PMLevyCOLPEm Resource

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To: Habib, Donald
Cc: Kitchen, Robert
Subject: Draft Responses and AP1000 DCD Mark-ups for Levy Condensate Return RAI Questions 06.03-10 through 06.03-12
Attachments: APP-FSAR-GF-006 rev D nonprop.pdf

Don

As discussed earlier with you, Duke Energy is forwarding the subject draft RAI question responses for NRC staff review in order to facilitate discussion during the public meeting call on Tuesday, June 17. Note that there is no proprietary information contained in the attachment.

If you have any questions, please let us know.

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REQUEST FOR INFORMATION (RFI)

Note: Configuration management is controlled by the referenced technical document(s). RFIs are to be used for limited applications in accordance with the requirements contained in Section 5.0 of APP-GW-GFP-001.	
RFI Number: APP-FSAR-GF-006 rev 0	**Date:**
Project Number (for Shaw use only): N/A	**Project Name and Unit:** APOG
Request Initiated By: Richard Grumbir / Robert Kitchen	**Discipline:** Nuclear systems
Company Name: APOG LLC	
RFI Title: 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	

Referenced Technical Document(s): ***(Required Field)***	**Rev:**
APP-GW-GLR-161	1
APP-GW-GL-700	19

Information Requested:

Provide a response to the following question posed by the Nuclear Regulatory Commission staff in their Request for Additional Information (RAI) number 7439 submitted to Duke Energy in response to the condensate return COL Application update submitted for Levy County Units 1 and 2.

Question 06.03-10

As stated in section 4.3.3.5 of the Utility Requirements Document (URD) and restated in Section 2.3.2 of the staff's safety evaluation, a design requirement for the passive decay heat removal system is to have sufficient water capacity in the passive decay heat water pools to permit 72 hours of operation after SCRAM without the need for refill (ADAMS Accession No. ML070600372). Based upon the licensing guidance in the URD, NUREG-1242, SECY-94-084, and the Regulatory Treatment of Non-Safety Systems (RTNSS) as discussed in the Section 19.3 of the Standard Review Plan, in order for the Passive Residual Heat Removal Heat Exchanger (PRHR-HX) to meet the requirements of GDC 34 and GDC 44, the In-containment Refueling Water Storage Tank (IRWST) should have sufficient capacity to permit a minimum of 72 hours of operation after SCRAM following an accident without the need for refill. The submitted changes to the passive core cooling system regarding condensate return has caused staff to question the mission time for the PRHR-HX/IRWST. The staff requests the following:

- *What is the safety-related mission time for the PRHR-HX/IRWST following a non-LOCA accident?*
- *Provide the PRHR-HX tube plugging assumption used in the analysis of design basis accidents in Chapter 15 that credit use of the PRHR-HX.*
- *Provide the PRHR-HX tube plugging assumption used in the safe-shutdown analysis presented in Appendix E of Chapter 19.*
- *Update the FSAR to clarify the safety-related design basis for the PRHR-HX/IRWST regarding the 72 hour capacity of the IRWST for the mitgation of accidents.*

Question 06.03-11

In letter NPD-NRC-2014-005, dated February 07, 2014, Section 1.0 of Enclosure 2 states that among the "safety-related" design bases of the Passive Core cooling System (PXS) is the capability of 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. Compliance with safety-related design requirements is typically demonstrated through the use of conservative analyses or best estimate plus uncertainty evaluations. The best estimate shutdown temperature evaluation provided in Section 19E.4.10.2 has caused staff to question whether the treatment of uncertainty is adequate to demonstrate the safety-related design basis of the PRHR-HX having the capability to cool the RCS to 420 °F in 36 hours. Staff requests the following:

- *Provide the conservative assumptions used for the AP1000 Safe Shutdown Temperature Evaluation.*
- *If obtaining safe shutdown in 36 hours is a safety-related design requirement, update the FSAR with a conservative, design-basis analysis.*

- *If it is determined that obtaining safe shutdown in 36 hours is not a safety-related design requirement, provide justification and update the FSAR accordingly.*

Question 06.03-12
10 CFR Part 50, Appendix A, General Design Criteria 34 requires a system be provided with the safety function to transfer decay heat from the reactor core. In the AP1000 DCD, the passive residual heat removal heat exchanger (PRHR HX) is credited with performing this function in Chapters 6, 15, and 19.

In DCD Section 6.3.1.1, it is stated that for postulated non-LOCA events, "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 current submittal has raised questions about connotations associated with "indefinite operation" of the PRHR HX in this context.

