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

LEVY NUCLEAR PLANT, UNITS 1 AND 2
DOCKET NOS. 52-029 AND 52-030
SUPPLEMENT TO PARTIAL RESPONSE TO NRC RAI LETTERS 116, 117 and 118 – SRP SECTIONS 6.3 AND 15.2.6

References:
1. Letter from Donald Habib (NRC) to Christopher M. Fallon (DEF), dated March 6, 2014, "Request for Additional Information Letter No. 116 Related to SRP Sections 6.3 and 15.2.6."
2. Letter from Donald Habib (NRC) to Christopher M. Fallon (DEF), dated April 10, 2014, "Request for Additional Information Letter No. 117 Related to SRP Section 6.3."
3. Letter from Donald Habib (NRC) to Christopher M. Fallon (DEF), dated April 24, 2014, "Request for Additional Information Letter No. 118 Related to SRP Section 6.3."
4. Letter from Christopher M. Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated June 27, 2014, "Partial Response to NRC RAI Letters 116, 117 and 118 – SRP Sections 6.3 and 15.2.6," Serial: NPD-NRC-2014-021

Ladies and Gentlemen:

Duke Energy Florida, Inc. (DEF) hereby submits a supplemental response to the Nuclear Regulatory Commission's (NRC) request for additional information (RAI) cited in References 1, 2 and 3.

In a public teleconference with the NRC staff held on July 10, 2014, the staff described perceived inconsistencies in the proposed licensing basis markups in Enclosures 4 and 5 of Reference 4. For example, the proposed markup of UFSAR subsection 6.3.1.1.4 states, "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." However, while Section 6.3.1.1 enumerates the safety design basis of the **AP1000** plant, the analysis showing the PRHR HX can bring the plant to 420°F in 36 hours is not a design basis analysis. The staff requires clear separation of the design basis claims from non-design basis claims; and is of the opinion that the words "to about 420°F in 36 hours" retained in subsection 6.3.1.1.4 misrepresent the design basis capabilities of the **AP1000** plant. The NRC staff requested Duke Energy to provide a licensing basis markup that resolves this inconsistency, or provide justification that the existing licensing basis markup is fit for its intended purpose.

Enclosure 4 contains the proposed changes to the AP1000 DCD Tier 2 licensing basis associated with the responses to RAI Questions 06.03-10, 06.03-11 and 06.03-12 which have been revised to resolve the inconsistencies identified by the NRC. Enclosure 5 contains the

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revised Levy Nuclear Plant Part 2 and Part 4 COL application (COLA) revisions based on the Enclosure 4 DCD changes, which will be included in a future update of the COLA.

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

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

Executed on July 24, 2014

Sincerely,

Christopher M. Fallon
Vice President
Nuclear Development

Enclosures/Attachments:

1. Levy Nuclear Power Plant Units 1 and 2 Response to NRC Request for Additional Information Letters No. 116 Related to SRP Sections 06.03 and 15.02.06 for the Combined License Application, Dated March 6, 2014, No. 117 Related to SRP Section 06.03, Dated April 10, 2014, and No. 118 Related to SRP Section 06.03, dated April 24, 2014
   A. Responses to NRC RAIs 06.03-5 and 15.02.06-2 on Condensate Return Licensing Submittal (Proprietary)
   B. Responses to NRC RAIs 06.03-5 and 15.02.06-2 on Condensate Return Licensing Submittal (Nonproprietary)
2. Westinghouse Application Letter CAW-14-3961 and Affidavit
3. Proprietary Information Notice and Copyright Notice
4. AP1000 DCD Tier 2 Licensing Basis Document – Proposed Changes
5. Levy Nuclear Plant Units 1 and 2 - Part 2 and Part 4 COL Application Revisions

cc: U.S. NRC Region II, Regional Administrator
Mr. Donald Habib, U.S. NRC Project Manager

Enclosure 1
Levy Nuclear Power Plant Units 1 and 2 Response to NRC Request for Additional Information Letters No. 116 Related to SRP Sections 06.03 and 15.02.06 for the Combined License Application, Dated March 6, 2014, No. 117 Related to SRP Section 06.03, Dated April 10, 2014, and No. 118 Related to SRP Section 06.03, dated April 24, 2014 (See Reference 4 for this Enclosure)

Enclosure 2
Westinghouse Application Letter CAW-14-3961 and Affidavit
(See Reference 4 for this Enclosure)

Enclosure 3
Proprietary Information Notice and Copyright Notice
(See Reference 4 for this Enclosure)

Duke Energy
Enclosure 4
Levy Nuclear Plant Units 1 and 2

AP1000 DCD Tier 2 Licensing Basis Document - Proposed Changes
(14 pages including cover page)

AP1000 DCD Tier 2 Licensing Basis – Proposed Changes:

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.

Chapter 6 Table of Contents (actual page number changes are not shown)

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.

- The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, condensate collection features and the passive containment cooling system, are designed to remove decay heat following a design basis event. Automatic depressurization actuation is not expected; but may occur depending on the amount of reactor coolant system leakage and when normal systems are recovered (refer to subsection 6.3.1.1.4).

- The passive residual heat removal heat exchanger is designed to maintain acceptable reactor coolant system conditions for at least 72 hours following a non-LOCA event. The applicable post-accident safety evaluation criteria are discussed in Chapter 15. Operator action may be taken in accordance with emergency procedures to de-energize the loads on the Class 1E batteries to avoid unnecessary automatic actuation of the automatic depressurization system. Specific safe shutdown criteria are described in subsection 6.3.1.1.4.

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

- 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 and the automatic depressurization system, has the capability to establish long-term safe shutdown conditions in the reactor coolant system.

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

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

For loss of coolant accidents, 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.

6.3.1.1.6 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 6.3.1.3 includes specific nonsafety-related design requirements that help to confirm satisfactory system reliability.

6.3.1.2 Nonsafety Design Basis

6.3.1.2.1 Long-Term Core Decay Heat Removal

The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, the condensate return features and the passive containment cooling system, has the capability to maintain the reactor coolant system in the specified, long-term safe shutdown condition of 420°F for 14 days in a closed-loop mode of operation. The automatic depressurization system can be manually actuated by the operators at any time during extended passive residual heat removal heat exchanger operation to initiate open-loop cooling. The operator actions necessary to achieve safe shutdown using the passive residual heat removal heat exchanger in a closed-loop mode of operation involve preventing unnecessary actuation of the automatic depressurization system as detailed in 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 at least 72 hours. After the in-containment refueling water storage tank water reaches its saturation temperature (in several hours), the process of steaming to the containment initiates. Containment pressure will increase as steam is released from the in-containment refueling water storage tank. As the containment temperature increases, condensation begins to form on the subcooled metal and concrete surfaces inside containment. Condensation on these heat sink surfaces transfers energy to the bulk metal and concrete until they come into equilibrium with the containment atmosphere. Condensation that is not returned to the in-containment refueling water storage tank drains to the containment sump.

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

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. 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.

6.3.2.8 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.

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

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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.

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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.

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.

6.3.3.4.1 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.

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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 with a 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 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, a high and stable pressurizer level, and decreasing or stable reactor coolant system temperature. If automatic depressurization is not needed, the operator is directed to de-energize all loads on the Class 1E dc batteries. This action preserves the capability for the operator to initiate automatic depressurization at a later time based on assessment of the same parameters.

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 Identification of Causes and Accident Description

...

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" and in the corresponding table of the **AP1000** Probabilistic Risk Assessment (APP-GW-GL-022 rev 8):

...	

