Figure 15.4.8-4 not used

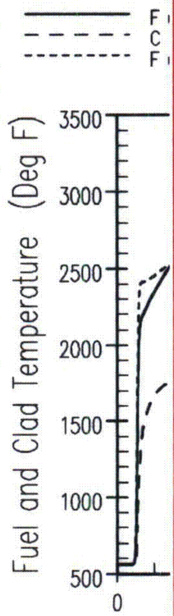

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Figure 15.4.8-4¶
Hot Spot Fuel, Average Fuel, and Outer Cladding Temperature Versus Time at End of Life, Hot Zero Power

AP1000 CORE REFERENCE REPORT
DCD (Rev. 19) Change Road Map

Change No.	Chapter 15 Section 15.5	Change Summary Description
[15.5-1]	15.5.1, Inadvertent Operation of the CMT During Power Operation	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), addition of the flow skirt, increased lower core support plate flow hole size, increased pressurizer volume, increased RV diameter for the neutron pad addition, containment backpressure effects on PRHR heat transfer, increased rod drop time for the safety analysis and the updated valve, nozzle and piping pressure loss coefficients. Editorial changes were made to the inadvertent CMT analyses to identify an operator action to open the safety related reactor vessel head vent to prevent filling the reactor coolant system water solid.
[15.5-2]	15.5.2, CVS Malfunction that Increases Reactor Coolant Inventory	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), addition of the flow skirt, increased lower core support plate flow hole size, increased pressurizer volume, increased RV diameter for the neutron pad addition, containment backpressure effects on PRHR heat transfer, increased rod drop time for the safety analysis and the updated valve, nozzle and piping pressure loss coefficients. Editorial changes were made to the inadvertent chemical and volume control analyses to identify an operator action to open the safety related reactor vessel head vent to prevent filling the reactor coolant system water solid.

15.5 Increase in Reactor Coolant Inventory

This section presents a discussion and analysis of the following events:

- Inadvertent operation of the core makeup tanks during power operation
- Chemical and volume control system malfunction that increases reactor coolant inventory

These Condition II events cause an increase in reactor coolant inventory.

15.5.1 Inadvertent Operation of the Core Makeup Tanks During Power Operation

Comment [B1]: [15.5-1]

15.5.1.1 Identification of the Causes and Accident Description

Spurious core makeup tank operation at power could be caused by an operator error, a false electrical actuation signal, or a valve malfunction. A spurious signal may originate from any of the safeguards ("S") actuation channels as described in Section 7.3. The AP1000 protection logic is such that a single failure cannot actuate both core makeup tanks without also actuating the passive residual heat removal (PRHR) heat exchanger. A scenario such as this is the spurious "S" signal event. However, if one core makeup tank is inadvertently actuated by a single failure, the event may progress with the plant at power until a reactor trip is reached. For the plant under automatic rod control, a reactor trip on high-3 pressurizer water level reactor trip is expected to occur followed by the PRHR actuation and eventually by an "S" signal, which would then actuate the second core makeup tank. When a consequential loss of offsite power is assumed, this event is more conservative than the spurious "S" signal event.

The inadvertent opening of the core makeup tank discharge valves, due to operator error or valve failure, results in significant core makeup tank injection flow leading to a boration similar to that resulting from a chemical and volume control system malfunction event. If the automatic rod control system is operable, it will begin to withdraw rods from the core to counteract the reactivity effects of the boration. As a result, the core makeup tank will continue injection and slowly increase the pressurizer level until the high -2 pressurizer level setpoint is reached and continues until the high-3 pressurizer level trip setpoint is reached. In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, a loss of offsite power is assumed to occur as a consequence of reactor trip. The primary effect of this assumption is the coastdown of the reactor coolant pumps. The core makeup tank injection will increase as the steam generator outlet temperature increases resulting in a lower density in the CMT balance line. This event will then proceed similarly to a spurious "S" signal or chemical and volume control system malfunction event. However, this event is more limiting primarily due to the higher pressurizer level at the time of reactor trip and to the significant heat up of the injected fluid during the pre-trip phase of the accident. Thus, the inadvertent core makeup tank actuation event with a consequential loss of offsite power is analyzed here.

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Upon receipt of the high-3 pressurizer level reactor trip signal, the reactor is tripped; then the turbine is tripped after a 5-second delay and 3-seconds after turbine trip, a consequential loss of offsite power is assumed. The basis for the 3-second delay is described in subsection 15.0.14. The high-3 pressurizer level signal also actuates the PRHR heat exchanger and blocks the pressurizer heaters, but a 15-second delay is built in to prevent unnecessary actuation of the PRHR heat exchanger if offsite power is maintained.

Following reactor trip, the reactor power drops and the average reactor coolant system temperature decreases with subsequent coolant shrinkage. However, due to the assumed loss of offsite power, the reactor coolant cold leg temperature, in the loop without PRHR, increases and the core makeup tank starts injecting cold water into the reactor coolant system at a much higher rate. The primary coolant system shrinkage is counteracted by the core makeup tank injection, and the pressurizer water volume starts to increase because of the heatup of the cold injected fluid by the decay heat. The high-3 pressurizer level setpoint is once again reached, and after a 15-second delay, the signal is sent to actuate the PRHR heat exchanger and block the pressurizer heaters.

The PRHR heat exchanger extracts heat from the reactor coolant system leading to an "S" signal on a Low T_{cold} signal. The PRHR heat exchanger may inject asymmetrically into the steam generator outlet plenum such that a higher percentage of the PRHR flow is in one of two cold legs coming from the steam generator on the PRHR loop. To account for this, the analysis assumes that the Low T_{cold} setpoint is reached coincident with PRHR heat exchanger actuation. This actuates the second core makeup tank sooner in the transient, which is more limiting with respect to filling the pressurizer.

Both core makeup tanks inject mass into the reactor coolant system and the pressurizer level continues to increase until the operators take action to end the pressurizer level increase transient. The operators are assumed to be alerted to a potential filling event on the high-2 pressurizer level signal, which occurs well before the reactor trip on the first of two high-3 pressurizer level signals. The operator action assumed in the analysis is to open the reactor vessel head vent following receipt of the second high-3 pressurizer level signal; this action is at least 30 minutes (45 minutes as analyzed) after the operator has been alerted by the high-2 pressurizer level signal. When the head vent is opened, the pressurizer level increase slows and ultimately the level begins to decrease.

This event is a Condition II incident (a fault of moderate frequency) as defined in subsection 15.0.1.

15.5.1.2 Analysis of Effects and Consequences

The plant response to an inadvertent core makeup tank actuation is analyzed by using a modified version of the computer program LOFTRAN (Reference 1) described in subsection 15.0.11.2. The code simulates the neutron kinetics, reactor coolant system, pressurizer,

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pressurizer safety valves, pressurizer spray, steam generator, steam generator safety valves, PRHR heat exchanger, and core makeup tanks. The program computes pertinent plant variables, including temperatures, pressures, and power level.

Reactor power and average temperature drop immediately following the trip, and the operating conditions never approach the core limits. The analysis demonstrates that no reactor coolant system overpressurization occurs.

Core makeup tank and PRHR system performance is conservatively simulated. Core makeup tank enthalpies have been maximized. This is conservative because it minimizes the cooling provided by the core makeup tanks as flow recirculates and thereby increases the peak pressurizer water volume during the transient. Core makeup tank injection and balance lines pressure drop is minimized. This maximizes the core makeup tank flow injected in the primary system. During this event, the core makeup tanks remain filled with water. The volume of injection flow leaving the core makeup tanks is offset by an equal volume of recirculation flow that enters the core makeup tanks via the balance lines. PRHR heat transfer capability has been minimized.

Plant characteristics and initial conditions are further discussed in subsection 15.0.3.

- Initial operating conditions

  The initial reactor power is assumed to be 101 percent of nominal. The initial pressurizer pressure is assumed to be 50 psi below nominal. The initial reactor coolant system average temperature is assumed to be 8°F below nominal.

- Control systems

  The pressurizer spray system and automatic rod control system are conservatively assumed to operate. The pressurizer heaters are automatically blocked on a high-3 pressurizer level signal, so they cannot add heat to the system during the period of thermal expansion that produces the peak pressurizer water volume. Thus, the pressurizer heaters are assumed to be inoperable during this event. Other control systems are conservatively not assumed to function during the transient.

- Moderator and Doppler coefficients of reactivity

  A least-negative moderator temperature coefficient, a low (absolute value) Doppler power coefficient, and a minimum boron worth are assumed. With these minimum feedback parameters and the operability of the pressurizer spray system and automatic rod control system assumed, the reactivity effects of the boron injection from the core makeup tanks is counteracted. As a result, the high-3 pressurizer signal is the first reactor trip signal generated during the transient.

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- Boron injection

  The transient is initiated by an inadvertent opening of the discharge valves of one of the two core makeup tanks. The core makeup tank injects 3400 ppm borated water.

- Protection and safety monitoring system actuations

  The operators are assumed to be alerted of the pressurizer level increase transient on the high-2 pressurizer level signal. Reactor trip is initiated by the first of two high-3 pressurizer level signals. The second high-3 pressurizer level signal triggers the operators to open the reactor vessel head vent; this action is at least 30 minutes after the operator has been alerted by the high-2 pressurizer level signal.

  The core decay heat is removed by the PRHR heat exchanger. The worst single failure is assumed to occur in the outlet line of the PRHR heat exchanger. One of the two parallel isolation valves is assumed to fail to open.

Plant systems and equipment available to mitigate the effect of the accident are discussed in subsection 15.0.8 and listed in Table 15.0-6.

15.5.1.3 Results

Figures 15.5.1-1 through 15.5.1-8 show the transient response to the inadvertent operation of one of the two core makeup tanks during power operation. The inadvertent opening of the core makeup tank discharge valves occurs at 10 seconds. As the core makeup tank continues to add inventory to the primary system, the pressurizer level begins to increase until the high-2 pressurizer level setpoint is reached (556.1 seconds) and continues until the high-3 pressurizer level reactor trip setpoint is reached at about 2,589.3 seconds. After a 2-second delay, the neutron flux starts decreasing due to the reactor trip, which is followed by turbine trip after a 5-second turbine trip delay. Following reactor trip, the reactor power drops and the average reactor coolant system temperature decreases with subsequent coolant shrinkage. Due to the assumed loss of offsite power, the reactor coolant pumps trip at about 2,599.3 seconds. The cold leg temperature increases and the core makeup tank starts injecting cold water into the reactor coolant system at a higher rate due to the increased driving head resulting from the density decrease in the balance line and due to the reduced pressure drop between the cold leg and the injection line connection on the reactor vessel following the trip of the reactor coolant pumps. The post-trip primary coolant system shrinkage is counteracted by the core makeup tank injection, and the pressurizer water volume starts to increase because of the heatup of the cold injected fluid by the decay heat. The high-3 pressurizer level setpoint is once again reached at 2,736.6 seconds, and after a 15-second delay, the signal is sent to actuate the PRHR heat exchanger and block the pressurizer heaters.

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Following a conservative 17-second delay, the valves are assumed to open to actuate the PRHR heat exchanger at 2,768.6 seconds.

If the PRHR heat exchanger coolant asymmetrically injects into the steam generator outlet plenum, then one cold leg could reach the Low T_{cold} "S" setpoint more quickly than if the flow were split evenly. To conservatively account for this effect, the Low T_{cold} "S" signal is modeled to actuate simultaneously with the actuation of the PRHR heat exchanger (2,768.6 seconds). The Low T_{cold} "S" signal activates the second core makeup tank, which then begins injecting additional mass into the reactor coolant system. Previous analyses have demonstrated that a more limiting pressurizer fill transient is calculated the earlier the second core makeup tank is actuated.

As the second core makeup tank begins injecting, the pressurizer level continues to increase. The operators are assumed to be alerted by the high-2 pressurizer level signal (556.1 seconds) that a pressurizer level increase transient is underway, and it is assumed that the operators are ready to take corrective action at least 30 minutes later. In this analysis, since pressurizer level continues to increase, the high-3 pressurizer level reactor trip setpoint is reached within this time. The operator action assumed in this case is to open the reactor vessel head vent to preclude overfill following receipt of the second high-3 pressurizer level signal (3,256.1 seconds); this action is at least 30 minutes (45 minutes as analyzed) after the operator has been alerted by the high-2 pressurizer level signal.

The safety related reactor vessel head vent is opened by the operators and the pressurizerwater level increase slows and eventually the level begins to decrease. This demonstrates that the capacity of the reactor vessel head vent is sufficient to preclude pressurizer overfill as a result of an inadvertent actuation of a core makeup tank.

During the event, the departure from nucleate boiling ratio (DNBR) never drops significantly below the initial value due to the addition of highly borated water from the core makeup tanks to the reactor coolant system. At the time of reactor trip core power and heat flux drop rapidly and the DNBR is well above the design limit value defined in Section 4.4.

The calculated sequence of events is shown in Table 15.5-1.

As noted above, the limiting case presented here models explicit operator action 45 minutes after receipt of the high-2 pressurizer level signal. For pressurizer level increase events, the operator would take action to reduce the increase in coolant inventory. As the pressurizer water level would increase above the high pressurizer water level that normally isolates chemical and volume control system makeup (high-2), the normal letdown line could be placed into service to reduce the increase in coolant inventory. If letdown could not be placed

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into service, the operator could use the safety related reactor vessel head vent valves to reduce the increase in coolant inventory (this is explicitly modeled in the case presented here). For these events, following the procedures outlined in the Emergency Response Guidelines AFR-I.1, there is sufficient time for the operator to mitigate the consequences of this event.

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15.5.1.4 Conclusions

The results of this analysis show that inadvertent operation of the core makeup tanks during power operation does not adversely affect the core, the reactor coolant system, or the steam system. Water is not relieved from the pressurizer safety valves. DNBR always remains above the design limit values, and reactor coolant system and steam generator pressures remain below 110 percent of their design values.

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15.5.2 Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory

Comment [B2]: [15.2-2]

15.5.2.1 Identification of Causes and Accident Description

An increase of reactor coolant inventory, which results from addition of cold unborated water to the reactor coolant system, is analyzed in subsection 15.4.6.

In this subsection 15.5.2, the increase of reactor coolant system inventory due to the addition of borated water is analyzed.

The increase of reactor coolant system coolant inventory may be due to the spurious operation of one or both of the chemical and volume control system pumps or by the closure of the letdown path. If the chemical and volume control system is injecting highly borated water into the reactor coolant system, the reactor experiences a negative reactivity excursion due to the injected boron, causing a decrease in reactor power and subsequent coolant shrinkage. The load decreases due to the effect of reduced steam pressure after the turbine control valve fully opens.

At high chemical and volume control system boron concentration, low reactivity feedback conditions, and reactor in manual rod control, an "S" signal will be generated by either the low T_{cold} or low steam line pressure setpoints before the chemical and volume control system can inject a significant amount of water into the reactor coolant system. In this case, the chemical and volume control system malfunction event proceeds similarly to, and is only slightly more limiting than, a spurious "S" signal event. If the automatic rod control is modeled and the pressurizer spray functions properly to prevent a high pressure reactor trip signal, no "S" signals are generated and this specific event is terminated by automatic isolation of the chemical and volume control system on the safety-related high-2 pressurizer level setpoint.

Under typical operating conditions for the AP1000, the boron concentration of the injected chemical and volume control system water is equal to that of the reactor coolant system. If the chemical and volume control system is functioning in this manner and the pressurizer spray system functions properly to prevent a high pressure reactor trip signal, no "S" signals are generated and this specific event is also terminated by automatic isolation of the chemical and volume control system on the safety-related high-2 pressurizer level setpoint.

While these scenarios are the most probable outcomes of a chemical and volume control system malfunction, several combinations of boron concentration, feedback conditions, and plant system interactions have been identified which can result in more limiting scenarios with respect to pressurizer overfill. The key factors that make this event more limiting than a spurious "S" signal event are that the reactor coolant system is at a lower average temperature, higher pressure, and a higher pressurizer level at the time an "S" signal is generated. These factors produce a greater volume of higher density water and, thus, a larger reactor coolant system mass at the time of the "S" signal. In addition, at lower reactor coolant system average temperature, the PRHR is less effective in removing decay heat, which results in greater expansion of the cold water injected by the core makeup tanks.

The limiting analysis scenario minimizes reactor coolant system average temperature, maximizes reactor coolant system mass, and maximizes pressurizer water volume at the time of an "S" signal. This scenario is as follows:

- Both of the chemical and volume control system pumps spuriously begin delivering flow at a boron concentration slightly higher than that of the reactor coolant system. (Assuming that a chemical and volume control system malfunction results in both chemical and volume control system pumps delivering flow is a conservative assumption. One chemical and volume control system pump is automatically controlled and one is manually controlled.)

- The non-safety-related pressurizer spray is assumed to be available, so that a high pressurizer pressure reactor trip is prevented.

Due to the boron addition in the core, the plant cools down until an "S" signal is generated on low cold leg temperature. On the "S" signal, the reactor is tripped, the core makeup tank discharge valves are opened, the reactor coolant pumps are tripped, the pressurizer heaters are blocked, and the main feedwater lines, steam lines, and chemical and volume control system are isolated. After a conservative 17-second delay, the PRHR heat exchanger is actuated.

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Normally, the reactor coolant pumps would be tripped 15 seconds after the receipt of the "S" signal. However, to meet the requirements of GDC 17 of 10 CFR Part 50, Appendix A, a loss of offsite power is assumed to occur as a consequence of reactor trip. The primary effect of this assumption is the coastdown of the reactor coolant pumps. Following reactor trip and a

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5-second timer delay the turbine is tripped 3-seconds after a turbine trip a consequential loss of offsite power is assumed. The basis for the 3-second delay is described in subsection 15.0.14. As a result, the reactor coolant pumps are conservatively assumed to trip about 10 seconds before they would otherwise trip due to the "S" signal.

This event is a Condition II incident (a fault of moderate frequency) as defined in subsection 15.0.1.

15.5.2.2 Analysis of Effects and Consequences

The malfunction of the chemical and volume control system is analyzed by using a modified version of the computer program LOFTRAN (Reference 1) described in subsection 15.0.11.2. The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generator, steam generator safety valves, PRHR heat exchanger, and core makeup tanks. The program computes pertinent plant variables including temperatures, pressures, and power level.

Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits. The analysis demonstrates that no reactor coolant system overpressurization or loss of reactor coolant system water occurs.

The assumptions are as follows:

- Initial operating conditions

  The initial reactor power is assumed to be 101 percent of nominal. The initial pressurizer pressure is assumed to be 50 psi above nominal. The initial reactor coolant system average temperature is assumed to be 8°F above nominal.

- Moderator and Doppler coefficients of reactivity

  A least-negative moderator temperature coefficient, a low (absolute value) Doppler power coefficient, and a minimum boron worth are assumed. For a different set of reactivity feedback parameters, a different chemical and volume control system boron concentration can result in an identical transient.

- Reactor control

  Rod control is not modeled.

- Pressurizer heaters

  The pressurizer heaters are automatically blocked on an "S" signal, and do not add heat to the system during the period of fluid thermal expansion that produces the peak

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pressurizer water volume. Thus, the pressurizer heaters are assumed to be inoperable during this event.

- Pressurizer spray

  The spray system controls the pressurizer pressure so that a high pressurizer pressure reactor trip is prevented.

- Boron injection

  After 10 seconds at steady state, the chemical and volume control system pumps start injecting borated water, which is slightly above the reactor coolant system boron concentration. Upon receipt of an "S" signal, the core makeup tanks begin injecting 3400 ppm borated water. The chemical and volume control system pumps are isolated on high-2 pressurizer level. In this analysis the boron concentration of the chemical and volume control system is iterated upon until the high-2 pressurizer level and the low T_{cold} "S" setpoint are reached at the same time. This begins core makeup tank injection when the chemical and volume control system pumps are isolated, which is conservative with respect to filling the pressurizer.

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- Turbine load

  The turbine load is assumed constant until the turbine D-EHC drives the control valve wide open. Then the turbine load drops as steam pressure drops.

- Protection and safety monitoring system actuations

  If the automatic rod control system is modeled and the pressurizer spray system functions properly, no reactor trip signal is expected to occur. Instead, the event is terminated by automatic isolation of the chemical and volume control system on the safety grade high-2 pressurizer level setpoint. If the automatic rod control system is not active and the pressurizer spray system is assumed to be available, reactor trip may be initiated on either low T_{cold} "S" or a low steam line pressure "S" signal.

  The core decay heat is removed by the PRHR heat exchanger. The worst single failure is assumed to occur in the outlet line of the PRHR heat exchanger. One of the two parallel isolation valves is assumed to fail to open.

  Plant systems and equipment available to mitigate the effect of the accident are discussed in subsection 15.0.8 and listed in Table 15.0-6.

15.5.2.3 Results

Figures 15.5.2-1 through 15.5.2-9 show the transient response to a chemical and volume control system malfunction that results in an increase of reactor coolant system inventory.

As the chemical and volume control system injection flow increases reactor coolant system inventory, pressurizer water volume begins increasing while the primary system is cooling down. At 2,271.3 seconds, the low T_{cold} setpoint is reached, the reactor trips on the resulting "S" signal, and the control rods start moving into the core. At the same time, the high-2 pressurizer level setpoint is reached and after a conservative delay, the chemical and volume control system injection is isolated.

The turbine is tripped as a result of the reactor trip following a 5-second turbine trip timer delay. After a 3-second delay following turbine trip, a consequential loss of offsite power is assumed and the reactor coolant pumps trip. The basis for the 3-second delay is described in subsection 15.0.14. Soon after reactor trip, the pressurizer heaters are blocked and the main feedwater lines, steam lines, and chemical and volume control system are isolated. After a conservative 17-second delay, the PRHR heat exchanger is actuated and the core makeup tank discharge valves are opened. The core makeup tanks work in recirculation mode, meaning they are always filled with water because cold borated water injected through the injection lines is replaced by hot water coming from the cold leg balance lines.

The operation of the PRHR heat exchanger and the core makeup tanks cools down the plant. Due to the swelling of the core makeup tank water, the pressurizer level continues to increase. The operators are assumed to be alerted by the high-2pressurizer level signal (2,270.8 seconds) that a pressurizer level increase transient is underway, and it is assumed that the operators are ready to take corrective action at least 30 minutes later. The specific operator action assumed in this case is to open the reactor vessel head vent to preclude pressurizer overfill following the high-3 pressurizer level signal (4,070.8 seconds); this action is at least 30 minutes after the operator has been alerted by the high-2 pressurizer level signal.

The safety related reactor vessel head vent is opened by the operators and the pressurizer water level increase slows and eventually the level begins to decrease. This demonstrates that the capacity of the reactor vessel head vent is sufficient to preclude pressurizer overfill as a result of a chemical and volume control system malfunction that causes an increase in reactor coolant inventory.

