents will be available for inspection by the NRC.:

1. **AP1000** Condensate Return Test Report, TR-SEE-III-12-01
2. Containment Response Analysis for the Long Term PRHR Operation, APP-PXS-M3C-071
3. Condensate Return to IRWST for Long Term PRHR Operation, APP-PXS-M3C-072
4. **AP1000** Safe Shutdown Temperature Evaluation, APP-SSAR-GSC-536

The following summarizes the content and purpose of each of these reports and explains the evaluation process established to provide required input for the IRWST condensate return rate analysis.

1. **AP1000** Condensate Return Test Report, TR-SEE-III-12-01

   This document reports the results of experiments performed to evaluate the condensate return rate along the vertical section of the containment vessel wall above the IRWST gutter and the amount of condensate captured by the IRWST gutter. The test configuration, and results are summarized in Section 4.0.

2. Containment Response Analysis for the Long Term PRHR Operation, APP-PXS-M3C-071

   The WGOTHIC **AP1000** containment model was developed to perform the containment peak pressure and temperature response calculations for the design analyses presented in Chapter 6 of the DCD. The approved methodology for that application uses conservative assumptions that tend to reduce the steam condensation heat and mass transfer rates and increase the calculated containment pressure/temperature response. To this end, the model does not take credit for all of the passive heat sinks that are located inside the containment vessel.

   The purpose of the calculations documented in APP-PXS-M3C-071 is to quantify the following thermodynamic phenomena:

   - Containment pressure and temperature,
   - Losses due to condensation on passive heat sinks,
   - Losses due to containment leakage,
   - Mass of steam which remains in containment free volume,
   - IRWST steaming rate,
   - PXS condensate return flow rate and temperature, and
   - CMT Compartment temperature.

   The WGOTHIC **AP1000** containment model is the best tool available for performing these calculations. However, for this application, the heat transfer areas for all of the passive heat sinks in the model must be increased to account for those that were not included for the containment peak pressure/temperature application. In addition, sensitivity studies must be performed to identify the most conservative initial and boundary conditions for this application. The changes that are required to be made to the WGOTHIC **AP1000** containment model for this application are described in further detail in APP-PXS-M3C-071.

   The WGOTHIC basedeck uses conservative inputs and initial conditions for the design basis analysis. Sensitivity analyses were performed with several sets of initial conditions/inputs in order to determine the most conservative set of input for the design basis case. The following input/boundary conditions were analyzed:

- Heat Input to the PRHR HX,
- PCS flow, PCS water temperature, PCS water coverage,
- Containment vessel heat transfer rates,
- PCS actuation time,
- IRWST water level, water temperature,
- Containment initial pressure, temperature, relative humidity,
- Mass of the heat sinks inside the containment, and
- PRHR Tube Plugging sensitivities.

The resulting evaluation in APP-PXS-M3C-071 provides input to APP-PXS-M3C-072 and to APP-SSAR-GSC-536.

3. Condensate Return to IRWST for Long Term PRHR Operation, APP-PXS-M3C-072

   The purpose of the APP-PXS-M3C-072 calculation note is to perform the Condensate Return evaluation. This calculation takes inputs from APP-PXS-M3C-071, including;

- Containment pressure and temperature,
- Losses due to condensation on passive heat sinks,
- Losses due to containment leakage, and
- Mass of steam which remains in containment free volume.

This calculation provides the mass of steam that remains in the free containment volume which contributes to pressurization, mass of the steam that will condense on the heat sinks, mass of the steam that will be lost through containment leakage, and inventory losses that will occur on the containment vessel head and containment vessel sidewalls considering test data reported in TR-SEE-III-12-01. [

.]^{a,c} This calculation considers the following phenomena previously identified.

- Dripping from containment inside surface (upper dome), “rain out phenomenon”
- Dripping from containment inside surface (upper dome) due to the containment plates misalignment (interference dripping)
- Obstacle-induced dripping from the containment dome
- Obstacle-induced dripping from the containment sidewalls.
- Water entrainment from the IRWST

4. **AP1000** Safe Shutdown Temperature Evaluation, APP-SSAR-GSC-536

   The purpose of APP-SSAR-GSC-536 is to perform the Safe Shutdown Temperature Evaluation. This cooldown analysis evaluates two specific cases, time to Safe Shutdown (420°F in 36 hours) and the Design Basis Analysis Extension case (72 hours with Chapter 15 acceptance criteria).

   The inputs for this analysis are taken from APP-PXS-M3C-071, including:

- Containment Pressure,
- Containment Temperature,
- IRWST steaming rate,

- PXS condensate return flow rate and temperature, and
- CMT Compartment temperature.

The Safe Shutdown Temperature evaluation, which is performed with the LOFTRAN computer code and is documented in Chapter 19E of DCD Revision 19, demonstrates the capability of the PRHR HX to reduce the core average temperature to 420 °F within 36 hours after shutdown following any design basis transient. The loss of normal feedwater with coincident loss of AC power is the most limiting transient with respect to removal of core decay heat; and is used to demonstrate that the passive safety systems can bring the plant to a stable, safe condition.

In addition to the 36 hour Safe Shutdown temperature evaluation, this analysis includes an extended design basis accident case. This case is considered to demonstrate that the Chapter 15 non-LOCA acceptance criteria are met for 72 hours (including maximum primary and secondary pressures, minimum departure from nucleate boiling ratio (DNBR), and maximum pressurizer water volume).

The Safe Shutdown temperature evaluation relies on the IRWST to reduce the core average temperature, and is therefore sensitive to condensate return rate. This evaluation is being revised to reflect updates incorporated in the approved DCD Revision 19 design, as well as the condensate return insights and design changes described in Sections 4.0 and 5.0 of this report.

A.3 Not used

Appendix B

Marked up pages of the plant-specific DCD

Tier 1 Changes

Table 2.2.3-1

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
Passive Residual Heat Removal Heat Exchanger (PRHR HX)	PXS-ME-01	Yes	Yes	-	- / -	-	- / -	-	-
Accumulator Tank A	PXS-MT-01A	Yes	Yes	-	- / -	-	- / -	-	-
Accumulator Tank B	PXS-MT-01B	Yes	Yes	-	- / -	-	- / -	-	-
Core Makeup Tank (CMT) A	PXS-MT-02A	Yes	Yes	-	- / -	-	- / -	-	-
CMT B	PXS-MT-02B	Yes	Yes	-	- / -	-	- / -	-	-
IRWST	PXS-MT-03	No	Yes	-	- / -	-	- / -	-	-
IRWST Screen A	PXS-MY-Y01A	No	Yes	-	- / -	-	- / -	-	-
IRWST Screen B	PXS-MY-Y01B	No	Yes	-	- / -	-	- / -	-	-
IRWST Screen C	PXS-MY-Y01C	No	Yes	-	- / -	-	- / -	-	-
Containment Recirculation Screen A	PXS-MY-Y02A	No	Yes	-	- / -	-	- / -	-	-
Containment Recirculation Screen B	PXS-MY-Y02B	No	Yes	-	- / -	-	- / -	-	-
pH Adjustment Basket 3A	PXS-MY-Y03A	No	Yes	-	- / -	-	- / -	-	-
pH Adjustment Basket 3B	PXS-MY-Y03B	No	Yes	-	- / -	-	- / -	-	-
pH Adjustment Basket 4A	PXS-MY-Y04A	No	Yes		- / -		- / -		
pH Adjustment Basket 4B	PXS-MY-Y04B	No	Yes		- / -		- / -		
Downspout Screen 1A	PXS-MY-Y81	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 1B	PXS-MY-Y82	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 1C	PXS-MY-Y83	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 1D	PXS-MY-Y84	No	Yes	-	- / -	-	- / -	-	-

Note: Dash (-) indicates not applicable.

Table 2.2.3-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. Harsh Envir.**	**Safety-Related Display**	**Control PMS/ DAS**	**Active Function**	**Loss of Motive Power Position**
Downspout Screen 2A	PXS-MY-Y85	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2B	PXS-MY-Y86	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2C	PXS-MY-Y87	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2D	PXS-MY-Y88	No	Yes	-	- / -	-	- / -	-	-
CMT A Inlet Isolation Motor-operated Valve	PXS-PL-V002A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/No	None	As Is
CMT B Inlet Isolation Motor-operated Valve	PXS-PL-V002B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/No	None	As Is
CMT A Discharge Isolation Valve	PXS-PL-V014A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
CMT B Discharge Isolation Valve	PXS-PL-V014B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
CMT A Discharge Isolation Valve	PXS-PL-V015A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
CMT B Discharge Isolation Valve	PXS-PL-V015B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
CMT A Discharge Check Valve	PXS-PL-V016A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
CMT B Discharge Check Valve	PXS-PL-V016B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-

Note: Dash (-) indicates not applicable.

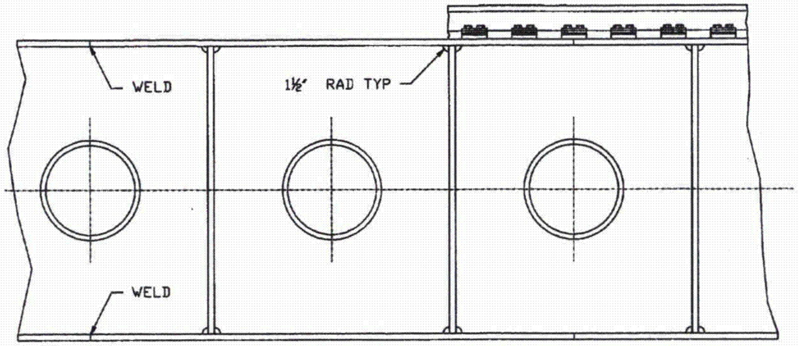
Table 2.2.3-2 (cont.)				
Line Name	**Line Number**	**ASME Code Section III**	**Leak Before Break**	**Functional Capability Required**
IRWST injection line B to DVI line B	PXS-L125B, PXS-L127B	Yes	Yes	Yes
	PXS-L123B, PXS-L124B, PXS-L118B, PXS-L117B, PXS-L116B, PXS-L114B, PXS-L112B, PXS-L120	Yes	No	Yes
IRWST screen cross-connect line	PXS-L180A, PXS-L180B	Yes	No	Yes
Containment recirculation line A	PXS-L113A, PXS-L131A, PXS-L132A	Yes	No	Yes
Containment recirculation line B	PXS-L113B, PXS-L131B, PXS-L132B	Yes	No	Yes
IRWST gutter drain line	PXS-L142A, PXS-L142B	Yes	No	Yes
	PXS-L141A, PXS-L141B	Yes	No	No
Downspout drain lines from polar crane girder and internal stiffener to collection box A	PXS-L301A, PXS-L302A, PXS-L303A, PXS-L304A, PXS-L305A, PXS-L306A, PXS-L307A, PXS-L308A, PXS-L309A, PXS-L310A	Yes	No	Yes
Downspout drain lines from polar crane girder and internal stiffener to collection box B	PXS-L301B, PXS-L302B, PXS-L303B, PXS-L304B, PXS-L305B, PXS-L306B, PXS-L307B, PXS-L308B, PXS-L309B, PXS-L310B	Yes	No	Yes

Tier 2 Changes

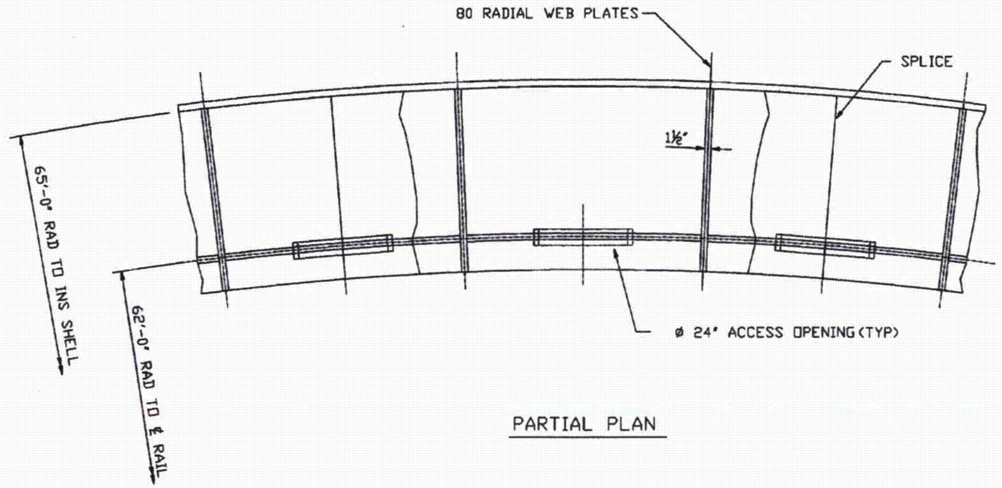
Table 3.2-3 (Sheet 20 of 7~~5~~3)