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

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 analysis demonstrates that the passive systems can bring the plant to a safe, stable condition and maintain this condition so that no transients will result in the specified acceptable fuel design limit and pressure boundary design limit being violated and that no high-energy piping failure being initiated from this condition results in 10 CFR 50.46 (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 specified safe shutdown condition following a non-LOCA event. An analysis of the loss of main feedwater with loss of ac power event demonstrates that the passive systems can bring the plant to this condition following postulated transients. A non-bounding, conservative analysis is represented in 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.

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

19E.9 References

14. Not used.

Enclosure 5
Duke Energy
Levy Nuclear Plant Units 1 and 2
Part 2 and Part 4 COL Application Revisions
(31 pages including cover page)

Associated LNP COL Application Revisions:

The following revisions to the LNP COL application represent an integrated list of revisions based on revisions identified in Enclosure 5 of Serial: NPD-NRC-2014-012 and Enclosure 1 of Serial: NPD-NRC-2014-017 as well as revisions identified in this letter. These revisions will be incorporated into the next update of the LNP COLA.

COLA Part 2, FSAR

COLA Part 2, FSAR Chapter 1, Table 1.8-201, Summary of FSAR Departures from the DCD, will be revised to add additional FSAR changes to the list of FSAR Section or Subsection references for departure LNP COL 3.2-1, to read as follows:

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

		Table 19E.4.10-201, Figures 19E.4.10-201 through 19E.4.10-204, 19E.9

COLA Part 2, FSAR Chapter 1, Table 1.8-201, Summary of FSAR Departures from the DCD, will be revised to add FSAR changes to the list of FSAR Section or Subsection references for departure LNP COL 6.3-1, to read as follows:

Departure Number	Departure Description Summary	FSAR Section or Subsection
LNP DEP 6.3-1	The DCD states that the PRHR HX can maintain safe shutdown conditions for non-LOCA accidents "indefinitely." A quantitative duration of greater than 14 days has been adopted based on that time being long enough to minimize the need to switch to passive feed and bleed cooling except for very unlikely or extreme hazard events. The following are the departures from the DCD: Subsection 5.4.14.1, Subsections 6.3.1.1.1, 6.3.1.2, 6.3.1.3, 6.3.2.1.1, 6.3.3.4.1, Subsection 7.4.1.1, Table 9.5.1-1 (Sheet 11), Subsection 15.2.6.1, Table 19.59-18 (Sheet 6), Subsection 19E.4.10.2	5.4.14.1, 6.3.1.1.1, 6.3.1.2, 6.3.1.3, 6.3.2.1.1, 6.3.3.4.1, 7.4.1.1, Table 9.5.1-201, 15.2.6.1, Table 19.59-202, 19E.4.10.2

COLA Part 2, FSAR Sections 5.4, 6.3, 7.4, 9.5, 14.3, Chapter 15, Chapter 16 and Chapter 19 will be revised to add the departures identified in Table 1.8-201 with a LMA of LNP DEP 3.2-1 or 6.3-1, as presented below.

1. COLA Part 2, FSAR Chapter 5, will be revised to add new Subsection 5.4.11.2, with a LMA of LNP DEP 3.2-1, to read:

   5.4.11.2 System Description

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

   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 6.3-1.

2. COLA Part 2, FSAR Chapter 5, will be revised to add new Subsection 5.4.14.1 to read:

   5.4.14.1 Design Bases

   Replace the first sentence of the first paragraph of DCD Subsection 5.4.14.1 with the following, with a LMA of LNP DEP 6.3-1:

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

   Combine the second and third paragraphs of DCD Subsection 5.4.14.1 and revise to read as follows, with LMAs of LNP DEP 3.2-1 and LNP DEP 6.3-1:

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

3. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.1.1.1, with a LMA of LNP DEP 3.2-1, to read:

   6.3.1.1.1 Emergency Core Decay Heat Removal

   Add new second and third bullets in the first paragraph of DCD Subsection 6.3.1.1.1 to read as follows:

   - 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.

Replace the fourth bullet (old second bullet) in the first paragraph of DCD Subsection 6.3.1.1.1 with the following:

- 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.

4. Continue to revise the first paragraph of DCD Subsection 6.3.1.1.1 by deleting entirely the fifth bullet (old third bullet). Show as "(Deleted - new fifth bullet (old third bullet))" with LMAs of LNP DEP 3.2-1 and 6.3-1.

5. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.1.1.4, with a LMA of LNP DEP 3.2-1, to read:

   6.3.1.1.4 Safe Shutdown

   Replace the first two paragraphs of DCD Subsection 6.3.1.1.4 with the following three paragraphs, to read:

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

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

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

Replace the first sentence of the fifth paragraph (old fourth paragraph) of DCD Subsection 6.3.1.1.4 with the following:

The basis used to define the passive core cooling system functional requirements is derived from Section 7.4 of the Standard Review Plan.

Add a last sentence to the fifth paragraph (old fourth paragraph) of DCD Subsection 6.3.1.1.4, to read as follows:

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. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.1.1.6, with a LMA of LNP DEP 3.2-1, to read:

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

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

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

   6.3.1.2 Nonsafety Design Basis

   6.3.1.2.1 Long-Term Core Decay Heat Removal

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

8. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.1.3, title only, to reflect the numbering change of DCD Subsection 6.3.1.2 to 6.3.1.3, with LMAs of LNP DEP 3.2-1 and LNP DEP 6.3-1, to read as follows:

   6.3.1.3 Power Generation Design Basis

9. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.1, with a LMA of LNP DEP 3.2-1, to read:

   6.3.2.1 Schematic Piping and Instrumentation Diagrams

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

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

10. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.1.1 to read:

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

   Replace the seventh and eighth paragraphs of DCD Subsection 6.3.2.1.1 with the following, with LMAs of LNP DEP 3.2-1 and LNP DEP 6.3-1:

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

Condensation occurs on the steel containment vessel, which is cooled by the passive containment cooling system. Most of the condensate formed on the containment vessel wall is collected in a safety-related gutter arrangement. A gutter is located near the operating deck elevation, and a downspout piping system is connected at the polar crane girder and internal stiffener, to collect steam condensate inside the containment during passive containment cooling system operation and return it to the in-containment refueling water storage tank. The gutter and downspouts normally drain to the containment sump, but when the passive residual heat removal heat exchanger actuates, safety-related isolation valves in the gutter drain line shut and the gutter overflow returns directly to the in-containment refueling water storage tank. Recovery of the condensate maintains the passive residual heat removal heat exchanger heat sink for an extended period of time. Revise the first and second sentences of the ninth paragraph of DCD Subsection 6.3.2.1.1 to read as follows, with a LMA of LNP DEP 3.2-1:

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

Add a new tenth paragraph to DCD Subsection 6.3.2.1.1 to read as follows, with a LMA of LNP DEP 3.2-1:

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

11. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.2.7, with a LMA of LNP DEP 3.2-1, to read:

6.3.2.2.7 IRWST and Containment Recirculation Screens
Replace the first paragraph of DCD Subsection 6.3.2.2.7 with the following:

The passive core cooling system has two different sets of screens that are used to prevent debris from entering the reactor and blocking core cooling passages during a LOCA: IRWST screens and containment recirculation screens. The screens are AP1000 Equipment Class C and are designed to meet seismic Category I requirements. The structural 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 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

12. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.2.7.1, with a LMA of LNP DEP 3.2-1, to read:

    6.3.2.2.7.1 General Screen Design Criteria

    Replace the first paragraph of DCD Subsection 6.3.2.2.7.1 with the following:

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

13. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.2.7.2, with a LMA of LNP DEP 3.2-1, to read:

    6.3.2.2.7.2 IRWST Screens

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

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

14. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.8, with a LMA of LNP DEP 3.2-1, to read:

    6.3.2.8 Manual Actions

Add a new third paragraph of DCD Subsection 6.3.2.8 to read as follows:

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.