During the event, the DNBR never drops significantly below the initial value since both the chemical and volume control system and the core makeup tanks add borated water to the reactor coolant system. At the time of reactor trip, core power and heat flux drop rapidly and the DNBR is well above the design limit value defined in Section 4.4.

The calculated sequence of events is shown in Table 15.5-1.

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The limiting case presented here models operator action to open the reactor vessel head vent following receipt of the high-3 pressurizer level signal; this action is at least 30 minutes after the operator has been alerted by the high-2 pressurizer level signal. For pressurizer level increase events, the operator could take other actions to reduce the increase in coolant inventory. As the pressurizer water level would increase above the high pressurizer water level that normally isolates chemical and volume control system makeup, the normal letdown line could be placed into service to reduce the increase in coolant inventory. If letdown could not be placed into service, the operator would use the safety-related reactor vessel head vent valves to reduce the increase in coolant inventory. For these events, following operations procedures, there is sufficient time for the operator to mitigate the consequences of this event.

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15.5.2.4 Conclusions

The results of this analysis show that a chemical and volume control system malfunction does not adversely affect the core, the reactor coolant system, or the steam system. Water is not relieved from the pressurizer safety valves. DNBR remains above the design limit values, and reactor coolant system and steam generator pressures remain below 110 percent of their design values.

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If the automatic rod control system and the pressurizer spray systems are assumed to function, no reactor trip signal is expected to occur. Instead, the event would be terminated by automatic isolation of the chemical and volume control system on the safety grade high-2 pressurizer level setpoint. If manual rod control is assumed and the pressurizer spray system is assumed to be unavailable, reactor trip may be initiated on either a high pressurizer pressure, low T_{cold} "S", or a low steamline pressure "S" signal.

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15.5.3 Boiling Water Reactor Transients

This subsection is not applicable to the AP1000.

15.5.4 Combined License Information

This subsection has no requirement for additional information to be provided in support of the Combined License application.

15.5.5 References

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), April 1984.

Table 15.5-1 (Sheet 1 of 2)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN AN INCREASE IN REACTOR COOLANT INVENTORY

Accident	Event	Time (seconds)
Inadvertent operation of the core makeup tanks during power operation	Core makeup tank discharge valves open	10
	High-2 pressurizer level setpoint reached	556.1
	High-3 pressurizer level setpoint reached	2,589.3
	Rod motion begins	2,591.3
	Loss of offsite power	2,599.3
	Reactor coolant pumps trip	2,599.3
	High-3 pressurizer level setpoint reached	2,735.6
	PRHR heat exchanger actuated	2,768.6
	Low T_{cold} "S" setpoint is reached	2,768.6
	Second CMT starts recirculating	2,768.6
	Main steam and feed lines are isolated	2,780.6
	Operators open the reactor vessel head vent after the high-3 pressurizer level signal is reached (at least 30 minutes after high-2 pressurizer level setpoint is reached)	3,256.1
	Peak pressurizer water volume occurs	5,460.0

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Table 15.5-1 (Sheet 2 of 2) **TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN AN INCREASE IN REACTOR COOLANT INVENTORY**		
Accident	**Event**	**Time (seconds)**
Chemical and volume control system malfunction that increases reactor coolant inventory	Chemical and volume control system charging pumps start	10.0
	Low T_{cold} "S" signal and high-2 pressurizer level signals are reached	2,270.8
	Core makeup tank discharge valves open	2,271.4
	Rod motion begins	2,272.8
	Loss of offsite power	2,280.8
	Reactor coolant pumps trip	2,280.8
	Main steam and feed lines are isolated	2,283.4
	PRHR heat exchanger actuated	2,288.4
	Chemical and volume control system charging pumps are isolated	2,308.9
	Operators open the reactor vessel head vent after the high-3 pressurizer level signal is reached (at least 30 minutes after high-2 pressurizer level setpoint is reached)	4,070.8
	Peak pressurizer water volume occurs	5,078.0
	Pressurizer water volume begins to decrease	5,484.0

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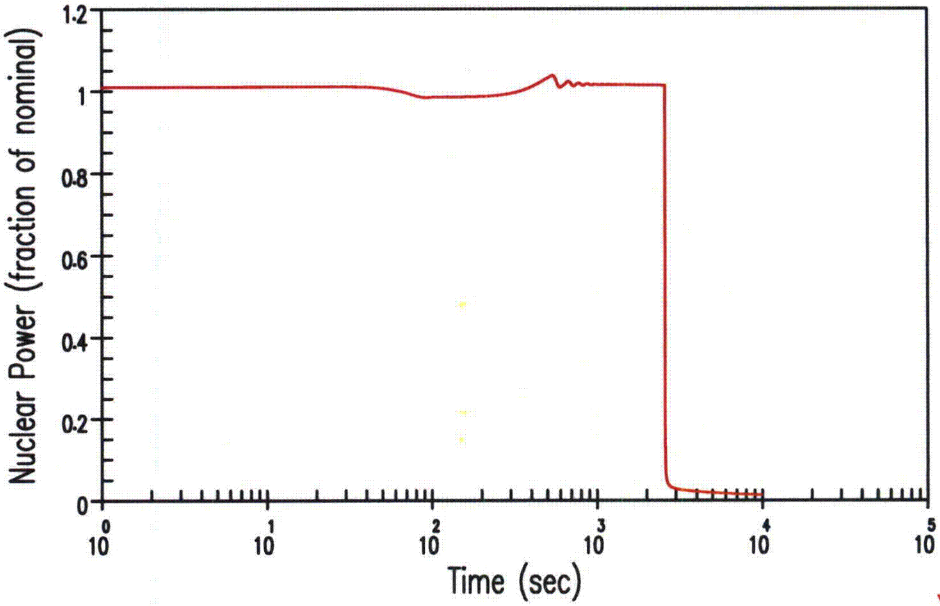

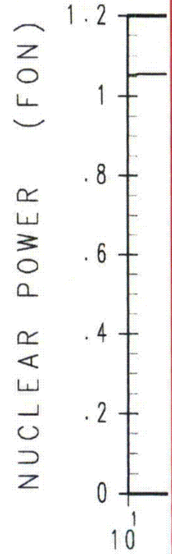

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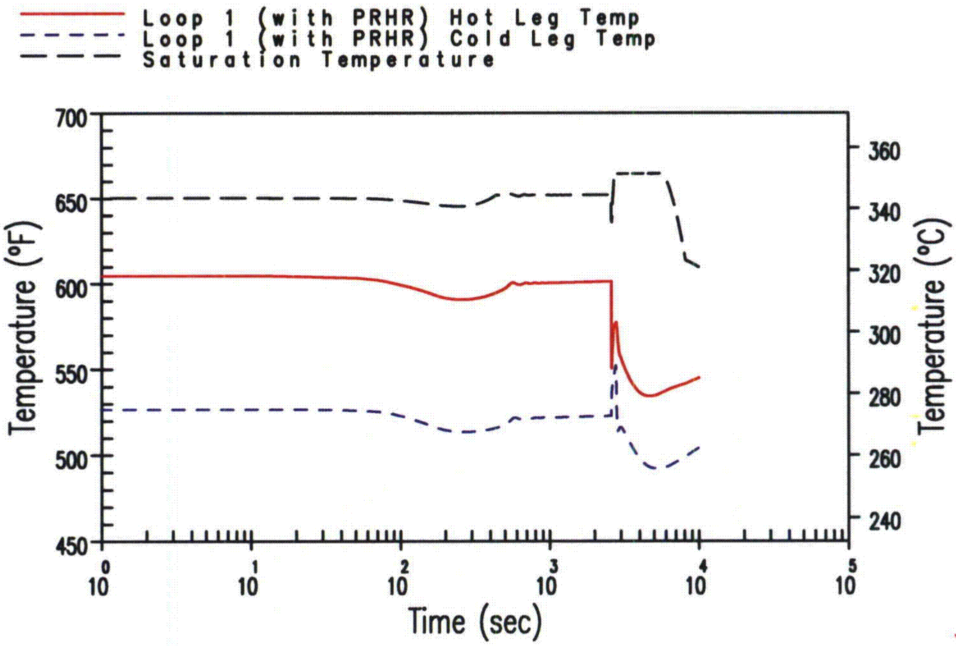
Figure 15.5.1-1

Core Nuclear Power Transient for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves



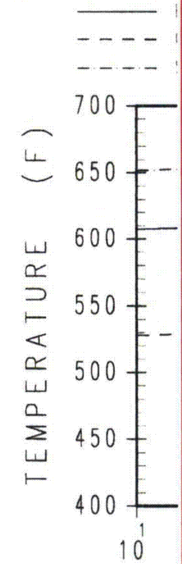


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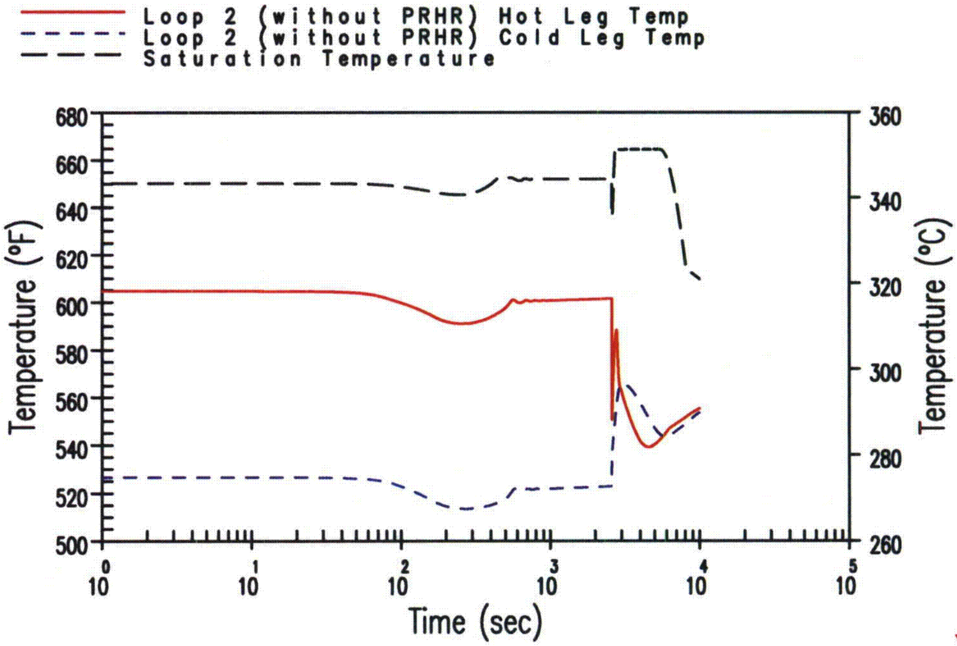
Figure 15.5.1-2

RCS Temperature Transient in Loop Containing the PRHR for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves



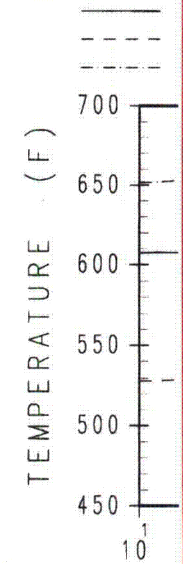


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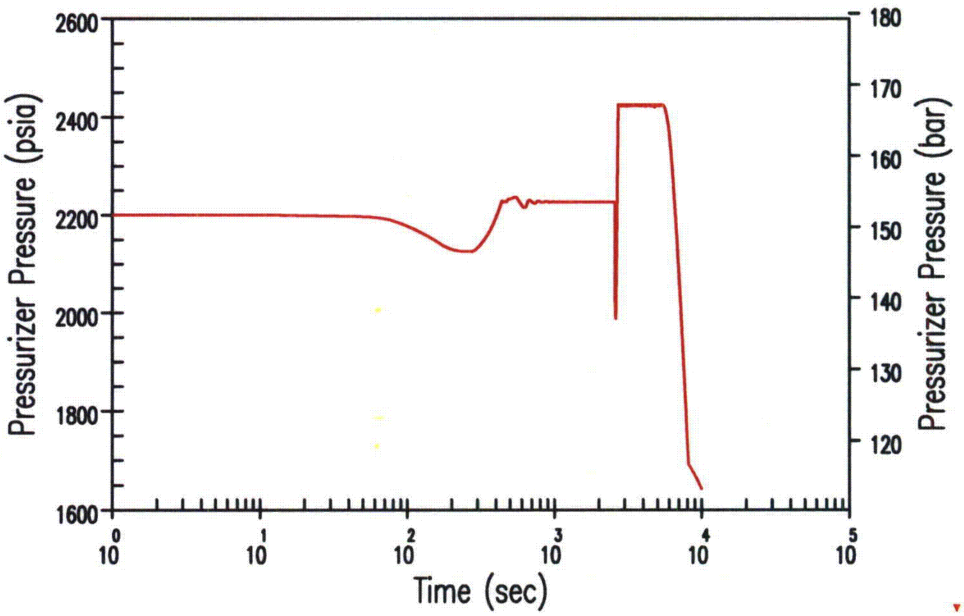
Figure 15.5.1-3

RCS Temperature Transient in Loop Not Containing the PRHR for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves



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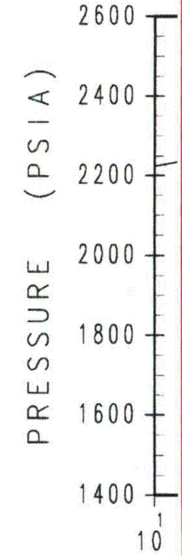
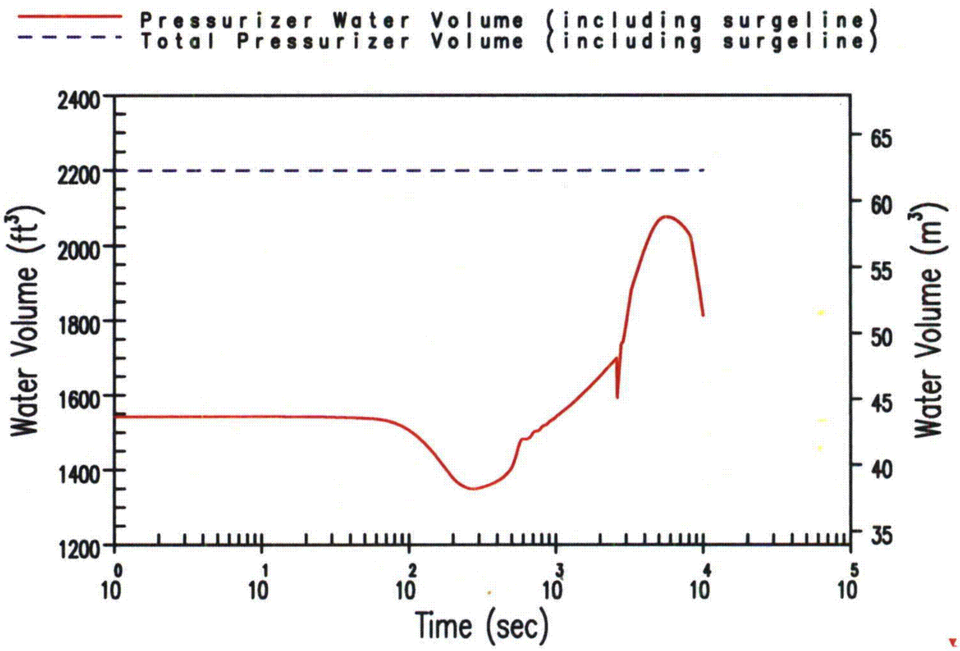


Figure 15.5.1-4

Pressurizer Pressure Transient for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves



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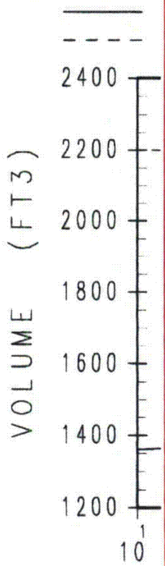

Figure 15.5.1-5

Pressurizer Water Volume Transient for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

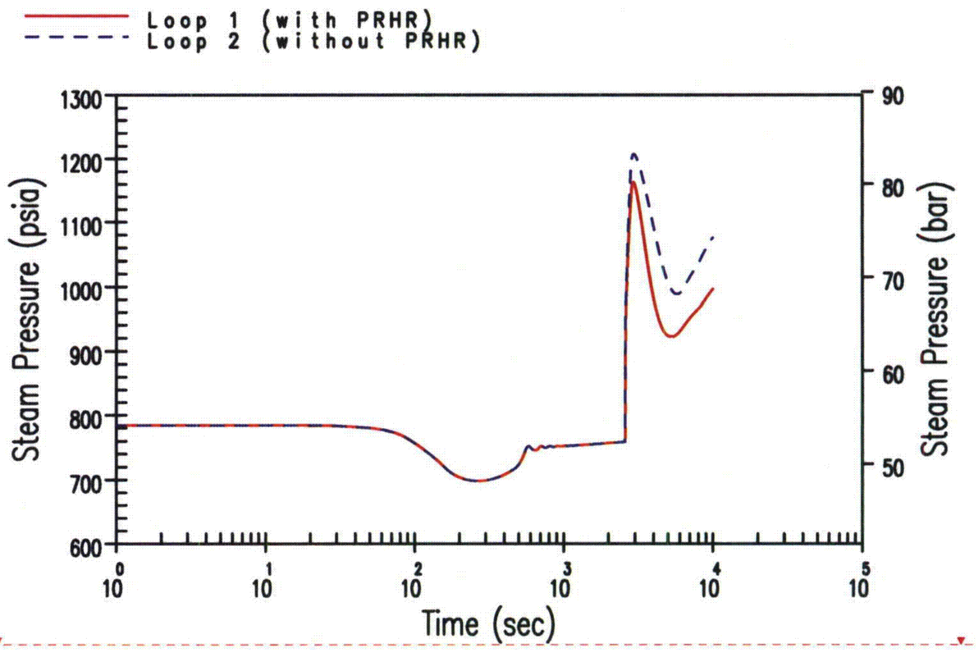

Figure 15.5.1-6

Steam Generator Pressure Transient for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

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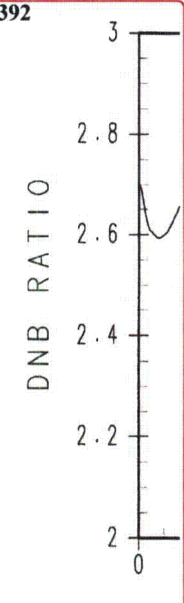

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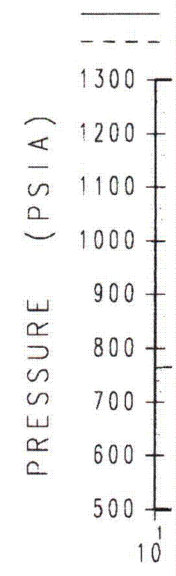

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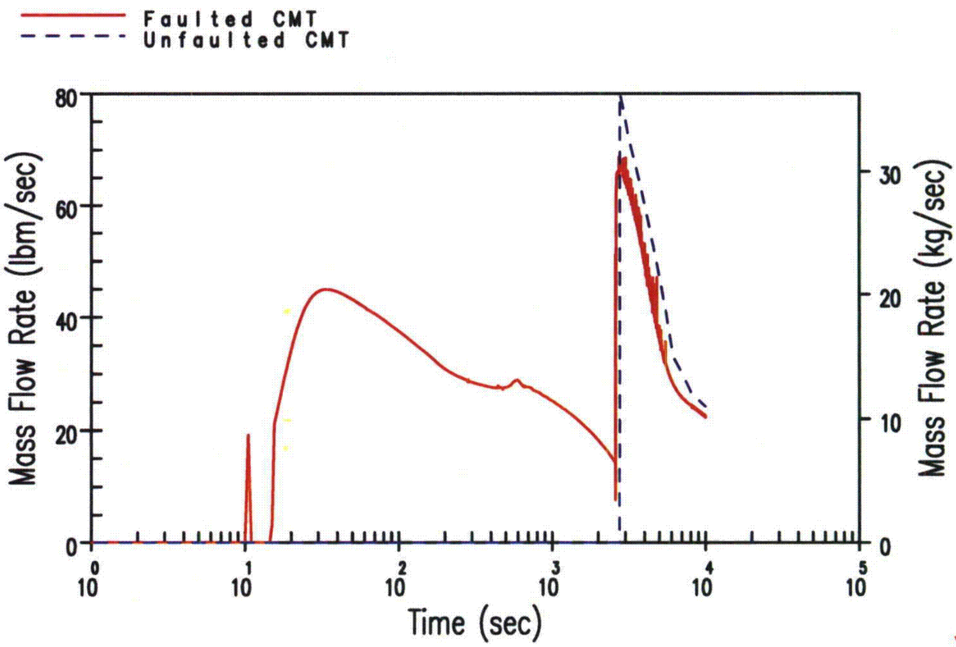

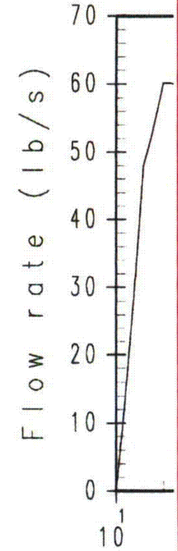

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Figure 15.5.1-7

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CMT Flow Rate Transient
for Inadvertent Operation of the Emergency Core Cooling System
Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

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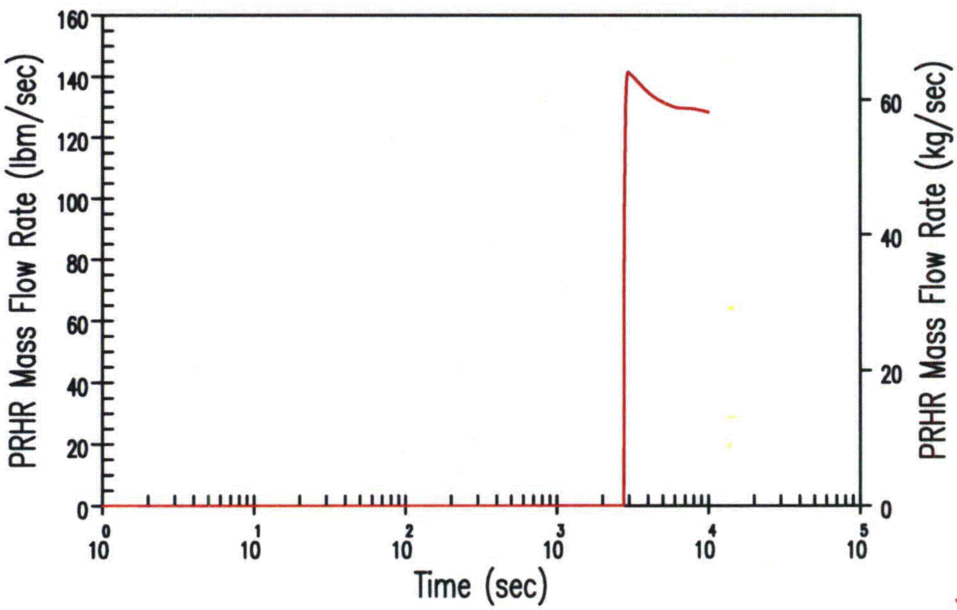