DCD Section 6.3.4 states that "the passive core cooling system can maintain safe shutdown conditions for 72 hours after an event without operator action and without both nonsafety-related onsite and offsite power." Pursuant to staff guidance in Section 4.3.3.5 of the Utility Requirements Document and consistent with Regulatory Treatment of Non-Safety Systems (RTNSS) as discussed in the Section 19.3 of the Standard Review Plan, in order for the PRHR HX to meet the requirements of GDC 34, the system should have sufficient capacity to permit a minimum of 72 hours of operation without operator action following an accident. A preliminary analysis of the calculations available for staff audit indicate the system, with the proposed changes, appears to be capable of performing its safety function for substantially longer than 72 hours.

Staff seeks to clarify the intent of the phrase "indefinite operation" in the context of the proposed design change. Please provide, in an RAI response, a more detailed explanation on the intent and meaning of "indefinite operation" as it applies to the non-LOCA events.

Response Required By (Date): 5/15/2014

Impacts: None

Response Information

Response Information:

06.03-10 Response:

PRHR HX Safety-Related Mission Time: The **AP1000** plant meets the requirements of the Commission's regulations as well as the EPRI URD design requirements. The URD requires advanced light water reactor designs provide for mitigation of transients and accidents to meet the requirements of GDCs 34 and 44 for at least 72 hours without the need for operator action to replenish the water pools providing decay heat sink function (the passive containment cooling water storage tank or the in-containment refuelling water storage tank). The plant is equipped with diverse means of meeting this requirement – closed-loop cooling using the PRHR HX and open-loop cooling pairing automatic depressurization system (ADS) valve operation and IRWST injection.

- The preferred means of providing long-term residual heat removal following a non-LOCA event is closed-loop cooling using the PRHR HX. As described in DCD/UFSAR subsection 6.3.2.1.1, "Emergency Core Decay Heat Removal at High Pressure and Temperature Conditions," the PRHR HX provides emergency core decay heat removal for events not involving a loss of coolant. Closed-loop PRHR HX cooling is the primary means of meeting the Condition II success criteria: preventing fuel rod failures, preventing reactor coolant system (RCS) failures, and preventing secondary system overpressurization. Section 7.4 of the DCD/UFSAR notes the "safe shutdown conditions" for the RCS are those in which the reactor is subcritical, stable, and borated; RCS average temperature is less than or equal to no-load average temperature; and adequate RCS inventory and core cooling are established. The PRHR HX operates to bring the RCS to and maintain the RCS in an acceptable, stable condition for at least 72 hours after a non-LOCA event to allow ample time for decision-making and initiation of recovery actions. It is proposed subsection 6.3.1.1.1 of the plant-specific DCD/UFSAR, in which the passive core cooling system safety-related emergency core decay heat removal design bases are enumerated, be changed to clarify the safety-related mission time for the PRHR HX. This clarification would specify that, during this 72 hour time period, the applicable Chapter 15 design basis safety evaluation criteria would be met. Fulfilment of this design requirement will be demonstrated using conservative, design basis assumptions and conditions.
- PRHR HX Tube Plugging: A design change was implemented to reduce the allowable number of plugged tubes for the PRHR HX from that making up 8 percent of the heat transfer area to a number of tubes

making up 5 percent of the heat transfer area. This reduction in the allowable tube plugging was implemented in the revised analyses supporting the shutdown temperature evaluation presented in Appendix E of Chapter 19 of the submittal. The tube plugging reduction is not implemented in the Chapter 15 analyses. The PRHR performance modelled in the existing Chapter 15 analyses assumes 8 percent of the tubes are plugged. This tube plugging assumption is more conservative because plugging a larger portion of the tubes reduces heat removal by the heat exchanger, which presents greater challenge to meeting the Condition II event safety evaluation success criteria. It should be noted, however, that for those design basis events in which high PRHR HX capacity is conservative (i.e., 0% tubes plugged is more conservative for the main steam line break event) the analysis assumes all of the PRHR HX tubes are available for heat transfer.

- 72 Hour Mitigation of Accidents: As described in the first part of this response, the PXS is automatically actuated to bring the plant to a safe shutdown condition after a design basis event. The PRHR HX is automatically actuated to provide core decay heat removal to meet the Condition II success criteria during design basis accidents; and with limited operator action to prolong its operation, can maintain acceptable RCS conditions for at least 72 hours. These are safety-related design requirements.

  The duration the PXS can maintain a safe, stable condition is predicated on PCS water storage tank capacity, which is verified and tested during preoperational testing as described in DCD/UFSAR subsection 6.2.2.4.2 and in accordance with surveillance requirements of subsection 3.6.6 of the Technical Specifications. No licensing basis changes are required to reflect this information.