Add a new first sentence to the fourth paragraph (old third paragraph) of DCD Subsection 6.3.2.8, to read as follows:

Section 7.4 describes the anticipated operator actions to block unnecessary automatic depressurization system actuation.

15. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.3, with a LMA of LNP DEP 3.2-1, to read:

6.3.3 Performance Evaluation

Replace the seventh paragraph of DCD Subsection 6.3.3 with the following:

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.

Add a new eighth paragraph to DCD Subsection 6.3.3, as follows:

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.

Add the following as the last sentence to the tenth paragraph (old ninth paragraph) of DCD Subsection 6.3.3, as follows:

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.

Add a new eleventh paragraph to DCD Subsection 6.3.3, as follows:

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 bringing the plant to a safe shutdown condition and maintaining that condition.

16. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.3.2.1.1 (new DCD Subsection 6.3.3.2.1.1), with a LMA of LNP DEP 3.2-1, to read:

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.

17. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.3.4.1, with a LMA of LNP DEP 6.3-1, to read:

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

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

    This allows it to function as a heat sink.

18. COLA Part 2, FSAR Section 6.3 will be revised to add a departure from DCD Figure 6.3-1 as Figure 6.3-201, as shown in Sheets 1 through 3 of Figure 6.3-201 in the attachment to this enclosure, with a LMA of LNP DEP 3.2-1. These sheets replace the figure added as Figure 6.3-201 in LNP COLA Revision 6.

19. COLA Part 2, FSAR Chapter 7, will be revised to add new Subsection 7.4.1.1, to read:

    This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

    7.4.1.1 Safe Shutdown Using Safety-Related Systems

    Revise the second sentence of the sixth paragraph of DCD Subsection 7.4.1.1 to read as follows, with a LMA of LNP DEP 6.3-1:

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

Revise the last sentence of the eighth paragraph of DCD Subsection 7.4.1.1 to read as follows, with LMAs of LNP DEP 3.2-1 and LNP DEP 6.3-1:

The system provides core decay heat removal in this configuration with a limited increase in the containment water level.

Revise the ninth paragraph of DCD Subsection 7.4.1.1 to read as follows, with a LMA of LNP DEP 3.2-1:

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

Revise the last three sentences of the eleventh paragraph of DCD Subsection 7.4.1.1 to read as follows, with a LMA of LNP DEP 3.2-1:

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

20. COLA Part 2, FSAR Section 9.5 will be revised to add a departure from DCD Table 9.5.1-1, AP1000 Fire Protection Program Compliance with BTP CMEB 9.5-1, Sheet 11 of 33, as new FSAR Table 9.5.1-201, Sheet 1, with a LMA of LNP DEP 6.3-1. This Table shall also be added to the list of tables in Chapter 9. Table 9.5.1-201, Sheet 1, is shown in the attachment to this enclosure.

21. COLA Part 2, FSAR Section 14.3 will be revised to add a departure from DCD Table 14.3-2, Design Basis Accident Analysis, Sheets 7 and 8 of 17, as new FSAR Table 14.3-202, Sheets 1 and 2, with a LMA of LNP DEP 3.2-1. This Table shall also be added to the list of tables in Chapter 14. Table 14.3-202, Sheets 1 and 2, are shown in the attachment to this enclosure.

22. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.0.13, with a LMA of LNP DEP 3.2-1, to read:

15.0.13 Operator Actions

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

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

23. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.2.6.1, with a LMA of LNP DEP 6.3-1, to read:

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

15.2.6.1 Identification of Causes and Accident Description

Revise the seventh sentence of the fourth paragraph of DCD Subsection 15.2.6.1 to read as follows:

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

24. COLA Part 2, FSAR Section 19.59 will be revised to add a departure from DCD Table 19.59-18, PRA Based Insights, Sheet 6 of 25, as new FSAR Table 19.59-202, Sheet 1, with a LMA of LNP DEP 6.3-1. This Table shall also be added to the list of tables in Chapter 19. Table 19.59-202, Sheet 1, is shown in the attachment to this enclosure.

25. COLA Part 2, FSAR Chapter 19, Appendix 19E Shutdown Evaluation, will be revised as follows, with a LMA of LNP DEP 3.2-1:

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

19E.4.10.2 Shutdown Temperature Evaluation

Revise the first and second paragraphs of DCD Subsection 19E.4.10.2 to read as follows:

As discussed in subsection 6.3.1.1.4, the passive residual heat removal heat exchanger is required to be able to cool the reactor coolant system to a safe, stable condition after shutdown following a non-LOCA event. The following summarizes a non-bounding, conservative analysis, which demonstrates the passive residual heat removal heat exchanger can meet this criterion and cool the RCS to the specified, safe shutdown condition of 420°F within 36 hours. This analysis demonstrates that the passive systems can bring the plant to a safe, stable condition and maintain this condition so that no transients will result in the specified acceptable fuel design limit and pressure boundary design limit being violated and that no high-energy piping failure being initiated from this condition results in 10 CFR 50.46 (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 specified safe shutdown condition following a non-LOCA event. An analysis of the loss of main feedwater with loss of ac power event demonstrates that the passive systems can bring the plant to this condition following postulated transients. A non-bounding, conservative analysis is represented in 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.

Add new paragraphs 3 and 4 to DCD Subsection 19E.4.10.2 to read as follows:

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.

Revise the first sentence of the fifth paragraph (old third paragraph) of DCD Subsection 19E.4.10.2 to read as follows:

Summarizing this transient, the loss of normal ac power (offsite and onsite) occurs, followed by the reactor trip.

26. COLA Part 2, FSAR Chapter 19, Appendix 19E Shutdown Evaluation, will continue to be revised as follows, with a LMA of LNP DEP 3.2-1:

Revise paragraphs 6 and 7 (old paragraphs 4 and 5) of DCD Subsection 19E.4.10.2 to read as follows:

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.

27. COLA Part 2, FSAR Chapter 19, Appendix 19E, will continue to be revised as follows, with LMAs of LNP DEP 3.2-1 and LNP DEP 6.3-1:

Add a new eighth paragraph to DCD Subsection 19E.4.10.2 to read as follows:

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 more than 14 days. One important consideration with regard to the duration closed-loop cooling can be maintained is the RCS leak rate. This duration of closed-loop cooling can be achieved with expected RCS leak rates. For abnormal leak rates, it may become necessary to initiate open-loop cooling earlier than 14 days.

28. COLA Part 2, FSAR Section 19E.4.10 will be revised to add a departure from DCD 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, as new FSAR Table 19E.4.10-201, with a LMA of LNP DEP 3.2-1. This Table shall also be added to the list of tables for Chapter 19. Table 19E.4.10-201 is shown in the attachment to this enclosure.

29. COLA Part 2, FSAR Chapter 19 will be revised to add a new Subsection 19E.9, with a LMA of LNP DEP 3.2-1, to read:

19E.9 References

14. Not used.

30. COLA Part 2, FSAR Section 19E will be revised to add a departure from DCD Figures 19E.4.10-1 through 19E.4.10-4 as Figures 19E.4.10-201 through 19E.4.10-204, with a LMA of LNP DEP 3.2-1. These figures shall also be added to the list of figures for Chapter 19. Figures 19E.4.10-201 through 19E.4.10-204 are shown in the attachment to this enclosure.

COLA Part 4, Technical Specifications

31. Revise LCO 11 for Part 4, TS Bases B 3.3.3, last sentence of the first paragraph, to read as follows, with a LMA of LNP DEP 3.2-1:

    The condensate is returned to the IRWST via a gutter and downspouts.