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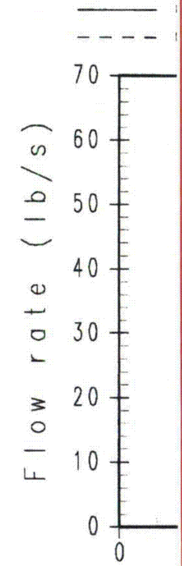

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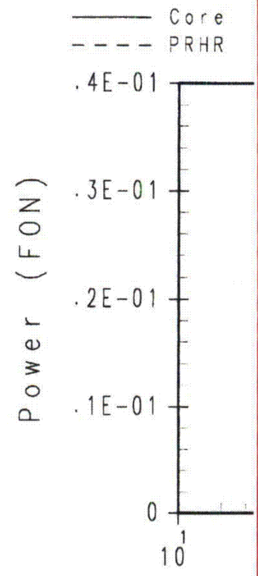

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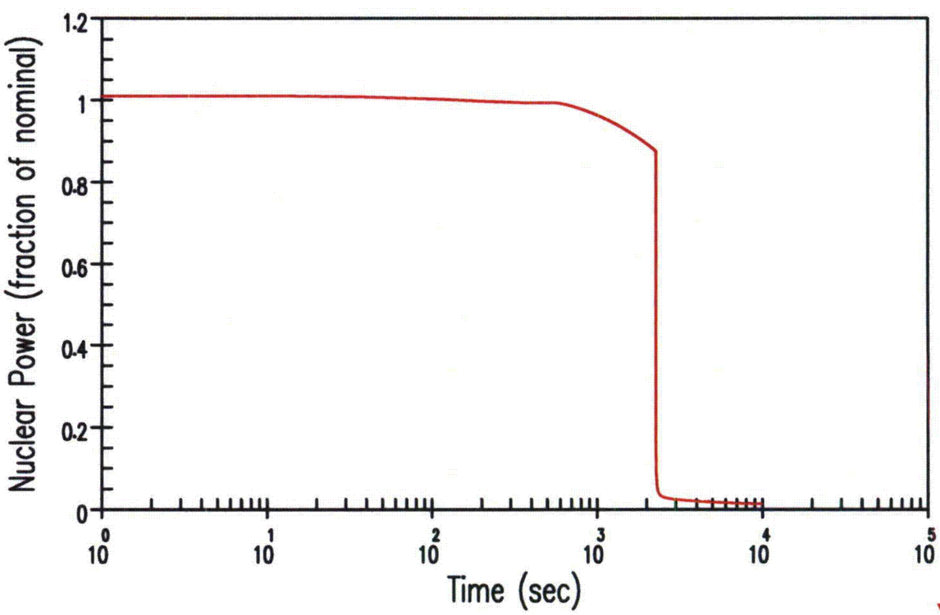
Figure 15.5.1-8

PRHR Flow Rate Transient
for Inadvertent Operation of the Emergency Core Cooling System
Due to a Spurious Opening of the Core Makeup Tank Discharge Valves



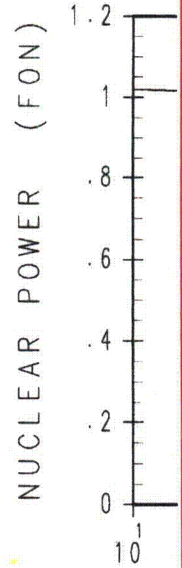


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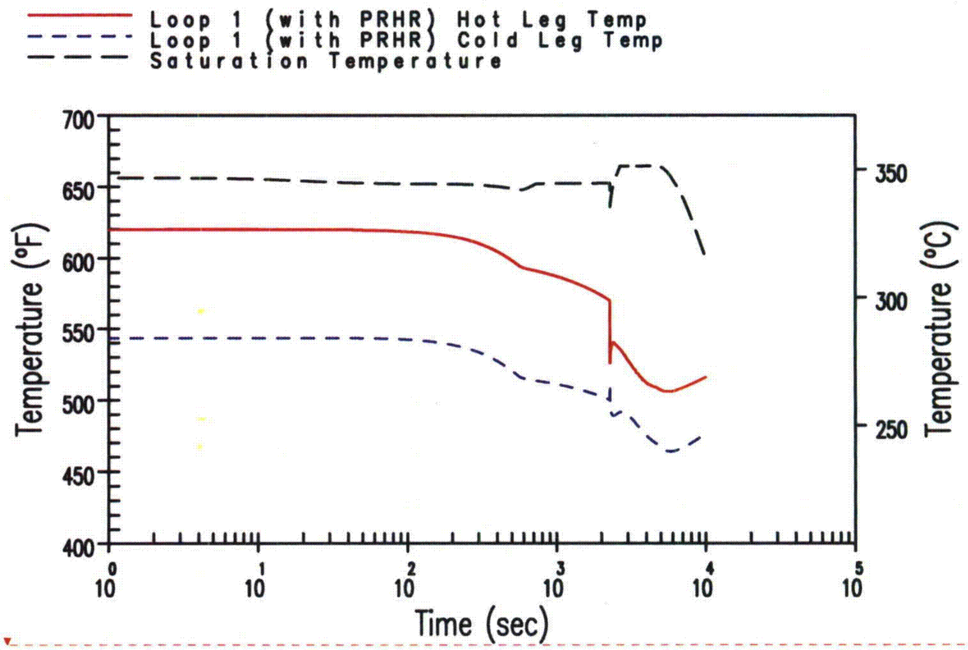
Figure 15.5.2-1

Core Nuclear Power Transient for Chemical and Volume Control System Malfunction



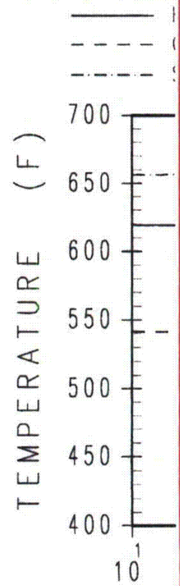


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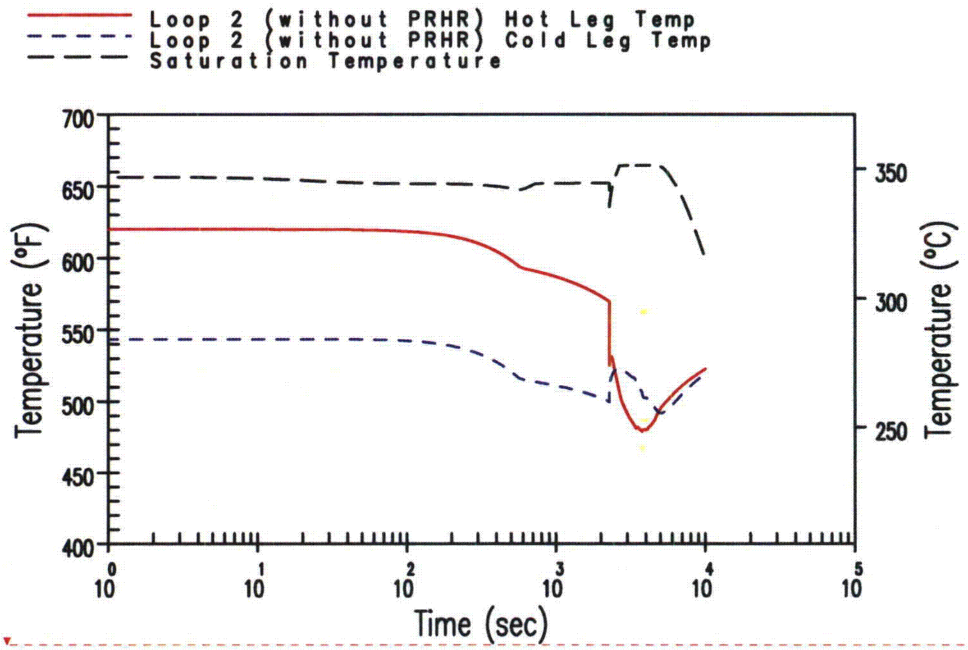
Figure 15.5.2-2

RCS Temperature Transient in Loop Containing the PRHR for Chemical and Volume Control System Malfunction



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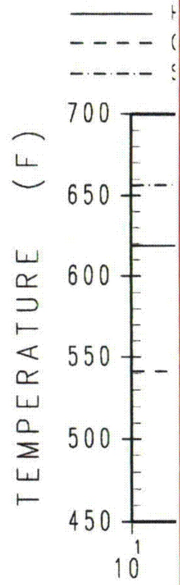
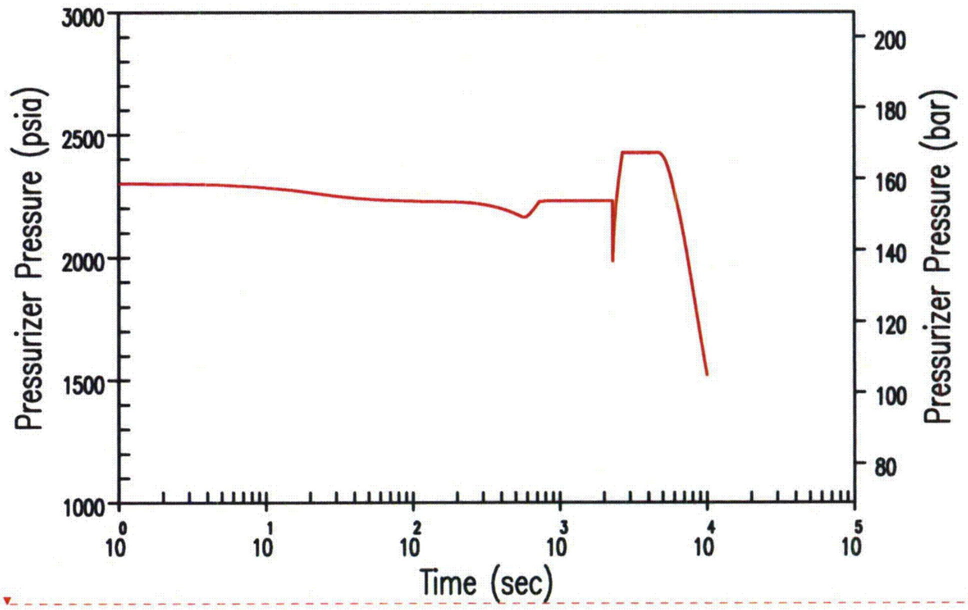


Figure 15.5.2-3

RCS Temperature Transient in Loop Not Containing the PRHR for Chemical and Volume Control System Malfunction



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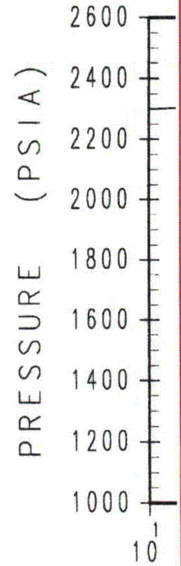


Figure 15.5.2-4

Pressurizer Pressure Transient
for Chemical and Volume Control System Malfunction

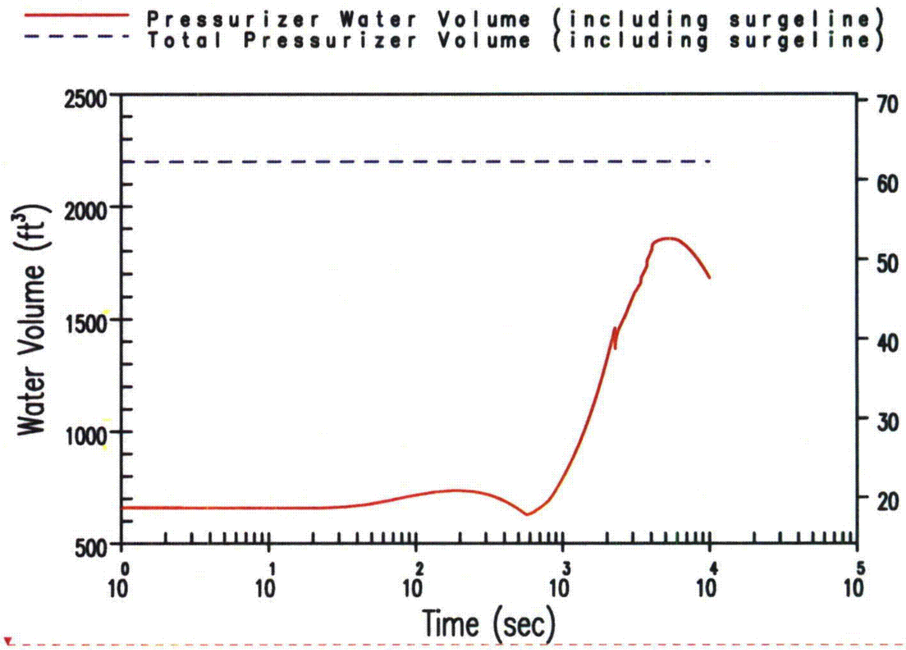
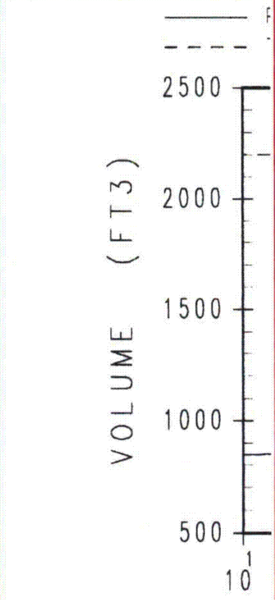


Figure 15.5.2-5

Pressurizer Water Volume Transient
for Chemical and Volume Control System Malfunction



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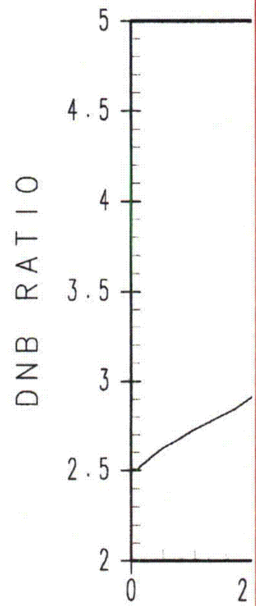

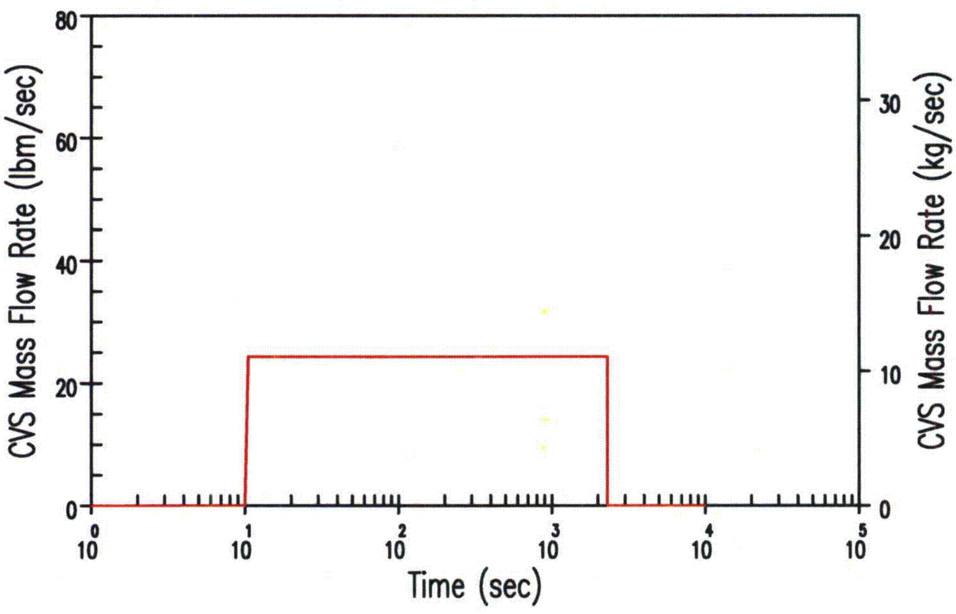

Figure 15.5.2-6

CVS Flow Rate Transient
for Chemical and Volume Control System Malfunction

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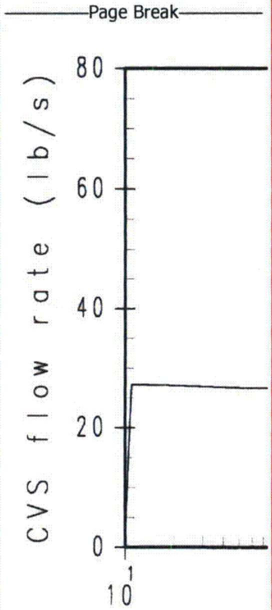
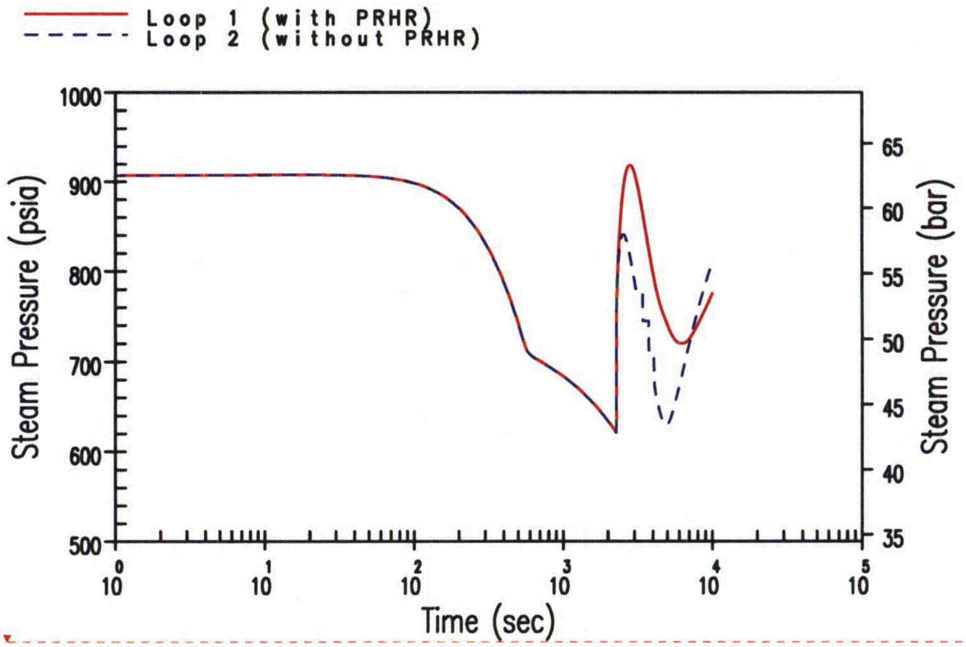


Figure 15.5.2-7¶



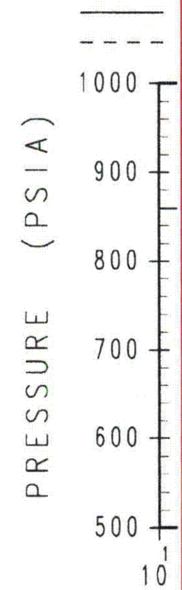


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Figure 15.5.2-7

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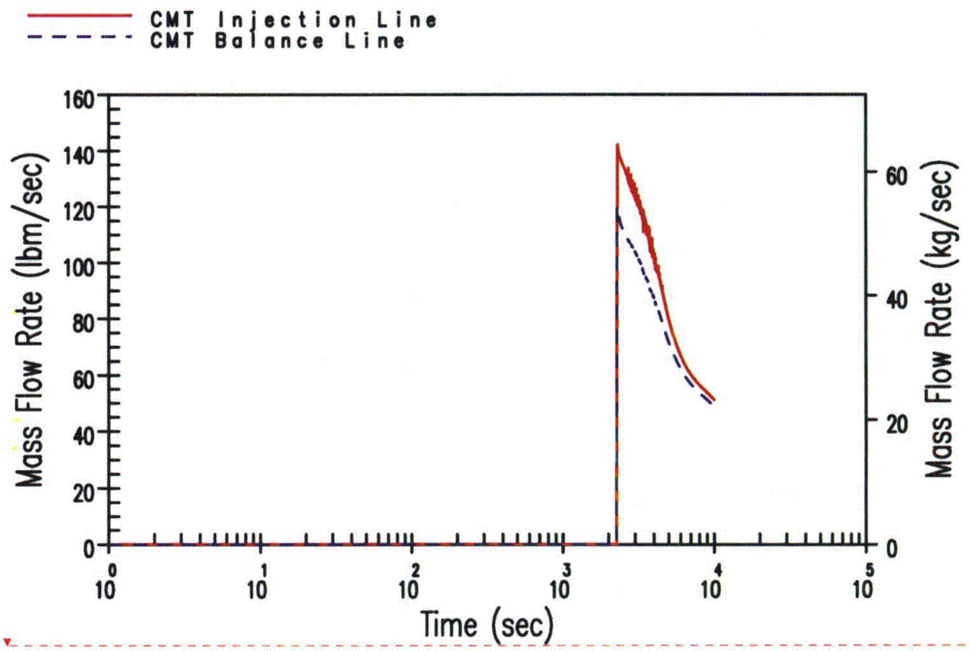
Steam Generator Pressure Transient
for Chemical and Volume Control System Malfunction



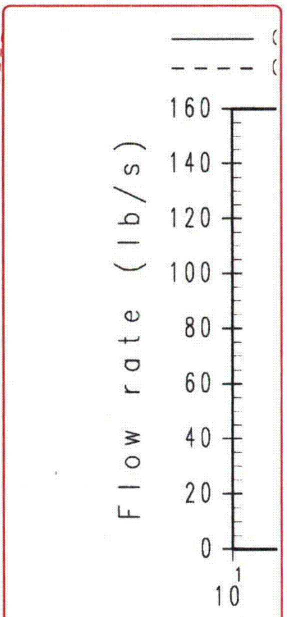


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Figure 15.5.2-8

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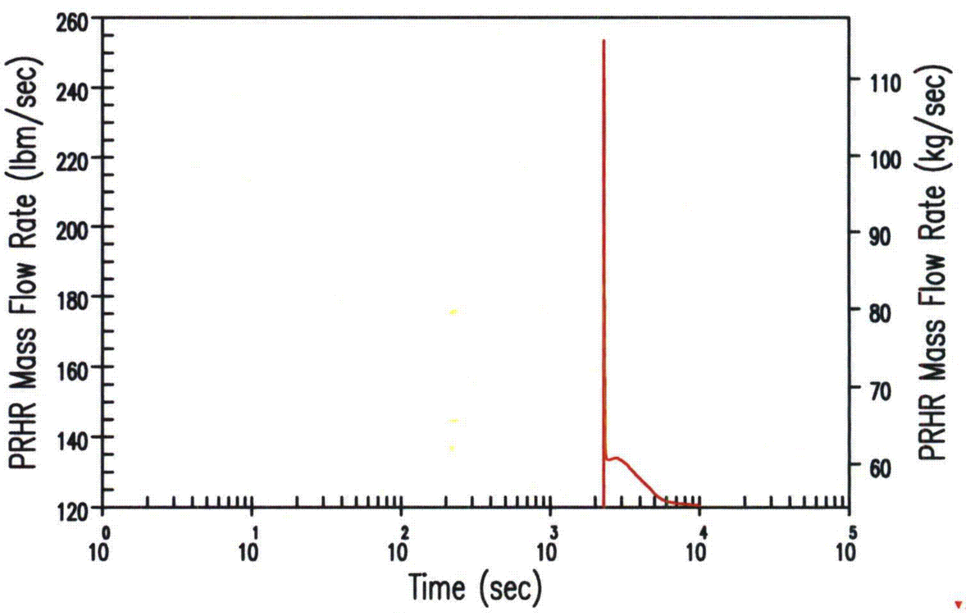
CMT Injection Line and Balance Line Flow Transient for Chemical and Volume Control System Malfunction