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

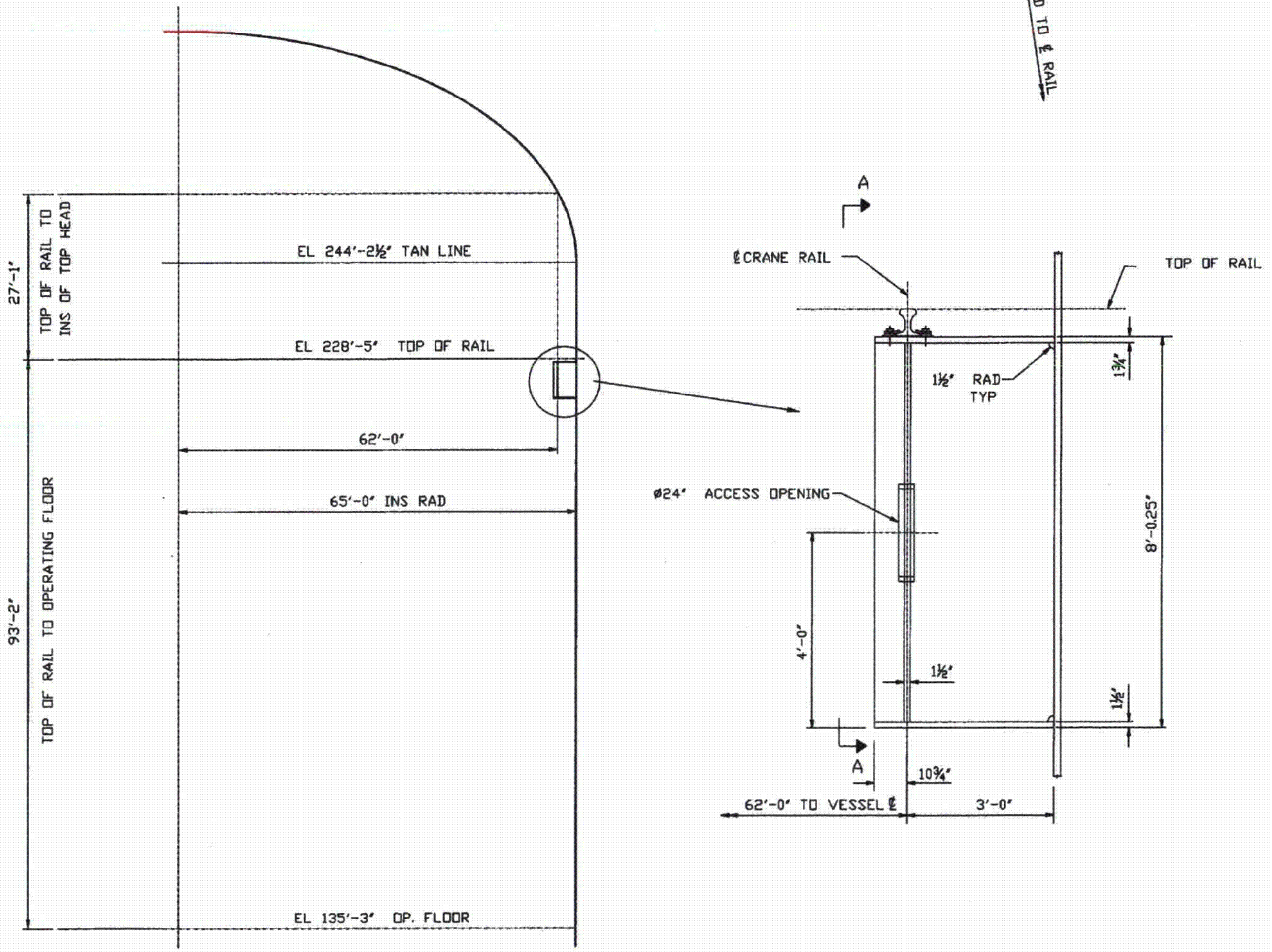
Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-MY-Y81	Downspout Screen 1A	C	I	Manufacturer Std.	
PXS-MY-Y82	Downspout Screen 1B	C	I	Manufacturer Std.	
PXS-MY-Y83	Downspout Screen 1C	C	I	Manufacturer Std.	
PXS-MY-Y84	Downspout Screen 1D	C	I	Manufacturer Std.	
PXS-MY-Y85	Downspout Screen 2A	C	I	Manufacturer Std.	
PXS-MY-Y86	Downspout Screen 2B	C	I	Manufacturer Std.	
PXS-MY-Y87	Downspout Screen 2C	C	I	Manufacturer Std.	
PXS-MY-Y88	Downspout Screen 2D	C	I	Manufacturer Std.	
PXS-PL-V002A	CMT A CL Inlet Isolation	A	I	ASME III-1	
PXS-PL-V002B	CMT B CL Inlet Isolation	A	I	ASME III-1	
PXS-PL-V010A	CMT A Upper Sample	B	I	ASME III-2	
PXS-PL-V010B	CMT B Upper Sample	B	I	ASME III-2	
PXS-PL-V011A	CMT A Lower Sample	B	I	ASME III-2	
PXS-PL-V011B	CMT B Lower Sample	B	I	ASME III-2	
PXS-PL-V012A	CMT A Drain	A	I	ASME III-1	
PXS-PL-V012B	CMT B Drain	A	I	ASME III-1	
PXS-PL-V013A	CMT A Discharge Manual Isolation	A	I	ASME III-1	
PXS-PL-V013B	CMT B Discharge Manual Isolation	A	I	ASME III-1	



SECTION A-A



PARTIAL PLAN



gases in the pressurizer following an accident. These functions are discussed in subsections 5.2.2, 5.4.12, and in Section 6.3. The AP1000 does not have power operated relief valves connected to the pressurizer.

The discharge of the safety valves is directed through a rupture disk to containment atmosphere.

The discharge of the first-, second-, and third-stage automatic depressurization system valves is directed to the in-containment refueling water storage tank. For the automatic depressurization system valves, the following discussion considers only the gas venting function. Only the first stage automatic depressurization valves are used to vent non-condensible gases following an accident. The sizing considerations and design basis for the in-containment refueling water storage tank for the depressurization function are discussed throughout Section 6.3. The provisions to minimize the differential pressure between the containment atmosphere and the interior of the in-containment refueling water storage tank are also discussed in subsection 6.3.2.

The safety valve on the normal residual heat removal system, which provides low temperature overpressure protection, discharges into the containment atmosphere. See subsection 5.4.7 for a discussion of the connections to and location of the safety valve in the normal residual heat removal system.

5.4.11.1 Design Bases

The containment has the capability to absorb the pressure increase and heat load resulting from the discharge of the safety valves to containment atmosphere. The in-containment refueling water storage tank has the capability to absorb the pressure increase and heat load from the discharge, including the water seal, steam and gases, from a first-stage automatic depressurization system valve when used to vent noncondensable gases from the pressurizer following an accident. The venting of noncondensable gases from the pressurizer following an accident is not a safety-related function.

5.4.11.2 System Description

Each safety valve discharge is directed to a rupture disk at the end of the discharge piping. A small pipe is connected to the discharge piping to drain away condensed steam leaking past the safety valve. The discharge is directed away from any safety related equipment, structures, or supports that could be damaged to the extent that emergency plant shutdown is prevented by such a discharge.

The discharge from each of two groups of automatic depressurization system valves is connected to a separate sparger below the water level in the in-containment refueling water storage tank. The piping and instrumentation diagram for the connection between the automatic depressurization system valves and the in-containment refueling water storage tank is shown in Figure 6.3-~~2~~1. The in-containment refueling water storage tank is a stainless steel lined compartment integrated into the containment interior structure. The discharge of water, steam, and gases from the first-stage automatic depressurization system valves when used to vent noncondensable gases does not result in pressure in excess of the in-containment refueling water storage tank design pressure. Additionally, vents on the top of the tank protect the tank from overpressure, as described in subsection 6.3.2.

Overflow provisions prevent overfilling of the tank. The overflow is directed into the refueling cavity. The in-containment refueling water storage tank does not have a cover gas and does not require a connection to the waste gas processing system. The normal residual heat removal system provides nonsafety-related cooling of the in-containment refueling water storage tank.

5.4.11.3 Safety Evaluation

The design of the control for the reactor coolant system and the volume of the pressurizer is such that a discharge from the safety valves is not expected. The containment design pressure, which is based on loss of coolant accident considerations, is greatly in excess of the pressure that would result from the discharge of a pressurizer safety valve. The heat load resulting from a discharge of a pressurizer safety valve is considerably less than the capacity of the passive containment cooling system or the fan coolers. See Section 6.2.

Venting of noncondensable gases, including entrained steam and water from the loop seals in the lines to the automatic depressurizations system valves, from the pressurizer into spargers below the water line in the in-containment refueling water storage tank does not result in a significant increase in the pressure or water temperature. The in-containment refueling water storage tank is not susceptible to vacuum conditions resulting from the cooling of hot water in the tank, as described in subsection 6.3.2. The in-containment refueling water storage tank has capacity in excess of that required for venting of noncondensable gases from the pressurizer following an accident.

5.4.11.4 Instrumentation Requirements

The instrumentation for the safety valve discharge pipe, containment, and in-containment refueling water storage tank are discussed in subsections 5.2.5, 5.4.9, and in Sections 6.2 and 6.3, respectively. Separate instrumentation for the monitoring of the discharge of noncondensable gases is not required.

5.4.11.5 Inspection and Testing Requirements

Sections 6.2 and 6.3 discuss the requirements for inspection and testing of the containment and in-containment refueling water storage tank, including operational testing of the spargers. Separate testing is not required for the noncondensable gas venting function.

5.4.12 Reactor Coolant System High Point Vents

The requirements for high point vents are provided for the AP1000 by the reactor vessel head vent valves and the automatic depressurization system valves. The primary function of the reactor vessel head vent is for use during plant startup to properly fill the reactor coolant system and vessel head. Both reactor vessel head vent valves and the automatic depressurization system valves may be activated and controlled from the main control room. The AP1000 does not require use of a reactor vessel head vent to provide safety-related core cooling following a postulated accident.

The reactor vessel head vent valves (Figure 5.4-8) can remove noncondensable gases or steam from the reactor vessel head to mitigate a possible condition of inadequate core cooling or

transients for normal operation, anticipated transients and postulated accident conditions are discussed in subsection 3.9.1.

Stress intensities resulting from design loads do not exceed the limits specified in ASME Code, Section III. The rules for the evaluation of the faulted conditions are defined in Appendix F of the ASME Code, Section III. Only those stress limits applicable for an elastic system analysis are used for the external load analysis.

5.4.13.4 Material Corrosion/Erosion Evaluation

Those portions of the core makeup tank in contact with reactor coolant are fabricated from or clad with stainless steel. The water chemistry of the core makeup tank, comparable to reactor coolant, causes minimal corrosion of the stainless steel. Erosion is not an issue, since there is normally no flow. A periodic analysis of the coolant chemical composition verifies that the reactor coolant quality meets the specifications, as discussed in subsection 5.2.3.

Contamination of stainless steel and nickel-chromium-iron alloys by copper, low-melting-temperature alloys, mercury, and lead is prohibited. The material selection, water chemistry specification, and residual stress in the piping minimize the potential for stress corrosion cracking, as discussed in subsection 5.2.3.

5.4.13.5 Test and Inspections

Charpy V-notch tests and drop-weight fracture toughness tests are performed as required. Orientation of test specimens is according to the ASME Code, Section III, except that the material is not considered to be subjected to high irradiation.

Compliance with the sensitization requirement is demonstrated by passing the susceptibility to intergranular attack test of ASTM A-262, Practice E, including the oxalic acid screening test according to Practice A. Inservice inspection requirements for Class 1 are discussed in Section 5.2.4.

In addition, materials and welds are inspected according to the requirements of the ASME Code, Section III Class 1.

5.4.14 Passive Residual Heat Removal Heat Exchanger

The passive residual heat removal heat exchanger (PRHR HX) is the component of the passive core cooling system that removes core decay heat for any postulated non-loss of coolant accident event where a loss of cooling capability via the steam generators occurs. Section 6.3 discusses the operation of the passive residual heat removal heat exchanger in the passive core cooling system.

5.4.14.1 Design Bases

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

returned to the tank. The passive residual heat removal heat exchanger is designed to withstand the design environment of 2500 psia and 650°F.

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

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

The passive residual heat removal heat exchanger is designed and fabricated according to the ASME Code, Section III, as a Class 1 component. Those portions of the passive residual heat exchanger that support the primary-side pressure boundary and falls under the jurisdiction of ASME Code, Section III, Subsection NF are AP1000 equipment Class A (ANS Safety Class 1, Quality Group A). Stresses for ASME Code, Section III equipment and supports are maintained within the limits of Section III of the Code. Section 5.2 provides ASME Code, Section III and material requirements. Subsection 5.2.4 discusses inservice inspection.

Materials of construction are specified to minimize corrosion/erosion and to provide compatibility with the operating environment, including the expected radiation level. Subsection 5.2.3 discusses the welding, cutting, heat treating and other processes used to minimize sensitization of stainless steel.

5.4.14.2 Design Description

The passive residual heat removal heat exchanger consists of an upper and lower tubesheet mounted through the wall of the in-containment refueling water storage tank. A series of 0.75-inch outer diameter C-shaped tubes connect the tubesheets shown in Figure 6.3-5. The primary coolant passes through the tubes, which transfer decay heat to the in-containment refueling water storage tank water and generate enough thermal driving head to maintain the flow through the heat exchanger during natural circulation. The design minimizes the diameter of the tubesheets and allows ample flow area between the tubes in the in-containment refueling water storage tank.

The horizontal lengths of the tubes and lateral support spacing in the vertical section allow for the potential temperature difference between the tubes at cold conditions and the tubes at hot conditions. The tubes are supported in the in-containment refueling water storage tank interior with a frame structure.

The passive residual heat removal heat exchanger is welded to the in-containment refueling water storage tank.

6.3 Passive Core Cooling System

The primary function of the passive core cooling system is to provide emergency core cooling following postulated design basis events. To accomplish this primary function, the passive core cooling system is designed to perform the following functions:

- Emergency core decay heat removal

  Provide core decay heat removal during transients, accidents or whenever the normal heat removal paths are lost. This heat removal function is available at reactor coolant system conditions including shutdowns. During refueling operations, when the IRWST is drained into the refueling cavity, other passive means of core decay heat removal are utilized. Subsection 6.3.3.4.4 provides a description of how this is accomplished.

- Reactor coolant system emergency makeup and boration

  Provide reactor coolant system makeup and boration during transients or accidents when the normal reactor coolant system makeup supply from the chemical and volume control system is unavailable or is insufficient.

- Safety injection

  Provide safety injection to the reactor coolant system to provide adequate core cooling for the complete range of loss of coolant accidents, up to and including the double-ended rupture of the largest primary loop reactor coolant system piping.

- Containment pH control

  Provide for chemical addition to the containment during post-accident conditions to establish floodup chemistry conditions that support radionuclide retention with high radioactivity in containment and to prevent corrosion of containment equipment during long-term floodup conditions.

The passive core cooling system is designed to operate without the use of active equipment such as pumps and ac power sources. The passive core cooling system depends on reliable passive components and processes such as gravity injection and expansion of compressed gases. The passive core cooling system does require a one-time alignment of valves upon actuation of the specific components.

6.3.1 Design Basis

The passive core cooling system is designed to perform its safety-related functions based on the following considerations:

- It has component redundancy to provide confidence that its safety-related functions are performed, even in the unlikely event of the most limiting single failure occurring coincident with postulated design basis events.

- Components are designed and fabricated according to industry standard quality groups commensurate with its intended safety-related functions.
- It is tested and inspected at appropriate intervals, as defined by the ASME Code, Section XI, and by technical specifications.
- It performs its intended safety-related functions following events such as fire, internal missiles or pipe breaks.
- It is protected from the effects of external events such as earthquakes, tornadoes, and floods.
- It is designed to be sufficiently reliable, considering redundancy and diversity, to support the plant core melt frequency and significant release frequency goals.

6.3.1.1 Safety Design Basis

The passive core cooling system is designed to provide emergency core cooling during events involving increases and decreases in secondary side heat removal and decreases in reactor coolant system inventory. Subsection 6.3.3 provides a description of the design basis events. The performance criteria are provided in subsection 6.3.1 and also described in Chapter 15, under the respective event sections.

6.3.1.1.1 Emergency Core Decay Heat Removal

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

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

the steam generated in the in-containment refueling water storage tank is condensed on the containment vessel and returned by gravity via the in-containment refueling water storage tank condensate return gutter and downspouts.