32. Revise the first two sentences of the third paragraph for Part 4, TS Bases B 3.5.4, Background, to read as follows, with a LMA of LNP DEP 3.2-1:

    In order to preserve the IRWST water for long-term PRHR HX operation, downspouts and a gutter are provided to collect and return water to the IRWST that has condensed on the inside surface of the containment shell. During normal plant operation, any water collected by the downspouts or gutter is directed to the normal containment sump.

33. Revise SR 3.5.4.7 of Part 4, TS Bases B 3.5.4, Surveillance Requirements, to read as follows, with a LMA of LNP DEP 3.2-1:

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

Attachments:

Figure 6.3-201, Sheets 1 through 3
Table 9.5.1-201
Table 14.3-202, Sheets 1 and 2
Table 19.59-202
Table 19E.4.10-201
Figure 19E.4.10-201
Figure 19E.4.10-202
Figure 19E.4.10-203
Figure 19E.4.10-204

Attachments to Enclosure 5
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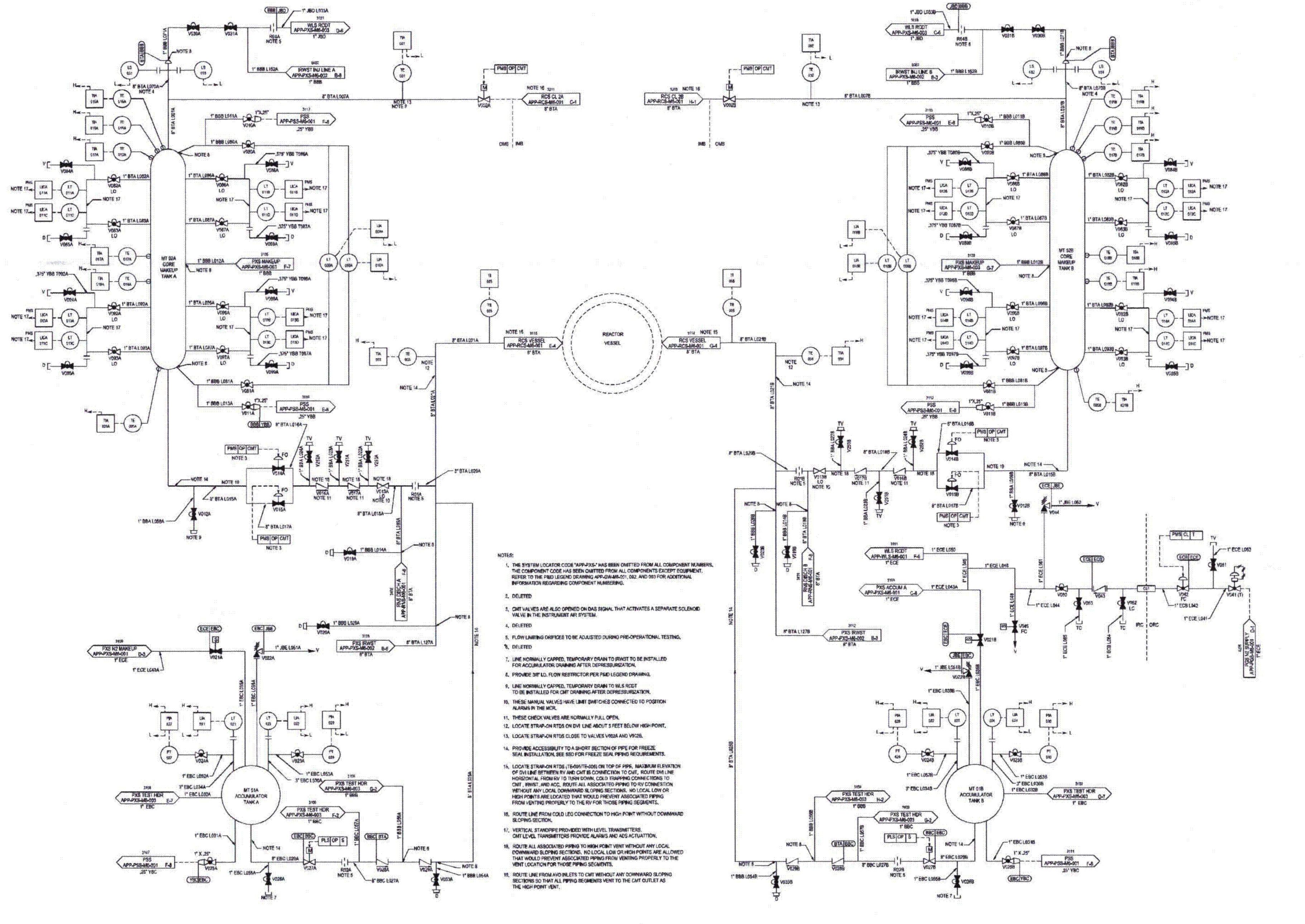
Figure 6.3-201, Sheets 1 through 3
Table 9.5.1-201
Table 14.3-202, Sheets 1 and 2
Table 19.59-202
Table 19E.4.10-201
Figure 19E.4.10-201
Figure 19E.4.10-202
Figure 19E.4.10-203
Figure 19E.4.10-204



NOTES:

1. THE SYSTEM LOCATOR CODE "APP-PXS-" HAS BEEN OMITTED FROM ALL COMPONENT NUMBERS. THE COMPONENT CODE HAS BEEN OMITTED FROM ALL COMPONENTS EXCEPT EQUIPMENT. REFER TO THE P&ID LEGEND DRAWING APP-GW-M6-001, 002, AND 003 FOR ADDITIONAL INFORMATION REGARDING COMPONENT NUMBERING.
2. DELETED
3. CMT VALVES ARE ALSO OPENED ON DAS SIGNAL THAT ACTIVATES A SEPARATE SOLENOID VALVE IN THE INSTRUMENT AIR SYSTEM.
4. DELETED
5. FLOW LIMITING ORIFICES TO BE ADJUSTED DURING PRE-OPERATIONAL TESTING.
6. DELETED
7. LINE NORMALLY CAPPED, TEMPORARY DRAIN TO IRWST TO BE INSTALLED FOR ACCUMULATOR DRAINING AFTER DEPRESSURIZATION.
8. PROVIDE 3/8" I.D. FLOW RESTRICTOR PER P&ID LEGEND DRAWING.
9. LINE NORMALLY CAPPED, TEMPORARY DRAIN TO WLS RCDT TO BE INSTALLED FOR CMT DRAINING AFTER DEPRESSURIZATION.
10. THESE MANUAL VALVES HAVE LIMIT SWITCHES CONNECTED TO POSITION ALARMS IN THE MCR.
11. THESE CHECK VALVES ARE NORMALLY FULL OPEN.
12. LOCATE STRAP-ON RTDS ON DVI LINE ABOUT 3 FEET BELOW HIGH POINT.
13. LOCATE STRAP-ON RTDS CLOSE TO VALVES V002A AND V002B.
14. PROVIDE ACCESSIBILITY TO A SHORT SECTION OF PIPE FOR FREEZE SEAL INSTALLATION. SEE SSD FOR FREEZE SEAL PIPING REQUIREMENTS.
15. 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 CONNESTION WITHOUT ANY LOCAL DOWNWARD SLOPING SECTIONS. NO LOCAL LOW OR HIGH POINTS ARE LOCATED THAT WOULD PREVENT ASSOCIATED PIPING FROM VENTING PROPERLY TO THE RV FOR THOSE PIPING SEGMENTS.
16. ROUTE LINE FROM COLD LEG CONNECTION TO HIGH POINT WITHOUT DOWNWARD SLOPING SECTION.
17. VERTICAL STANDPIPE PROVIDED WITH LEVEL TRANSMITTERS. CMT LEVEL TRANSMITTERS PROVIDE ALARMS AND ADS ACTUATTION.
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 AVO INLETS TO CMT WITHOUT ANY DOWNWARD SLOPING SECTIONS SO THAT ALL PIPING SEGMENTS VENT TO THE CMT OUTLET AS THE HIGH POINT VENT.