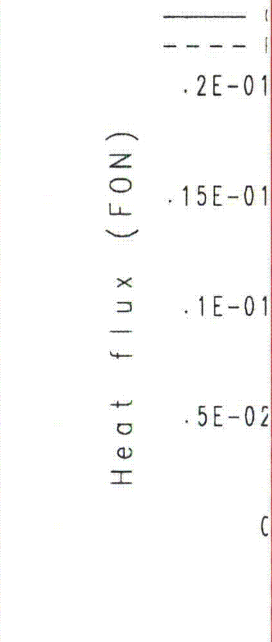


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Figure 15.5.2-9

PRHR Flow Rate Transient
for Chemical and Volume Control System Malfunction

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AP1000 CORE REFERENCE REPORT
DCD (Rev. 19) Change Road Map

Change No.	Chapter 15 Section 15.6	Change Summary Description
[15.6-1]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), addition of the flow skirt, increased lower core support plate flow hole size, increased pressurizer volume, increased RV diameter for the neutron pad addition, use of the digital ΔT signal, increased rod drop time for the safety analysis and the updated valve, nozzle and piping pressure loss coefficients.
[15.6-2]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	Clarification. The ADS actuation sequence includes progression of the valves from 1-3 with associated delay timers in between such that the max valve stroke times plus delay timers ensure each valve set doesn't actuate before the other valve set.
[15.6-3]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	Additional text was added to provide clarity. It was not a change in the analysis or design. The DCD description itself was updated.
[15.6-4]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	Updated the description of the valve parameters. The previous value represents the max opening time. The max opening times of the ADS valves were previously revised. However, there are delay timers in place, so if the max stroke time changes the delay timers can be adjusted accordingly so the analysis is not affected. To reduce the number of possible future changes the minimum stroke time was listed, which is a hard functional requirement for the valve performance.
[15.6-5]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	By adding the comments in the preceding paragraph it was possible to omit these sections.
[15.6-6]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	Additional detail on why loss of AC power need not be considered for an RCS depressurization event has been added. The previous description did not contain sufficient detail.
[15.6-7]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	Because of the RCP delay on reactor trip the inadvertent ADS valve operation does not challenge DNB. Therefore LOFTRAN is sufficient to conclude DNB margin is maintained.
[15.6-8]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	As stated in 10CFR 50 GDC 17 analysis of coincident loss of AC power for a RCS depressurization event is not required based on the Turbine and RCP response to this scenario.

Change No.	Chapter 15 Section 15.6	Change Summary Description
[15.6-9]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	With a loss of AC power, the OTΔT is the trip signal. Now, the Low Pressurizer Pressure is the actuated protection signal.
[15.6-10]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	See Change No. [15.6-8]
[15.6-11]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	See Change No. [15.6-8]
[15.6-12]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	See Change No. [15.6-9]
[15.6-13]	15.6.2, Failure of Small Lines Carrying Primary Coolant Outside Containment	Editorial Changes. It is more accurate to describe the initial iodine and noble gas primary coolant concentrations as based on their respective technical specifications (i.e. equilibrium operating limits) because the technical specification limits do not necessarily correspond to the design fuel defect level. This is consistent with the modeling used in the analyses. The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), increased pressurizer volume, increased RV diameter for the neutron pad addition, use of the digital ΔT signal, increased rod drop time for the Safety analysis and the updated valve, nozzle and piping pressure loss coefficients.
[15.6-14]	15.6.3, Steam Generator Tube Rupture	Editorial Changes. The analysis was revised to incorporate updates to the NSSS model and also incorporate the resolution to the containment backpressure issue. The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), increased pressurizer volume, increased RV diameter for the neutron pad addition, increase MSSV inlet piping diameter (increased 1.2 inches), increased rod drop time for the Safety analysis and the updated valve, nozzle and piping pressure loss coefficients.
[15.6-15]	15.6.3.3 Radiological Consequences (SGTR)	It is more accurate to describe the initial iodine and noble gas primary coolant concentrations as based on their respective technical specifications (i.e. equilibrium operating limits) because the technical specification limits do not necessarily correspond to the design fuel defect level. This is consistent with the modeling used in the analyses. Doses were updated based on the revised analyses.

Change No.	Chapter 15 Section 15.6	Change Summary Description
[15.6-16]	15.6.5.3 LOCA (Radiological Consequences Only)	Editorial Changes. The analyses are based on a 1% power measurement uncertainty. It is more accurate to describe the initial iodine and noble gas primary coolant concentrations as based on their respective technical specifications (i.e. equilibrium operating limits) because the technical specification limits do not necessarily correspond to the design fuel defect level. This is consistent with the modeling used in the analyses. Doses and limiting 2-hour intervals updated based on revised source terms for the Advanced First Core.
[15.6-17]	15.6.5.4A, Large-break LOCA Analysis Methodology and Results	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), addition of the flow skirt, increased lower core support plate flow hole size, increased pressurizer volume, increased RV diameter for the neutron pad addition, use of the digital ΔT signal, the updated reactor coolant pump flywheel material and the updated valve, nozzle and piping pressure loss coefficients.
[15.6-18]	15.6.5.4A, Large-break LOCA Analysis Methodology and Results	Reference 3 (AP600 SER) was added since Advanced Plant specific restrictions which were originally identified in the AP600 SER, and were carried to the AP1000 SER issued in 2005 and remain valid with application of ASTRUM methodology for US licensing.
[15.6-19]	15.6.5.4A.3, Signal Logic for Large-break LOCA	Editorial changes incorporated to clarify section.
[15.6-20]	15.6.5.4A.5, Large-break LOCA Analysis Results	Reference added consistent with comment 15.6.5.4A-2
[15.6-21]	15.6.5.4A.6, Description of AP1000 Large-Break LOCA Transient	Editorial changes incorporated to clarify section.
[15.6-22]	15.6.5.4A.6, Description of AP1000 Large-Break LOCA Transient	Pump delay updated consistent with current timer value and timer uncertainty.
[15.6-23]	15.6.5.4A.6, Description of AP1000 Large-Break LOCA Transient	Value updated due to the additional 1.3 seconds pump delay.
[15.6-24]	15.6.5.4A.6, Description of AP1000 Large-Break LOCA Transient	Values updated due to ASTRUM methodology.
[15.6-25]	15.6.5.4A.6, Description of AP1000 Large-Break LOCA Transient	Section reworded to present the most limiting case. ASTRUM is statistical and based on probabilities. Therefore the results can change slightly each time the spectrum transient is performed. However, the most limiting transient is always chosen.

Change No.	Chapter 15 Section 15.6	Change Summary Description
[15.6-26]	15.6.5.4A.6, Description of AP1000 Large-Break LOCA Transient	Values updated due to ASTRUM methodology.
[15.6-27]	15.6.5.4B.1 Description of Small-break LOCA Transient	Editorial changes incorporated to clarify section.
[15.6-28]	15.6.5.4B, Small-break LOCA Analyses	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), increased pressurizer volume, increased RV diameter for the neutron pad addition, use of the digital ΔT signal, the updated reactor coolant pump flywheel material and the updated valve, nozzle and piping pressure loss coefficients.
[15.6-29]	15.6.5.4B.1 Description of Small-break LOCA Transient	Editorial changes incorporated to clarify section.
[15.6-30]	15.6.5.4B.1 Description of Small-break LOCA Transient	Minimum value replaced with the nominal value since the ASTRUM methodology uses the range of input values. Therefore the nominal value is more representative.
[15.6-31]	15.6.5.4B.2.1 NOTRUMP Computer Code	Main feedwater flow can support a 1% uncertainty. It is permissible to only model the uncertainty associated with the calorimetric measurement. In reality the main feedwater flow measurement supports a calorimetric uncertainty of 1%.
[15.6-32]	15.6.5.4B.2.1 NOTRUMP Computer Code	Value updated consistent with the 5.3 second pump delay plus a 2 second signal processing delay.
[15.6-33]	15.6.5.4B.2.1.1 AP1000 Model-Detailed Noding	This resistance increase is due to finalized fuel design and RCS piping design. The overall change is small from 70% to 82%.
[15.6-34]	15.6.5.4B.2.3 Critical Heat Flux Assessment During Accumulator Injection	Values updated to account for the revised pressurizer diameter and height and updated line resistance calculations.
[15.6-35]	15.6.5.4B.2.3 Critical Heat Flux Assessment During Accumulator Injection	See Change No. [15.6-34]

Change No.	Chapter 15 Section 15.6	Change Summary Description
[15.6-36]	15.6.5.4B.3.1 Introduction	Editorial changes incorporated to clarify section.
[15.6-37]	15.6.5.4B.3.3 Inadvertent Actuation of Automatic Depressurization System	See Change No. [15.6-31]
[15.6-38]	15.6.5.4B.3.3 Inadvertent Actuation of Automatic Depressurization System	Timer delays have been updated as a result of changes to the valve stroke time. The timer delays were updated to make the valve stroke time changes transparent to the analyses.
[15.6-39]	15.6.5.4B.3.3 Inadvertent Actuation of Automatic Depressurization System	See Change No. [15.6-32]
[15.6-40]	15.6.5.4B.3.4 2-inch Cold Leg Break in the Core Makeup Tank Loop	See Change No. [15.6-32]
[15.6-41]	15.6.5.4B.3.4 2-inch Cold Leg Break in the Core Makeup Tank Loop	Since the PXS is not the RCS, the PXS mass should not be considered in the RCS.
[15.6-42]	15.6.5.4B.3.5 Direct Vessel Injection Line Break	Added to clarify which CMT is being discussed.
[15.6-43]	15.6.5.4B.3.5 Direct Vessel Injection Line Break	See Change No. [15.6-32]
[15.6-44]	15.6.5.4B.3.5 Direct Vessel Injection Line Break	Added to clarify what is being depicted in the cited figure.
[15.6-45]	15.6.5.4B.3.5 Direct Vessel Injection Line Break	Added to provide additional clarification.
[15.6-46]	15.6.5.4B.3.5 Direct Vessel Injection Line Break	Updates of detailed line resistances causes more injection flow, or less core exit flow from ADS 1-3 could cause downcomer level to remain fairly constant during this time period.
[15.6-47]	15.6.5.4B.3.5 Direct Vessel Injection Line Break	Added to provide additional clarification.
[15.6-48]	15.6.5.4B.3.6 10-inch Cold Leg Break	See Change No. [15.6-32]

Change No.	Chapter 15 Section 15.6	Change Summary Description
[15.6-49]	15.6.5.4B.3.6 10-inch Cold Leg Break	Due to increased ADS-4 entrainment from increased resistance calculation shown above.
[15.6-50]	15.6.5.4B.3.6 10-inch Cold Leg Break	The predictor for the onset of core boiling (x>90%) does not occur in the updated transient, therefore this paragraph is no longer applicable.
[15.6-51]	15.6.5.4B.3.6 10-inch Cold Leg Break	Updated to reflect the results of the revised analysis.
[15.6-52]	15.6.5.4B.3.7 Direct Vessel Injection Line Break (Entrainment Sensitivity)	Wording updated to provide additional clarification.
[15.6-53]	15.6.5.4B.4, Conclusions	Added to clarify that this is only applicable to small break LOCAs.
[15.6-54]	15.6.5.4B.4, Conclusions	Compilation of the integrated design changes for this analysis. Namely, RCP delay times, updated line resistances, PZR geometry change. The integrated changes were not evaluated separately, therefore it is not possible to pinpoint which change contributed to the variances, only that the analysis was done in accordance with the approved licensed methodology.
[15.6-55]	15.6.5.4B.4, Conclusions	Updated based on results of DEDVI entrainment study.
[15.6-56]	15.6.5.4C.2, DEDVI Line Break with ADS Stage 4 Single Failure, Passive Core Cooling System Only Case; Continuous Case	Value updated because of IRWST initial conditions and piping conditions.
[15.6-57]	15.6.5.4C.2, DEDVI Line Break with ADS Stage 4 Single Failure, Passive Core Cooling System Only Case; Continuous Case	See Change No. [15.6-41] and [15.6-56].
[15.6-58]	15.6.5.4C.2, DEDVI Line Break with ADS Stage 4 Single Failure, Passive Core Cooling System Only Case; Continuous Case	The DCD is not an appropriate place for the Sensitivity runs provided here and have therefore been removed.
[15.6-59]	15.6.5.4C.3, DEDVI Break and Wall-to-Wall Floodup; Containment Recirculation	The longer equilibration time reduces uncertainty in the equilibrium conditions.

Change No.	Chapter 15 Section 15.6	Change Summary Description
[15.6-60]	15.6.5.4C.3, DEDVI Break and Wall-to-Wall Floodup; Containment Recirculation	See Change No. [15.6-41] and [15.6-56].
[15.6-61]	15.6.5.4C.3, DEDVI Break and Wall-to-Wall Floodup; Containment Recirculation	See Change No. [15.6-41] and [15.6-56].
[15.6-62]	Tables and Figures	Tables and figures have been updated to reflect the results of the revised analysis. Unless noted below, refer to the individual sections for additional details regarding changes incorporated.
[15.6-63]	Table 15.6.2-1	A more conservative method of calculating the flashing fraction was applied. Vessel outlet temperature was used in place of vessel average temperature. This is conservative.
[15.6-64]	Table 15.6.3-1	Sequence of Events updated to reflect revised SGTR analysis
[15.6-65]	Table 15.6.3-2	SGTR Mass releases updated to reflect mass releases from revised SGTR analysis.
[15.6-66]	Table 15.6.3-3	Spike duration recalculated based on revised source terms. RCS mass updated based on revised NSSS models. Steam release duration updated based on revised analysis. Ruptured and intact SG masses data updated based on updated values modeled in the analysis. Alkali metal partition factor updated to be consistent with moisture carryover.
[15.6-67]	Table 15.6.5-2 (sheets 1 through 3)	Coolant mass updated based on revised NSSS models. Containment purge rate updated to reflect the value modeled in the analysis.
[15.6-68]	Table 15.6.5-3	Doses and limiting 2-hour intervals updated based on revised source terms for the Advanced First Core.
[15.6-69]	Figure 15.6.3-1 through 15.6.3-10	Figures are updated based on the revised SGTR analysis.
[15.6-70]	15.6.5.4A	The Large-Break LOCA section was updated in Revision 1 to address the effects of thermal conduictivity degradation as described in response to CRR-008.
[15.6-71]	15.6.5.4B	The Small-Break LOCA section was updated in Revision 1 to address a change in the assumptions used in the analysis. A discussion on the changes contained in this section are described in Section 5.
[15.6-72]	Table 15.6.5-3	The results in this table have been removed. Additional information on this change is described in Section 5.

15.6 Decrease in Reactor Coolant Inventory

This section discusses the following events that result in a decrease in reactor coolant inventory:

- An inadvertent opening of a pressurizer safety valve or inadvertent operation of the automatic depressurization system (ADS)
- A break in an instrument line or other lines from the reactor coolant pressure boundary that penetrate the containment
- A steam generator tube failure
- A loss-of-coolant accident (LOCA) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary

The applicable accidents in this category have been analyzed. It has been determined that the most severe radiological consequences result from the major LOCA described in subsection 15.6.5. The LOCA, chemical and volume control system letdown line break outside the containment and the steam generator tube rupture (SGTR) accidents are analyzed for radiological consequences. Other accidents described in this section are bounded by these accidents.

15.6.1 Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS

Comment [B1]: [15.6-1]

15.6.1.1 Identification of Causes and Accident Description

Two types of inadvertent depressurization are discussed in this section. One covers the inadvertent operation of automatic depressurization system (ADS) valves. The other covers inadvertent opening of a pressurizer safety valve.

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An inadvertent depressurization of the reactor coolant system can occur as a result of an inadvertent opening of a pressurizer safety valve or ADS valves. Initially, the event results in a rapidly decreasing reactor coolant system pressure. The pressure decrease causes a decrease in power via the moderator density feedback. The average coolant temperature decreases slowly, but the pressurizer level increases until reactor trip.

The reactor may be tripped by the following reactor protection system signals:

- Overtemperature ΔT
- Pressurizer low pressure

The ADS is designed such that inadvertent operation of the ADS is classified as a Condition III event, an infrequent fault. An inadvertent opening of a pressurizer safety valve is a Condition II event, a fault of moderate frequency.

The ADS system consists of four stages of depressurization valves. The ADS stages are interlocked. For example, Stage 1 is initiated first and subsequent stages are not actuated until previous stages have completed actuation. Each stage includes two redundant parallel valve paths with two valves in series in each path such that no single failure prevents operation of the ADS stage when it is called upon to actuate and the spurious opening of a single ADS valve does not initiate ADS flow. Since each ADS path includes two valves in series, no mechanical failure could result in an inadvertent operation of an ADS stage. The ADS Stage 4 squib valves cannot be opened while the reactor coolant system is at nominal operating pressure. To actuate the ADS manually from the main control room, the operators actuate two separate controls positioned at some distance apart on the main control board. Therefore, one unintended operator action does not cause ADS actuation.

ADS Stage 1 has a minimum opening time of 20 seconds and a maximum effective flow area of 7 in^2 (maximum). ADS Stages 2 and 3 have a minimum opening time of 60 seconds and a maximum effective flow area of 28 in^2 Q.

For this analysis, multiple failures and or errors are assumed which actuate both Stage 1 ADS paths. Although ADS Stages 2 and 3 have larger depressurization valves, the opening time of the Stage 1 depressurization valves is faster. This results in a more severe reactor coolant system depressurization due to ADS operation with the reactor at power.

Inadvertent opening of a pressurizer safety valve can only be postulated due to a mechanical failure. Although a pressurizer safety valve is smaller than the combined two Stage 1 ADS valves, the pressurizer safety valve is postulated to open in a short time.

Analyses are presented in this section for the inadvertent opening of a pressurizer safety valve and the inadvertent opening of two paths of Stage 1 of the ADS. These analyses are performed to demonstrate that the departure from nucleate boiling ratio (DNBR) does not decrease below the design limit values (see Section 4.4) while the reactor is at power.

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, the effects of a possible consequential loss of AC power during an RCS Depressurization event have been evaluated to not adversely impact the analysis results. This conclusion is based on a review of the time sequence associated with a consequential loss of AC power in comparison to the reactor shutdown time for an RCS Depressurization event.. The primary effect of the loss of AC power is to cause the Reactor Coolant Pumps (RCPs) to coast down. The Protection & Safety Monitoring System (PMS) includes a five second minimum delay between the reactor trip and the turbine

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trip. In addition, a three second delay between the turbine trip and the loss of offsite AC power is assumed, consistent with the discussion of Section 15.0.14. Considering these delays between the time of the reactor trip and RCP coastdown due to the loss of AC power, it is clear that the plant shutdown sequence will have passed the critical point and the control rods will have been completely inserted before the RCPs begin to coast down. Therefore, the consequential loss of AC power does not adversely impact this analysis because the plant will be shut down well before the RCPs begin to coast down.

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15.6.1.2 Analysis of Effects and Consequences

15.6.1.2.1 Method of Analysis

The accidental depressurization transient is analyzed by using the computer code LOFTRAN (References 14 and 15). The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, main steam isolation valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

Plant characteristics and initial conditions are discussed in subsection 15.0.3. The following assumptions are made to give conservative results in calculating the DNBR during the transient:

Comment [B7]: [15.6-7]

Deleted: For reactor coolant system depressurization analyses that include a primary coolant flow coastdown caused by a consequential loss of offsite power, a combination of three computer codes is used to perform the DNBR analyses. First the LOFTRAN code is used to perform the plant system transient. The FACTRAN code (Reference 18) is then used to calculate the core heat flux based on nuclear power and reactor coolant flow from LOFTRAN. Finally, the VIPRE-01 code (see Section 4.4) is used to calculate the DNBR using heat flux from FACTRAN and flow from LOFTRAN.

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- Initial conditions are discussed in subsection 15.0.3. Uncertainties in initial conditions are included in the DNBR limit as discussed in WCAP-11397-P-A (Reference 16).

- A least negative moderator temperature coefficient is assumed. The spatial effect of voids resulting from local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape.

- A large (absolute value) Doppler coefficient of reactivity is used such that the resulting amount of positive feedback is conservatively high to retard any power decrease.

Plant systems and equipment necessary to mitigate the effects of reactor coolant system depressurization are discussed in subsection 15.0.8 and are listed in Table 15.0-6.

Normal reactor control systems are not required to function. The rod control system is assumed to be in the automatic mode to maintain the core at full power until the reactor trip protection function is reached. This is a worst case assumption. The reactor protection system functions to trip the reactor on the appropriate signal. No single active failure prevents the reactor protection system from functioning properly.

15.6.1.2.2 Results

The system response to an inadvertent opening of a pressurizer safety valve is shown in Figures 15.6.1-1 through 15.6.1-4. The calculated sequence of events for the inadvertent opening of a pressurizer safety valve scenario is shown in Table 15.6.1-1.

A pressurizer safety valve is assumed to step open at the start of the event. The reactor coolant system then depressurizes until the low pressurizer pressure reactor trip setpoint is reached. Figure 15.6.1-3 shows the pressurizer pressure transient.

Prior to tripping of the reactor, the core power remains relatively constant (Figure 15.6.1-1). The minimum DNBR during the event occurs shortly after the rods begin to be inserted into the core (Figure 15.6.1-2). The DNBR remains above the design limit values as discussed in Section 4.4 throughout the transient.

The system response for inadvertent operation of the ADS is shown in Figures 15.6.1-5 through 15.6.1-8. The sequence of events is provided in Table 15.6.1-1. The system response for inadvertent operation of the ADS is very similar to that obtained for inadvertent opening of a pressurizer safety valve.

15.6.1.3 Conclusion

The results of the analysis show that the low pressurizer pressure reactor protection system signal provides adequate protection against the reactor coolant system depressurization events. The calculated DNBR remains above the design limit defined in Section 4.4. The long-term plant responses due to a stuck-open ADS valve or pressurizer safety valve, which cannot be isolated, are bounded by the small-break LOCA analysis.