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

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

6.3.1.1.2 Reactor Coolant System Emergency Makeup and Boration

For postulated non-LOCA events, sufficient core makeup water inventory is automatically provided to keep the core covered and to allow for decay heat removal. In addition, this makeup prevents actuation of the automatic depressurization system for a significant time.

For postulated events resulting in an inadvertent cooldown of the reactor coolant system, such as a steam line break, sufficient borated water is automatically provided to makeup for reactor coolant system shrinkage. The borated water also counteracts the reactivity increase caused by the resulting system cooldown.

For a Condition II steam line break described in Chapter 15, return to power is acceptable if there is no core damage. For this event, the automatic depressurization system is not actuated.

For a large steam line break, the peak return to power is limited so that the offsite dose limits are satisfied. Following either of these events, the reactor is automatically brought to a subcritical condition.

For safe shutdown, the passive core cooling system is designed to supply sufficient boron to the reactor coolant system to maintain the technical specification shutdown margin for cold, post-depressurization conditions, with the most reactive rod fully withdrawn from the core. The automatic depressurization system is not expected to actuate for these events.

6.3.1.1.3 Safety Injection

The passive core cooling system provides sufficient water to the reactor coolant system to mitigate the effects of a loss of coolant accident. In the event of a large loss of coolant accident, up to and including the rupture of a hot or cold leg pipe, where essentially all of the reactor coolant volume is initially displaced, the passive core cooling system rapidly refills the reactor

vessel, refloods the core, and continuously removes the core decay heat. A large break is a rupture with a total cross-sectional area equal to or greater than one square foot. Although the criteria for mechanistic pipe break are used to limit the size of pipe rupture considered in the design and evaluation of piping systems, as described in subsection 3.6.3, such criteria are not used in the design of the passive core cooling system.

Sufficient water is provided to the reactor vessel following a postulated loss of coolant accident so that the performance criteria for emergency core cooling systems, described in Chapter 15, are satisfied.

The automatic depressurization system valves, provided as part of the reactor coolant system, are designed so that together with the passive core cooling system they:

- Satisfy the small loss of coolant accident performance requirements
- Provide effective core cooling for loss of coolant accidents from when the passive core cooling system is actuated through the long-term cooling mode.

6.3.1.1.4 Safe Shutdown

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

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

In most sequences the operators would return the plant to normal system operations and terminate passive system operation within several hours in accordance with the plant emergency operating procedures. In scenarios when ac power sources are unavailable for approximately 22 hours, the automatic depressurization system will automatically actuate. However, after initial plant cooldown following a non-LOCA event, operators will assess plant conditions and have the option to perform recovery actions to further cool and depressurize the reactor coolant system in a closed-loop mode of operation, i.e., without actuation of the automatic depressurization system.

After verifying the reactor coolant system is in an acceptable, stable condition such that automatic depressurization is not needed, the operators may take action to extend passive residual heat removal heat exchanger operation by de-energizing the loads on the Class 1E dc batteries powering the protection and monitoring system actuation cabinets. After operators have taken action to extend its operation, the passive residual heat removal heat exchanger, in conjunction with the passive containment cooling system, has the capability to maintain safe, stable shutdown conditions. The automatic depressurization system remains available to maintain safe shutdown conditions at a later time.

For loss of coolant accidents, when the core makeup tank level reaches the automatic depressurization system actuation setpoint and other postulated events where ac power sources are lost but the passive residual heat removal heat exchanger operation is not extended or exhausted~~, or when the core makeup tank levels reach the automatic depressurization system actuation setpoint,~~ the automatic depressurization system will be initiated.~~initiates.~~ This results in injection from the accumulators and subsequently from the in-containment refueling water storage tank, once the reactor coolant system is nearly depressurized. For these conditions, the reactor coolant system depressurizes to saturated conditions at about 250°F within 24 hours. The passive core cooling system can maintain this safe shutdown condition indefinitely for the plant.

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

6.3.1.1.5 Containment pH Control

The passive core cooling system is capable of maintaining the desired post-accident pH conditions in the recirculation water after containment floodup. The pH adjustment is capable of maintaining containment pH within a range of 7.0 to 9.5, to enhance radionuclide retention in the containment and to prevent stress corrosion cracking of containment components during long-term containment floodup.

6.3.1.1.6 Reliability Requirements

The passive core cooling system satisfies a variety of reliability requirements, including redundancy (such as for components, power supplies, actuation signals, and instrumentation), equipment testing to confirm operability, procurement of qualified components, and provisions for periodic maintenance. In addition, the system provides protection in a number of areas including:

- Single active and passive component failures
- Spurious failures
- Physical damage from fires, flooding, missiles, pipe whip, and accident loads

- Environmental conditions such as high-temperature steam and containment floodup

Subsection 6.3.1.~~2~~3 includes specific nonsafety-related design requirements that help to confirm satisfactory system reliability.

6.3.1.2 ~~Power Generation Design Basis~~Nonsafety Design Basis

6.3.1.2.1 Long Term Core Decay Heat Removal

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

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

6.3.1.3 Power Generation Design Basis

The passive core cooling system is designed to be sufficiently reliable to support the probabilistic risk analysis goals for core damage frequency and severe release frequency. In assessing the reliability for probabilistic risk analysis purposes, more realistic analysis is used for both the passive core cooling system performance and for plant response.

In the event of a small loss of coolant accident, the passive core cooling system limits the increase in peak clad temperature and core uncovery with design basis assumptions. For pipe ruptures of less than eight-inch nominal diameter size, the passive core cooling system is designed to prevent core uncovery with best estimate assumptions.

The passive residual heat removal heat exchanger and the in-containment refueling water storage tank are designed to delay significant steam release to the containment for at least one hour.

The frequency of automatic depressurization system actuation is limited to a low probability to reduce safety risks and to minimize plant outages. Equipment is located so that it is not flooded or it is designed so that it is not damaged by the flooding. Major plant equipment is designed for multiple occurrences without damage.

The pH control equipment is designed to minimize the potential for and the impact of inadvertent actuation.

The passive core cooling system is capable of supporting the required testing and maintenance, including capabilities to isolate and drain equipment.

6.3.2 System Design

The passive core cooling system is a seismic Category I, safety-related system. It consists of two core makeup tanks, two accumulators, the in-containment refueling water storage tank, the passive residual heat removal heat exchanger, pH adjustment baskets, and associated piping, valves, instrumentation, and other related equipment. The automatic depressurization system valves and spargers, which are part of the reactor coolant system, also provide important passive core cooling functions.

The passive core cooling system is designed to provide adequate core cooling in the event of design basis events. The redundant onsite safety-related class 1E dc and UPS system provides power such that protection is provided for a loss of ac power sources, coincident with an event, assuming a single failure has occurred.

6.3.2.1 Schematic Piping and Instrumentation Diagrams

Figures 6.3-1 ~~and 6.3-2~~ shows the piping and instrumentation drawings of the passive core cooling system. Simplified flow diagrams are shown in Figures 6.3-3 and 6.3-4. The accident analysis results of events analyzed in Chapter 15 provide a summary of the expected fluid conditions in the passive core cooling system for the various locations shown on the simplified flow diagrams, for the specific plant conditions identified -- safety injection and decay heat removal.

The passive core cooling system is designed to supply the core cooling flow rates to the reactor coolant system specified in Chapter 15 for the accident analyses. The accident analyses flow rates and heat removal rates are calculated by assuming a range of component parameters, including best estimate and conservatively high and low values.

The passive core cooling system design is based on the six major components, listed in subsection 6.3.2.2, that function together in various combinations to support the four passive core cooling system functions:

- Emergency decay heat removal
- Emergency reactor makeup/boration
- Safety injection
- Containment pH control

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

For events not involving a loss of coolant, the emergency core decay heat removal is provided by the passive core cooling system via the passive residual heat removal heat exchanger. The heat

exchanger consists of a bank of C-tubes, connected to a tubesheet and channel head arrangement at the top (inlet) and bottom (outlet). The passive residual heat removal heat exchanger connects to the reactor coolant system through an inlet line from one reactor coolant system hot leg (through a tee from one of the fourth stage automatic depressurization lines) and an outlet line to the associated steam generator cold leg plenum (reactor coolant pump suction).

The inlet line is normally open and connects to the upper passive residual heat removal heat exchanger channel head. The inlet line is connected to the top of the hot leg and is routed continuously upward to the high point near the heat exchanger inlet. The normal water temperature in the inlet line will be hotter than the discharge line.

The outlet line contains normally closed air-operated valves that open on loss of air pressure or on control signal actuation. The alignment of the passive residual heat removal heat exchanger (with a normally open inlet motor-operated valve and normally closed outlet air-operated valves) maintains the heat exchanger full of reactor coolant at reactor coolant system pressure. The water temperature in the heat exchanger is about the same as the water in the in-containment refueling water storage tank, so that a thermal driving head is established and maintained during plant operation.

The heat exchanger is elevated above the reactor coolant system loops to induce natural circulation flow through the heat exchanger when the reactor coolant pumps are not available. The passive residual heat removal heat exchanger piping arrangement also allows actuation of the heat exchanger with reactor coolant pumps operating. When the reactor coolant pumps are operating, they provide forced flow in the same direction as natural circulation flow through the heat exchanger. If the pumps are operating and subsequently trip, then natural circulation continues to provide the driving head for heat exchanger flow.

The heat exchanger is located in the in-containment refueling water storage tank, which provides the heat sink for the heat exchanger.

Although gas accumulation is not expected, there is a vertical pipe stub on the top of the inlet piping high point that serves as a gas collection chamber. Level detectors indicate when gases have collected in this area. There are provisions to allow the operators to open manual valves to locally vent these gases to the in-containment refueling water storage tank.

The passive residual heat removal heat exchanger, in conjunction with in-containment refueling water storage tank, condensate return features, and the passive containment cooling system, can provide core cooling for greater than 14 days.~~an indefinite period of time.~~ After the in-containment refueling water storage tank water reaches its saturation temperature (in ~~about 2~~ several hours), the process of steaming to the containment initiates. Containment pressure will increase as steam is released from the in-containment refueling water storage tank. As the containment temperature increases, condensation begins to form on the subcooled metal and concrete surfaces inside containment. Condensation on these heat sink surfaces transfers energy to the bulk metal and concrete until they come into equilibrium with the containment atmosphere. Condensation that is not returned to the in-containment refueling water storage tank drains to the containment sump.

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

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

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

6.3.2.1.2 Reactor Coolant System Emergency Makeup and Boration

The core makeup tanks provide reactor coolant system makeup and boration during events not involving loss of coolant when the normal makeup system is unavailable or insufficient. There are two core makeup tanks located inside the containment at an elevation slightly above the reactor coolant loops. During normal operation, the core makeup tanks are completely full of cold, borated water. The boration capability of these tanks provides adequate core shutdown margin following a steam line break.

The core makeup tanks are connected to the reactor coolant system through a discharge injection line and an inlet pressure balance line connected to a cold leg. The discharge line is blocked by two normally closed, parallel air-operated isolation valves that open on a loss of air pressure or electrical power, or on control signal actuation. The core makeup tank discharge isolation valves are diverse from the passive residual heat removal heat exchanger outlet isolation valves discussed above. They use different globe valve body styles and different air operator types.

in-containment refueling water storage tank, thereby promoting more effective steam condensation.

The first three stages of automatic depressurization system valves discharge through the spargers and are designed to pass sufficient depressurization venting flow, with an acceptable pressure drop, to support the depressurization system performance requirements. The installation of the spargers prevents undesirable and/or excessive dynamic loads on the in-containment refueling water storage tank and other structures.

Each sparger is sized to discharge at a flow rate that supports automatic depressurization system performance, which in turn, allows adequate passive core cooling system injection.

6.3.2.2.7 IRWST and Containment Recirculation Screens

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

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

The operation of the passive core cooling system following a LOCA is described in subsection 6.3.2.1.3. Proper screen design, plant layout, and other factors prevent clogging of these screens by debris during accident operations.

6.3.2.2.7.1 General Screen Design Criteria

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

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

- Separate, large screens are provided for each function.

- Screens are located well below containment floodup level. Each screen provides the function of a trash rack and a fine screen. A debris curb is provided to prevent high density debris from being swept along the floor to the screen face.

- Floors slope away from screens (not required for AP1000).

- Drains do not impinge on screens.

shown that a head loss of 4.1 psi at these flows is acceptable based on long-term core cooling sensitivity analysis.

6.3.2.2.7.2 IRWST Screens

The IRWST screens are located inside the IRWST at the bottom of the tank. Figure 6.3-6 shows a plan view and Figure 6.3-7 shows a section view of these screens. Three separate screens are provided in the IRWST, one at either end of the tank and one in the center. A cross-connect pipe connects all three IRWST screens to distribute flow. The IRWST is closed off from the containment; its vents and overflows are normally closed by louvers. The potential for introducing debris inadvertently during plant operations is limited. A cleanliness program (refer to subsection 6.3.8.1) controls foreign debris from being introduced into the tank during maintenance and inspection operations. The Technical Specifications require visual inspections of the screens during every refueling outage.

The IRWST design eliminates sources of debris from inside the tank. Insulation is not used in the tank. Air filters are not used in the IRWST vents or overflows. Wetted surfaces in the IRWST are corrosion resistant such as stainless steel or nickel alloys; the use of these materials prevents the formation of significant amounts of corrosion products. In addition, the water is required to be clean because it is used to fill the refueling cavity for refueling; filtering and demineralizing by the spent fuel pit cooling system is provided during and after refueling.

During a LOCA, steam vented from the reactor coolant system condenses on the containment shell and~~,~~ drains down the shell to the polar crane girder or internal stiffener where it is drained via downspouts to the IRWST. Steam that condenses below the internal stiffener drains down the shell ~~to the operating deck elevation~~ and is collected in a gutter near the operating deck elevation. It is very unlikely that debris generated by a LOCA can reach the downspouts or the gutter because of their~~its~~ locations. Each downspout inlet is covered with a coarse screen that prevents larger debris from entering the downspout. The gutter is covered with a trash rack which prevents larger debris from clogging the gutter or entering the IRWST through the two 4-inch drain pipes. The inorganic zinc coating applied to the inside surface of the containment shell is safety – Service Level I, and will stay in place and will not detach.

The design of the IRWST screens reduces the chance of debris reaching the screens. The screens are oriented vertically such that debris that settles out of the water does not fall on the screens. The lowest screening surface of the IRWST screens is located 6 inches above the IRWST floor to prevent high density debris from being swept along the floor by water flow to the IRWST screens. The screen design provides the trash rack function. This is accomplished by the screens having a large surface area to prevent a single object from blocking a large portion of the screen and by the screens having a robust design to preclude an object from damaging the screen and causing by-pass. The screen prevents debris larger than 0.0625 inch from being injected into the reactor coolant system and blocking fuel cooling passages. The screen is a type that has sufficient surface area to accommodate debris that could be trapped on the screen. The design of the IRWST screens is described further in APP-GW-GLN-147 (Reference 4).