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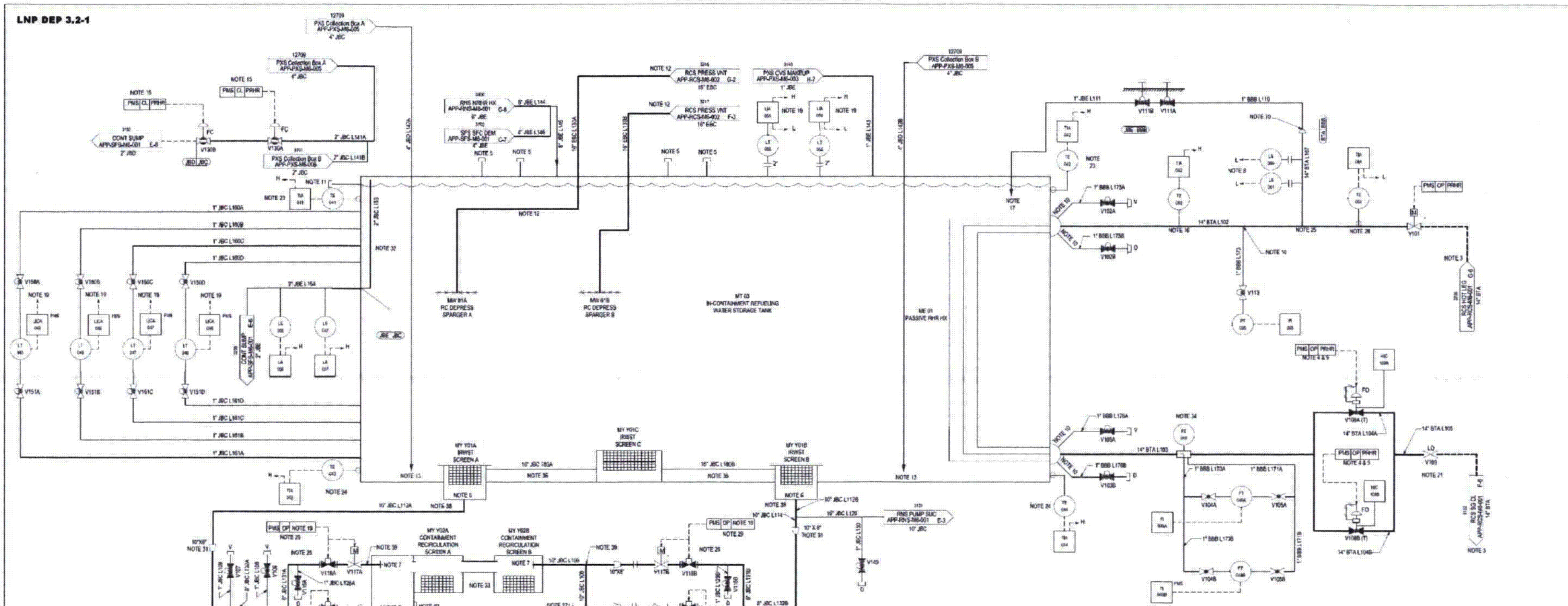
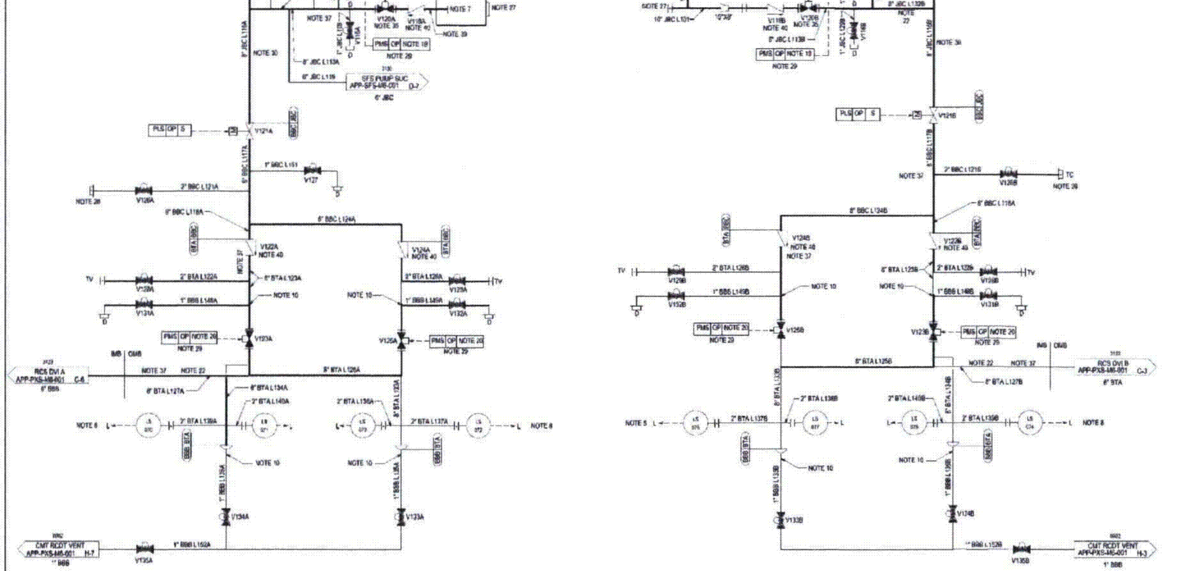
Figure 6.3-201 (Sheet 1 of 3)
Passive Core Cooling System
Piping and Instrumentation Diagram Rev. X





NOTES:

1. THE SYSTEM LOCATOR CODE "APP-PXS-" HAS BEEN OMITTED FROM ALL COMPONENT NUMBERS. THE COMPONENT CODE HAS BEEN OMITTED FROM ALL COMPONENTS EXCEPT EQUIPMENT. REFER TO THE P&ID LEGEND DRAWING APP-GW-M6-001, 002, AND 003 FOR ADDITIONAL INFORMATION REGARDING COMPONENT NUMBERING.

3. 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 HIGHT POINTS ARE ALLOWED IN EITHER THE INLET OR OUTLET LINES THAT WOULD PREVENT ASSOCIATED PIPING FROM VENTING PROPERLY TO THE LOCAL VENT PATH.

4. PRHR VALVES ARE ALSO OPENED ON DAS SIGNAL THAT ACTUATES A SEPARATE SOLENOID VALVE IN THE INSTRUMENT AIR SYSTEM.

5. IRWST HAS MULTIPLE VENTS. REFER TO SSD FOR LOCATION/SIZE REQUIREMENTS. EACH VENT HAS AN AIRTIGHT COVER THAT OPENS WITH A LOW DP.

6. LOCATE BOTH RCS GRAVITY INJECTION LINE CONNECTIONS SUCH THAT THEY DRAIN THE WHOLE IRWST. ENTRANCE PIPE HAS A ROUNDED CORNER WITH A RADIUS OF CURVATURE GREATER THAN OR EQUAL TO 1.5 INCH.