  It is proposed the UFSAR sections describing the PRHR HX capabilities be updated to provide a statement concerning the safety-related mission time of the PRHR HX. A description of the plant cooldown by the safety-related systems to maintain a safe, stable condition for at least 72 hours after a design basis event without actuation of the automatic depressurization system will be evaluated and discussed in the licensing basis. The duration the PRHR HX can maintain the Condition II success criteria is dependent on the efficiency of the condensate return features. The analysis will assume design basis conditions. In addition, it will be noted in the evaluation, that the event assumes operators have taken manual action to maintain the closed-loop cooling mode of operation beyond the automatic depressurization system actuation time. A description of the event and summary of the associated analysis will be incorporated into the design substantiation of DCD/UFSAR subsection 6.3.3.2.1 as a new subsection 6.3.3.2.1.1.

Changes to the licensing basis will be made to clarify the safety-related mission time of the PRHR HX in the closed loop mode of operation. The proposed changes are shown in Enclosure #.

06.03-11 Response:

The safe shutdown temperature evaluation is modelled on a best estimate basis with several conservatisms incorporated, which account for uncertainties. These conservatisms include not only conservative modelling assumptions; but conservatisms accounted for in development of the condensate return rate to the IRWST.

- The conservative assumptions used for the Shutdown Temperature Evaluation summarized in Chapter 19E include the following:
  - Condensate return rate
    - Testing uncertainty was accounted for in the development of the condensate return fraction by applying an efficiency across interferences that bounded the highest loss fractions observed by experiment.
  - The LOFTRAN analysis of the Loss of normal feedwater (LONF)/Loss of ac power (LOAC) event incorporated the following conservatisms:
    - The steam generator mass at the time of reactor trip is modelled conservatively. Bounding steam generator level trips were credited for the low steam generator narrow range and wide range level trips.
      - Using the conservative low level reactor trip setpoint results in a steam generator mass at the time of reactor trip that is only 42 percent of the initial mass at the beginning of the transient.
    - Heat transfer from the RCS thick metal to the containment atmosphere is not credited.
      - This eliminates the effect of this heat removal mechanism, which maximizes the PRHR HX heat input and IRWST steaming.

- Maximum initial fuel pellet average temperatures are modelled for the entire core.
  - Maximum core stored energy
- Uncertainties on initial RCS pressure and temperature.
  - Nominal full power initial reactor vessel average coolant temperature plus uncertainty.
  - Nominal initial pressurizer pressure minus uncertainty was modelled to minimize the energy release from the RCS through the pressurizer safety valves.
  - Maximum initial pressurizer water volume at full power plus uncertainty.
- PRHR HX and IRWST
  - Consistent with the Chapter 15 non-LOCA safety analysis, the vertical form of the Rohsenow heat transfer equation was used to determine the heat flux between the PRHR HX tube wall and the IRWST in nucleate boiling for the entire transient.
  - Maximum allowable number of PRHR HX tubes is assumed to be plugged.
  - The PRHR HX actuates on a conservatively low steam generator wide range level of approximately 13 percent of wide range span.
- The ambient air, which acts as the ultimate heat sink, is assumed to be a constant 115°F.
  - Natural temperature fluctuations of the ambient air reflecting night time and day time are not modelled.
- The initial containment temperature and the temperature of all the structures and components inside containment are assumed to be conservatively low when compared to the high ambient air temperature.
  - The lower containment temperature increases the amount of condensate formed on heat sinks, which does not return to the IRWST.

The capability to bring the plant to the specified long-term safe shutdown condition of 420°F within 36 hours after an event is considered a safety-related design requirement. However, it is considered reasonable to demonstrate the capability for the PRHR HX to perform this function using the thermal hydraulic analysis summarized in Chapter 19E, which is a non-bounding conservative analysis, for the following reasons:

- The PRHR HX is an extremely reliable, safety-related component.
  - The PRHR HX, its heat transfer capability and the materials of its construction are based on proven design.
  - The PRHR HX and the components supporting its operation are simple, single-failure resistant, and fail to their safe position when all power sources are lost.
  - The PRHR HX and the components supporting its operation are safety-related, designed to applicable nuclear safety codes and standards, and qualified to operate in a harsh environment.
  - The PRHR HX is automatically actuated. Two diverse systems can actuate the PRHR HX: the PMS and the diverse actuation system (DAS).
  - The PRHR HX can be initiated at full RCS design temperature and pressure. (RCS cooldown is not necessary for actuation of this feature.)
- The PRHR HX is backed up by a separate, diverse, safety-related residual heat removal system.
  - The ADS actuation combined with passive safety injection can bring the plant to the specified safe shutdown condition.
  - The ADS and passive safety injection use only safety-related components designed to applicable nuclear safety codes and standards, and qualified to operate in a harsh environment.
  - ADS / passive safety injection are automatically actuated by the PMS, and can be manually actuated by the DAS.
  - ADS / passive safety injection require only safety-related Class 1E dc power to assume their safe position.
  - The ADS / passive safety injection can be initiated at full RCS design temperature and pressure. (RCS cooldown is not necessary for actuation of this feature.)
- The probability the PRHR HX would be unable to perform the specified safe shutdown function, and open-loop cooling would be required to bring the plant to safe shutdown after a non-LOCA event is

remote – on the order of 1E-06 per reactor year – much lower than the frequency of other initiating events that require ADS actuation, such as small break LOCA.