The screen flow area is conservatively designed considering the operation of the nonsafety-related normal residual heat removal system pumps which produce a higher flow than the safety-

that reposition to initiate safety-related system functions, the valve repositioning times are less than the times assumed in the accident analyses. These lag times refer to the time after initiation of the safeguards actuation signal.

It is acceptable for the core makeup tank injection to be delayed several minutes following actuation due to high initial steam condensation rates in the tank.

6.3.2.5.4 Potential Boron Precipitation

Boron precipitation in the reactor vessel is prevented by sufficient flow of passive core cooling system water through the core to limit the increase in boron concentration of the water remaining in the reactor vessel. Water along with steam leaves the core and exits the RCS through the fourth stage ADS lines. These valves connect to the hot leg and open in about 20 minutes after a loss of coolant accident or an automatic depressurization system actuation.

6.3.2.5.5 Safe Shutdown

During a safe shutdown, the passive core cooling system provides redundancy for boration, makeup, and heat removal functions. Section 7.4 provides additional information about safe shutdown.

6.3.2.6 Protection Provisions

The measures taken to protect the system from damage that might result from various events are described in other sections, as listed below.

- Protection from dynamic effects is presented in Section 3.6.
- Protection from missiles is presented in Section 3.5.
- Protection from seismic damage is presented in Sections 3.7, 3.8, 3.9, and 3.10.
- Protection from fire is presented subsection 9.5.1.
- Environmental qualification of equipment is presented in Section 3.11.
- Thermal stresses on the reactor coolant system are presented in Section 5.2.

6.3.2.7 Provisions for Performance Testing

The passive core cooling system includes the capability for determination of the integrity of the pressure boundary formed by series passive core cooling system check valves. Additional information on testing can be found in subsection 6.3.6.

6.3.2.8 Manual Actions

The passive core cooling system is automatically actuated for those events as presented in subsection 6.3.3. Following actuation, the passive core cooling system continues to operate in the injection mode until the transition to recirculation initiates automatically following containment floodup.

Although the passive core cooling system operates automatically, operator actions would be beneficial, in some cases, in reducing the consequences of an event. For example, in a steam generator tube rupture with no operator action, the protection and safety monitoring system automatically terminates the leak, prevents steam generator overfill, and limits the offsite doses. However, the operator can initiate actions, similar to those taken in current plants, to identify and isolate the faulted steam generator, cool down and depressurize the reactor coolant system to terminate the break flow to the steam generator, and stabilize plant conditions.

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

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

6.3.3 Performance Evaluation

The events described in subsection 6.3.1 result in passive core cooling system actuation and are mitigated within the performance criteria. For the purpose of evaluation in Chapters 15 and 19, the events that result in passive core cooling system actuation are categorized as follows:

A. Increase in heat removal by the secondary system

   1. Inadvertent opening of a steam generator power-operated atmospheric steam relief or safety valve

   2. Steam system piping failure

B. Decrease in heat removal by the secondary system

   1. Loss of Main Feedwater Flow
   2. Feedwater system piping failure

C. Decrease in reactor coolant system inventory

   1. Steam generator tube rupture

   2. Loss of coolant accident from a spectrum of postulated reactor coolant system piping failures

3. Loss of coolant due to a rod cluster control assembly ejection accident

   (This event is enveloped by the reactor coolant system piping failures.)

D. Shutdown Events (Chapter 19)

1. Loss of Startup Feedwater
2. Loss of normal residual heat removal system with reactor coolant system pressure boundary intact
3. Loss of normal residual heat removal system during mid-loop operation
4. Loss of normal residual heat removal system with refueling cavity flooded

The events listed in groups A and B are non-LOCA events where the primary protection is provided by the passive core cooling system passive residual heat removal heat exchanger. For these events, the passive residual heat removal heat exchanger is actuated by the protection and monitoring system for the following conditions:

- Steam generator low narrow range level, coincident with startup feedwater low flow
- Steam generator low wide range level
- Core makeup tank actuation
- Automatic depressurization actuation
- Pressurizer water level - High 3
- Manual actuation

The events listed in group C above are events involving the loss of reactor coolant where the primary protection is by the core makeup tanks and accumulators. For these events the core makeup tanks are actuated by the protection and monitoring system for the following conditions:

- Pressurizer low pressure
- Pressurizer low level
- Steam line low pressure
- Containment high pressure
- Cold leg low temperature
- Steam generator low wide range level, coincident with reactor coolant system high hot leg temperature
- Manual actuation

In addition to initiating passive core cooling system operation, these signals initiate other safeguards automatic actions including reactor trip, reactor coolant pump trip, feedwater isolation, and containment isolation. The passive core cooling system actuation signals are described in Section 7.3.

The core makeup tanks and passive residual heat removal heat exchangers are also actuated by the Diverse Actuation System as described in subsection 7.7.1.11.

Upon receipt of an actuation signal, the actions described in subsection 6.3.2.1 are automatically initiated to align the appropriate features of the passive core cooling system.

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

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

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

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

The transient analyses summarized in Chapter 15 are extended long enough to demonstrate the applicable safety evaluation criteria are met. It is expected that normal systems would be

available such that operators could terminate the passive safety systems and proceed with an orderly shutdown. However, as discussed in subsection 6.3.1.1.4, the passive systems are capable of bring the plant to a safe shutdown condition and maintaining that condition.

The events listed in group D occur during shutdown conditions that are characterized by slow plant responses and mild thermal-hydraulic transients. In addition, some of the passive core cooling system features need to be isolated to allow the plant to be in these conditions or to perform maintenance on the system. The protection and monitoring system automatically actuates gravity injection from the IRWST to provide core cooling during shutdown conditions prior to refueling cavity floodup. In addition, the operator can also manually actuate other passive core cooling system equipment, such as the passive residual heat removal heat exchanger, to provide core cooling during shutdown conditions when the equipment does not automatically actuate.

6.3.3.1 Increase in Heat Removal by the Secondary System

A number of events that could result in an increase in heat removal from the reactor coolant system by the secondary system have been postulated. For each event, consideration has been given to operation of nonsafety-related systems that could affect the event results. The operation of the startup feedwater system and the chemical and volume control system makeup pumps can affect these events. Analyses of these events, both with and without these nonsafety-related systems operating, are presented in Section 15.1. For those events resulting in passive core cooling system actuation, the following summarizes passive core cooling system performance.

6.3.3.1.1 Inadvertent Opening of a Steam Generator Relief or Safety Valve

Subsection 15.1.4 provides a description of an inadvertent opening of a steam generator relief or safety valve, including criteria and analytical results.

For this event, upon generation of a safeguards actuation signal the reactor is tripped, the core makeup tanks are actuated, and the reactor coolant pumps are tripped. Since the core makeup tanks are actuated, the passive residual heat removal heat exchanger is also actuated. The main steam lines are also isolated to prevent blowdown of more than one steam generator. The core makeup tanks operate with water recirculation injection to provide borated water to the reactor vessel downcomer plenum for reactor coolant system inventory and reactivity control. The trip of the reactor initially brings the reactor sub-critical. The rapid reactor coolant system cool down may result in the reactor returning to critical because the rate of positive reactivity addition (reactor coolant system temperature reduction) exceeds the rate of negative reactivity addition (boron from the core makeup tank). As the event continues, the reactor coolant system cooldown will slow down such that the continued core makeup tank boration will return the reactor sub-critical. The departure from nucleate boiling design basis is met, thereby preventing fuel damage.

During this event, the startup feedwater system is assumed to malfunction so that it injects water at the maximum flow rate. This injection continues until feedwater isolation occurs on low reactor coolant system temperature. The feedwater isolation signal terminates the feedwater addition from the startup feedwater system. The passive residual heat removal heat exchanger is

also assumed to function in this event. This heat removal mechanism continues throughout the duration of the event.

For this event, the core makeup tanks operate in the water recirculation mode, providing boration and injection flow without draining. Therefore, the automatic depressurization system is not actuated on the lowering of the core makeup tank level.

Subsequent to stabilizing plant conditions and satisfying passive core cooling system termination criteria, the operator terminates passive core cooling system operation and initiates normal plant shutdown operations.

6.3.3.1.2 Steam System Pipe Failure

The most severe core conditions resulting from a steam system piping failure are associated with a double-ended rupture of a main steam line, occurring at zero power. Effects of smaller piping failures at higher power levels are bounded by the double-ended rupture at zero power. Subsection 15.1.5 provides a description of this event, including criteria and analytical results.

For this event, the passive core cooling system functions as described in subsection 6.3.3.1.1 for the inadvertent opening of a steam generator relief or safety valve. However, this piping failure constitutes a more severe cooldown transient. The malfunctioning of the startup feedwater system is considered as it was in the inadvertent steam generator depressurization. The trip of the reactor initially brings the reactor sub-critical. The rapid reactor coolant system cool down may result in the reactor returning to critical because the rate of positive reactivity addition (reactor coolant system temperature reduction) exceeds the rate of negative reactivity addition (boron from the core makeup tank). As the event continues, the reactor coolant system cooldown will slow down such that the continued core makeup tank boration will return the reactor sub-critical. The departure from nucleate boiling design basis is met.

For this event, the reactor coolant system may depressurize sufficiently to permit the accumulators to deliver makeup water to the reactor coolant system. The core makeup tanks inject via water recirculation without draining. Therefore, the automatic depressurization system is not actuated on the lowering of the core makeup tank level. Subsequent to stabilizing plant conditions and satisfying passive core cooling system termination criteria, the operator terminates passive core cooling system operation and initiates a normal plant shutdown.

6.3.3.2 Decrease in Heat Removal by the Secondary System

A number of events have been postulated that could result in a decrease in heat removal from the reactor coolant system by the secondary system. For each event, consideration has been given to operation of nonsafety-related systems that could affect the consequences of an event. The operation of the startup feedwater system and the chemical and volume control system makeup pumps can affect these events. Analyses of these events, both with and without these nonsafety-related systems operating, are presented in Section 15.2. For those events resulting in passive core cooling system actuation, the following summarizes passive core cooling system performance.

6.3.3.2.1 Loss of Main Feedwater

The most severe core conditions resulting from a loss of main feedwater system flow are associated with a loss of flow at full power. The heat-up transient effects of loss of flow at reduced power levels are bounded by the loss of flow at full power. Subsection 15.2.7 provides a description of this event, including criteria and analytical results.

For this event, the passive residual heat removal heat exchanger is actuated. If the core makeup tanks are not initially actuated, they actuate later when passive residual heat exchanger cooling sufficiently reduces pressurizer level. The passive residual heat removal heat exchanger serves to remove core decay heat and the core makeup tanks inject a borated water solution directly into the reactor vessel downcomer annulus. Since the reactor coolant pumps are tripped on actuation of the core makeup tanks, the passive residual heat removal heat exchanger operates under natural circulation conditions. The core makeup tanks operate via water recirculation, without draining, to maintain reactor coolant system inventory. Therefore, the automatic depressurization system is not actuated on the lowering of the core makeup tank level. Since the event is characterized by a heat-up transient, the injection of negative reactivity is not required and is not taken credit for in the analysis to control core reactivity.

The reactor coolant system does not depressurize to permit the accumulators to deliver makeup water to the reactor coolant system. Subsequent to stabilizing plant conditions and satisfying passive core cooling system termination criteria, the operator terminates passive core cooling system operation and initiates a normal plant shutdown.

6.3.3.2.1.1 Loss of AC Power to the Plant Auxiliaries

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

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

The loss of main feedwater with loss of ac power event is analyzed for a 72 hour period, assuming operators extend closed-loop cooling beyond the time the automatic depressurization system would be actuated by the protection and safety monitoring system. This event mirrors the loss of ac power to the plant auxiliaries event described in subsection 15.2.6, but the loss of ac power extends to 72 hours. In this event, operation of the passive residual heat removal heat

exchanger continues for 72 hours and maintains acceptable reactor coolant system conditions such that the applicable Condition II safety evaluation criteria are met.

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

6.3.3.2.2 Feedwater System Pipe Failure

The most severe core conditions resulting from a feedwater system piping failure are associated with a double-ended rupture of a feed line at full power. Depending on break size and power level, a feedwater system pipe failure could cause either a reactor coolant system cooldown transient or a reactor coolant system heat-up transient. Only the reactor coolant system heat-up transient is evaluated as a feedwater system pipe failure, since the spectrum of cooldown transients is bounded by the steam system pipe failure analyses. The heat-up transient effects of smaller piping failures at reduced power levels are bounded by the double-ended feed line rupture at full power. Subsection 15.2.8 provides a description of this event, including criteria and analytical results.

For this event, the passive residual heat removal heat exchanger and the core makeup tanks are actuated. The passive residual heat removal heat exchanger serves to remove core decay heat, and the core makeup tanks inject a borated water solution directly into the reactor vessel downcomer. Since the reactor coolant pumps are tripped on actuation of the core makeup tanks, the passive residual heat removal heat exchanger operates under natural circulation conditions. The core makeup tanks operate via water recirculation to maintain reactor coolant system inventory. Since the event is characterized by a heat-up transient, the injection of negative reactivity is not required and is not taken credit for in the analysis to control core reactivity.

The reactor coolant system does not depressurize to permit the accumulators to deliver makeup water to the reactor coolant system. Subsequent to stabilizing plant conditions and satisfying passive core cooling system termination criteria, the operator terminates passive core cooling system operation and initiates normal plant shutdown operations.

6.3.3.3 Decrease in Reactor Coolant System Inventory

A number of events have been postulated that could result in a decrease in reactor coolant system inventory. For each event, consideration has been given to operation of nonsafety-related systems that could affect the consequences of the event. The operation of the startup feedwater system and the chemical and volume control system makeup pumps can affect these events. Analyses of these events, both with and without these nonsafety-related systems operating, are presented in Section 15.6. For those events which result in passive core cooling system actuation, the following summarizes passive core cooling system performance.

6.3.3.4 Shutdown Events

The passive core cooling system components are available whenever the reactor is critical and when reactor coolant energy is sufficiently high to require passive safety injection. During low-temperature physics testing, the core decay heat levels are low and there is a negligible amount of stored energy in the reactor coolant. Therefore, an event comparable in severity to events occurring at operating conditions is not possible and passive core cooling system equipment is not required. The possibility of a loss of coolant accident during plant startup and shutdown has been considered.

During shutdown conditions, some of the passive core cooling system equipment is isolated. In addition, since the normal residual heat removal system is not a safety-related system, its loss is considered.