8. LEVEL SWITCHES SHOULD BE AT THE SAME ELEVATION.

9. PRHR SIGNAL FROM PMS OVERRIDES MODULATING SIGNAL FROM POSITIONER AND OPENS VALVE.

10. PROVIDE 3/8" I.D. FLOW RESTRICTOR PER WESTINGHOUSE P&ID LEGEND DRAWING (APP-GW-M6-001).

11. IRWST HAS MULTIPLE OVERFLOWS WHICH DISCHARGE TO THE REFUELING CAVITY LOWER PIT. REFER TO SSD FOR LOCATION/SIZE REQUIREMENTS. EACH OVERFLOW HAS AN AIRTIGHT COVER THAT OPENS WITH A LOW DP.

13. 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 CAUSES DRAIN TO OVERFLOW INTO IRWST. OVERFLOW LINE TERMINATES HORIZONTALLY 12 FEET AWAY FROM IRWST SCREEN FACE AND WITHIN 6 FEET OF THE IRWST FLOOR. GUTTER IS COVERED BY TRASH RACK.

14. 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 AUTUATED FLOAT, LEVEL SENSORS.

15. IRWST GUTTER ISOLATION VALVES ALSO CLOSE ON DAS SIGNAL THAT ACTUATES A SEPARATE SOLENOID VALVE IN THE INSTRUMENT AIR SYSTEM.

16. LOCATE STRAP-ON RTD ON PIPE APPROXIMATELY 3 FEET BELOW HIGH POINT.

17. PIPE SHOULD TERMINATE 6" BELOW NORMAL WATER LEVEL IN IRWST.

18. DELETED

19. IRWST HAS H, L, L-2, L-3 ALARMS. L-3 SETPOINT OPENS CONTAINMENT RECIRCULATION VALVES.

20. ADS STAGE 4 ACTUATION SIGNAL OR A LOW HL LEVEL SIGNAL OPENS IRWST INJECTION VALVES.

21. THIS MANUAL VALVE HAS LIMIT SWITCHES CONNECTED TO POSITION ALARMS IN MCR.

22. PROVIDE ACCESSIBILITY TO A SHORT SECTION OF PIPE FOR FREEZE SEAL INSTALLATION. SEE SSD FOR FREEZE SEAL PIPING REQUIREMENTS.

26. ROUTE 2" TEST CONNECTION LINE TO CONVENIENT LOCATION FOR HOOKUP TO A TEST CART.

27. REMOVABLE COVER TO ALLOW FOR CHECK VALVE INSERVICE TESTING WITH MECHANICAL TEST DEVICE.

28. LOCATE RTD JUST DOWNSTREAM OF VALVE V101.

29. VALVES, V118A/B, V120A/B, V123A/B, V125A/B ALSO OPEN ON A DAS SIGNAL.

30. LOCATE IRWST INJECTION LINES (FROM IRWST TO RV) AND CONTAINMENT RECIRCULATION LINES (FROM CONTAINMENT TO IRWST INJECTION LINES) AT AN ELEVATION BELOW THE BOTTOM OR IRWST.

31. EXTEND 10" PIPE FROM IRWST SUMP INTO PXS VALVE ROOM, CHANGE TO 8" AS SOON AS ENTERING PXS ROOM.

32. TANK OVERFLOW: CHANGE SAFETY CLASSIFICATION AT TANK PENETRATION. CONNECT WATER SENSORS IN BOTTOM OF HORIZONTAL PIPE RUN.

33. TWO SEPARATE CONTAINMENT RECIRCULATION SCREENS PROVIDED. THE TWO SCREENS ARE INTERCONNECTED BY A LINE WITH A MINIMUM FLOW AREA OF 1 SQUARE FOOT.

34. FLOW METER ELEMENT INCLUDES INTEGRAL 0.375" FLOW RESTRICTOR FOR CLASS A TO B TRANSITION SIMILAR TO P&ID LEGEND SHEET 2 NOTE 7.

35. VALVE V120A/B HAS A CLASS 2500 RATING. THESE VALVES ARE EMPLOYED TO ADD VALVE DESIGN DIVERSITY TO THE CONTAINMENT RECIRCULATION LINE.

36. ENDS OR IRWST SCREEN CROSS-CONNECT PIPE HAVE ROUNDED CORNERS WITH A RADIUS OF CURVATURE GREATER THEN OR EQUAL TO 2.25".

37. 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 LOCATIONS FOR THOSE PIPING SEGMENTS.

38. ROUTE LINE FROM REDUCER TO IRWST SCREEN INLET WITHOUT A DOWNWARD SLOPING SECTION. PIPING SEGMENT MUST VENT TO IRWST OUTLET AS THE HIGH POINT.

39. ROUTE LINE FROM CONTAINMENT RECIRCULATION SCREEN TO PARALLEL SQUIB VALVES DOWNHILL WITHOUT ANY SLOPING SECTIONS SO THAT ALL PIPING SEGMENT CAN VENT TO CONTAINMENT THROUGH THE RECIRCULATION SCREENS.

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Figure 6.3-201 (Sheet 2 of 3)
Passive Core Cooling System
Piping and Instrumentation Diagram

Rev. X

LNP DEP 3.2-1

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Figure 6.3-201 (Sheet 3 of 3)
Passive Core Cooling System
Piping and Instrumentation Diagram
Rev. X

LNP DEP 6.3-1

Table 9.5.1-201

AP1000 FIRE PROTECTION PROGRAM COMPLIANCE WITH BTP CMEB 9.5-1

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

LNP DEP 3.2-1	Table 14.3-202 (Sheet 1 of 2) **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 6.3-1	The PRHR inlet line (hot leg to high point) has no downward sloping sections.	

LNP DEP 3.2-1 Table 14.3-202 (Sheet 2 of 2)

DESIGN BASIS ACCIDENT ANALYSIS

Reference	Design Feature	Value
Figure 6.3-1	The maximum elevation of the IRWST injection lines (from the connection to the IRWST to the reactor vessel) and the containment recirculation lines (from the containment to the IRWST injection lines) is less than the bottom inside surface of the IRWST.	
Figure 6.3-1	The maximum elevation of the PRHR outlet line (from the PRHR to the SG) is less than the PRHR lower channel head top inside surface.	
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.	

LNP DEP 6.3-1	Table 19.59-202
	AP1000 PRA-BASED INSIGHTS

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

LNP DEP 3.2-1

Table 19E.4.10-201

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

Event	Time (seconds)
Feedwater is Lost	10.0
Low Steam Generator Water Level (Narrow-Range) Reactor Trip Setpoint Reached	≤ 60
Rods Begin to Drop	≤ 61
Low Steam Generator Water Level (Wide-Range) Reached	≤ 230
PRHR HX Actuation on Low Steam Generator Water Level (Narrow-Range Coincident with Low Startup Feedwater Flow)	≤ 240
Low T_{cold} Setpoint Reached	≤ 2400
Steam Line Isolation on Low T_{cold} Signal	≤ 2400
CMTs Actuated on Low T_{cold} Signal	≤ 2400
IRWST Reaches Saturation Temperature	≤ 15,500
Heat Extracted by PRHR HX Matches Core Decay Heat	≤ 34,500
CMTs Stop Recirculating	--
Cold Leg Temperature Reaches 420°F (loop with PRHR)	≤ 48.600
Core Average Temperature Reaches 420°F	≤ 124,400