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

The small lines carrying primary coolant outside containment are the reactor coolant system sample line and the discharge line from the chemical and volume control system to the liquid radwaste system. These lines are used only periodically. No instrument lines carry primary coolant outside the containment.

When excess primary coolant is generated because of boron dilution operations, the chemical and volume control system purification flow is diverted out of containment to the liquid radwaste system. Before passing outside containment, the flow stream passes through the chemical and volume control system heat exchangers and mixed bed demineralizer. The flow leaving the containment is at a temperature of less than 140°F and has been cleaned by the demineralizer. The flow out a postulated break in this line is limited to the chemical and volume control system purification flow rate of 100 gpm. Considering the low temperature of the flow and the reduced

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iodine activity because of demineralization, this event is not analyzed. The postulated sample line break is more limiting.

The sample line isolation valves inside and outside containment are open only when sampling. The failure of the sample line is postulated to occur between the isolation valve outside the containment and the sample panel. Because the isolation valves are open only when sampling, the loss of sample flow provides indication of the break to plant personnel. In addition, a break in a sample line results in activity release and a resulting actuation of area and air radiation monitors. The loss of coolant reduces the pressurizer level and creates a demand for makeup to the reactor coolant system. Upon indication of a sample line break, the operator would take action to isolate the break.

The sample line includes a flow restrictor at the point of sample to limit the break flow to less than 130 gpm. The liquid sampling lines are 1/4 inch tubing which further restricts the break flow of a sampling line outside containment. Offsite doses are based on a conservative break flow of 130 gpm with isolation after 30 minutes.

15.6.2.1 Source Term

The only significant radionuclide releases are the iodines and the noble gases. The analysis assumes that the reactor coolant iodine is at the maximum Technical Specification level for continuous operation. In addition, it is assumed that an iodine spike occurs at the time of the accident. The reactor coolant noble gas concentrations are assumed to be those associated with equilibrium operating limits for primary coolant noble gas activity.

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15.6.2.2 Release Pathway

The reactor coolant that is spilled from the break is assumed to be at high temperature and pressure. A large portion of the flow flashes to steam, and the iodine in the flashed liquid is assumed to become airborne.

The iodine and noble gases are assumed to be released directly to the environment with no credit for depletion, although a large fraction of the airborne iodine is expected to deposit on building surfaces. No credit is assumed for radioactive decay after release.

15.6.2.3 Dose Calculation Models

The models used to calculate doses are provided in Appendix 15A.

15.6.2.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.6.2-1.

15.6.2.5 Identification of Conservatisms

The assumptions used contain the following significant conservatisms:

- The reactor coolant activities are based on conservative assumptions (See Table 15.6.2-1); whereas, the expected activities based on the fuel defect level are far less (see Section 11.1).
- It is unlikely that the conservatively selected meteorological conditions would be present at the time of the accident.

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15.6.2.6 Doses

Using the assumptions from Table 15.6.2-1, the calculated total effective dose equivalent (TEDE) doses are determined to be 1.3 rem at the exclusion area boundary and 0.6 rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. The phrase "a small fraction" is taken as being ten percent or less.

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At the time the accident occurs, there is the potential for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because pool boiling would not occur until after 2 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE and, when this is added to the dose calculated for the small line break outside containment, the resulting total dose remains less than the value reported above.

15.6.3 Steam Generator Tube Rupture

Comment [B14]: [15.6-14]

15.6.3.1 Identification of Cause and Accident Description

15.6.3.1.1 Introduction

The accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited number of defective fuel rods within the allowance of the Technical Specifications. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the reactor coolant system. In the event of a coincident loss of offsite power, or a failure of the condenser steam dump, discharge of radioactivity to the atmosphere takes place via the steam generator power-operated relief valves or the safety valves.

The assumption of a complete tube severance is conservative because the steam generator tube material (Alloy 690) is a corrosion-resistant and ductile material. The more probable mode of tube failure is one or more smaller leaks of undetermined origin. Activity in the secondary side is subject to continual surveillance, and an accumulation of such leaks, which exceeds the limits established in the Technical Specifications, is not permitted during operation.

The AP1000 design provides automatic protective actions to mitigate the consequences of an SGTR. The automatic actions include reactor trip, actuation of the passive residual heat removal (PRHR) heat exchanger, initiation of core makeup tank flow, termination of pressurizer heater operation, and isolation of chemical and volume control system flow and startup feedwater flow on high-2 steam generator level or high steam generator level coincident with reactor trip (P-4). These protective actions result in automatic cooldown and depressurization of the reactor coolant system, termination of the break flow and release of steam to the atmosphere, and long-term maintenance of stable conditions in the reactor coolant system. These protection systems serve to prevent steam generator overfill (see discussion in subsections 15.6.3.1.2 and 15.6.3.1.3) and to maintain offsite radiation doses within the allowable guideline values for a design basis SGTR. The operator may take actions that would provide a more rapid mitigation of the consequences of an SGTR.

Because of the series of alarms described next, the operator can readily determine when an SGTR occurs, identify and isolate the ruptured steam generator, and complete the required recovery actions to stabilize the plant and terminate the primary-to-secondary break flow. The recovery procedures are completed on a time scale that terminates break flow to the secondary system before steam generator overfill occurs and limits the offsite doses to acceptable levels without actuation of the ADS. Indications and controls are provided to enable the operator to carry out these functions.

15.6.3.1.2 Sequence of Events for a Steam Generator Tube Rupture

The following sequence of events occur following an SGTR:

- Pressurizer low pressure and low level alarms are actuated and chemical and volume control system makeup flow and pressurizer heater heat addition starts or increases in an attempt to maintain pressurizer level and pressure. On the secondary side, main feedwater flow to the affected steam generator is reduced because the primary-to-secondary break flow increases steam generator level.

- The condenser air removal discharge radiation monitor, steam generator blowdown radiation monitor, and/or main steam line radiation monitor alarm indicate an increase in radioactivity in the secondary system.

- Continued loss of reactor coolant inventory leads to a reactor trip generated by a low pressurizer pressure or over-temperature ΔT signal. Following reactor trip, the SGTR leads to a decrease in reactor coolant pressure and pressurizer level, counteracted by chemical and volume control system flow and pressurizer heater operation. A safeguards ("S") signal from low pressurizer pressure, actuates the core makeup tanks. The "S" signal automatically terminates the normal feedwater supply and trips the reactor coolant pumps. The core makeup tank actuation signal will actuate the PRHR heat exchanger and trip pressurizer heaters. Startup feedwater flow is initiated on a low steam generator narrow range level signal and controls the steam generator levels to the programmed level.

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Deleted: and PRHR heat exchanger actuation is initiated by low pressurizer pressure or low-2 pressurizer level.

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- The reactor trip automatically trips the turbine, and if offsite power is available, the steam dump valves open permitting steam dump to the condenser. In the event of a loss of offsite power or loss of the condenser, the steam dump valves automatically close to protect the condenser. The steam generator pressure rapidly increases resulting in steam discharge to the atmosphere through the steam generator power-operated relief valves and/or the safety valves.

- Following reactor trip and core makeup tank and PRHR actuation, the PRHR heat exchanger operation – combined with startup feedwater flow, borated core makeup tank flow, and chemical and volume control system flow – provides a heat sink that absorbs the decay heat. This reduces the amount of steam generated in the steam generators and steam bypass to the condenser. In the case of loss of offsite power, this reduces steam relief to the atmosphere.

- Injection of the chemical and volume control system and core makeup tank flow stabilizes reactor coolant system pressure and pressurizer water level, and the reactor coolant system pressure trends toward an equilibrium value, where the total injected flow rate equals the break flow rate.

15.6.3.1.3 Steam Generator Tube Rupture Automatic Recovery Actions

The AP1000 incorporates several protection system and passive design features that automatically terminate a steam generator tube leak and stabilize the reactor coolant system, in the highly unlikely event that the operators do not perform recovery actions. Following an SGTR, the injecting chemical and volume control system flow (and pressurizer heater heat addition if the pressure control system is operating) maintains the primary-to-secondary break flow and the ruptured steam generator secondary level increases as break flow accumulates in the steam generator. Eventually, the ruptured steam generator secondary level reaches the high and high-2 steam generator narrow range level setpoint, which is near the top of the narrow range level span.

The AP1000 protection system automatically provides several safety-related actions to cool down and depressurize the reactor coolant system, terminate the break flow and steam release to

the atmosphere, and stabilize the reactor coolant system in a safe condition. The safety-related actions include initiation of the PRHR system heat exchanger, isolation of the chemical and volume control system pumps and pressurizer heaters, and isolation of the startup feedwater pumps. In addition, the protection and safety monitoring system provides a safety-related signal to trip the redundant, nonsafety related pressurizer heater breakers.

Actuating the PRHR heat exchanger transfers core decay heat to the in-containment reactor water storage tank (IRWST) and initiates a cooldown (and a consequential depressurization) of the reactor coolant system.

Isolation of the chemical and volume control system pumps and pressurizer heaters minimizes the repressurization of the primary system. This allows primary pressure to equilibrate with the secondary pressure, which effectively terminates the primary-to-secondary break flow. Because the core makeup tank continues to inject when needed to provide boration following isolation of the chemical and volume control system pumps, isolating the chemical and volume control system pumps does not present a safety concern.

Isolation of the startup feedwater provides protection against a failure of the startup feedwater control system, which could potentially result in the ruptured steam generator being overfilled.

With decay heat removal by the PRHR heat exchanger, steam generator steaming through the power-operated relief valves ceases and steam generator secondary level is maintained.

15.6.3.1.4 Steam Generator Tube Rupture Assuming Operator Recovery Actions

In the event of an SGTR, the operators can diagnose the accident and perform recovery actions to stabilize the plant, terminate the primary-to-secondary leakage, and proceed with orderly shutdown of the reactor before actuation of the automatic protection systems. The operator actions for SGTR recovery are provided in the plant emergency operating procedures. The major operator actions include the following:

- Identify the ruptured steam generator – The ruptured steam generator can be identified by an unexpected increase in steam generator narrow range level or a high radiation indication from any main steam line monitor, steam generator blowdown line monitor, or steam generator sample.

- Isolate the ruptured steam generator – Once the steam generator with the ruptured tube is identified, recovery actions begin by isolating steam flow from and stopping feedwater flow to the ruptured steam generator.

- Cooldown of the reactor coolant system using the intact steam generator or the PRHR system – After isolation of the ruptured steam generator, the reactor coolant system is cooled

as rapidly as possible to less than the saturation temperature corresponding to the ruptured steam generator pressure. This provides adequate subcooling in the reactor coolant system after depressurization of the reactor coolant system to the ruptured steam generator pressure in subsequent actions.

- Depressurize the reactor coolant system to restore reactor coolant inventory – When the cooldown is completed, the chemical and volume control system and core makeup tank injection flow increases the reactor coolant system pressure until break flow matches the total injection flow. Consequently, these flows must be terminated or controlled to stop primary-to-secondary leakage. However, adequate reactor coolant inventory must first be provided. This includes both sufficient reactor coolant subcooling and pressurizer inventory to maintain a reliable pressurizer level indication after the injection flow is stopped.

Because leakage from the primary side continues after the injection flow is stopped, until reactor coolant system and ruptured steam generator pressures equalize, the reactor coolant system is depressurized to provide sufficient inventory to verify that the pressurizer level remains on span after the pressures equalize.

- Termination of the injection flow to stop primary to secondary leakage – The previous actions establish adequate reactor coolant system subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to verify that injection flow is no longer needed. When these actions are completed, core makeup tank and chemical and volume control system flow is stopped to terminate primary-to-secondary leakage. Primary-to-secondary leakage continues after the injection flow is stopped until the reactor coolant system and ruptured steam generator pressures equalize. Chemical and volume control system makeup flow, letdown, pressurizer heaters, and decay heat removal via the intact steam generator or the PRHR heat exchanger are then controlled to prevent repressurization of the reactor coolant system and reinitiation of leakage into the ruptured steam generator.

Following the injection flow termination, the plant conditions stabilize and the primary-to-secondary break flow terminates. At this time, a series of operator actions is performed to prepare the plant for cooldown to cold shutdown conditions. The actions taken depend on the available plant systems and the plan for further plant repair and operation.

15.6.3.2 Analysis of Effects and Consequences

An SGTR results in the leakage of contaminated reactor coolant into the secondary system and subsequent release of a portion of the activity to the atmosphere. An analysis is performed to demonstrate that the offsite radiological consequences resulting from an SGTR are within the allowable guidelines.

One of the concerns for an SGTR is the possibility of steam generator overfill because this can potentially result in a significant increase in the offsite radiological consequences. Automatic protection and passive design features are incorporated into the AP1000 design to automatically terminate the break flow to prevent overfill during an SGTR. These features include actuation of the PRHR system, isolation of chemical and volume control system flow, and isolation of startup feedwater.

An analysis is performed, without modeling expected operator actions to isolate the ruptured steam generator and cool down and depressurize the reactor coolant system, to demonstrate the role that the AP1000 design features have in preventing steam generator overfill. The limiting single failure for the overfill analysis is assumed to be the failure of the startup feedwater control valve to throttle flow when nominal steam generator level is reached. Other conservative assumptions that maximize steam generator secondary volume (such as high initial steam generator level, minimum initial reactor coolant system pressure, loss of offsite power, maximum chemical and volume control system injection flow, maximum pressurizer heater addition, maximum startup feedwater flow, and minimum startup feedwater delay time) are also assumed.

The results of this analysis demonstrate the effectiveness of the AP1000 protection system and passive system design features and support the conclusion that an SGTR event would not result in steam generator overfill.

For determining the offsite radiological consequences, an SGTR analysis is performed assuming the limiting single failure and limiting initial conditions relative to offsite doses. Because steam generator overfill is prevented for the AP1000, the results of this analysis represent the limiting radiological consequences for an SGTR.

A thermal-hydraulic analysis is performed to determine the plant response for a design basis SGTR, the integrated primary-to-secondary break flow, and the mass releases from the ruptured and intact steam generators to the condenser and to the atmosphere. This information is then used to calculate the radioactivity release to the environment and the resulting radiological consequences.

15.6.3.2.1 Method of Analysis

15.6.3.2.1.1 Computer Program

The plant response following an SGTR until the primary-to-secondary break flow is terminated is analyzed with the LOFTTR2 program (Reference 21). The LOFTTR2 program is modified to model the PRHR system, core makeup tanks, and protection system actions appropriate for the AP1000. These modifications to LOFTTR2 are described in WCAP-14234, Revision 1 (Reference 14).

15.6.3.2.1.2 Analysis Assumptions

The accident modeled is a double-ended break of one steam generator tube located at the top of the tube sheet on the outlet (cold leg) side of the steam generator. The location of the break on the cold leg side of the steam generator results in higher initial primary-to-secondary leakage than a break on the hot side of the steam generator.

The reactor is assumed to be operating at full power at the time of the accident, and the initial secondary mass is assumed to correspond to operation at nominal steam generator mass minus an allowance for uncertainties. Offsite power is assumed to be lost and the rods are assumed to be inserted at the start of the event because continued operation of the reactor coolant pumps has been determined to reduce flashing of primary-to-secondary break flow and, consequently, lower offsite radiological doses. Maximum chemical and volume control system flows and pressurizer heater heat addition are assumed immediately (even though offsite power is not available) to conservatively maximize primary-to-secondary leakage. The steam dump system is assumed to be inoperable, consistent with the loss of offsite power assumption, because this results in steam release from the steam generator power-operated relief valves to the atmosphere following reactor trip. The chemical and volume control system and pressurizer heater modeling is conservatively chosen to delay the low pressurizer pressure "S" and the low-2 pressurizer level signal and associated protection system actions.

The limiting single failure is assumed to be the failure of the ruptured steam generator power-operated relief valve. Failure of this valve in the open position causes an uncontrolled depressurization of the ruptured steam generator, which increases primary-to-secondary leakage and the mass release to the atmosphere.

It is assumed that the ruptured steam generator power-operated relief valve fails open when the low-2 pressurizer level signal is generated. This results in the maximum integrated flashed primary-to-secondary break flow.

The valve is subsequently isolated when the associated block valve is automatically closed on a low steam line pressure protection system signal.

No operator actions are modeled in this limiting analysis, and the plant protection system provides the protection for the plant. Not modeling operator actions is conservative because the operators are expected to have sufficient time to recover from the accident and supplement the automatic protection system. In particular, the operator would take action to reduce the primary pressure before the high steam generator level coincident with reactor trip (P-4) chemical and volume control and startup feedwater system shutoff signals are generated. It is also expected that the operator can close the block valve to the ruptured steam generator power-operated relief valve in much shorter time than the automatic protection signal. The operators can quickly

diagnose a power-operated relief valve failure based on the rapid depressurization of the steam generator and increase in steam flow. They can then close the block valve from the control panel.

Consistent with the assumed loss of offsite power, the main feedwater pumps coast down and no startup feedwater is assumed to conservatively minimize steam generator secondary inventory and thus maximize secondary activity concentration and steam release.

15.6.3.2.1.3 Results

The sequence of events for this transient is presented in Table 15.6.3-1. The system responses to the SGTR accident are shown in Figures 15.6.3-1 to 15.6.3-10.

Offsite power is lost concurrent with the rupture of the tube. The reactor trips due to the loss of offsite power. The main feedwater pumps are assumed to coast down following reactor trip. The startup feedwater pumps are conservatively assumed not to start. Following the tube rupture, reactor coolant flows from the primary into the secondary side of the ruptured steam generator. In response to this loss of reactor coolant, pressurizer level and reactor coolant system pressure decreases as shown in Figures 15.6.3-1 and 15.6.3-2. As a result of the decreasing pressurizer level and pressure, two chemical and volume control system pumps are automatically initiated to provide makeup flow and the pressurizer heaters turn on.

After reactor trip, core power rapidly decreases to decay heat levels and the core inlet to outlet temperature differential decreases. The turbine stop valves close, and steam flow to the turbine is terminated. The steam dump system is conservatively assumed to be inoperable. The secondary side pressure increases rapidly after reactor trip until the steam generator power-operated relief valves (and safety valves, if their setpoints are reached) lift to dissipate the energy, as shown in Figure 15.6.3-3.

Maximum heat addition to the pressurizer from the pressurizer heaters increases the primary pressure.

As the leak flow continues to deplete primary inventory, low pressurizer level "S" and core makeup tank and PRHR actuation signals are reached. Power to the pressurizer heaters is shut off so that they will not provide additional heat to the primary should the pressurizer level return. The ruptured steam generator power-operated relief valve is assumed to fail open at this time.

The failure causes the intact and ruptured steam generators to rapidly depressurize (Figure 15.6.3-3). This results in an initial increase in primary-to-secondary leakage and a decrease in the reactor coolant system temperatures. Both the intact and ruptured steam generators depressurize because the steam generators communicate through the open steam line isolation valves.

The decrease in the reactor coolant system temperature results in a decrease in the pressurizer level and reactor coolant system pressure (Figures 15.6.3-1 and 15.6.3-2). Depressurization of the primary and secondary systems continues until the low steam line pressure setpoint is reached. As a result, the steam line isolation valves and intact and ruptured steam generator power-operated relief block valves are closed.

Following closure of the block valves, the primary and secondary pressures and the ruptured steam generator secondary water volume and mass increase as break flow accumulates. This increase continues until the steam generator secondary level reaches the high narrow range level when the chemical and volume control and startup feedwater systems are isolated.

With continued reactor coolant system cooldown, depressurization provided by the PRHR heat exchanger, and with the chemical and volume control system isolated, primary system pressure eventually falls to match the secondary pressure. The break flow terminates as shown in Figure 15.6.3-5, and the system is stabilized in a safe condition. As shown in Figure 15.6.3-8, steam release through the intact loop, unfaulted power-operated relief valve does not occur following PRHR initiation because the PRHR is capable of removing the core decay heat.

As shown in Figure 15.6.3-9, the core makeup tank flow trends toward zero because the gravity head diminishes as the core makeup tank temperature approaches the reactor coolant system temperature due to the continued balance line flow. The core makeup tank remains full, and ADS actuation does not occur.

The ruptured steam generator water volume is shown in Figure 15.6.3-6. The water volume in the ruptured steam generator when the break flow is terminated is significantly less than the total steam generator volume of greater than 9000 ft^3

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The design basis SGTR event does not result in fuel failures. In the event of an SGTR, the reactor coolant system depressurizes due to the primary-to-secondary leakage through the ruptured steam generator tube. This depressurization reduces the calculated DNBR. The depressurization prior to reactor trip for the SGTR has been compared to the depressurization for the reactor coolant system depressurization accidents analyzed in subsection 15.6.1. The rate of depressurization is much slower for the SGTR than for the reactor coolant system depressurization accidents. Following reactor trip, the DNBR increases rapidly. Thus, the conclusion of subsection 15.6.1, that the calculated DNBR remains above the limit, is extended to the SGTR analysis, justifying the assumption of no failed fuel.

15.6.3.2.1.4 Mass Releases

The mass release of an SGTR event is determined for use in evaluating the exclusion area boundary and low population zone radiation exposure. The steam releases from the ruptured and

intact steam generators and the primary-to-secondary leakage into the ruptured steam generator are determined from the LOFTTR2 results for the period from the initiation of the accident until the leakage is terminated.

Following reactor trip, the releases to the atmosphere are through the steam generator power-operated relief valves (and steam generator safety valves for a short period). Steam relief through the power-operated relief valves continues until RNS conditions are met. The mass releases for the SGTR event are presented in Table 15.6.3-2.

15.6.3.3 Radiological Consequences

Comment [B15]: [15.6-15]

The evaluation of the radiological consequences of the postulated SGTR assumes that the reactor is operating with a limited number of fuel rods containing cladding defects and that leaking steam generator tubes result in a buildup of activity in the secondary coolant.

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Following the rupture, any noble gases carried from the primary coolant into the ruptured steam generator via the break flow are released directly to the environment. The iodine and alkali metal activity entering the secondary side is also available for release, with the amount of release dependent on the flashing fraction of the reactor coolant and on the partition coefficient in the steam generator. In addition to the activity released through the ruptured loop, there is also a small amount of activity released through the intact loop.

15.6.3.3.1 Source Term

The significant radionuclide releases from the SGTR are the noble gases, alkali metals and the iodines that become airborne and are released to the environment as a result of the accident.

The analysis considers two different reactor coolant iodine source terms, both of which consider the iodine spiking phenomenon. In one case, the initial iodine concentrations are assumed to be those associated with the equilibrium operating limits for primary coolant iodine activity. The iodine spike is assumed to be initiated by the accident with the spike causing an increasing level of iodine in the reactor coolant.