As a result, gravity injection is automatically actuated when required during shutdown conditions prior to refueling cavity floodup, as discussed in subsection 6.3.3.3.2. The operator can also manually actuate other passive core cooling system equipment, such as the passive residual heat removal heat exchanger, if required for accident mitigation during shutdown conditions when the equipment does not automatically actuate.

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

During normal cooldowns, the steam generators are supplied by the startup feedwater pumps and steam from the steam generator is directed to either the main condenser or to the atmosphere. There are two nonsafety-related startup feedwater pumps, each of which is capable of providing sufficient feedwater flow to both steam generators to remove decay heat. These pumps are also automatically loaded on the nonsafety-related diesel-generators in the event offsite power is lost. Since these pumps are nonsafety-related, their failure is considered.

In the event of a loss of startup feedwater, the passive residual heat removal heat exchanger is automatically actuated on low steam generator water level and provides safety-related heat removal. The passive residual heat removal heat exchanger can maintain the reactor coolant system temperature, as well as provide for reactor coolant system cooldown to conditions where the normal residual heat removal system can be operated.

Since the chemical and volume control system makeup pumps are nonsafety-related, they may not be available. In this case, the core makeup tanks automatically actuate as the cooldown continues and the pressurizer level decreases. The core makeup tanks operate in a water recirculation mode to maintain reactor coolant system inventory while the passive residual heat removal heat exchanger is operating.

The in-containment refueling water storage tank provides the heat sink for the passive residual heat removal heat exchanger. Initially, the heat addition increases the water temperature. Within one to two hours, the water reaches saturation temperature and begins to boil. The steam generated in the in-containment refueling water storage tank discharges to containment. Because the containment integrity is maintained during cooldown Modes 3 and 4, the passive containment cooling system provides the safety-related ultimate heat sink. Therefore, most of the steam

generated in the in-containment refueling water storage tank is condensed on the inside of the containment vessel and drains back into the in-containment refueling water storage tank via the condensate return gutter arrangement. This allows it to ~~indefinitely~~ function as a heat sink for greater than 14 days.

6.3.3.4.2 Loss of Normal Residual Heat Removal Cooling With The Reactor Coolant System Pressure Boundary Intact

During normal shutdown conditions, the normal residual heat removal system is placed into service at about 350°F to accomplish reactor coolant system cooldown to refueling temperatures. The normal residual heat removal system piping is safety-related and meets seismic Category I requirements to prevent pipe breaks that could result in a significant loss of reactor coolant during system operation. The pump motors and the electrical power supplies are nonsafety-related.

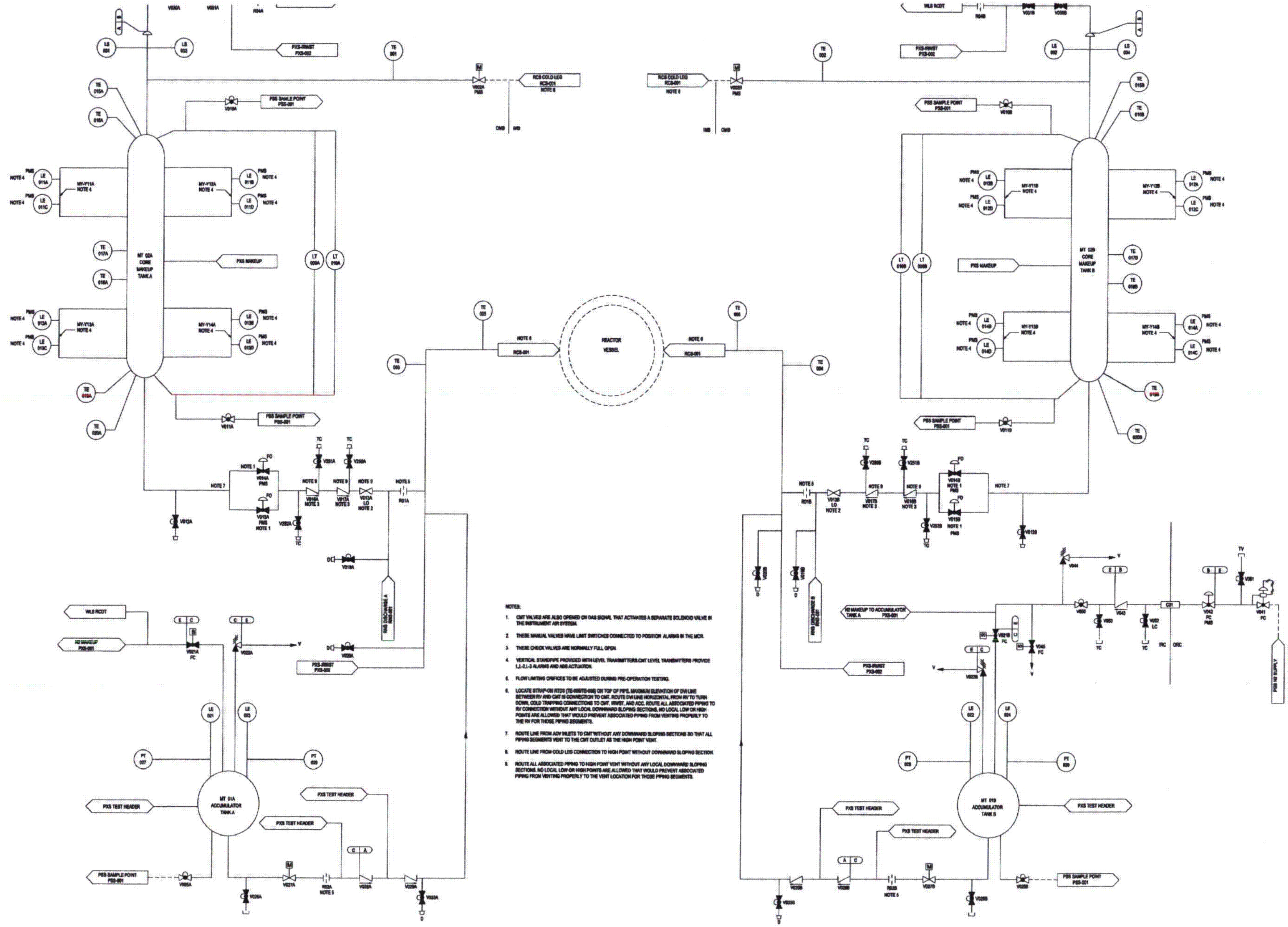
The system is designed so that with single failure of an active system component, it can maintain the plant in a hot shutdown condition (<350°F). It is also possible to perform a reactor coolant system cooldown, but at a slower rate than with full system capability. Heat removed by the normal residual heat removal system is transferred to the component cooling water system and then to the service water system. The heat removal path is powered by the nonsafety-related diesel-generators in the event that offsite power is lost.

Since the normal residual heat removal pumps are nonsafety-related, they may not be available. In this case, the reactor coolant system pressure boundary remains intact and the passive residual heat removal heat exchanger provides the safety-related heat removal flow path.

The normal residual heat removal system is operated once the reactor coolant system temperature is too low to support sufficient steam production for decay heat removal. With a loss of shutdown cooling, the reactor coolant system temperature does not increase sufficiently to initiate steam generator steaming and to reduce steam generator level. This is because the steam generators are normally filled, with a nitrogen purge established, during shutdown conditions. The loss of cooling would result in the heat up of the reactor coolant system and a pressure increase resulting in the normal residual heat removal system relief valve opening. This loss of fluid would result in a decrease in the pressurizer level; which a low pressurizer level signal automatically actuates the core make tanks and the passive residual heat removal heat exchanger. The passive residual heat removal heat exchanger could also be manually actuated.

The passive residual heat removal heat exchanger is capable of functioning at low reactor coolant system temperatures and pressures, but it may not be able to maintain the initial reactor coolant system temperature. It can remove sufficient heat to maintain the reactor coolant system within the normal residual heat removal system design limits (400°F). This permits the normal residual heat removal system to be placed back in operation when it becomes available.

For this event, the reactor coolant system temperature is expected to increase and expand into the pressurizer. Reactor coolant system injection should not be required. The makeup pumps are aligned for automatic operation in the event that pressurizer level decreases, due to leakage. However, since they are nonsafety-related, they are considered unavailable for reactor coolant



NOTES:

1. CMT VALVES ARE ALSO OPENED ON DAS SIGNAL THAT ACTIVATES A SEPARATE SOLENOID VALVE IN THE INSTRUMENT AIR SYSTEM.
2. THESE MANUAL VALVES HAVE LIMIT SWITCHES CONNECTED TO POSITION ALARMS IN THE MCR.
3. THESE CHECK VALVES ARE NORMALLY FULL OPEN.
4. VERTICAL STANDPIPE PROVIDED WITH LEVEL TRANSMITTERS.CMT LEVEL TRANSMITTERS PROVIDE L1-2,L-3 ALARMS AND ADS ACTUATION.
5. FLOW LIMITING ORIFICES TO BE ADJUSTED DURING PRE-OPERATION TESTING.
6. LOCATE STRAP-ON RTDS (TE-005/TE-006) ON TOP OF PIPE. MAXIMUM ELEVATION OF DVI LINE BETWEEN RV AND CMT IS CONNECTION TO CMT. ROUTE DVI LINE HORIZONTAL FROM RV TO TURN DOWN, COLD TRAPPING CONNECTIONS TO CMT, IRWST, AND ACC. ROUTE ALL ASSOCIATED PIPING TO RV CONNECTION WITHOUT ANY LOCAL DOWNWARD SLOPING SECTIONS. NO LOCAL LOW OR HIGH POINTS ARE ALLOWED THAT WOULD PREVENT ASSOCIATED PIPING FROM VENTING PROPERLY TO THE RV FOR THOSE PIPING SEGMENTS.
7. ROUTE LINE FROM ADV INLETS TO CMT WITHOUT ANY DOWNWARD SLOPING SECTIONS SO THAT ALL PIPING SEGMENTS VENT TO THE CMT OUTLET AS THE HIGH POINT VENT.
8. ROUTE LINE FROM COLD LEG CONNECTION TO HIGH POINT WITHOUT DOWNWARD SLOPING SECTION.
9. ROUTE ALL ASSOCIATED PIPING TO HIGH POINT VENT WITHOUT ANY LOCAL DOWNWARD SLOPING SECTIONS. NO LOCAL LOW OR HIGH POINTS ARE ALLOWED THAT WOULD PREVENT ASSOCIATED PIPING FROM VENTING PROPERLY TO THE VENT LOCATION FOR THOSE PIPING SEGMENTS.

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

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

[tlw20]

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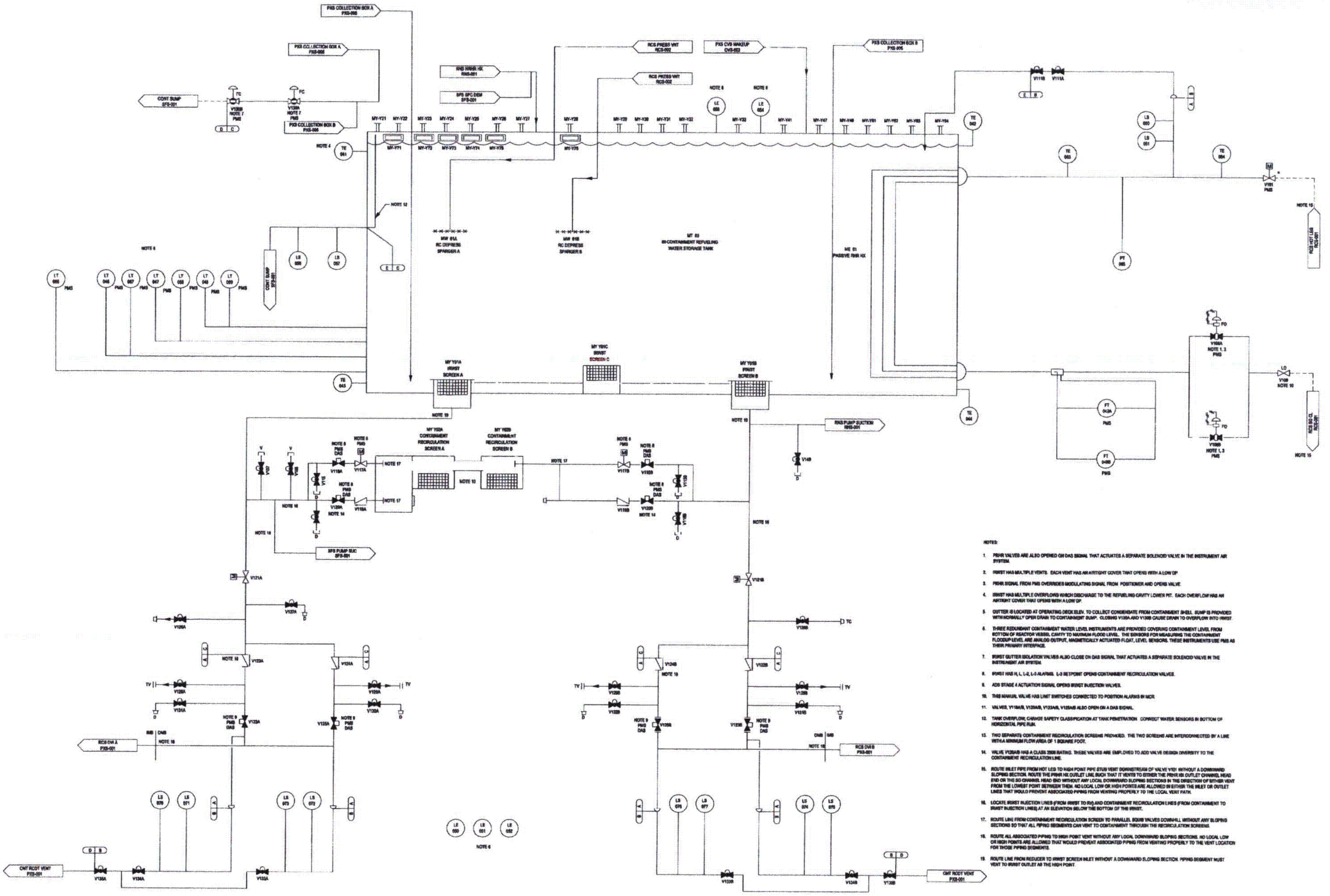