LNP DEP 3.2-1

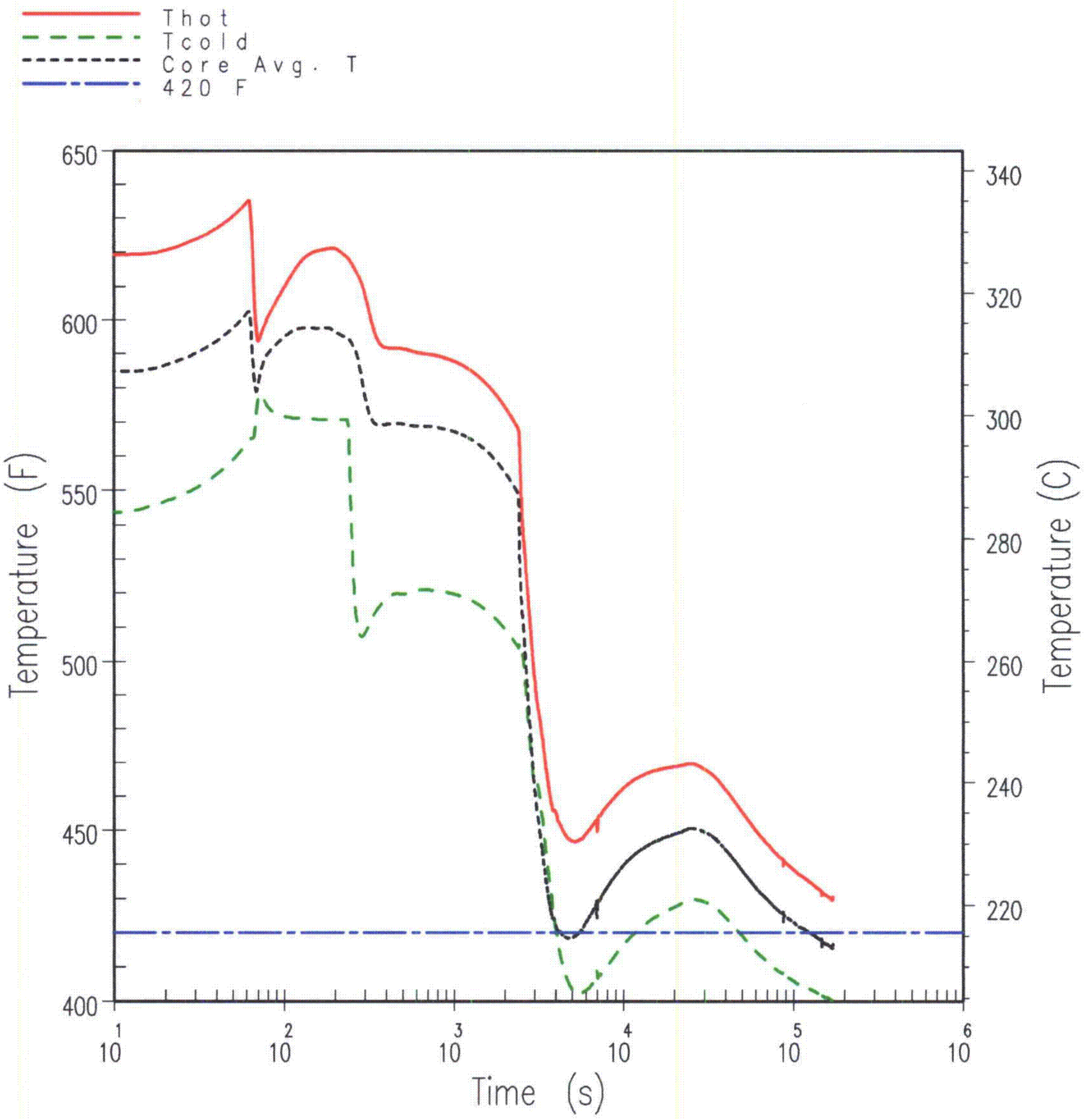

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Figure 19E.4.10-201
Shutdown Temperature Evaluation,
RCS Temperature
Rev. X