The second case assumes that the iodine spike occurs before the accident and that the maximum reactor coolant iodine concentration exists at the time the accident occurs. The reactor coolant noble gas concentrations are assumed to be those associated with equilibrium operating limits for primary coolant noble gas activity. The reactor coolant alkali metal concentrations are assumed to be those associated with the design fuel defect level.

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The secondary coolant iodine and alkali metal activity is assumed to be 10 percent of the maximum equilibrium primary coolant activity.

15.6.3.3.2 Release Pathways

The noble gas activity contained in the reactor coolant that leaks into the intact steam generator and enters the ruptured steam generator through the break is assumed to be released immediately as long as a pathway to the environment exists. There are three components to the modeling of iodine and alkali metal releases:

- Intact loop steaming, with credit for partitioning of iodines and alkali metals (includes continued primary-to-secondary leakage at the maximum rate allowable by the Technical Specifications)
- Ruptured loop steaming, with credit for partitioning of iodines and alkali metals (includes modeling of increasing activity in the secondary coolant due to the break flow)
- Release of flashed reactor coolant through the ruptured loop, with no credit for scrubbing (this conservatively assumes that break location is at the top of the tube bundle)

Credit is taken for decay of radionuclides until release to the environment. After release to the environment, no consideration is given to radioactive decay or to cloud depletion of iodines by ground deposition during transport offsite.

15.6.3.3.3 Dose Calculation Models

The models used to calculate doses are provided in Appendix 15A.

15.6.3.3.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.6.3-3.

15.6.3.3.5 Identification of Conservatisms

The assumptions used in the analysis contain a number of significant conservatisms, such as:

- The reactor coolant activities are based on conservative assumptions whereas, the activities based on the expected fuel defect level are far less (see Section 11.1).
- It is unlikely that the conservatively selected meteorological conditions are present at the time of the accident.

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15.6.3.3.6 Doses

Using the assumptions from Table 15.6.3-3, the calculated TEDE doses for the case in which the iodine spike is assumed to be initiated by the accident are determined to be 0.6 rem at the

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exclusion area boundary for the limiting 2-hour interval (0-2 hours) 0.5 rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A "small fraction" is defined, consistent with the Standard Review Plan, as being ten percent or less.

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For the case in which the SGTR is assumed to occur coincident with a pre-existing iodine spike, the TEDE doses are determined to be 1.3 rem at the exclusion area boundary for the limiting 2-hour interval (0 to 2 hours) and 0.6 rem at the low population zone outer boundary. These doses are within the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34.

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At the time the accident occurs, there is the potential for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour exclusion area boundary dose because pool boiling would not occur until after 2.0 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE and, when this is added to the doses calculated for the steam generator tube rupture, the resulting total doses remain as reported above.

15.6.3.4 Conclusions

The results of the SGTR analysis show that the overfill protection logic and the passive system design features provide protection to prevent steam generator overfill. Following an SGTR accident, the operators can identify and isolate the ruptured steam generator and complete the required actions to terminate the primary-to-secondary break flow before steam generator overfill or ADS actuation occurs.

Even when no operator actions are assumed, the AP1000 protection system and passive design features initiate automatic actions that can terminate a steam generator tube leak and stabilize the reactor coolant system in a safe condition while preventing steam generator overfill and ADS actuation.

The resulting offsite radiological doses for the limiting case analyzed are within the dose acceptance limits.

15.6.4 Spectrum of Boiling Water Reactor Steam System Piping Failures Outside of Containment

This section is not applicable to the AP1000.

15.6.5 Loss-of-coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

15.6.5.1 Identification of Causes and Frequency Classification

A LOCA is the result of a pipe rupture of the reactor coolant system pressure boundary. For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is considered a Condition IV event (a limiting fault) because it is not expected to occur during the lifetime of the plant but is postulated as a conservative design basis (see subsection 15.0.1).

A minor pipe break (small break), as considered in this subsection, is defined as a rupture of the reactor coolant pressure boundary (Section 5.2) with a total cross-sectional area less than 1.0 ft^2 in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This is considered a Condition III event because it is an infrequent fault that may occur during the life of the plant.

The acceptance criteria for the LOCA are described in 10 CFR 50.46 (Reference 1) as follows:

- The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
- Localized cladding oxidation shall not exceed 17 percent of the total cladding thickness before oxidation.
- The amount of hydrogen generated from fuel element cladding reacting chemically with water or steam shall not exceed 1 percent of the total amount if all metal cladding were to react.
- The core remains amenable to cooling for any calculated change in core geometry.
- The core temperature is maintained at a low value, and decay heat is removed for the extended period of time required by the long-lived radioactivity remaining in the core.

These criteria are established to provide significant margin in emergency core cooling system performance following a LOCA.

For the AP1000, the small breaks (less than 1.0 ft^2) yield results with more margin than large breaks.

15.6.5.2 Basis and Methodology for LOCA Analyses

Should a major break occur, depressurization of the reactor coolant system results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low-pressure trip setpoint is reached. A safeguards actuation ("S") signal is generated when the appropriate setpoint is reached. These measures limit the consequences of the accident in two ways:

- Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. Insertion of control rods to shut down the reactor is neglected in the large-break analysis.

- Injection of borated water provides core cooling and prevents excessive cladding temperatures.

The acceptability of the computer codes approved for AP600 LOCA analyses for the AP1000 application is documented in Reference 24. The acceptability of additional computer codes for the AP1000 Best-Estimate Large-Break LOCA analysis is documented in Reference 34.

15.6.5.2.1 Description of Large-break LOCA Transient

Before the break occurs, the unit is in an equilibrium condition in which the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay stored energy in the fuel, hot internals, and vessel continues to be transferred to the reactor coolant. At the beginning of the blowdown phase, the entire reactor coolant system contains subcooled liquid, which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break, the core heat transfer is based upon local fluid conditions. Transition boiling and dispersed flow film boiling are the major heat transfer mechanisms.

The heat transfer between the reactor coolant system and the secondary system may be in either direction, depending upon the relative temperatures. In the case of continued heat addition to the secondary system, secondary system pressure increases and the main steam safety valves may lift to limit the pressure. The safety injection signal actuates a feedwater isolation signal, which isolates normal feedwater flow by closing the main feedwater isolation valves.

The reactor coolant pumps trip automatically during the accident following an "S" signal. The effects of pump coastdown are included in the blowdown. The blowdown phase of the transient ends when the reactor coolant system pressure (initially assumed at 2250 psia) falls to a value approaching that of the containment atmosphere.

When the "S" signal occurs, the core makeup tank isolation valves are opened. The core makeup tank begins to inject subcooled borated water into the reactor vessel through the direct vessel injection lines.

Subsection 15.6.5.4C presents calculations that show the effective post-LOCA long-term cooling of the AP1000 by passive means.

15.6.5.2.2 Description of Small-break LOCA Transient

The AP1000 includes passive safety features to prevent or minimize core uncovery during small-break LOCAs. The passive safety design approach of the AP1000 is to depressurize the reactor coolant system if the break or leak is greater than the makeup capability of the charging system. By depressurizing the reactor system, large volumes of borated water in the accumulators and in the IRWST become available for cooling the core. This analysis demonstrates that, with a single failure, the passive systems are capable of depressurizing the reactor coolant system while maintaining acceptable core conditions and establishing stable delivery of cooling water from the IRWST.

During a small-break LOCA, the AP1000 reactor coolant system depressurizes to the pressurizer low-pressure setpoint, actuating a reactor trip signal. The passive core cooling system is aligned for delivery following the generation of an "S" signal when the pressurizer low-pressure setpoint is reached. The passive core cooling system includes two core makeup tanks, two accumulators, a large IRWST, and the PRHR heat exchanger.

The core makeup tanks operate at reactor coolant system pressure. They provide high-pressure safety injection in the event of a small-break LOCA. The core makeup tanks share a common discharge line with the accumulators and IRWST; they are filled with borated water to provide core shutdown margin. The injection of the core makeup tanks is provided by gravity head of the colder water in the core makeup tanks. The core makeup tanks are located above the reactor coolant loops, and each is equipped with a pressure balancing line from a cold leg to the top of the tank.

The pressurized accumulators provide additional borated water to the reactor coolant system in the event of a LOCA. Nominally, these 2000-ft^3 tanks are filled with 1700 ft^3 of water and 300 ft^3 of nitrogen at an initial pressure of 700 psig. Once sufficient reactor coolant system depressurization occurs, either as a result of a LOCA or the actuation of the ADS, accumulator injection commences.

The IRWST provides an additional source of water for long-term core cooling. To attain injection from the IRWST, the reactor coolant system pressure must be lowered to approximately

13 psi above containment pressure. For this pressure to be achieved during a small-break LOCA, the ADS system is initiated.

The ADS consists of a series of valves, connected to the pressurizer and hot legs, which provide a phased depressurization of the reactor coolant system. As the reactor system loses inventory through the break, the core makeup tanks provide flow to the reactor vessel. When the level in the core makeup tank drops to the 67.5-percent level, the ADS valves open to accelerate the reactor coolant system depressurization rate. The ADS Stage 1 4-inch valves open at the 67.5-percent level; the 8-inch Stage 2 and the 8-inch Stage 3 valves open in a timed sequence thereafter. The flow from the first three stages of the ADS is discharged into the IRWST through a sparger system. The fourth stages of the ADS are connected to the reactor coolant system hot legs and discharge to containment atmosphere. The ADS Stage 4 valves are activated when the core makeup tank level reaches the 20-percent level.

As the reactor coolant system depressurizes and mass is lost out the break, mass is added to the reactor vessel from the core makeup tanks and the accumulators. When the system is depressurized below the IRWST delivery pressure, flow from the IRWST continues to maintain the core in a coolable state. Calculations described in subsection 15.6.5.4B indicate that acceptable core cooling is provided for the small-break LOCA transients. Subsection 15.6.5.4C calculations show that effective post-LOCA core cooling is provided in the long term by passive means.

15.6.5.3 Radiological Consequences

Comment [B16]: [15.6-16]

Although the analysis of the core response during a LOCA (see subsection 15.6.5.4) shows that core integrity is maintained, for the evaluation of the radiological consequences of the accident, it is assumed that major core degradation and melting occur.

The dose calculations take into account the release of activity by way of the containment purge line prior to its isolation near the beginning of the accident and the release of activity resulting from containment leakage. Purge of the containment for hydrogen control is not an intended mode of operation and is not considered in the dose analysis. While the normal residual heat removal system is capable of post-LOCA cooling, it is not a safety-related system and may not be available following the accident. If it is operable, it would be used only if the source term is not far above the normal shutdown primary coolant source term. It is assumed that core cooling is accomplished by the passive core cooling system, which does not pass coolant outside of containment. Thus, there is no recirculation leakage release path to be modeled.

15.6.5.3.1 Source Term

The release of activity to the containment consists of two parts. The initial release is the activity contained in the reactor coolant system. This is followed by the release of core activity.

15.6.5.3.1.1 Primary Coolant Release

The reactor coolant is assumed to have activity levels consistent with operation at the Technical Specification limits of 280 μCi/gm dose equivalent Xe-133 and 1.0 μCi/gm dose equivalent I-131.

Based on NUREG-1465 (Reference 19), for a plant using leak-before-break methodology, the release of coolant into the containment can be assumed to last for 10 minutes. The AP1000 is a leak-before-break plant, and the water in the reactor coolant system is assumed to blow down into the containment over a period of 10 minutes. The flow rate is assumed to be constant over the 10-minute period. As the reactor coolant enters the containment, the noble gases and half of the iodine activity are assumed to be released into the containment atmosphere.

15.6.5.3.1.2 Core Release

The release of activity from the fuel takes place in two stages as summarized in Table 15.6.5-1. First is the gap release which is assumed to occur at the end of the primary coolant release phase (i.e., at ten minutes into the accident) and continue over a period of half an hour. The second stage is that of the in-vessel core melt in which the bulk of the activity releases associated with the accident occur. The source term model is based on NUREG-1465 and Regulatory Guide 1.183 (Reference 20).

The core fission product inventory at the time of the accident is based on operation near the end of a fuel cycle at 101-percent power and is provided in Table 15A-3 of Appendix 15A. The main feedwater flow measurement supports a 1-percent power uncertainty. Consistent with NUREG-1465, there are three groups of nuclides considered in the gap activity releases: noble gases, iodines, and alkali metals (cesium and rubidium). For the core melt phase, there are five additional nuclide groups for a total of eight. The five additional nuclide groups are the tellurium group, the noble metals group, the cerium group, the lanthanide group, and barium and strontium. The specific nuclides included in the source term are as shown in Table 15A-3.

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Gap Activity Release

Consistent with NUREG-1465 guidance for a plant using leak-before-break methodology, the gap release phase begins after the primary coolant release phase ends at ten minutes and has a duration of 0.5 hour.

In-vessel Core Release

After the gap activity release phase, there is an in-vessel release phase which lasts for 1.3 hours and which releases activity to the containment due to core melting. The fractions of the core activity released to the containment atmosphere during this phase are from NUREG-1465:

Noble gases	0.95
Iodines	0.35
Alkali metals	0.25
Tellurium group	0.05
Noble metals	0.0025
Ba and Sr	0.02
Cerium group	0.0005
Lanthanide group	0.0002

Consistent with NUREG-1465, the releases are assumed to occur at a constant rate over the 1.3-hour phase duration.

15.6.5.3.1.3 Iodine Form

The iodine form is consistent with the NUREG-1465 model. The model shows the iodine to be predominantly in the form of nonvolatile cesium iodide with a small fraction existing as elemental iodine. Additionally, the model assumes that a portion of the elemental iodine reacts with organic materials in the containment to form organic iodine compounds. The resulting iodine species split is as follows:

- Particulate 0.95
- Elemental 0.0485
- Organic 0.0015

If the post-LOCA cooling solution has a pH of less than 6.0, part of the cesium iodide may be converted to the elemental iodine form. The passive core cooling system provides sufficient trisodium phosphate to the post-LOCA cooling solution to maintain the solution pH at 7.0 or greater following a LOCA (see subsection 6.3.2.1.4).

15.6.5.3.2 In-containment Activity Removal Processes

The AP1000 does not include active systems for the removal of activity from the containment atmosphere. The containment atmosphere is depleted of elemental iodine and of particulates as a result of natural processes within the containment.

Elemental iodine is removed by deposition onto surfaces. Particulates are removed by sedimentation, diffusiophoresis (deposition driven by steam condensation), and thermophoresis

(deposition driven by heat transfer). No removal of organic iodine is assumed. Appendix 15B provides a discussion of the models and assumptions used in calculating the removal coefficients.

15.6.5.3.3 Release Pathways

The release pathways are the containment purge line and containment leakage. The activity releases are assumed to be ground level releases.

During the initial part of the accident, before the containment is isolated, it is assumed that containment purge is in operation and that activity is released through this pathway until the purge valves are closed. No credit is taken for the filters in the purge exhaust line.

The majority of the releases due to the LOCA are the result of containment leakage. The containment is assumed to leak at its design leak rate for the first 24 hours and at half that rate for the remainder of the analysis period.

15.6.5.3.4 Offsite Dose Calculation Models

The offsite dose calculation models are provided in Appendix 15A. The models address the determination of the TEDE doses from the combined acute doses and the committed effective dose equivalent doses.

The exclusion area boundary dose is calculated for the 2-hour period over which the highest doses would be accrued by an individual located at the exclusion area boundary. Because of the delays associated with the core damage for this accident, the first 2 hours of the accident are not the worst 2-hour interval for accumulating a dose.

The low population zone boundary dose is calculated for the nominal 30-day duration of the accident.

For both the exclusion area boundary and low population zone dose determinations, the calculated doses are compared to the dose guideline of 25 rem TEDE from 10 CFR Part 50.34.

15.6.5.3.5 Main Control Room Dose Model

There are two approaches used for modeling the activity entering the main control room. If power is available, the normal heating, ventilation, and air-conditioning (HVAC) system will switch over to a supplemental filtration mode (Section 9.4). The normal HVAC system is not a safety-class system but provides defense in depth.

Alternatively, if the normal HVAC is inoperable or, if operable, the supplemental filtration train does not function properly resulting in increasing levels of airborne iodine in the main control

room, the emergency habitability system (Section 6.4) would be actuated when high iodine activity is detected. The emergency habitability system provides passive pressurization of the main control room from a bottled air supply to prevent inleakage of contaminated air to the main control room. The bottled air also induces flow through the passive air filtration system which filters contaminated air in the main control room. There is a 72-hour supply of air in the emergency habitability system. After this time, the main control room is assumed to be opened and unfiltered air is drawn into the main control room by way of an ancillary fan. After 7 days, offsite support is assumed to be available to reestablish operability of the control room habitability system by replenishing the compressed air supply. As a defense-in-depth measure, the nonsafety-related normal control room HVAC would be brought back into operation with the supplemental filtration train if power is available.

The main control room is accessed by a vestibule entrance, which restricts the volume of contaminated air that can enter the main control room from ingress and egress. The design of the emergency habitability system (VES) provides 65 scfm ±5 scfm to the control room and maintains it in a pressurized state. The path for the purge flow out of the main control room is through the vestibule entrance and this should result in a dilution of the activity in the vestibule and a reduction in the amount of activity that might enter the main control room. However, no additional credit is taken for dilution of the vestibule via the purge. The projected inleakage into the main control room through ingress/egress is 5 cfm. An additional 10 cfm of unfiltered inleakage is conservatively assumed from other sources.

Activity entering the main control room is assumed to be uniformly dispersed. With the VES in operation, airborne activity is removed from the main control room via the passive recirculation filtration portion of the VES.

The main control room dose calculation models are provided in Appendix 15A for the determination of doses resulting from activity which enters the main control room envelope.

15.6.5.3.6 Analytical Assumptions and Parameters

The analytical assumptions and parameters used in the radiological consequences analysis are listed in Table 15.6.5-2.

15.6.5.3.7 Identification of Conservatisms

The LOCA radiological consequences analysis assumptions include a number of conservatisms. Some of these conservatisms are discussed in the following subsections.

15.6.5.3.7.1 Primary Coolant Source Term

The source term is based on conservative assumptions whereas, the activities based on the expected fuel defect level are far less.

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15.6.5.3.7.2 Core Release Source Term

The assumed core melt is a major conservatism associated with the analysis. In the event of a postulated LOCA, no major core damage is expected. Release of activity from the core is limited to a fraction of the core gap activity.

15.6.5.3.7.3 Atmospheric Dispersion Factors

The atmospheric dispersion factors assumed to be present during the course of the accident are conservatively selected. Actual meteorological conditions are expected to result in significantly higher dispersion of the released activity.

15.6.5.3.8 LOCA Doses

15.6.5.3.8.1 Offsite Doses

The doses calculated for the exclusion area boundary and the low population zone boundary are listed in Table 15.6.5-3. The doses are within the 10 CFR 50.34 dose guideline of 25 rem TEDE.

The reported exclusion area boundary doses are for the time period of 1.3 to 3.3 hours. This is the 2-hour interval that has the highest calculated doses. The dose that would be incurred over the first 2 hours of the accident is well below the reported dose.

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At the time the LOCA occurs, there is the potential for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because pool boiling would not occur until after the limiting 2 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE and, when this is added to the dose calculated for the LOCA, the resulting total dose remains less than that reported in Table 15.6.5-3.

15.6.5.3.8.2 Doses to Operators in the Main Control Room

The doses calculated for the main control room personnel due to airborne activity entering the main control room are listed in Table 15.6.5-3. Also listed on Table 15.6.5-3 are the doses due to direct shine from the activity in the adjacent buildings and sky-shine from the radiation that

streams out the top of the containment shield building and is reflected back down by air-scattering. The total of the three dose paths is within the dose criteria of 5 rem TEDE as defined in GDC 19.

As discussed above for the offsite doses, there is the potential for a dose to the operators in the main control room due to iodine releases from postulated spent fuel boiling. The calculated dose from this source is less than 0.01 rem TEDE and is reported in Table 15.6.5-3.

15.6.5.4 Core and System Performance

Subsection 15.6.5.4A describes the large-break LOCA analysis methodology and results. Subsections 15.6.5.4B.1.0 through 15.6.5.4B.4.0 describe the small-break LOCA analysis methodology and results.

15.6.5.4A Large-break LOCA Analysis Methodology and Results

Comment [B17]: [15.6-17]

Comment [B18]: [15.6-70]

Westinghouse applies the WCOBRA/TRAC computer code to perform best-estimate large-break LOCA analyses in compliance with 10 CFR 50 (Reference 5). WCOBRA/TRAC is a thermal-hydraulic computer code that calculates realistic fluid conditions in a PWR during the blowdown and reflood of a postulated large-break LOCA. The methodology used for the AP1000 analysis is documented in WCAP-12945-P-A, WCAP-14171, Revision 2, WCAP-16009-P-A (References 10, 11, 32), and Reference 31.

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The NRC staff has reviewed and approved the ASTRUM best-estimate LOCA methodology (ASTRUM methodology), as documented in the SER attached in front of Reference 32 for estimating the 95th percentile PCT for two-loop, three-loop and four-loop Westinghouse pressurized water reactors (PWRs) and the AP600. Application of the ASTRUM methodology for the AP1000 plant was submitted to the NRC staff per Reference 34. The NRC staff has reviewed and approved the ASTRUM methodology for estimating the 95th percentile PCT for the AP1000 plant, as documented in Reference 35. In the ASTRUM methodology, the WCOBRA/TRAC code is used to calculate the effects of initial conditions, power distributions, and global models, and the HOTSPOT code is used to calculate the effects of local models.

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Deleted: In Reference 3, the NRC staff has reviewed and approved a best-estimate LOCA methodology, as documented in Reference 11, for estimating the 95th percentile PCT for the AP600. In the Reference 32 and Reference 11 methodologies,

In the ASTRUM uncertainty methodology (Reference 32), as used in the AP1000 LB LOCA analysis, global models and initial-condition, power-distribution, and local uncertainties are sampled independently for each of 124 runs over the same ranges of uncertainty and distributions as in References 10, 32, and 33, as described in References 34 and 31. The sampled global models, initial conditions, and power-distribution uncertainties become inputs to each of the WCOBRA/ TRAC calculations. The thermal-hydraulic boundary conditions for the hot rod are input to the local uncertainties calculation performed by the HOTSPOT code.

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Results from the calculations are ranked by PCT from highest to lowest. A similar procedure is repeated for maximum local oxidation (MLO) and core wide oxidation (CWO). In order statistics as applied in the ASTRUM methodology, the limiting case for a parameter, such as peak cladding temperature (PCT), is a conservative estimate of the 95th percentile with 95 percent confidence. The limiting PCT, limiting MLO, and CWO may come from the same case or as many as three different cases because each parameter is assumed to be independent of the other two. The assumption of independence of the calculated licensing parameters is a conservative assumption because there is a dependence of MLO and CWO on cladding temperature.