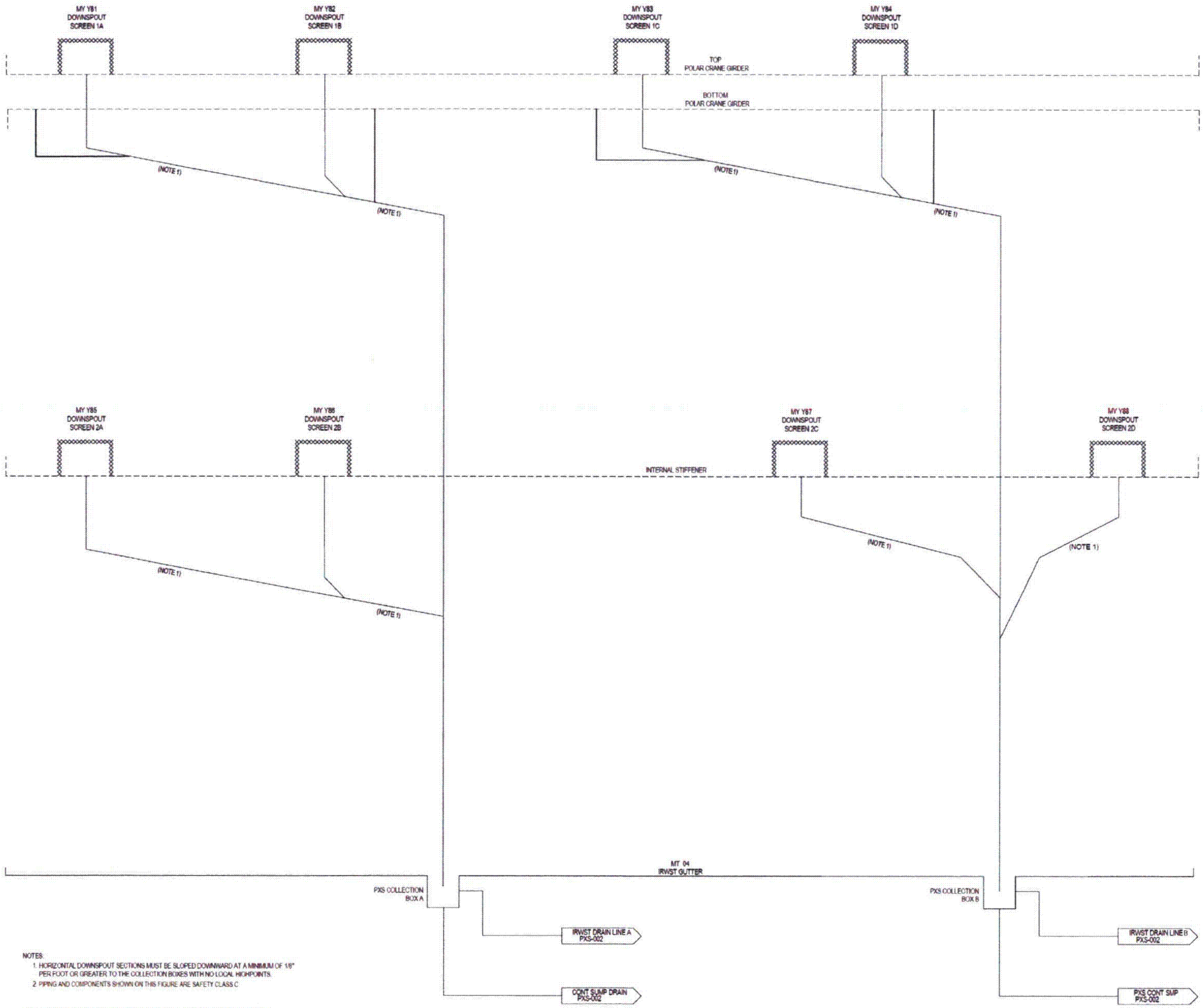
NOTES:

1. PRHR VALVES ARE ALSO OPENED ON DAS SIGNAL THAT ACTUATES A SEPARATE SOLENOID VALVE IN THE INSTRUMENT AIR SYSTEM.
2. IRWST HAS MULTIPLE VENTS. EACH VENT HAS AN AIRTIGHT COVER THAT OPENS WITH A LOW DP.
3. PRHR SIGNAL FROM PMS OVERRIDES MODULATING SIGNAL FROM POSITIONER AND OPENS VALVE.
4. IRWST HAS MULTIPLE OVERFLOWS WHICH DISCHARGE TO THE REFUELING CAVITY LOWER PIT. EACH OVERFLOW HAS AN AIRTIGHT COVER THAT OPENS WITH A LOW DP.
5. GUTTER IS LOCATED AT OPERATING DECK ELEV. TO COLLECT CONDENSATE FROM CONTAINMENT SHELL. SUMP IS PROVIDED WITH NORMALLY OPEN DRAIN TO CONTAINMENT SUMP. CLOSING V130A AND V130B CAUSE DRAIN TO OVERFLOW INTO IRWST.
6. THREE REDUNDANT CONTAINMENT WATER LEVEL INSTRUMENTS ARE PROVIDED COVERING CONTAINMENT LEVEL FROM BOTTOM OF REACTOR VESSEL CAVITY TO MAXIMUM FLOOD LEVEL. THE SENSORS FOR MEASURING THE CONTAINMENT FLOODUP LEVEL ARE ANALOG OUTPUT, MAGNETICALLY ACTUATED FLOAT, LEVEL SENSORS. THESE INSTRUMENTS USE PMS AS THEIR PRIMARY INTERFACE.
7. IRWST GUTTER ISOLATION VALVES ALSO CLOSE ON DAS SIGNAL THAT ACTUATES A SEPARATE SOLENOID VALVE IN THE INSTRUMENT AIR SYSTEM.
8. IRWST HAS H, L, L-2, L-3 ALARMS. L-3 SETPOINT OPENS CONTAINMENT RECIRCULATION VALVES.
9. ADS STAGE 4 ACTUATION SIGNAL OPENS IRWST INJECTION VALVES.
10. THIS MANUAL VALVE HAS LIMIT SWITCHES CONNECTED TO POSITION ALARMS IN MCR.
11. VALVES, V118A/B, V120A/B, V123A/B, V125A/B ALSO OPEN ON A DAS SIGNAL.
12. TANK OVERFLOW, CHANGE SAFETY CLASSIFICATION AT TANK PENETRATION. CONNECT WATER SENSORS IN BOTTOM OF HORIZONTAL PIPE RUN.
13. TWO SEPARATE CONTAINMENT RECIRCULATION SCREENS PROVIDED. THE TWO SCREENS ARE INTERCONNECTED BY A LINE WITH A MINIMUM FLOW AREA OF 1 SQUARE FOOT.
14. VALVE V120A/B HAS A CLASS 2500 RATING. THESE VALVES ARE EMPLOYED TO ADD VALVE DESIGN DIVERSITY TO THE CONTAINMENT RECIRCULATION LINE.
15. ROUTE INLET PIPE FROM HOT LEG TO HIGH POINT PIPE STUB VENT DOWNSTREAM OF VALVE V101 WITHOUT A DOWNWARD SLOPING SECTION. ROUTE THE PRHR HX OUTLET LINE SUCH THAT IT VENTS TO EITHER THE PRHR HX OUTLET CHANNEL HEAD END OR THE SG CHANNEL HEAD END WITHOUT ANY LOCAL DOWNWARD SLOPING SECTIONS IN THE DIRECTION OF EITHER VENT FROM THE LOWEST POINT BETWEEN THEM. NO LOCAL LOW OR HIGH POINTS ARE ALLOWED IN EITHER THE INLET OR OUTLET LINES THAT WOULD PREVENT ASSOCIATED PIPING FROM VENTING PROPERLY TO THE LOCAL VENT PATH.
16. LOCATE IRWST INJECTION LINES (FROM IRWST TO RV) AND CONTAINMENT RECIRCULATION LINES (FROM CONTAINMENT TO IRWST INJECTION LINES) AT AN ELEVATION BELOW THE BOTTOM OF THE IRWST.
17. ROUTE LINE FROM CONTAINMENT RECIRCULATION SCREEN TO PARALLEL SQUIB VALVES DOWNHILL WITHOUT ANY SLOPING SECTIONS SO THAT ALL PIPING SEGMENTS CAN VENT TO CONTAINMENT THROUGH THE RECIRCULATION SCREENS.
18. ROUTE ALL ASSOCIATED PIPING TO HIGH POINT VENT WITHOUT ANY LOCAL DOWNWARD SLOPING SECTIONS. NO LOCAL LOW OR HIGH POINTS ARE ALLOWED THAT WOULD PREVENT ASSOCIATED PIPING FROM VENTING PROPERLY TO THE VENT LOCATION FOR THOSE PIPING SEGMENTS.
19. ROUTE LINE FROM REDUCER TO IRWST SCREEN INLET WITHOUT A DOWNWARD SLOPING SECTION. PIPING SEGMENT MUST VENT TO IRWST OUTLET AS THE HIGH POINT.

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

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

[tlw21]



NOTES:

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

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

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

[g22]

Simpli
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Figure 6.3-2 not used.

(Renumbered as Figure 6.3-1, Sheet 2)

7.4 Systems Required for Safe Shutdown

Systems to establish safe shutdown conditions perform two basic functions. First, they provide the necessary reactivity control to maintain the core in a subcritical condition. Boration capability is provided to compensate for xenon decay and to maintain the required core shutdown margin. Second, these systems must provide residual heat removal capability to maintain adequate core cooling.

The designation of systems required for safe shutdown depends on identifying those systems that provide the following capabilities for maintaining a safe shutdown:

- Decay heat removal
- Reactor coolant system inventory control
- Reactor coolant system pressure control
- Reactivity control

There are two different safe shutdown conditions that are expected following a transient or accident condition. Short-term safe shutdown refers to the plant conditions from the start of an event until about 36 hours later. Long-term safe shutdown refers to the plant conditions after this 36-hour period.

The short-term safe shutdown conditions include maintaining the reactor subcritical, the reactor coolant average temperature less than or equal to no load temperature, and adequate coolant inventory and core cooling. These shutdown conditions shall be achieved following any of the design basis events using safety-related equipment. The specific safe shutdown condition achieved is a function of the particular accident sequence.

The long-term safe shutdown conditions are the same as the short-term conditions except that the coolant temperature shall be less than 420°F. This long-term condition must be achieved within 36 hours and following a non-LOCA event using the PRHR HX as shown in Chapter 19E. These safe shutdown conditions can be maintained by the PRHR HX for greater than 14 days based on a non-bounding, conservative analysis that only credits using safety-related equipment. In addition, these safe shutdown conditions can be maintained indefinitely using the ADS and passive injection and recirculation as discussed in 7.4.1.1. Also refer to Chapter 6.3.1.1 for additional discussion on safe shutdown requirements. ~~safety-related equipment.~~

There are no systems specifically and solely dedicated as safe shutdown systems. However, there are a number of plant systems that are available to establish and maintain safe shutdown conditions. Normally, in the event of a turbine or reactor trip, nonsafety-related plant systems automatically function to place the plant in short-term safe shutdown, as described in subsection 7.4.1.2. During the short-term safe shutdown condition, an adequate heat sink is provided to remove reactor core residual heat and boration control is available. Redundancy of systems and components is provided to enable continued maintenance of the short-term safe shutdown condition. Additional redundant nonsafety-related systems are normally available to manually perform a plant depressurization and cooldown.

The engineered safety systems are designed to establish and maintain safe shutdown conditions for the plant. Nonsafety-related systems are not required for safe shutdown of the plant.

This section focuses on safety-related systems used to establish and maintain safe shutdown conditions. The discussion of safe shutdown does not include accident response and/or mitigation since the standard review plan for this section addresses safe shutdown not related to accident mitigation. However, safe shutdown conditions are also established and maintained by these safety-related systems following accident conditions. For example, the control rods are released to initially place the plant in a shutdown condition to mitigate the consequences of various accidents. The passive core cooling system, on the other hand, is used to provide core cooling in an accident, but it is also one of the principal systems used for safe shutdown. Only those specific engineered safety systems listed in Table 7.4-1 are used to establish and maintain safe shutdown of the plant. These engineered safety systems automatically function to place the plant in a safe shutdown condition without operator action.

The instrumentation functions necessary for safe shutdown are available through instrumentation channels associated with the safety-related systems in the primary plant. These channels automatically actuate the protective functions provided by the safety-related systems. Manual actuation of the associated safety-related systems is also provided.

The instrumentation systems discussed in this section are those which are required during nonaccident conditions to align the safety-related systems and perform the specified safe shutdown functions.

The specific systems available for safe shutdown are discussed in subsection 7.4.2 and are listed in Table 7.4-1.

Maintenance of safe shutdown conditions with these systems, and the associated instrumentation and controls, includes consideration of the accident consequences that might challenge safe shutdown conditions. The accident consequences that are germane are those that tend to degrade the capabilities for coolant circulation, boration, heat removal, and depressurization. Safe shutdown is achieved following any of the accidents analyzed in Chapter 15. The specific safe shutdown condition reached is a function of the particular accident sequence.

The instrumentation and controls discussed in subsection 7.4.1 are used to control and/or monitor shutdown. These safety-related systems allow the maintenance of safe shutdown, even under accident conditions that tend toward a return to criticality or a loss of heat sink.

In addition to the operation of safety-related systems used for safe shutdown, as described in subsection 7.4.1, the following are part of the safe shutdown provisions:

- The turbine is tripped. (This can be accomplished at the turbine as well as from the main control room.)
- The reactor is tripped. (This can be accomplished at the reactor trip switchgear as well as from the main control room.)
- Support of engineered safety systems actuation is provided by safety-related onsite dc power.

7.4.1 Safe Shutdown

7.4.1.1 Safe Shutdown Using Safety-Related Systems

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

Since this discussion only considers the use of safety-related systems, offsite electrical power sources are assumed to be lost at the start of the event. This results in a loss of the reactor coolant pumps. Even though the reactor coolant pumps are tripped during the initiation of certain engineered safety system actuation, it is assumed that no engineered safety system actuation signal is generated for this initiating event. With loss of the reactor coolant pumps, reactor coolant system natural circulation flow initiates and transfers core heat to the steam generators. Since feedwater flow is lost, the existing steam generator water inventory provides initial decay heat removal capability.

The initial loss of main ac power results in the Class 1E dc batteries automatically supplying power to the Class 1E dc power distribution network and the four Class 1E 120 Vac instrumentation divisions via the inverters.

The initial response of the passive safety systems is to actuate the passive residual heat removal heat exchanger due to low steam generator water level. The passive residual heat removal heat exchanger removes decay heat from the core by transferring this heat to the in-containment refueling water storage tank.

The passive residual heat removal heat exchanger removes core decay heat, cooling the reactor coolant system. As reactor coolant system cooldown continues, the reactor coolant system pressure decreases due to contraction of the reactor coolant system inventory since the pressurizer heaters are de-energized. An engineered safety system actuation signal occurs when reactor coolant system pressure decreases below a setpoint. This actuates the core makeup tanks, if they had not been previously actuated due to low pressurizer level. The core makeup tanks provide borated water injection to the reactor coolant system.