- A loss of offsite power is the most frequent event that could challenge long-term PRHR HX operation. (The loss of normal feedwater with consequent loss of offsite power modelled in the shutdown temperature evaluation is less frequent.) A long-term loss of offsite power event would most likely be marked by the following:
  - Loss of offsite power (initiating event frequency of 1.20E-01 /year, DCD Rev. 19 PRA)
  - Failure of both diesel generators to operate (probability of 2.09E-02 /demand, DCD Rev. 19 PRA)
  - Failure to recover offsite power within 24 hours (probability of 1.79E−02 /demand, NUREG/CR-6890)
  - Decay heat as high as 2σ above nominal (probability of 2.3E-02 /demand, ANS-5.1-1979)
- In the unlikely event all of the conditions listed above occur, the PRHR HX would still successfully perform its safety mission, bringing the RCS to an acceptable, stable condition. In this scenario, the RCS would be stable with a temperature higher than 420°F. In this condition, the health and safety of the plant workers and the public will not be at risk. However, for the purpose of this evaluation, it is assumed the ADS could be actuated earlier than expected.
- This evaluation shows the probability the operators would actuate ADS to provide long-term core cooling due to PRHR HX performance that does not trend as expected is significantly smaller than for other sequences during which ADS would automatically be actuated. For example, the initiating event frequency for a small break LOCA, for which ADS actuation occurs as part of short-term accident mitigation, is 5.00E-04 /year (DCD Rev. 19 PRA).

- Establishing an RCS temperature of 420°F is not a prerequisite for maintaining acceptable, stable RCS conditions.
  - Maintaining a stable, post-accident condition with an RCS temperature higher than 420°F would not result in exceeding any of the safety evaluation criteria evaluated in the conservative Chapter 15 analyses.
    - The core would still be cooled.
    - RCS pressure would still be a small fraction of its design pressure.
    - Peak cladding temperatures, departure from nucleate boiling, and pressurizer level would be maintained within acceptable limits of the evaluation criteria.
  - The capability to maintain hot standby conditions for at least 8 hours after an event is a URD design requirement. A hot standby condition has also been acknowledged by the staff as a safe condition.
  - The Chapter 15 analyses demonstrate the plant is safe using fully conservative thermal hydraulic analyses. The limiting loss of main (normal) feedwater with consequential loss of ac power event was extended to 72 hours as described in the response to request for information 7440, question 15.02.06-2. This analysis demonstrates the Chapter 15 evaluation criteria are met with extended PRHR HX operation. A summary of this transient is added in a new subsection to subsection 6.3.3.2.1 of the licensing basis.

As stated in Section 7.4 of the DCD/UFSAR, the “safe shutdown conditions” for the RCS are those in which the reactor is subcritical, stable, and borated; RCS average temperature is less than or equal to no-load average temperature; and adequate RCS inventory and core cooling are established. The capability to bring the plant to a safe shutdown condition after an event is a safety-related design requirement. As discussed in the response to question 06.03-10, the **AP1000** plant is equipped with diverse, safety-related means of meeting this requirement.

The protection and safety monitoring system (PMS) provides for automatic actuation of the PRHR HX and the ADS to bring the plant to a safe shutdown condition within 36 hours after a design basis event. The automatically actuated means of achieving a safe, stable shutdown condition after an event uses closed-loop and open-loop cooling: PRHR HX operation followed by automatic depressurization and steam relief through the ADS valves combined with passive injection of the IRWST water inventory into the RCS. The open-loop cooling mode uses single failure-tolerant, safety-related components that are diverse from the closed-loop mode of cooling. The open-loop cooling mode can be initiated at full RCS design pressure and temperature, a significant design improvement as compared to existing plants. The open-loop mode of cooling is automatically actuated by the PMS after a design basis event without the need for operator action for the first 72 hours. This mode of cooling

can bring the plant to and maintain the plant in a safe shutdown condition indefinitely.