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For the AP1000 large-break LOCA analysis, a plant-specific adaptation of the ASTRUM methodology is applied as described in Reference 31. The plant-specific adaptation explicitly models the effects of thermal conductivity degradation and peaking factor burndown. The best-estimate large-break LOCA analysis complies with the stipulated applicability limits in the Reference 3, Reference 32, and Reference 35 approvals. The post-LOCA long-term core cooling and core boron concentration analyses discussed in subsection 15.6.5.4C are applicable to the large-break LOCA transient.

Comment [B19]: [15.6-18]

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15.6.5.4A.1 General Description of WCOBRA/TRAC Modeling

WCOBRA/TRAC is the best-estimate thermal-hydraulic computer code used to calculate realistic fluid conditions in the PWR during blowdown and reflood of a postulated large-break LOCA.

The WCOBRA/TRAC Code Qualification Document (Reference 10) contains a complete description of the code models and justifies their applicability to PWR large-break LOCA analysis.

Table 15.6.5-4 lists AP1000-specific parameters identified for use in the large-break LOCA analysis. WCOBRA/TRAC studies were performed for AP1000 to establish sensitivities to parameter variations. These studies included effects of ranging steam generator tube plugging, ranging the relative power in the low-power assemblies, loss of offsite power coincident with the break initiation, and break location. The calculated results were used to identify bounding conditions, which are then used in the AP1000 uncertainty calculations.

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The WCOBRA/TRAC vessel nodalization is developed from plant design drawings to divide the vessel into 10 vertical sections. The bottom of section 1 is the inside vessel bottom, and the top of section 10 is the inside top of the vessel upper head. In addition to the major downcomer and core flow paths, the modeled bypass flow paths are the upper head cooling spray, guide thimbles, and core bypass. After defining the elevations for each section, a noding scheme is defined for the WCOBRA/TRAC model as shown in Reference 34. WCOBRA/TRAC assumes a vertical flow path for vertically stacked channels, unless specified otherwise in the input. Positive flow

for the vertically connected channels (and cells) is upward. Several of the 10 sections are divided vertically into 2 or more levels; these levels are referred to as cells within a channel.

The WCOBRA/TRAC loop model represents the major primary, secondary, and passive safety systems components. Both loops are explicitly modeled, including the hot leg, the steam generator, and the two cold legs and associated pumps. The loop designated "1" has the pressurizer and the PRHR system connections, and loop "2" cold legs have the core makeup tank pressure balance line connections. The reactor coolant pump models contain the AP1000 homologous curves together with appropriate two-phase head and torque multipliers and degradation data. AP1000 values for pump coastdown characteristics are also applied. The passive safety features are modeled using design data for elevations, liquid volumes, and line losses. Because the ADS is not actuated until long after the time of PCT in large-break LOCA events, it is not modeled in detail.

15.6.5.4A.2 Steady-state Calculation

A WCOBRA/TRAC LOCA calculation is initiated from a point at which the flows, temperatures, powers, and pressures are at their approximate steady-state values before the postulated break occurs. Steady-state WCOBRA/TRAC calculations are run for a brief time period to verify that the calculated conditions are steady and that the desired reactor conditions are achieved.

The values used to set the steady-state plant conditions reflect the AP1000 parameters for reactor coolant pump flows, core power, and steam generator tube plugging levels. The fuel parameters provide the steady-state fuel temperatures, pressures, and gap conductances as a function of fuel burnup and linear power, accounting for the effects of thermal conductivity degradation as described in Reference 31. The calculated fuel temperatures from WCOBRA/TRAC are adjusted to match the specified fuel data by adjusting the gap heat transfer coefficient between the pellet and the cladding. Once the vessel fluid temperatures, flows, pressures, loop pressure drop, and core parameters are in agreement with the desired values and are steady, a suitable initial condition is achieved.

15.6.5.4A.3 Signal Logic for Large-break LOCA

Comment [B20]: [15.6-19]

The reactor trip signal occurs due to compensated pressurizer pressure within the first seconds of the large-break transient however control rod insertion is not modeled in WCOBRA/TRAC and no effects of control rod insertion on reactivity ensue. A safeguards "S" signal occurs due to containment high pressure of 6.7 psig at 2.2 seconds of large-break LOCA transients.

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As a consequence of this signal, after appropriate delays, the PRHR and core makeup tank isolation valves open, containment isolation occurs, and the reactor coolant pump automatic trip timer begins. The rapid depressurization of the primary system during a large-break LOCA leads

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to the initiation of accumulator injection early in the large-break transient. The accumulator flow diminishes core makeup tank delivery to such an extent that the core makeup tank level does not approach the ADS Stage 1 valve actuation point until after the accumulator tank is empty. The accumulator empties long after the blowdown portion of the large-break LOCA transient is complete. Actuation of the ADS on CMT water level does not occur until long after the AP1000 PCT is calculated to occur.

15.6.5.4A.4 Transient Calculation

Once the steady-state calculation is found to be acceptable, the transient calculation is initiated. The semi-implicit pipe break model is added to the desired break location. Cold-leg breaks are analyzed because the hot-leg break location is nonlimiting in the large-break LOCA best-estimate methodology. The break size and type are sampled consistent with the WCAP-16009-P-A (Reference 32) methodology. The containment backpressure is specified consistent with WCAP-16009-P-A (Reference 32) methodology. The steady-state calculation is restarted with the above changes to begin the transient.

Table 15.6.5-5 shows a general sequence of events following a large cold-leg break LOCA and the relationship of these events to the blowdown and reflood portion of the transient.

15.6.5.4A.5 Large-break LOCA Analysis Results

For the AP1000 large-break LOCA analysis, a plant-specific adaptation of the ASTRUM best-estimate LOCA analysis methodology is applied, as described in Reference 31. The AP1000 large-break LOCA analysis complies with the restrictions in Reference 3, Reference 32, and Reference 35. AP1000 sensitivity calculations evaluated the sensitivity to the modeling of the CMT and PRHR relative to the reference transient configuration. A case in which the CMT was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was lower than the PCT of the reference transient configuration. Also, a case in which the PRHR was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was 2°F higher than the PCT of the reference transient configuration. The ASTRUM methodology samples the parameters ranged in the global model matrix of calculations, and the final 95 percent uncertainty calculations have been performed for AP1000. Further, local and core-wide cladding oxidation values have been determined using the plant-specific adaptation of the approved Reference 32 methodology as described in Reference 31.

In the AP1000 ASTRUM analysis, the limiting PCT and limiting MLO results were from two different uncertainty calculations. Both the limiting PCT case and the limiting MLO case were double ended guillotine breaks. Figures 15.6.5.4A-1 through 15.6.5.4A-12 present the parameters of principal interest for the limiting PCT case. Values of the following parameters are presented:

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- Highest calculated cladding temperature at any elevation for the five fuel rods modeled
- Hot rod cladding temperature transient at the limiting elevation for PCT
- Core fluid mass flows at the top of the core for the fuel assemblies modeled in WCOBRA/TRAC
- Pressurizer pressure
- Break flow rates
- Core and downcomer collapsed liquid levels
- Accumulator water flow rates
- Core makeup tank flow rates

15.6.5.4A.6 Description of AP1000 Large-Break LOCA Transient

Comment [B22]: [15.6-21]

A description of the limiting PCT case from the AP1000 ASTRUM analysis follows. The limiting PCT case is a double ended guillotine break. The sequence of events is presented in Table 15.6.5-6. The break was modeled to occur in one of the cold legs in the loop containing the core makeup tanks. After the break opens, the vessel rapidly depressurizes and the core flow quickly reverses. The hot assembly fuel rods dry out and begin to heat up (Figures 15.6.5.4A-1 and 15.6.5.4A-2) after the initial flow reversal (Figure 15.6.5.4A-3).

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In Figure 15.6.5.4A-1, "Hot Rod" refers to the hot fuel rod at the maximum linear heat rate for the run, "Hot Assembly" refers to the average fuel rod in the hot assembly that contains the hot rod, "Support Column/Open Hole" refers to the fuel rod in average assemblies under support columns or open holes, "Guide Tubes" refers to the fuel rod in average assemblies under guide tubes, and "Low Power" refers to the fuel rod in the low power peripheral fuel assemblies.

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The steam generator secondaries are assumed to be isolated immediately at the inception of the break, which maximizes their stored energy. The massive size of the break causes an immediate, rapid pressurization of the containment. At 2.2 seconds, an "S" signal is generated due to High-2 containment pressure. Applying the pertinent signal processing delay means that the valves isolating the core makeup tanks from the direct vessel injection line and the PRHR begin to open at 4.2 seconds into the transient. The reactor coolant pumps automatically trip after a 5.3 second delay from the actuation of the core makeup tank isolation valves, which is 9.5 seconds into the transient. Core shutdown occurs due to voiding; no credit is taken for the control rod insertion effect.

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The system depressurizes rapidly (Figure 15.6.5.4A-4) as the initial mass inventory is depleted due to break flow. The pressurizer drains completely approximately 30 seconds into the transient, and accumulator injection commences 13 seconds into the transient (Figure 15.6.5.4A-5). Accumulator actuation shuts off core makeup tank flow (Figure 15.6.5.4A-6), which has been occurring since the isolation valve opened. The CMT liquid level remains well above the ADS

Comment [B24]: [15.6-23]

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Stage 1 actuation setpoint throughout the AP1000 LBLOCA cladding temperature excursion, even though CMT injection begins again around 200 seconds.

The dynamics of the 95th percentile estimator PCT case are shown in terms of the flow rates of liquid, vapor, and entrained liquid at the top of the core (Figures 15.6.5.4A-7 through 15.6.5.4A-9) for the peripheral, open hole/support column average power interior, and guide tube average power interior assemblies (the corresponding figure for the hot assembly is Figure 15.6.5.4A-3).

Figure 15.6.5.4A-7 demonstrates that liquid downflow exists through the top of the peripheral core assemblies from approximately 1 to 3 seconds and again from 9 to 20 seconds in the 95th percentile estimator PCT case. The power of the fuel in this region is significantly lower than that of the fuel in the open hole/support column and guide tube locations (Table 15.6.5-4), so liquid downflow occurs earlier on the periphery than in the average power assemblies. Once the upper head begins to flash, liquid drains directly down the guide tubes and that fraction that is able to penetrate into the core does so, at a maximum flow rate exceeding 1000 lbm/sec of total liquid flow between 5-23 seconds (Figure 15.6.5.4A-8).

Figure 15.6.5.4A-9 presents the open hole/support column assembly top of core flow behavior. In this case, liquid downflow into the support column/open hole assemblies is delayed relative to downflow into the guide tubes; there is continuous liquid flow from approximately 10 seconds until 22 seconds; the entrained liquid flow continues to be significant until 28 seconds as fluid drains through the upper core plate holes into the upper plenum.

The timing of the initial downflow into the hot assembly is similar to that of the downflow into the open hole/support column average assemblies. Around 10 seconds into the transient, liquid that has built up in the global region above the hot assembly begins to flow into the hot assembly (Figure 15.6.5.4A-3). Significant flow of continuous liquid into the hot assembly exists between 10 to 20 seconds. The liquid flow is not enough to quench the hot rod and hot assembly rod or the average rods at all elevations (Figure 15.6.5.4A-1) although some cooling is achieved.

After 13 seconds into the transient, the accumulator begins to inject water into the upper downcomer region, most of which is initially bypassed to the break. The break flow rate diminishes as the transient progresses (Figure 15.6.5.4A-10). At 27.5 seconds, the accumulator injection begins to refill the lower plenum. At approximately 40.0 seconds, the lower plenum fills to the point that water begins to reflood the core from below (Figure 15.6.5.4A-11). The void fraction at the core bottom begins to decrease, and as time passes, core cooling increases substantially. Figure 15.6.5.4A-11 presents the collapsed liquid levels in the core; Figure 15.6.5.4A-12 presents the collapsed liquid levels in the downcomer. The cladding temperature begins to decrease once the core water level has risen high enough in the core.

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15.6.5.4A.7 Global Model Sensitivity Studies and Uncertainty Evaluation

Section 15.6.5.4A discusses the treatment of the global model parameters and the uncertainty evaluation in the ASTRUM methodology.

15.6.5.4A.8 Large-Break LOCA Conclusions

Comment [B28]: [15.6-27]

In accordance with 10 CFR 50.46, the conclusions of the best-estimate large-break LOCA analysis are that there is a high level probability that the following criteria are met.

1. The calculated maximum fuel element cladding temperature (i.e., peak cladding temperature (PCT)) will not exceed 2200°F.

2. The calculated total oxidation of the cladding (i.e., maximum cladding oxidation) will nowhere exceed 0.17 times the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam (i.e., maximum hydrogen generation) will not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. The calculated changes in core geometry are such that the core remains amenable to cooling.

   Note that criterion 4 has historically been satisfied by adherence to criteria 1 and 2, and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. Criteria 1 and 2 are satisfied for best-estimate large-break LOCA applications. The approved methodology specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the assemblies in the low power channel as defined in the WCOBRA/TRAC model. This situation has not been calculated to occur for the AP1000. Therefore, acceptance criterion 4 is satisfied.

5. After successful initial operation of the emergency core cooling system (ECCS), the core temperature will be maintained at an acceptably low value and decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

   Criterion 5 is satisfied if a coolable core geometry is maintained and the core is cooled continuously following the LOCA. The AP1000 passive core cooling system provides effective core cooling following a large-break LOCA event, even assuming the limiting single failure of a core makeup tank delivery line isolation valve. The large-break LOCA

transient has been extended beyond fuel rod quench to the time at which the CMT liquid level has decreased to the setpoint that actuates the fourth-stage ADS valves and IRWST injection. A significant increase in safety injection flow rate occurs when the IRWST becomes active. The analysis performed demonstrates that CMT injection is sufficient to maintain the mass inventory in the core and downcomer, from the period of fuel rod quench until IRWST injection. The AP1000 passive core cooling system provides effective post-LOCA long-term core cooling (Section 15.6.5.4C).

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Table 15.6.5-8 presents the calculated 95th percentile PCT, maximum cladding oxidation, maximum hydrogen generation, and core cooling results.

Based on the analysis, the Westinghouse Best-Estimate Large-Break LOCA methodology has shown that the acceptance criteria of 10 CFR 50.46 are satisfied for AP1000 when the burnup-related effects of thermal conductivity degradation and peaking factor burndown are considered.

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15.6.5.4B Small-break LOCA Analyses

Comment [B29]: [15.6-28]

Comment [B30]: [15.6-71]

Should a small break LOCA occur, depressurization of the reactor coolant system results in a pressure decrease in the pressurizer. The reactor trip signal occurs when the pressurizer low-pressure trip setpoint is reached. An "S" signal is generated when the appropriate setpoint is reached. These measures limit the consequences of the accident in two ways:

- Reactor trip leads to a rapid reduction of power to a residual level corresponding to fission product decay heat by the insertion of control rods to shut down the reactor.
- Injection of borated water provides core cooling and prevents excessive cladding temperatures.

15.6.5.4B.1 Description of Small-break LOCA Transient

Comment [B31]: [15.6-29]

The AP1000 plant design includes passive safety features to prevent or minimize core uncovery during small-break LOCAs. The passive safety design approach of the AP1000 is to depressurize the reactor coolant system if the break or leak is greater than the capability of the makeup system or if the non-safety makeup system fails to perform. By depressurizing the reactor system, large volumes of borated water in the accumulators and in the IRWST become available for cooling the core. These analyses demonstrate that, with a single failure of one of the ADS Stage 4 valves located off the non-pressurizer loop, the passive systems are capable of depressurizing the reactor coolant system while maintaining acceptable core conditions and establishing stable delivery of cooling water from the IRWST.

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During a small-break LOCA, the AP1000 reactor coolant system depressurizes to the pressurizer low-pressure setpoint, actuating a reactor trip signal. The passive core cooling system is aligned

for delivery following the generation of an "S" signal when the pressurizer low-pressure setpoint is reached. The passive core cooling system includes two core makeup tanks, two accumulators, a large IRWST, and the PRHR heat exchanger.

The core makeup tanks operate at reactor coolant system pressure. They provide high-pressure safety injection in the event of a small-break LOCA. The core makeup tanks share a common discharge line with the accumulators and IRWST; they are filled with borated water to provide core shutdown margin. Gravity head of the colder water in the core makeup tanks provides the injection of the core makeup tanks. The core makeup tanks are located above the reactor coolant loops, and each is equipped with a pressure balancing line from a cold leg to the top of the tank.

The pressurized accumulators provide additional borated water to the reactor coolant system in the event of a LOCA. Nominally, these 2000-ft^3 tanks are filled with 1700 ft^3 of water and 300 ft^3 of nitrogen at an initial pressure of 700 psig. Once sufficient reactor coolant system depressurization occurs, either as a result of a LOCA or the actuation of the ADS, accumulator injection begins.

The IRWST nominally provides an additional source of water for long-term core cooling. To attain injection from the IRWST, the reactor coolant system pressure must be lowered to approximately 13 psi above containment pressure. For this pressure to be achieved during a small-break LOCA, the actuation of the ADS valves is required.

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The ADS consists of a series of valves, connected to the pressurizer and hot legs, which provide a phased depressurization of the reactor coolant system. As the reactor system loses inventory through the break, the core makeup tanks provide flow to the reactor vessel. When the level in the core makeup tank drops to the 67.5-percent level, the ADS valves open to accelerate the reactor coolant system depressurization rate. The ADS Stage 1 4-inch valves open at the 67.5-percent level; the 8-inch Stage 2 and the 8-inch Stage 3 valves open in a timed sequence thereafter. The flow from the first three stages of the ADS is discharged into the IRWST through a sparger system. The fourth stages of the ADS are connected to the reactor coolant system hot legs and discharge to containment atmosphere. The ADS Stage 4 valves are activated when the core makeup tank level reaches the 20-percent level.

As the reactor system depressurizes and mass is lost out the break, mass is added to the reactor vessel from the core makeup tanks and the accumulators. When the system is depressurized below the IRWST delivery pressure, flow from the IRWST continues to maintain the core in a coolable state. Calculations described in this section indicate that acceptable core cooling is provided for the small-break LOCA transients.

15.6.5.4B.2 Small-break LOCA Analysis Methodology

Small-break LOCA response is evaluated for AP1000 with an evaluation model that conforms to 10 CFR 50 Appendix K. The elements of the AP1000 small-break LOCA evaluation model are the following:

- NOTRUMP computer code
- NOTRUMP homogeneous sensitivity model
- Critical heat flux assessment during accumulator injection
- SBLOCTA computer code

15.6.5.4B.2.1 NOTRUMP Computer Code

The NOTRUMP computer code is used in the analysis of LOCAs due to small-breaks in the reactor coolant system. The NOTRUMP computer code is a one-dimensional, general network code, which includes a number of advanced features. Among these features are the calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. The version of NOTRUMP used in AP1000 small-break LOCA calculations has been validated against applicable passive plant test data (Reference 22). The code has limited capability in modeling upper plenum and hot leg entrainment and did not predict the core collapsed level during the accumulator injection phase adequately. The NOTRUMP homogeneous sensitivity model (discussed in subsection 15.6.5.4B.2.2) and the critical heat flux assessment during the accumulator injection phase (discussed in subsection 15.6.5.4B.2.3) supplement the base NOTRUMP analysis to demonstrate the adequacy of the design.

In NOTRUMP, the reactor coolant system is nodalized into volumes interconnected by flow paths. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum applied throughout the system. A description of NOTRUMP is given in References 12 and 13. The AP600 modeling approach, described in Reference 17, is also used to develop the AP1000 model; NOTRUMP's applicability to AP1000 is documented in Reference 24.

The use of NOTRUMP in the analysis involves the representation of the reactor core as heated control volumes with an associated bubble rise model to permit a transient mixture height calculation. The multi-node capability of the program enables an explicit and detailed spatial representation of various system components. Table 15.6.5-9 lists important input parameters and initial conditions of the analysis.

A steady-state input deck for the AP1000 was set up to comply, where appropriate, with the standard small-break LOCA Evaluation Model methodology. Major features of the modeling of the AP1000 follow:

- Accumulators are modeled at an initial pressure of 715 psia.
- The flow through the ADS links is modeled using the Henry-Fauske, the homogeneous equilibrium (HEM), and the Murdock/Baumann critical flow models. The Henry-Fauske correlation is used for low-quality two--phase flow, and the HEM model, for high-quality flow, with a transition between the two beginning at 10-percent static quality. The Murdock-Bauman model is used if the ADS flow path is venting superheated steam.

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- Isolation and check valves used in the passive safety systems are modeled.
- The IRWST is modeled as two connected fluid nodes. The lower node is connected to the direct vessel injection line and is the source of injection water to the DVI lines driven by gravity head. The upper node acts as a sink for the ADS flow from the pressurizer and as a heat sink for the PRHR heat exchanger. These nodes are modeled as having an initial temperature of 120°F, a pressure of 14.7 psia, and the nominal full-power operation level of 28.8 feet. Therefore, the minimum head for IRWST injection is assumed. For the DEDVI simulations, a conservative 20 psia containment pressure was used based on containment pressurization calculations performed with the WGOTHIC containment model. In addition, the Inadvertent ADS actuation and the 2-inch cold leg break simulations each used a conservative, time-dependent containment pressure response also based on containment pressurization calculations performed with the WGOTHIC containment model as described in Section 13.8 of Reference 6.
- The PRHR system is modeled in accordance with the guidance provided in References 22 and 24. The PRHR isolation valve is modeled as opening with the maximum delay after the generation of an "S" signal to conservatively deny the cooling capability of the heat exchanger to the reactor coolant system for an extended period.
- The core power is initially set to 101 percent of the nominal core power. The reactor trip signal occurs when the pressurizer pressure falls below 1800 psia. A conservative delay time is modeled between the reactor trip signal and reactor trip. Decay heat is modeled according to the ANS-1971 (Reference 2) standard, with 20-percent uncertainty added.

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- The "S" signal is generated when the pressurizer pressure falls below 1700 psia. The isolation valves on the core makeup tank injection lines begin to open after the signal setpoint is reached; the valves are then assumed to open linearly. The main feedwater

isolation valves are ramped closed between 2 and 7 seconds after the "S" signal. The reactor coolant pumps are tripped 7.3 seconds after the "S" signal.

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- The ADS actuation signals are generated on low core makeup tank levels and the ADS timer delays. A list of the ADS parameters is given in Table 15.6.5-10 for AP1000. ADS Stages 1, 2, and 3 are modeled as discharging through spargers submerged in the IRWST at the appropriate depth.

- The Inadvertent ADS actuation and 2-inch cold leg break NOTRUMP simulations utilize a time-dependent containment pressure in the boundary node modeling of the containment. These conditions were generated by providing mass and energy releases from these AP1000 breaks to the AP1000 WGOTHIC containment model while the WGOTHIC code calculates the containment pressure response. The Inadvertent ADS actuation and 2-inch cold leg break NOTRUMP simulations then utilized the time-dependent pressure history curves as generated by WGOTHIC. The 10-inch cold leg break case models a pressure in the boundary node of the containment of 14.7 psia and the DEDVI line break models two cases with a constant 20 psia and 14.7 psia containment backpressure, respectively. The steam generator secondary is isolated 6 seconds after the reactor trip signal, due to closure of the turbine stop valves. The main steam safety valves actuate and remove energy from the steam generator secondary when pressure reaches 1235 psia.