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

The in-containment refueling water storage tank starts to boil about one to two hours after passive residual heat removal operation is initiated. Once boiling occurs, the in-containment refueling water storage tank begins steaming to containment, transferring heat to the air flowing on the outside of the containment shell. As steaming to containment continues, containment pressure slowly increases. As containment pressure slowly increases, an engineered safety system actuation signal is generated on containment high pressure, resulting in the initiation of passive containment cooling. This provides water flow on the outside of the containment shell to improve the heat removal performance from containment through evaporative cooling to the outside air.

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

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

Operation in this configuration may be limited in time duration by reactor coolant system leakage. The core makeup tanks can only supply a limited amount of makeup in the event there is reactor coolant system leakage. Eventually the volume of the water in the core makeup tanks will decrease to the first stage automatic depressurization setpoint. The time to reach this setpoint depends upon the reactor coolant system leak rate and the reactor coolant cooldown.

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

The automatic depressurization system can be manually initiated by the operator at any time, but no operator action is needed to provide safe shutdown conditions. Once the automatic depressurization system sequence initiates, the plant automatically transitions to lower pressure and temperature conditions that establish and maintain long-term safe shutdown of the plant.

When the automatic depressurization system is actuated, the first stage depressurization valves open and the reactor coolant system depressurization starts. The second and third stage depressurization valves open in sequence, based on automatic timers that are started upon the actuation of the first stage depressurization valves. As reactor coolant inventory continues to be lost, the core makeup tanks continue to inject. If the volume of the water in the core makeup tanks decrease to the fourth stage automatic depressurization setpoint, the fourth stage depressurization valves open. The water and steam vented from the reactor coolant system initially flows into the in-containment refueling water storage tank and overflows into the refueling canal. Eventually this overflows into the reactor vessel cavity, where any moisture from the fourth stage automatic depressurization system valves also collects from discharge in the loop

Table 9.5.1-1 (Sheet 11 of 3~~3~~4)

AP1000 FIRE PROTECTION PROGRAM COMPLIANCE WITH BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp(1)	Remarks
70. Drains in areas containing combustible liquids should have provisions for preventing the back flow of combustible liquids to safety-related areas through the interconnected drain systems.	C.5.a (14)	C	
71. Water drainage from areas that may contain radioactivity should be collected, sampled, and analyzed before discharge to the environment.	C.5.a(14)	WA	See Note 2. Capability is provided.
Safe Shutdown Capability			
72. Fire damage should be limited so that one train of systems necessary to achieve and maintain hot shutdown conditions from either the main control room or emergency control station is free of fire damage.	C.5.b(1)	C	
73. Fire damage should be limited so that systems necessary to achieve and maintain cold shutdown from either the control room or emergency control station can be repaired within 72 hours.	C.5.b (1)	AC	Safe shutdown following a fire is defined for the AP1000 plant as the ability to achieve and maintain the reactor coolant system (RCS) temperature below 215.6°C (420°F) without uncontrolled venting of the primary coolant from the RCS. This is a departure from the criteria applied to the evolutionary plant designs, and the existing plants where safe shutdown for fires applies to both hot and cold shutdown capability. With expected RCS leakage the AP1000 plant can maintain safe shutdown conditions ~~indefinitely~~for greater than 14 days. Therefore, repairs to systems necessary to reach cold shutdown need not be completed within 72 hours.

Table 14.3-2 (Sheet 7 of 17)

DESIGN BASIS ACCIDENT ANALYSIS

Reference		**Design Feature**	**Value**
Section	6.3.6.1.3	The bottom of the in-containment refueling water storage tank is located above the direct vessel injection nozzle centerline (ft).	≥ 3.4
Section	6.3.6.1.3	The pH baskets are located below plant elevation 107′ 2″.	
Figure	6.3-1	The passive core cooling system has two direct vessel injection lines.	
Table	6.3-2	The passive core cooling system has two core makeup tanks, each with a minimum required volume (ft^3).	2500
Table	6.3-2	The passive core cooling system has two accumulators, each with a minimum required volume (ft^3)	2,000
Table	6.3-2	The passive core cooling system has an in-containment refueling water storage tank with a minimum required water volume (ft^3)	73,900
Section	6.3.2.2.3	The containment floodup volume for a LOCA in PXS room B has a maximum volume (ft^3) (excluding the IRWST) below a containment elevation of 108 feet.	73,500
Table	6.3-2	Each sparger has a minimum discharge flow area (in^2).	≥ 274
Table	6.3-2	The passive core cooling system has two pH adjustment baskets each with a minimum required volume (ft^3).	280
Section	14.2.9.1.3f	The passive residual heat removal heat exchanger minimum natural circulation heat transfer rate (Btu/hr) - With 520°F hot leg and 80°F IRWST - With 420°F hot leg and 80°F IRWST	 ≥ 1.78 E+08 ≥ 1.11 E+08
Section	6.3.6.1.3	The centerline of the HX's upper channel head is located above the HL centerline (ft).	≥ 26.3
Figure	6.3-1	The CMT level sensors (PXS-11A/B/C/D, -12A/B/C/D, -13A/B/C/D, and -14A/B/C/D) upper level tap centerlines are located below the centerline of the upper level tap connection to the CMTs (in).	1″ ± 1″
Figure	6.3-1	The CMT inlet lines (cold leg to high point) have no downward sloping sections.	
Figure	6.3-1	The maximum elevation of the CMT injection lines between the connection to the CMT and the reactor vessel is the connection to the CMTs.	
Figure	6.3-~~2~~1	The PRHR inlet line (hot leg to high point) has no downward sloping sections.	

Table 14.3-2 (Sheet 8 of 17)

DESIGN BASIS ACCIDENT ANALYSIS

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

15.0.13 Operator Actions

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

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

15.0.14 Loss of Offsite ac Power

As required in GDC 17 of 10 CFR Part 50, Appendix A, anticipated operational occurrences and postulated accidents are analyzed assuming a loss of offsite ac power. The loss of offsite power is not considered as a single failure, and the analysis is performed without changing the event category. In the analyses, the loss of offsite ac power is considered to be a potential consequence of the event.

A loss of offsite ac power will be considered a consequence of an event due to disruption of the grid following a turbine trip during the event. Event analyses that do not result in a possible consequential disruption of offsite ac power do not assume offsite power is lost.

For those events where offsite ac power is lost, an appropriate time delay between turbine trip and the postulated loss of offsite ac power is assumed in the analyses. A time delay of 3 seconds is used. This time delay is based on the inherent stability of the offsite power grid as discussed in Section 8.2. Following the time delay, the effect of the loss of offsite ac power on plant auxiliary equipment – such as reactor coolant pumps, main feedwater pumps, condenser, startup feedwater pumps, and RCCAs – is considered in the analyses. Turbine trip occurs 5 seconds following a reactor trip condition being reached. This delay is part of the AP1000 reactor trip system.

Design basis LOCA analyses are governed by the GDC-17 requirement to consider the loss of offsite power. For the AP1000 design, in which all the safety-related systems are passive, the availability of offsite power is significant only regarding reactor coolant pump operation for LOCA events. A sensitivity study for AP1000 has shown that for large-break LOCAs, assuming the loss of offsite power coincident with the inception of the LOCA event is nonlimiting relative to assuming continued reactor coolant pump operation until the automatic reactor coolant pump trip occurs following an "S" signal less than 10 seconds into the transient. For small-break LOCA events, the AP1000 automatic reactor coolant pump trip feature prevents continued operation of the reactor coolant pumps from mixing the liquid and vapor present within a two-phase reactor coolant system inventory to increase the liquid break flow and deplete the reactor coolant system mass inventory rapidly. The automatic reactor coolant pump trip occurs early enough during AP1000 small-break LOCA transients that emergency core cooling system performance is not affected by the loss of offsite power assumption because the total break flow is approximately equivalent for reactor coolant pump trip occurring either at time zero or as a result of the "S"

the loss of the condenser vacuum. In addition, analyses for the other possible causes of a turbine trip, listed in subsection 15.2.3.1, are covered by subsection 15.2.3. Possible overfrequency effects, due to a turbine overspeed condition, are discussed in subsection 15.2.2.1 and are not a concern for this type of event.

15.2.6 Loss of ac Power to the Plant Auxiliaries

15.2.6.1 Identification of Causes and Accident Description

The loss of power to the plant auxiliaries is caused by a complete loss of the offsite grid accompanied by a turbine-generator trip. The onsite standby ac power system remains available but is not credited to mitigate the accident.

From the decay heat removal point of view, in the long term this transient is more severe than the turbine trip event analyzed in subsection 15.2.3 because, for this case, the decrease in heat removal by the secondary system is accompanied by a reactor coolant flow coastdown, which further reduces the capacity of the primary coolant to remove heat from the core. The reactor will trip:

- Upon reaching one of the trip setpoints in the primary or secondary systems as a result of the flow coastdown and decrease in secondary heat removal.

- Due to the loss of power to the control rod drive mechanisms as a result of the loss of power to the plant.

Following a loss of ac power with turbine and reactor trips, the sequence described below occurs:

- Plant vital instruments are supplied from the Class 1E and uninterruptable power supply.

- As the steam system pressure rises following the trip, the steam generator power-operated relief valves may be automatically opened to the atmosphere. The condenser is assumed not to be available for turbine bypass. If the steam flow rate through the power-operated relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.

- The onsite standby power system, if available, supplies ac power to the selected plant non-safety loads.

- As the no-load temperature is approached, the steam generator power-operated relief valves (or safety valves, if the power-operated relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot shutdown condition if the startup feedwater is available to supply water to the steam generators.

- If startup feedwater is not available, the PRHR heat exchanger is actuated.

During a plant transient, core decay heat removal is normally accomplished by the startup feedwater system if available, which is started automatically when low levels occur in either steam generator. If that system is not available, emergency core decay heat removal is provided

by the PRHR heat exchanger. The PRHR heat exchanger is a C-tube heat exchanger connected, through inlet and outlet headers, to the reactor coolant system. The inlet to the heat exchanger is from the reactor coolant system hot leg, and the return is to the steam generator outlet plenum. The heat exchanger is located above the core to provide natural circulation flow when the reactor coolant pumps are not operating. The IRWST provides the heat sink for the heat exchanger. The PRHR heat exchanger, in conjunction with the passive containment cooling system, provides core cooling and maintains ~~keeps~~ the reactor coolant system conditions to satisfy the evaluation criteria~~subcooled indefinitely~~. After the IRWST water reaches saturation, (in about two and half hours) steam starts to vent to the containment atmosphere. The condensation that collects on the containment steel shell (cooled by the passive containment cooling system) returns to the IRWST, maintaining fluid level for the PRHR heat exchanger heat sink. The analysis shows that the natural circulation flow in the reactor coolant system following a loss of ac power event is sufficient to remove residual heat from the core.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant and PRHR loops.

A loss of ac power to the plant auxiliaries is a Condition II event, a fault of moderate frequency. This event is more limiting with respect to long-term heat removal than the turbine trip initiated decrease in secondary heat removal without loss of ac power, which is discussed in subsection 15.2.3. A loss of offsite power to the plant auxiliaries will also result in a loss of normal feedwater.

The plant systems and equipment available to mitigate the consequences of a loss of ac power event are discussed in subsection 15.0.8 and listed in Table 15.0-6.

15.2.6.2 Analysis of Effects and Consequences

15.2.6.2.1 Method of Analysis

The analysis is performed to demonstrate the adequacy of the protection and safety monitoring system, the PRHR heat exchanger, and the reactor coolant system natural circulation capability in removing long-term (approximately 36,000 seconds) decay heat. This analysis also demonstrates the adequacy of these systems in preventing excessive heatup of the reactor coolant system with possible reactor coolant system overpressurization or loss of reactor coolant system water.

A modified version of the LOFTRAN code (Reference 2), described in WCAP-15644 (Reference 6), is used to simulate the system transient following a plant loss of offsite power. The simulation describes the plant neutron kinetics and reactor coolant system, including the natural circulation, pressurizer, and steam generator system responses. The digital program computes pertinent variables, including the steam generator level, pressurizer water level, and reactor coolant average temperature.

The assumptions used in this analysis minimize the energy removal capability of the PRHR heat exchanger and maximize the coolant system expansion.

Table 19.59-18 (Sheet 6 of 25)

AP1000 PRA-BASED INSIGHTS

Insight	Disposition
1e. (cont.)	
Capability exists and guidance is provided for the control room operator to identify a leak in the PRHR HX of 500 gpd. This limit is based on the assumption that a single crack leaking this amount would not lead to a PRHR HX tube rupture under the stress conditions involving the pressure and temperature gradients expected during design basis accidents, which the PRHR HX is designed to mitigate.	6.3.3 & 16.1
The positions of the inlet and outlet PRHR valves are indicated and alarmed in the control room.	6.3.7
PRHR air-operated valves are stroke-tested quarterly. The PRHR HX is tested to detect system performance degradation every 10 years.	3.9.6
PRHR is required by Technical Specifications to be available from Modes 1 through 5 with RCS pressure boundary intact.	16.1
The PRHR HX, in conjunction with the IRWST, condensate return features and the PCS, can provide core cooling for an ~~indefinite period of time~~ greater than 14 days. After the IRWST water reaches its saturation temperature, the process of steaming to the containment initiates. Condensation occurs on the steel containment vessel, and the condensate is collected in a safety-related gutter arrangement, which returns the condensate to the IRWST. The gutter normally drains to the containment sump, but when the PRHR HX actuates, safety-related isolation valves in the gutter drain line shut and the gutter overflow returns directly to the IRWST. The following design features provide proper re-alignment for the gutter system valves to direct water to the IRWST:	6.3.2.1.1 & 6.3.7.6
- IRWST gutter and its drain isolation valves are safety-related	
- These isolation valves are designed to fail closed on loss of compressed air, loss of Class 1E dc power, or loss of the PMS signal	
- These isolation valves are actuated automatically by PMS and DAS.	7.3.1.2.7
The PRHR subsystem provides a safety-related means of removing decay heat following loss of RNS cooling during shutdown conditions with the RCS intact.	16.1
2. The protection and safety monitoring system (PMS) provides a safety-related means of performing the following functions:	Tier 1 Information
- Initiates automatic and manual reactor trip	
- Automatic and manual actuation of engineered safety features (ESF).	
PMS monitors the safety-related functions during and following an accident as required by Regulatory Guide 1.97.	7.1.1

19E.4.10.2 Shutdown Temperature Evaluation

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

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

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

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

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

Once actuated, at about 2,700~~4600~~ seconds, the CMTs operate in recirculation mode, injecting cold borated water into the RCS. In the first part of their operation, due to the injection of cold water~~flow rate~~, the CMTs operate in conjunction with the PRHR HX to reduce RCS temperature. Due to the primary system cooldown, the PRHR heat transfer capability drops below the decay heat and the RCS cooldown is essentially driven by the CMT cold injection flow. However, at about ~~3,565~~6,000 seconds, the CMT cooling effect decreases and the RCS starts heating up again (Figure 19.E.4.10-1). The RCS temperature increases until the PRHR HX can match decay heat. At about ~~31,000~~ 46,700 seconds, the PRHR heat transfer matches decay heat and it continues to operate to reduce the RCS temperature to below 420°F within 36 hours. As seen from Figure 19E.4.10-1, the cold leg temperature in the loop with the PRHR is reduced to 420°F ~~at 82,600~~ 52,900 seconds, while the core average temperature reaches 420°F within ~~123,600~~ 120,900 seconds (approximately 34 hours).