The PRHR HX is designed to provide more than 72 hours of core decay heat removal to meet the Condition II event success criteria in conjunction with the passive containment cooling system and to bring the plant to a safe shutdown condition without reliance on the ADS (closed-loop operation). Likewise, the closed-loop cooling mode uses single failure-tolerant, safety-related components, can be initiated at full RCS design pressure and temperature and can bring the plant to an acceptable, stable condition after a non-LOCA event. The closed-loop cooling mode using the PRHR HX can be maintained for more than 72 hours after an event to maintain the non-LOCA event success criteria and meet the Commission's regulations.

Closed-loop cooling using the PRHR HX is the preferred cooling mode for several reasons. Prolonging PRHR HX operation plays an important role in minimizing the need for ADS actuation. The frequency of ADS actuation is limited to a low probability to reduce safety risks and to minimize plant outages. Per DCD/UFSAR subsection 1.2.1.4.1, "Engineered Safeguards Systems Design," the probability of significant flooding of the containment due to ADS actuation is limited to once in 600 reactor-years. Prolonging the duration and capability to maintain PRHR HX operation minimizes potential for depressurization transients and the associated outage time.

However, maintaining this closed-loop mode of PRHR HX operation for 72 hours is not the automatic post-accident protective action that would be initiated by the PMS after a non-LOCA event. As explained in DCD/UFSAR subsection 15.0.13, the PRHR HX automatically establishes an acceptable, stable condition after a non-LOCA event, at which the point plant operator is expected to take manual control of the plant and proceed with an orderly shutdown. In the event a loss of ac power or other associated non-LOCA event continued beyond several hours, operator assessment and action would be required to prolong PRHR HX operation beyond the automatic ADS actuation time. The Class 1E dc batteries supplying motive power to the ADS valves supply power for at least 24 hours. Upon loss of ac power, the input voltage to the Class 1E battery chargers would be lost, which would initiate an ADS actuation countdown timer within the PMS to ensure ample Class 1E motive power is available for actuation of the ADS. (See the description of this timer function in DCD/UFSAR subsection 6.3.7.7, "Automatic Depressurization System Actuation at 24 Hours.") The post-event operator assessment and action is described in DCD/UFSAR subsection 7.4.1.1, "Safe Shutdown Using Safety-Related Systems" and in the response to request for information 7440, question 15.02.06-1, part b.

ADS actuation may become required during long-term PRHR HX operation to cope with RCS leakage or in the case of a rare, severe non-LOCA event (massive tornado, beyond safe shutdown earthquake) after which there is a prolonged delay in recovery of an ac power source. The emergency procedures are based on realistic analyses of the dynamic plant response; and direct operators to make an assessment of plant conditions to determine whether the conditions for preserving and extending closed-loop cooling have been met. The operators will not de-energize the 24-hour battery powered loads if the realistic plant response is not indicative of unambiguous progression toward a sustainable safe shutdown condition. Likewise, if the 24-hour battery loads have been de-energized in order to extend closed-loop PRHR HX operation, and plant conditions unexpectedly degrade, the operators will re-energize the 24 hour battery loads and the plant will transition to the open-loop cooling mode, which will maintain safe shutdown conditions indefinitely.

The design basis analyses of non-LOCA events are not realistic descriptions of the expected plant response. The shutdown temperature evaluation is likewise not a realistic description of the expected plant response. As demonstrated in the early part of this response, conservatisms were incorporated into to the safe shutdown temperature evaluation to account for uncertainties. Extended closed-loop PRHR HX operation to bring the plant to 420°F within 36 hours is modelled in a non-bounding, conservative analysis in order to demonstrate a more realistic plant, and operator, response.

Operation of the PRHR HX for 72 hours was modelled using design basis assumptions as described in the response to request for information 7440, question 15.02.06-2 to demonstrate the ability to meet the Condition II success criteria for non-LOCA events, and the capability of the PRHR HX to establish and maintain acceptable RCS conditions for an extended period of time such that recovery actions (including extending PRHR HX operation) can be initiated.

06.03-12 Response:

The PRHR HX is credited with transferring decay heat from the reactor core to satisfy the acceptance criteria for Condition II events. The passive containment cooling water storage tank (PCCWST) holds 72 hours' worth of water to cool the containment vessel and transfer energy to the ultimate heat sink (the ambient atmosphere) after a design basis event. With simple operator action to maintain that water supply – either from the permanently installed ancillary equipment, or from an offsite, pumped water supply – the PRHR HX can supply adequate core cooling for a significant duration beyond 72 hours.