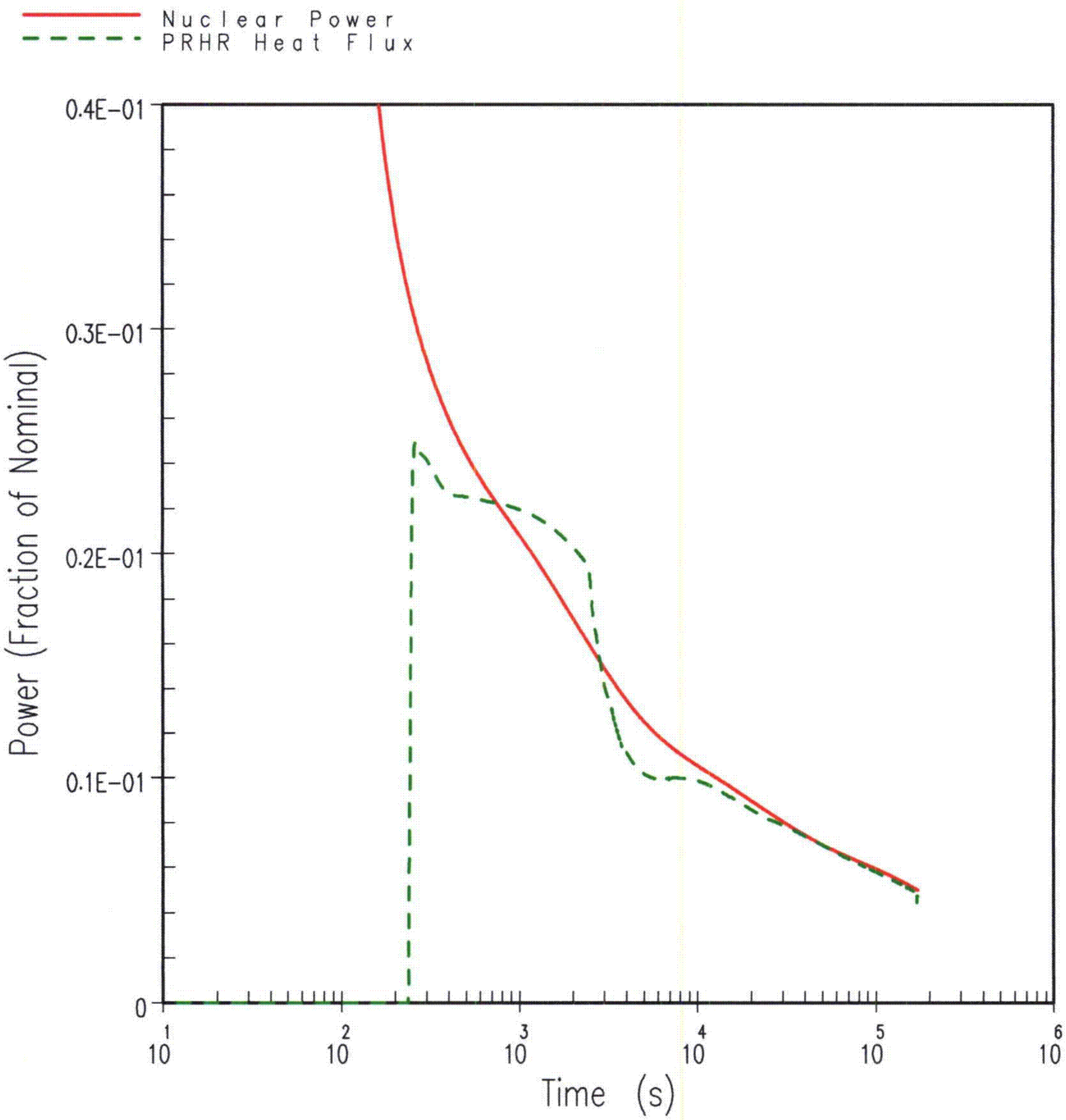

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Figure 19E.4.10-202

Shutdown Temperature Evaluation,
PRHR Heat Transfer

Rev. X

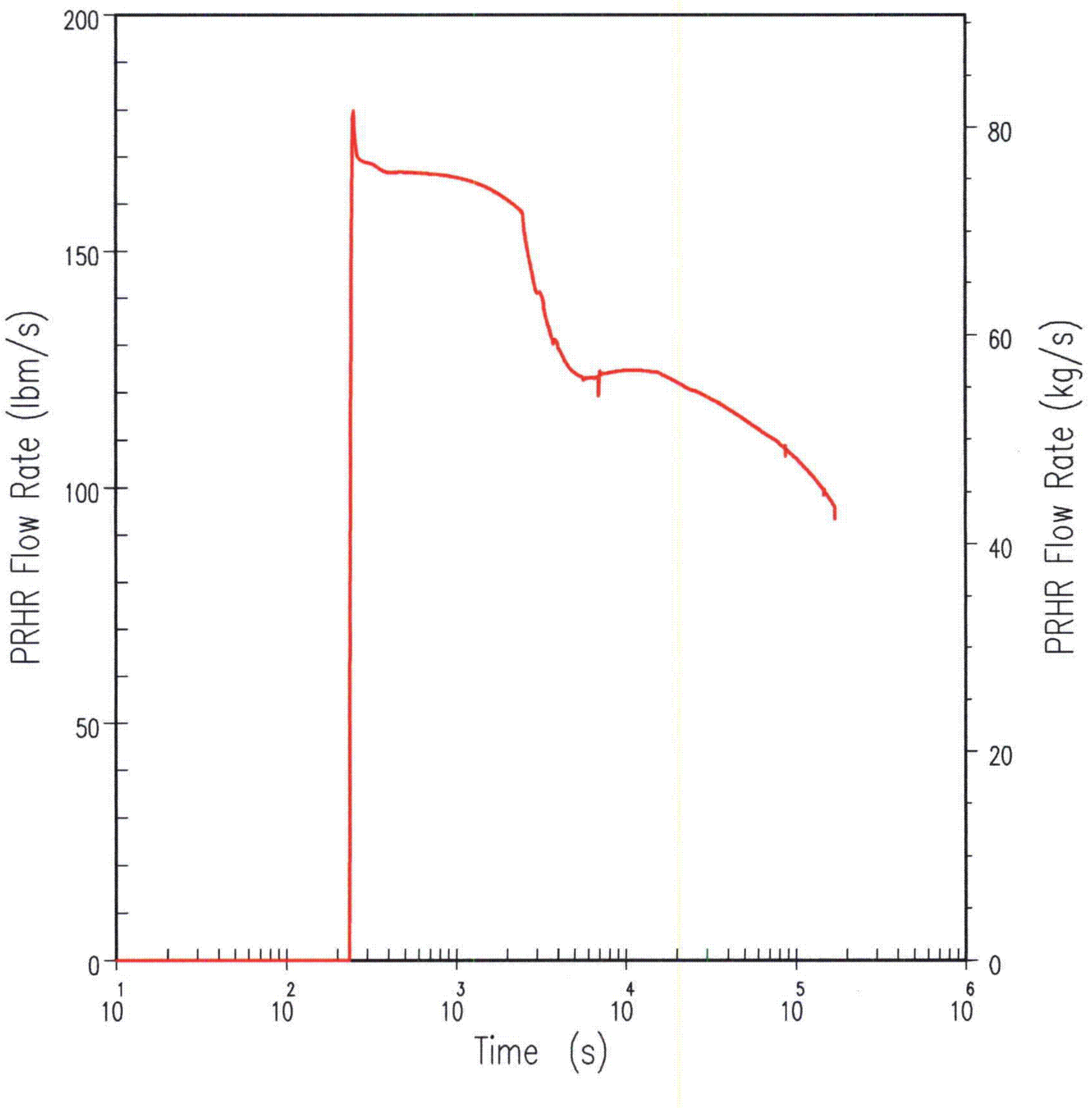

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Figure 19E.4.10-203
Shutdown Temperature Evaluation, PRHR Flow Rate

Rev. X

LNP DEP 3.2-1

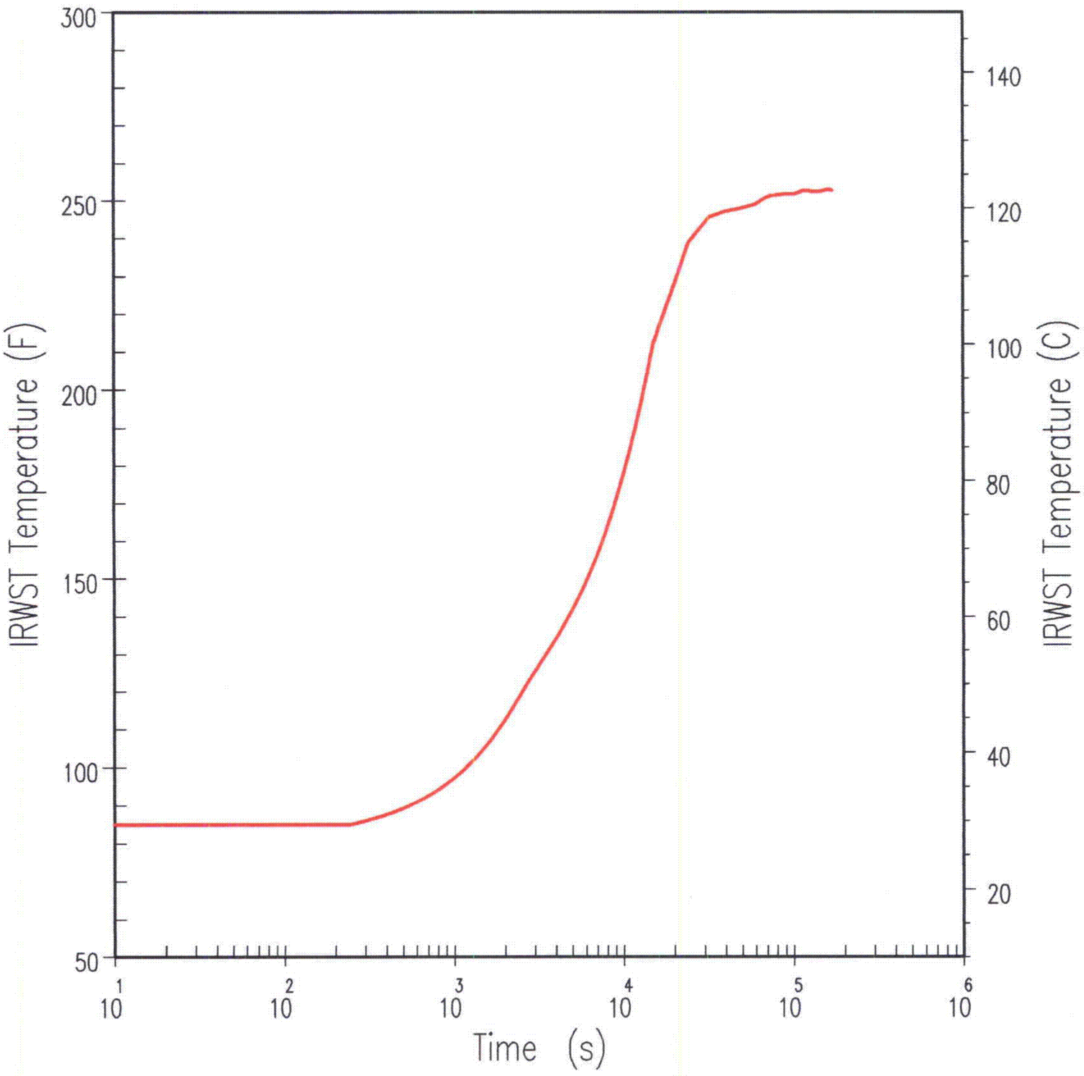

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Figure 19E.4.10-204
Shutdown Temperature Evaluation,
IRWST Heatup

Rev. X