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

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

19E.5 Technical Specifications

While the Technical Specification guidance provided in NUREG-1449 (Reference 2) relates to existing plant shutdown operation concerns, the underlying concerns relating to causes of events and recovery from those events during shutdown operations are applicable to the AP1000. Section 19E.5.1 summarizes the shutdown Technical Specifications. Section 19E.5.2 summarizes the AP1000's compliance with SECY-93-190 (Reference 16).

Table 19E.4.10-1

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

Event	Time (seconds)
Feedwater is Lost	10.0
Low Steam Generator Water Level (Narrow-Range) Reactor Trip Setpoint Reached	60.6~~72.4< 60~~
Rods Begin to Drop	62.6~~74.4< 61~~
Low Steam Generator Water Level (Wide-Range) Reached	209.5~~< 230~~
PRHR HX Actuation on Low Steam Generator Water Level (~~Wide~~Narrow-Range Coincident with Low Startup Feedwater Flow)	221.5~~129.4< 240~~
Low T_{cold} Setpoint Reached	2,752~~599.0< 2,400~~
Steam Line Isolation on Low T_{cold} Signal	2,764~~611.0< 2,400~~
CMTs Actuated on Low T_{cold} Signal	2,764~~617.0< 2,400~~
IRWST Reaches Saturation Temperature	15,900~~17,600< 15,500~~
Heat Extracted by PRHR HX Matches Core Decay Heat	46,700~~31,000< 34,500~~
~~CMTs Stop Recirculating~~	~~52,90043,500-~~
Cold Leg Temperature Reaches 420°F (loop with PRHR)	52,900 ~~82,600< 48,600~~
~~Hot Leg~~Core Average Temperature Reaches 420°F ~~(loop with PRHR)~~	120,900 ~~123,600< 124,400~~

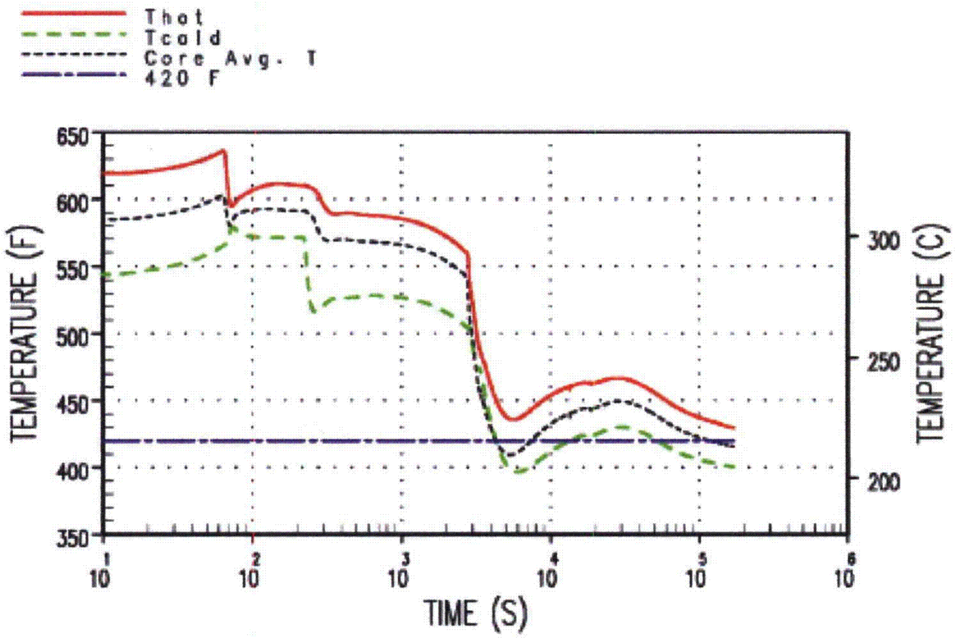

Figure 19E.4.10-1

Shutdown Temperature Evaluation, RCS Temperature

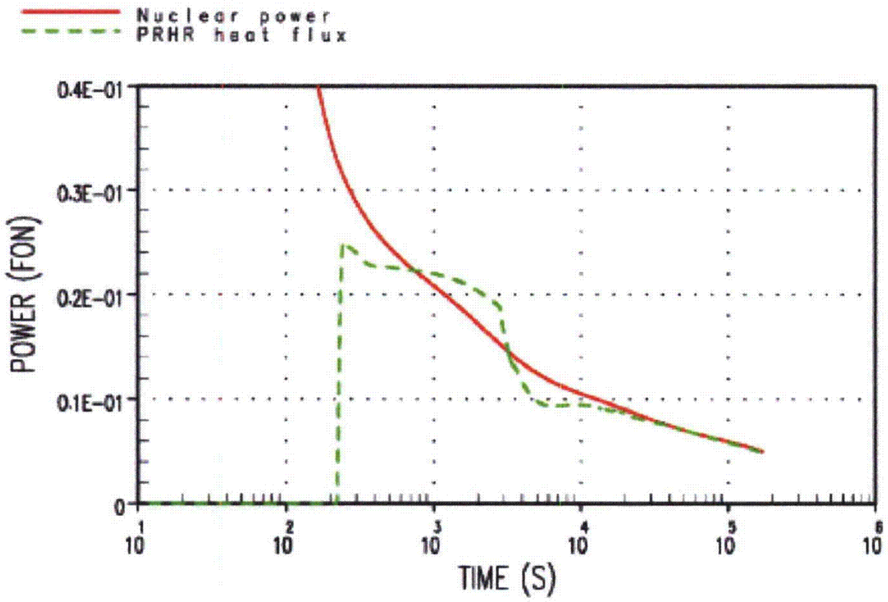

Figure 19E.4.10-2

Shutdown Temperature Evaluation, PRHR Heat Transfer

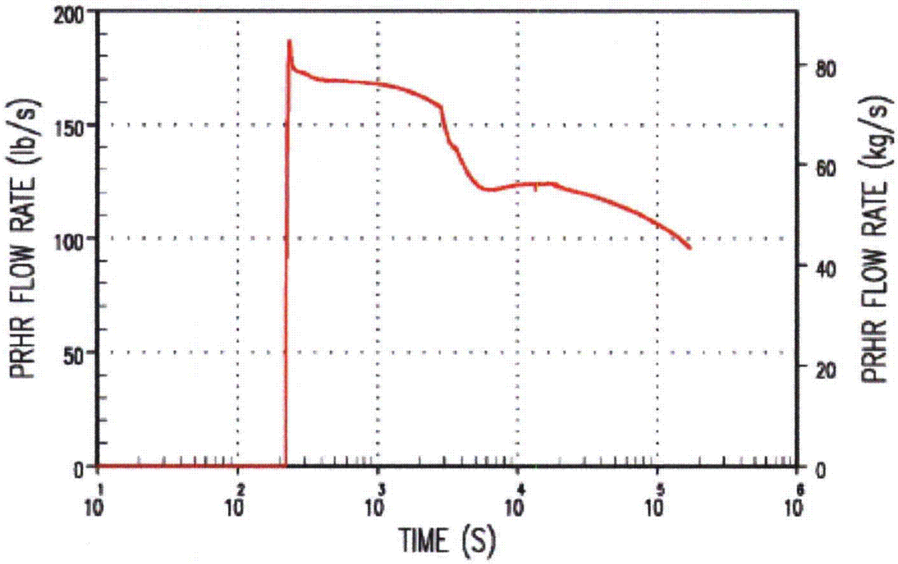

Figure 19E.4.10-3

Shutdown Temperature Evaluation, PRHR Flow Rate

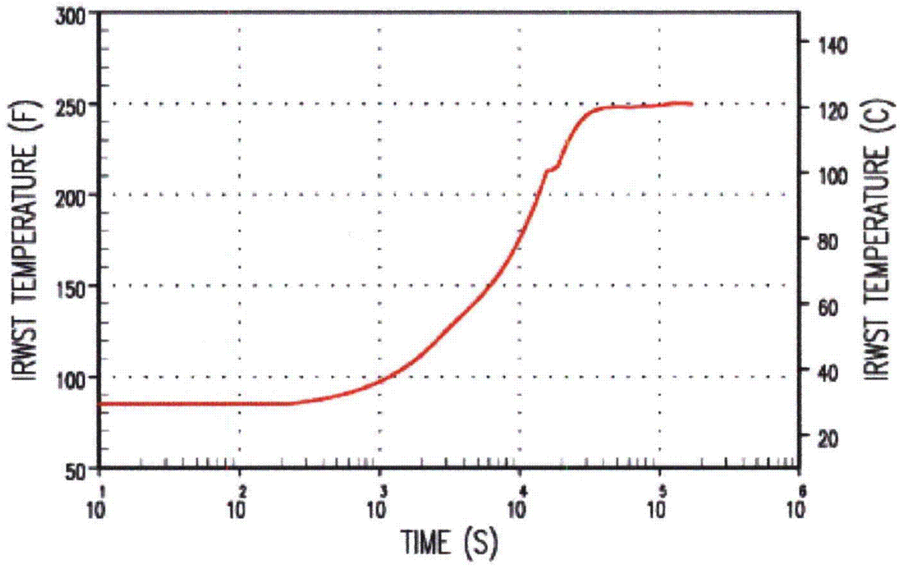

Figure 19E.4.10-4

Shutdown Temperature Evaluation, IRWST Heatup

Technical Specification Changes

LCO (continued)

10. Pressurizer Level and Associated Reference Leg Temperature

Pressurizer level is provided to monitor the RCS coolant inventory. During an accident, operation of the safeguards systems can be verified based on coolant inventory indicators.

The reference leg temperature is included in the Technical Specification since it is used to compensate the level signal.

11. In-Containment Refueling Water Storage Tank (IRWST) Water Level

The IRWST provides a long term heat sink for non-LOCA events and is a source of injection flow for LOCA events. When the IRWST is a heat sink, the level will change due to increased volume associated with the temperature increase. When saturation temperature is reached, the IRWST will begin steaming and initially lose mass to the containment atmosphere until condensation occurs on the steel containment shell which is cooled by the passive containment cooling system. The condensate is returned to the IRWST via a gutter and downspouts.

During a LOCA, the IRWST is available for injection. Depending on the severity of the event, when a fully depressurized RCS has been achieved, the IRWST will inject by gravity flow.

12. Passive Residual Heat Removal (PRHR) Flow and PRHR Outlet Temperature

PRHR Flow is provided to monitor primary system heat removal during accident conditions when the steam generators are not available. PRHR provides primary protection for non-LOCA events when the normal heat sink is lost.

PRHR outlet temperature is provided to monitor primary system heat removal during accident conditions when the steam generators are not available. PRHR provides primary protection for non-LOCA events when the normal heat sink is lost.

13, 14, 15, 16. Core Exit Temperature

Core Exit Temperature is provided for verification and long term surveillance of core cooling.

B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.4 Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Operating

BASES

BACKGROUND

The normal heat removal mechanism is the steam generators, which are supplied by the startup feedwater system. However, this path utilizes non-safety related components and systems, so its failure must be considered. In the event the steam generators are not available to remove decay heat for any reason, including loss of startup feedwater, the heat removal path is the PRHR HX (Ref. 1).

The principle component of the PRHR HX is a 100% capacity heat exchanger mounted in the In-containment Refueling Water Storage Tank (IRWST). The heat exchanger is connected to the Reactor Coolant System (RCS) by a inlet line from one RCS hot leg, and an outlet line to the associated steam generator cold leg channel head. The inlet line to the passive heat exchanger contains a normally open, motor operated isolation valve. The outlet line is isolated by two parallel, normally closed air operated valves, which fail open on loss of air pressure or control signal. There is a vertical collection point at the top of the common inlet piping high point which serves as a gas collector. It is provided with level detectors that indicate when noncondensible gases have collected in this area. There are provisions to manually vent these gases to the IRWST.

In order to preserve the IRWST water for long-~~-~~term PRHR HX operation, downspouts and a gutter are~~is~~ provided to collect and return water to the IRWST that has condensed on the inside surface of the containment shell. During normal plant operation, any water collected by the downspouts or gutter is directed to the normal containment sump. During PRHR HX operation, redundant series air operated valves are actuated to block the draining of condensate to the normal sump and to force the condensate into the IRWST. These valves fail closed on loss of air pressure or control signal.

The PRHR HX size and heat removal capability is selected to provide adequate core cooling for the limiting non-LOCA heatup Design Basis Accidents (DBAs) (Ref. 2). The Probability Risk Assessment (PRA) (Ref. 3) shows that PRHR HX is not required assuming that passive feed and bleed is available. Passive feed and bleed uses the Automatic Depressurization System (ADS) for bleed and the CMTs/accumulators/ IRWST for feed.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.4.7

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

REFERENCES

1. Section 6.3, "Passive Core Cooling System."
2. Chapter 15, "Safety Analysis."
3. AP1000 PRA.

Enclosure 3
Westinghouse Letter, Application for Withholding Proprietary Information from Public Disclosure (CAW-15-4240) with Affidavit
(6 pages including cover page)

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CAW-15-4240

10 July 2015

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: **APP-GW-GLR-161 Revision 3, "Changes to Passive Core Cooling System Condensate Return"**

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-15-4240 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by APOG.

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-15-240, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

Very truly yours,

Richard A. DeLong, Director
International Licensing & Regulatory Support

CAW-15-4240
10 July 2015

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, Richard A. DeLong, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

Richard A. DeLong, Director
International Licensing & Regulatory Support

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in APP-GW-GLR-161 Revision 3, "Changes to Passive Core Cooling System for Condensate Return" for submittal to the Commission, being transmitted by APOG letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with the topic of Condensate Return and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to:

(i) Provide the NRC and customers with technical information on the additional information on the condensate return evaluations.

(b) Further this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of providing more products and services.

(ii) Westinghouse can sell support and defense of industry guidelines and acceptance criteria for plant-specific applications.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar systems in commercial power reactors and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

Enclosure 4
Proprietary Information Notice and Copyright Notice
(2 pages including cover page)

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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