Closed-loop cooling capacity is described in the current licensing basis as indefinite. That determination, as well as the design basis analyses and the safe shutdown temperature evaluation contained in the DCD/UFSAR, were based on the constant condensate return fraction. The indefinite operation was described in terms of the apparent capacity of this equipment in relation to the duration of the design basis analysis runs. The limiting design basis event with respect to challenging long-term the PRHR HX operation is the loss of ac power to the plant auxiliaries event. As described in DCD/UFSAR subsection 15.2.6.1, the transient analysis of this event was run for 36,000 seconds (10 hours). Evaluation of PRHR HX capability on this time scale showed performance levels indicative of cooling capacity far beyond that analyzed, upward of 40 days or longer. The capacity of the closed-loop cooling operation was characterized as indefinite based on those results.

As previously described, with operation action to extend its operation, the PRHR HX can maintain acceptable RCS conditions for at least 72 hours after a design basis event. The duration the PRHR HX can maintain the specified safe shutdown condition was determined using better estimate assumptions with added conservatism to account for uncertainties. This evaluation indicates the PRHR HX can maintain the RCS temperature below 420°F for significantly longer than 14 days following a non-LOCA event. Sustaining this condition is predicated on the engagement of post-72 hour equipment to maintain the passive containment cooling system water storage tank water inventory beyond that provided by the 72-hour supply of water already in the tank.

Describing the duration of closed loop as indefinite can lead to potential misunderstandings. As a result, the licensing basis will be clarified to incorporate a quantitative basis for PRHR HX capacity. A capacity of 14 days of closed-loop cooling operation is considered adequate based on the following insights and operating experience:

- Condensate return to the IRWST may not be the limiting factor in the capacity for closed-loop operation. This mode of operation could be limited due to RCS leakage.
- The probability of a station blackout lasting longer than 14 days is remote. On average, offsite power is restored in about 2.5 hours after a loss of offsite power event. The duration of loss of offsite power events experienced by nuclear power plants in the United States range from a few hours to 5 days.
- The events that might result in extended loss of ac power are associated with severe, and likely beyond design basis, events, such as massive earthquakes or tornadoes. As an example, at Fukushima Daiichi, offsite power was restored in 12 days.

The capability to provide at least 14 days of closed-loop cooling provides a sufficiently long period to allow recovery of ac power and the defense-in-depth equipment needed to transition to cold shutdown after all but the rarest and most severe non-LOCA events. In the event ac power cannot be restored in this time, the plant can still safely transition to open-loop cooling.

The capacity to maintain closed-loop PRHR HX cooling for a period of greater than or equal to 14 days would exceed the regulatory requirements and expectations for long-term, closed-loop cooling. The **AP1000** plant is required to perform its safety functions for 72 hours after a design basis event without reliance on non-safety related systems, structures or components (SSCs) for support. The PRHR HX is adequate to perform this safety mission. Additionally, open-loop cooling is available to maintain acceptable RCS conditions indefinitely and independently of closed-loop cooling via the PRHR HX. Therefore the capability for the PRHR HX to maintain safe shutdown conditions for more than 14 days in a closed-loop mode of operation is considered a non-safety related design goal. This capability maintains the design goals for the probability for significant containment floodup (see the response to 06.03-11) and serves as investment protection by minimizing the probability of significant post-event outage time.

Although the capacity of the closed-loop mode of operation is not considered indefinite, the capability of the PXS to maintain adequate safe shutdown conditions indefinitely is preserved because, as explained in the response to question 06.03-11, the open-loop cooling mode of operation can be manually actuated at any time by the operators and is a safety-related means of providing core cooling.

Response Prepared By: Sylena E. Smith	
Response Preparer Signature: Electronically approved***	**Date:** see EDMS
Reviewed By: Terry L. Schulz	
Response Reviewer Signature:	**Date:** see EDMS
Verified By: Fletcher T. Wilkins	
Response Verifier Signature:	**Date:** see EDMS
Licensing Review By: Richard A. DeLong	

Licensing Reviewer Signature: Electronically approved***		**Date:** see EDMS
Approved By: John A. Green for Hang Yuan	**Response Approver Phone Number:** 412-374-5031	
Approver Signature: Electronically approved***		**Date:** see EDMS

***Electronically approved records are authenticated in the electronic document management system.

5.4.14.1 Design Bases

The passive residual heat removal heat exchanger automatically actuates to remove core decay heat for an extended period of time as discussed in Section 6.3, assuming the condensate from steam generated in the in-containment refueling water storage tank (IRWST) is returned to the tank. 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. In addition, the passive residual heat removal heat exchanger will cool the reactor coolant system, with reactor coolant pumps operating or in the natural circulation mode, so that the reactor coolant system pressure can be lowered to reduce stress levels in the system if required. See Section 6.3 for a discussion of the capability of the passive core cooling system.

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.

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 if normal systems are not recovered for many days (refer to section 6.3.1.1.4).

- Operator action may be taken in accordance with emergency procedures to de-energize the loads on the Class 1E batteries to avoid automatic actuation of the automatic depressurization system when unnecessary. The passive residual heat removal heat exchanger is designed to maintain acceptable reactor coolant system conditions for more than 72 hours following a non-LOCA event after such action. The applicable post-accident safety evaluation criteria are discussed in Chapter 15.

- The passive residual heat removal heat exchanger is capable of performing its post-accident safety functions, assuming the steam generated in the in-containment refueling water storage tank is condensed on the containment vessel and returned by gravity via the in-containment refueling water storage tank condensate return gutter and downspouts.

- 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.

6.3.1.1.4 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, the passive core cooling system, in conjunction with the passive containment cooling system, has diverse capability to establish long-term safe shutdown conditions in the reactor coolant system, eventually 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 acutated on low reactor coolant temperature or low pressurizer pressure or level. The passive core cooling system can maintain stable plant conditions for a long time in this mode of operation, depending on the reactor coolant leakage and the availability of ac power sources. For example, with a technical specification leak rate of 10 gpm, stable plant conditions can be maintained for at least 10 hours. With a smaller leak a longer time is available.

In most sequences the operators would return the plant to normal system operations and terminate passive system operation 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 establish safe 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 passive residual heat removal heat exchanger operation is not extended or is exhausted, the automatic depressurization system will be initiated. This results in injection from the accumulators and subsequently from the in-containment refueling water storage tank, once the reactor coolant system is nearly depressurized. For these conditions, the reactor coolant system depressurizes to saturated conditions at about 250°F within 24 hours. The passive core cooling system can maintain this safe shutdown condition indefinitely for the plant.

The basis used to define the passive core cooling system functional requirements is derived from Section 7.4 of the Standard Review Plan. The functional requirements are met over the range of anticipated events and single failure assumptions. The primary function of the passive core cooling system during a safe shutdown using only safety-related equipment is to provide a means for boration, injection, and core cooling. Details of the safe shutdown design bases are presented in 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.

6.3.1.2 Nonsafety Design Basis

6.3.1.2.1 Post-Accident Core Decay Heat Removal

The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, condensate return features and the passive containment cooling system, has the capability to maintain the reactor coolant system in a safe shutdown condition for 14 days in a closed-loop mode of operation. Actuation of the automatic depressurization system can be manually initiated 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 are detailed in subsection 7.4.1.1.

6.3.1.3 Power Generation Design Basis

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

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

The passive residual heat removal heat exchanger is used to maintain an acceptable, stable reactor coolant system condition. It transfers decay heat and sensible heat from the reactor coolant system to the in-containment refueling water storage tank, the containment atmosphere, the containment vessel, and finally to the ultimate heat sink–the atmosphere outside of containment. 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. If offsite ac power sources are not recovered after several hours, 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.

6.3.2.8 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.

Delaying actuation of the automatic depressurization system when it is not needed is another manual operator action that can reduce the consequences of an event. 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. However, the operators can take action to preserve passive residual heat removal heat exchanger operation beyond approximately 22 hours and maintain that mode of cooling for as long as acceptable reactor coolant system conditions can be maintained. This action reduces the probability of containment floodup and the resulting recovery activities.

Section 7.4 describes the anticipated operator actions for mitigation of a design basis event. Section 7.5 describes the post-accident monitoring instrumentation available to the operator in the main control room following an event.

The following, highlighted text will be added to subsection 6.3.3, "Performance Evaluation."

B. Decrease in heat removal by the secondary system

1. Loss of Main Feedwater Flow

2. Feedwater system piping failure

…

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 II safety evaluation criteria described in the relevant safety analyses. Subsection 6.3.3.2.1 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 to cooling the reactor coolant system to safe shutdown as described in subsection 19E.4.10.2. A conservative estimation 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, with some conservatism to account for uncertainties. This operator-controlled mode of operation provides a separate, diverse means of achieving safe shutdown conditions. 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 to low levels. This also allows plant conditions to be established for initiation of normal residual heat removal system operation.

…

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 required to be able to bring the plant to a safe shutdown condition and maintain that condition.

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 Main Feedwater with Extended Passive Residual Heat Removal Heat Exchanger Operation

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 delay its actuation. However, 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 long-term loss of main feedwater with consequential 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 main feedwater event described in subsection 6.3.3.2.1, but the loss of ac power extends to 72 hours. Operation of the passive residual heat removal heat exchanger continues beyond 72 hours to maintain acceptable reactor coolant system conditions such that the applicable Chapter 15 safety evaluation criteria are met.

A reactor coolant system leak could produce conditions that would preclude the operators from de-energizing the loads on the Class 1E dc batteries. If the reactor coolant system conditions were to degrade, the operators would re-energize the buses powered by the Class 1E batteries 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.

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

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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 function as a heat sink.

7.4.1.1 Safe Shutdown Using Safety-Related Systems

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 extended operation of the passive residual heat removal heat exchanger and the in-containment refueling water storage tank.

…

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 provides core decay heat removal in this configuration without an 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 an acceptable, stable reactor coolant system condition during a design basis event for at least 72 hours in a closed-loop mode of operation. A non-bounding, conservative analysis of extended operation in this mode shows the passive residual heat removal heat exchanger cools the reactor coolant system to 420°F in 36 hours.

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 pressurizer level. 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.

The following change would be made on Sheet 11 of Table 9.5.1-1, "AP1000 Fire Protection Program Compliance with BTP CMEB 9.5-1":

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

15.0.13 Operator Actions

For events where the PRHR heat exchanger is actuated, the plant automatically cools down to a safe, stable 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.

15.2.6.1 Operator Actions

…

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 reactor coolant system conditions to satisfy the evaluation criteria. 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.

The following change would be made on Sheet 6 of Table 19.59-18, “PRA-Based Insights”:

… The PRHR HX, in conjunction with the IRWST, condensate return features and the PCS, can provide core cooling for more than 72 hours. After the IRWST water reaches its saturation temperature, the process of steaming to the containment initiates…	6.3.2.1.1 & 6.3.7.6

19E.4.10.2 Shutdown Temperature Evaluation

As discussed in Section 6.3.1.1.4, the passive residual heat removal heat exchanger is required to be able to cool the reactor coolant system to 420°F or below within 36 hours 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. This analysis demonstrates that the passive systems can bring the plant to a stable safe condition and maintain this condition so that no transients will result in the specified acceptable fuel design limit and pressure boundary design limit being violated and that no high-energy piping failure being initiated from this condition results in 10 CFR 50.46 (Reference 15) criteria.

As discussed in 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 event. An analysis of the loss of main feedwater with consequential loss of ac power event demonstrates that the passive systems can bring the plant to a stable safe condition following postulated transients. A non-bounding, conservative analysis is represented in Figures 19E.4.10-1 through 19E.4.10-4. The progression of this event is outlined in Table 19E.4.10-1. Though some of the assumptions in this evaluation are based on nominal conditions, many of the analysis assumptions are bounding. Therefore, this summary represents a non-bounding, conservative evaluation of the plant response.

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 time-dependent condensate return rate was incorporated into the LOFTRAN computer code described in subsection 15.0.11.2 to demonstrate that the RCS could be cooled to 420°F within 36 hours.

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

Once actuated, at about 2,400 seconds, the CMTs operate in recirculation mode, injecting cold borated water into the RCS. In the first part of their operation, due to the injection of cold water, the CMTs operate in conjunction with the PRHR HX to reduce RCS temperature. Due to the primary system cooldown, the PRHR heat transfer capability drops below the decay heat and the RCS cooldown is essentially driven by the CMT cold injection flow. However, at about 5,000 seconds, the CMT cooling effect decreases and the RCS starts heating up again (Figure 19.E.4.10-1). The RCS temperature increases until the PRHR HX can match decay heat. At about 34,500 seconds, the PRHR heat transfer matches decay heat and it continues to operate to reduce the RCS temperature to below 420°F within 36 hours. As seen from Figure 19E.4.10-1, the cold leg temperature in the loop with the PRHR is reduced to 420°F within 48,600 seconds, while the core average temperature reaches 420°F within 124,400 seconds (approximately 34.6 hours).

As discussed in subsection 7.4.1.1, a timer is used to automatically actuate the automatic depressurization system if offsite and onsite power are lost for about 24 hours. This timer automates putting the open loop cooling features into service prior to draining the Class 1E dc 24-hour batteries that operate the ADS valves. At approximately 22 hours, if the plant conditions indicate that the ADS would not be needed until well after 24 hours, the operators are directed to de-energize all loads on the 24-hour batteries. This action will block actuation of the ADS and preserves the ability to align open loop cooling at a later time. Operation of the ADS in conjunction with the CMTs, accumulators, and IRWST reduces the RCS pressure and temperature to below 420°F. The ability to actuate ADS and IRWST injection provides a safety-related, backup mode of decay heat removal that is diverse to extended PRHR HX operation.

As discussed in subsection 6.3.3.2.3, the PRHR HX can operate in this mode for at least 72 hours to maintain RCS conditions within the applicable Chapter 15 safety evaluation criteria. In addition, the analysis supporting this section shows the PRHR HX is expected to maintain safe shutdown conditions for more than 14 days. One important consideration with regard to the duration closed-loop cooling can be maintained is the RCS leak rate. This duration of closed-loop cooing 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.