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Active single failures of the passive safeguards systems are considered. The limiting failure is judged to be one out of four ADS Stage 4 valves failing to open on demand, the failure that most severely impacts depressurization capability. The safety design approach of the AP1000 is to depressurize the reactor coolant system to the containment pressure in an orderly fashion such that the large reservoir of water stored in the IRWST is available for core cooling. The mass inventory plots provided for the breaks show the minimum inventory condition generally occurs at the start of IRWST injection. Penalizing the depressurization is the most conservative approach in postulating the single failure for such breaks.

The small-break LOCA spectrum analyzed for AP1000 includes a break that exhibits a minimum reactor vessel inventory early in the transient, before the accumulators become active: the DEDVI break at 20 psia containment backpressure. In this transient, the early mass inventory decrease is terminated by injection flow from the accumulators, and depressurization through the break enables accumulator injection to begin with no contribution from the actuation of ADS Stages 1, 2, and 3. For consistency, the conservative failure of one of the ADS Stage 4 valves located off the non-pressurizer loop, which adversely affects the depressurization necessary to achieve IRWST injection in small-break LOCAs, is assumed in all cases.

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15.6.5.4B.2.1.1 AP1000 Model-Detailed Noding

Refer to Reference 17 for details of the AP600 NOTRUMP modeling. The AP1000 model was developed in the same fashion with modifications to the AP600 model introduced as follows. A modification performed for AP1000 was the addition of two core nodes one foot each in length to reflect the added active fuel length of this design. The ADS-4 flow path resistances were increased to accommodate shortcomings in NOTRUMP identified during the integral test facility simulations, namely, the lack of a detailed momentum flux model in the ADS-4 discharge paths. A detailed calculation of the energy and momentum equations is performed for the ADS-4 piping over a range of flow and pressure conditions to provide a benchmark for the NOTRUMP ADS-4 flow path resistance. The methodology used to determine the resistance increase is described in Reference 24. By increasing the ADS-4 resistances, the onset of IRWST injection is more appropriately calculated. This methodology directly addresses the effect of momentum flux in ADS-4. The ADS-4 resistance increase utilized is computed for the NOTRUMP analyses in this section to be a 82 percent ADS-4 flow path resistance increase.

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15.6.5.4B.2.1.2 Plant Initial Conditions/Steady-State

A steady-state calculation is performed prior to initiating the transient portion of the calculation.

Table 15.6.5-9 contains the most important initial conditions for the transient calculations. The behaviors of the primary pressure and pressurizer level, steam generator pressures, and the core flow rate are stable at the end of the 100-second steady-state calculation.

15.6.5.4B.2.2 NOTRUMP Homogeneous Sensitivity Model

In order to address the uncertainties associated with entrainment in the upper plenum and hot leg following ADS-4 operation, a sensitivity study is performed with the limiting break with respect to these phenomena, effectively maximizing the amount of entrainment downstream of the core. This methodology is described and the results are presented for the double-ended direct vessel injection (DEDVI) line break in detail in Reference 24.

[*In order to maximize the entrainment downstream of the core for the limiting break with respect to entrainment, NOTRUMP is run with the regions of the upper plenum, hot leg, and ADS-4 lines in a homogeneous fluid condition, with slip = 1, to demonstrate that even with maximum entrainment, the 10 CFR 50.46 criteria are met.*]*

15.6.5.4B.2.3 Critical Heat Flux Assessment During Accumulator Injection

[*An assessment is performed of the peak core heat flux with respect to the critical heat flux during the later ADS depressurization time period for a double-ended rupture of the direct vessel injection line. This time period corresponds to the accumulator injection phase of the transient.*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

The predicted average mass flux at the core inlet and the reactor pressure from the NOTRUMP computer code base model analysis are used as input parameters to critical heat flux correlation as described in Reference 30. The requirements of 10 CFR 50.46 are met provided the maximum heat flux is less than the critical heat flux calculated by the correlation.]* NOTRUMP has been shown (Reference 24) to adequately predict mass flux and pressure for integral systems tests. The predicted mass flux at the core inlet is on the average constant and corresponds to 7.2 lbm ft^{-2} s^{-1} (~35 kg m^{-2} s^{-1}). The key thermal-hydraulic parameters at different times during the ADS depressurization time period are summarized in following table.

Time (sec)	UP Pressure (kPa)	UP Pressure (psia)	Mass Flux (kg/m^2s)	Core Average Heat Flux (kW/m^2)
400	1175	170	35	20.0
450	882	128	35	19.4
500	566	82	35	18.9
570	300	43	35	18.3

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For the critical heat flux assessment, the peak core heat flux is applied to simulate the hot assembly condition in a conservative manner. No credit is taken for increased flow in the hot assembly that is known to occur in rod bundles.

The correlation applied for this assessment is from vertical tube data (Reference 30) and recognizes two regimes depending on the mass flux. The main difference between the two is the mass flux dependence. They are as follows:

$$q^*_{CL} = q^*_{CF} + 0.01351\left(D^*\right)^{-0.473}(L/D)^{-0.533}\left|G^*\right|^{1.45} \quad \text{for low } G^*$$

and,

$$q^*_{CH} = q^*_{CF} + 0.05664\left(D^*\right)^{-0.247}(L/D)^{-0.501}\left|G^*\right|^{0.77} \quad \text{for high } G^*$$

The first term of above correlations is,

$$q^*_{CF} = 1.61\left(\frac{A}{Ah}\right)\frac{\left(D^*\right)^{0.5}}{\left[1+\left(\frac{\rho_g}{\rho_l}\right)^{0.25}\right]^2}$$

where A is the flow area and Ah is the heated area.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

The dimensionless CHF is calculated as,

$$q^*_{CHF} = \min(q^*_{CL}, q^*_{CH})$$

Dimensionless CHF, G, and D are defined as,

$$q^*_{CHF} = \frac{q''_{CHF}}{h_{fg}\sqrt{\lambda \rho_g g \Delta\rho}}$$

$$G^* = \frac{G}{\sqrt{\lambda \rho_g g \Delta\rho}}$$

$$D^* = \frac{D}{\lambda}$$

where λ is the length scale of the Taylor instability:

$$\lambda = \sqrt{\frac{\sigma}{g\Delta\rho}}$$

Conservative application of this correlation with the AP1000 parameters indicates that the peak AP1000 heat flux during this period is approximately 30 percent or more below the predicted critical heat flux.

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This CHF assessment addresses core cooling during a time period where the NOTRUMP computer code may not conservatively predict the core average void fraction. The requirements of 10 CFR 50.46 are met during this period since this CHF assessment indicates peak core heat flux is less than critical heat flux. Cladding temperatures will remain near the coolant saturation temperature, well below the 10 CFR 50.46 peak cladding temperature limit.

15.6.5.4B.2.4 SBLOCTA Computer Code

The LOCTA-IV computer code (Reference 4) was modified as described in Reference 13 to form SBLOCTA, a small-break LOCA specific version of the LOCTA-IV code. The SBLOCTA code calculates the cladding temperature and oxidation transients for the hot rod and hot assembly average rod, which represent the highest power rod and the average of the highest power fuel assembly in the core. Peak cladding temperature calculations are performed with the SBLOCTA code using boundary conditions from the NOTRUMP calculation. In addition to PCT,

SBLOCTA also calculates maximum local and axial average zirconium-water oxidation reaction based on the Baker-Just oxidation model. In the event that the NOTRUMP code predicts core uncovery in the core average channel, the NOTRUMP boundary conditions will be transferred to the SBLOCTA code to perform fuel rod heat-up calculations.

15.6.5.4B.3 Small-Break LOCA Analysis Results

Several small-break LOCA transients are analyzed using NOTRUMP, and the results of these calculations are presented. The transients documented herein analyze a single failure of one ADS Stage 4 valve on the non-pressurizer side, with the exception of the DEDVI entrainment study. The results demonstrate that the minimum reactor vessel mixture mass inventory condition occurs for the relatively small system pipe breaks. Larger breaks exhibit a greater margin-to-core uncovery.

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15.6.5.4B.3.1 Introduction

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The small-break LOCA safety design approach for AP1000 is to provide for a controlled depressurization of the primary system if the break cannot be terminated, or if the non safety-related charging system is postulated to be lost or cannot maintain acceptable plant conditions. Non safety-related systems are not modeled in this design basis analysis; the testing conducted in the SPES-2 facility has indicated that the mass inventory condition during small LOCAs is significantly improved when these non safety-related systems operate. The core makeup tank level activates primary system depressurization. The core makeup tank provides makeup to help compensate for the postulated break in the reactor coolant system. As the core makeup tank level drops, Stages 1 through 4 of the ADS valves are ramped open in sequence. The ADS valve descriptions for the AP1000 plant design are presented in Table 15.6.5-10. The reactor coolant system depressurizes due to the break and the ADS valves, while subcooled water from the core makeup tanks and accumulators enters the reactor vessel downcomer to maintain system inventory. Design basis maximum values of passive core cooling system resistances are applied to obtain a conservative prediction of system behavior during the small LOCA events.

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During controlled depressurization via the ADS, the accumulators and core makeup tanks maintain system inventory for small-break LOCAs. Once the reactor coolant system depressurizes, injection from the IRWST maintains long-term core cooling. For continued injection from the IRWST, the reactor coolant system must remain depressurized. To conservatively model this condition, design maximum resistance values are specified for the IRWST delivery lines.

A series of small-break LOCA calculations are performed to assess the AP1000 passive safety system design performance. In these calculations, the decay heat used is the ANS-1971

(Reference 2) plus 20 percent for uncertainty as specified in 10 CFR 50, Appendix K (Reference 1). This maximizes the core steam generation to be vented. The breaks analyzed in this document include the following:

Inadvertent ADS Actuation

A "no-break" small-break LOCA calculation that uses an inadvertent opening of the 4-inch nominal size ADS Stage 1 valves is a situation that minimizes the venting capability of the reactor coolant system. Only the ADS valve vent area is available; no additional vent area exists due to a break. This case examines whether sufficient vent area is available to completely depressurize the reactor coolant system and achieve injection from the IRWST to prevent/minimize core uncovery. The worst single failure for this situation is a failure of one of two ADS Stage 4 valves connected to the non-pressurizer side hot leg. The ADS Stage 4 valve is the largest ADS valve, and it vents directly to the containment with no additional backpressure from the spargers being submerged in the IRWST. The containment pressure is a conservative, time-dependent containment pressure response. This pressure response is based on iterative execution of the NOTRUMP and WGOTHIC codes. The NOTRUMP code provides the mass and energy releases from the AP1000 plant inadvertent ADS actuation simulation to the AP1000 plant WGOTHIC containment model, which calculates the containment pressure response.

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2-inch Break in a Cold Leg with Core Makeup Tank Balance Line Connections

A 2-inch equivalent diameter break is analyzed as a representative break, not specific to a particular pipe connection. The small size of the break leads to a long period of recirculatory flow from the cold leg into the core makeup tank. This delays the formation of a vapor space in the core makeup tank and therefore the actuation of the ADS. The containment pressure is a conservative, time-dependent containment pressure response. This pressure response is based on iterative execution of the NOTRUMP and WGOTHIC codes. The NOTRUMP code provides the mass and energy releases from the AP1000 plant 2-inch cold leg break simulation to the AP1000 plant WGOTHIC containment model, which calculates the containment pressure response.

Double-ended Rupture of the Direct Vessel Injection Line

The direct vessel injection line break evaluates the ability of the plant to recover from a moderately sized break with only half of the total emergency core cooling system capacity available. The vessel side of the break of the DEDVI line break is 4 inches in equivalent diameter. The double-ended nature of this break means that there are effectively two breaks modeled:

- Downcomer to containment. The direct vessel injection nozzle includes a venturi, which limits the available break area.

- Direct vessel injection line into containment from the cold leg balance line and the broken loop core makeup tank.

The containment pressure was conservatively assumed to pressurize to 20 psia. This pressure was selected based on iterative execution of the NOTRUMP and WGOTHIC codes. The NOTRUMP code provides the mass and energy releases from the AP1000 DEDVI break to the AP1000 WGOTHIC containment model while the WGOTHIC code calculates the containment pressure response. The containment pressure assumed in the NOTRUMP simulations was conservatively selected from the generated pressure history curves obtained from the WGOTHIC runs.

The peak core heat flux during the accumulator injection period is assessed relative to the predicted critical heat flux as discussed in subsection 15.6.5.4B.2.3.

An additional injection line break case is analyzed assuming containment pressure is at 14.7 psia.

Double-ended Rupture of the Direct Vessel Injection Line Entrainment Sensitivity

The sensitivity case is performed to assess the effect of higher than expected entrainment in the upper plenum and hot legs on the overall system response and core cooling. Subsection 15.6.5.4B.3.7 provides discussion on the applicability of this entrainment sensitivity.

10-inch Cold Leg Break

The 10-inch equivalent diameter break models a break size that approaches the upper limit size for small-break LOCAs.

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15.6.5.4B.3.2 Transient Results

The transient results are presented in tables and figures for the key AP1000 parameters of interest in the following sections.

15.6.5.4B.3.3 Inadvertent Actuation of Automatic Depressurization System

An inadvertent ADS signal is spuriously generated and the 4-inch ADS valves open. The plant, which is operating at 101-percent power, is depressurized via the ADS alone. Only safety-related systems are assumed to operate in this and other small-break LOCA cases. Additional ADS valves open; after a 48-second delay, the ADS Stage 2 8-inch valves open, and after an additional 120 seconds, the ADS Stage 3 valves open. At the 20-percent core makeup tank level, the operating ADS Stage 4A valve, which is connected to the PRHR inlet pipe, receives a signal to open. After a 60 second delay, both Stage 4B valves (one connected to the hot leg and the other connected to the PRHR inlet pipe) open. The path that fails to open as the assumed single active failure is the Stage 4A valve off the hot leg on the non-pressurizer side. The reactor steady-state

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initial conditions assumed can be found in Table 15.6.5-9. The sequence of events for the transient is given in Table 15.6.5-11.

This case uses a containment backpressure based on the containment pressure history that occurs as a result of the inadvertent ADS actuation. It represents a conservatively low estimate of the expected containment pressure response during the transient. The containment pressurizes for an inadvertent ADS actuation as a result of the ADS-4 discharge paths that vent directly to the containment atmosphere. The time-dependent containment pressure curve (Figure 15.6.5.4B-1(c)) was calculated using the mass and energy releases from the NOTRUMP small-break LOCA code, which were used as inputs in the WGOTHIC containment model.

Transient results are shown in Figures 15.6.5.4B-1(a) through 15.6.5.4B-16(b). The transient is initiated by the opening of the two ADS Stage 1 paths. Reactor trip, reactor coolant pump trip, and safety injection signals are generated via pressurizer low-pressure signals with appropriate delays. After generation of the reactor trip signal, the turbine stop valves begin to close. The main feedwater isolation valves begin to close 2 seconds after the "S" signal pressure setpoint is reached. The opening of the ADS valves and the reduction in core power due to reactor trip causes the primary pressure to fall rapidly (Figures 15.6.5.4B-1(a) and 15.6.5.4B-1(b)). Flow of fluid toward the open ADS paths causes the pressurizer to fill rapidly (Figure 15.6.5.4B-2), and the ADS flow becomes two-phase (Figures 15.6.5.4B-3 and -4(a)). The safety injection signal opens the valves isolating the core makeup tanks and circulation of cold water begins (Figures 15.6.5.4B-5 and -6). The mixture level (Figures 15.6.5.4B-7 and -8) in the core makeup tanks is relatively constant until the accumulators inject (Figures 15.6.5.4B-10 and -11). The reactor coolant pumps begin to coast down due to an automatic trip signal following a 7.3-second delay.

Continued mass flow through the ADS Stage 1, 2, and 3 valves drains the upper parts of the circuit (Figure 15.6.5.4B-4(b)). The steam generator tube cold leg sides start to drain, followed by the drop in mixture levels in the hot leg sides. As the ADS Stage 2 and 3 paths begin to open, increased ADS flow causes the primary pressure to fall rapidly (Figures 15.6.5.4B-1(a) and 15.6.5.4B-1(b)). Following the emptying of the steam generator tube cold leg sides, the cold legs have drained and a mixture level forms in the downcomer (Figure 15.6.5.4B-9).

The primary pressure falls below the pressure in the accumulators thus causing the accumulator check valves to open and accumulator delivery to begin (Figures 15.6.5.4B-10 and -11). The accumulators, and then the core makeup tanks inject until they empty. The ADS flow falls off as the primary pressure decreases. The flow from the accumulators raise the mixture levels in the upper plenum and downcomer (Figures 15.6.5.4B-16 and 15.6.5.4B-9).

As the levels in the core makeup tanks reach the ADS Stage 4 setpoint, one out of two paths is opened from the top of the hot leg (loop 2) and begins discharging fluid. After 30 seconds, the

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second path in loop 2 opens, as does a loop 1 Stage 4 path. Activating the Stage 4 paths (Figures 15.6.5.4B-12(a), -12(b) and -12(c)) leads to reduced flow through ADS Stages 1, 2, and 3 (Figure 15.6.5.4B-4(b)). The reduced flow allows the pressurizer level to fall, and these stages begin to discharge only steam. After the CMTs are empty (Figures 15.6.5.4B-7 and -8), IRWST injection (Figures 15.6.5.4B-13 and -14) does not begin until the pressure in the DVI line drops below the IRWST injection pressure, creating an injection gap (Table 15.6.5-11 and Figures 15.6.5.4B-5, -6, -13 and -14). The overall decrease in reactor vessel mixture inventory (Figure 15.6.5.4B-15(b)) is large enough to result in a short core uncovery (Figure 15.6.5.4B-16(a)). At 5000 seconds, the calculation is considered complete; IRWST delivery exceeds the ADS flows (which are removing the decay heat), and the reactor coolant system inventory and reactor vessel mixture inventory are slowly rising (Figure 15.6.5.4B-15(a) and -15(b)).

The inadvertent opening of the ADS Stage 1 transient confirms the minimum venting area capability to depressurize the reactor coolant system to the IRWST pressure. The analysis indicates that the ADS sizing is sufficient to depressurize the reactor coolant system assuming the worst single failure as the failure of a Stage 4 ADS path to open and decay heat equal to the 10 CFR 50 Appendix K (Reference 1) value of the ANS-1971 Standard (Reference 2) plus 20 percent, which over estimates the core steam generation rate. Even under these limiting conditions, IRWST injection is obtained, and the core mixture level recovers such that minimal cladding heatup occurs (Figure 15.6.5.4B-16(b)).

15.6.5.4B.3.4 2-inch Cold Leg Break in the Core Makeup Tank Loop

This case models a 2-inch (50.8 mm) break occurring in the cold leg connected to the balance line of CMT-1. The reactor steady-state initial conditions assumed for this transient can be found in Table 15.6.5-9. The event times for this transient are given in Table 15.6.5-12.

This case uses a containment backpressure based on the containment pressure history that occurs as a result of the 2-inch cold leg break. It represents a conservatively low estimate of the expected containment pressure response during the transient. The containment pressurizes for a 2-inch cold leg break as a result of the break and the ADS-4 discharge paths that vent directly to the containment atmosphere. The time-dependent containment pressure curve (Figure 15.6.5.4B-17(c)) was calculated using the mass and energy releases from the NOTRUMP small-break LOCA code, which were used as inputs in the WGOTHIC containment model.

Transient results are shown in Figures 15.6.5.4B-17(a) through 15.6.5.4B-35. The break opens at time zero, and the pressurizer pressure begins to fall as shown in Figures 15.6.5.4B-17(a) and 15.6.5.4B-17(b), as mass is lost out the break. The pressurizer mixture level initially decreases as given in Figure 15.6.5.4B-18. The liquid and vapor flow out of the break is shown in Figures 15.6.5.4B-32 and -33. The pressurizer pressure falls below the reactor trip set point, causing the reactor to trip (after the appropriate time delay) and causing isolation of the steam

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generator steam lines. The core makeup tank isolation valves on both delivery lines and the PRHR delivery line isolation valve open after an "S" signal occurs (with appropriate delays); the reactor coolant pumps trip after an "S" signal with a 7.3-second delay. The reactor coolant system is cooled by natural circulation with the steam generators removing the energy through their safety valves (as well as by the break) and via the PRHR. The PRHR heat removal and integrated heat removal are shown in Figure 15.6.5.4B-34 and Figure 15.6.5.4B-35. Once the core makeup tank isolation valves open, the core makeup tanks begin to inject borated water into the reactor coolant system as shown in Figures 15.6.5.4B-22 and -23.

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As time proceeds, the loops drain to the reactor vessel. The mixture level in the downcomer begins to drop as seen in Figure 15.6.5.4B-21. The core makeup tank reaches the 67.5-percent level, and after an appropriate delay, the ADS Stage 1 valves open. When the ADS is actuated, the mixture level increases in the pressurizer (Figure 15.6.5.4B-18) because an opening has been created at the top of the pressurizer. After these valves open, a more rapid depressurization occurs as seen in Figure 15.6.5.4B-17(a); the accumulator setpoint is reached and the accumulators begin to inject. The injection flow from the core makeup tanks are shown in Figures 15.6.5.4B-22 and -23, and from the accumulators, in Figures 15.6.5.4B-24 and -25.

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As Figures 15.6.5.4B-22 and -23 indicate, when the accumulators begin to inject, the flow from both core makeup tanks is reduced, and the flow is temporarily stopped due to the pressurization of the core makeup tanks injection lines by the accumulators.

The ADS Stage 2 valves open, maintaining the depressurization rate as shown in Figure 15.6.5.4B-17(a). ADS Stage 3 valves open, thereby increasing the system venting capability. Figures 15.6.5.4B-31(a), -31(b) and 31(c) indicate the instantaneous liquid, instantaneous vapor and integrated total mass discharged from the ADS Stage 1-3 valves. The ADS Stage 4 valves open when the core makeup tank water level is reduced to 20 percent. Figures 15.6.5.4B-28(a), -28(b) and -28(c) indicate the instantaneous liquid, instantaneous vapor and integrated total mass discharged from the ADS Stage 4 valves. After the ADS Stage 4 path opens, the pressurizer begins to drain mixture into the hot legs as seen in Figure 15.6.5.4B-18. After the CMTs are empty, IRWST injection does not begin until the pressure in the DVI line drops below the IRWST injection pressure, creating an injection gap (Table 15.6.5-12 and Figures 15.6.5.4B-22, -23, -26 and -27). The mass inventory shown in Figure 15.6.5.4B-29(a) considers the primary inventory to be the reactor coolant system proper, including the pressurizer; the mass present in the passive safety system components is not included. The mass inventory shown in Figure 15.6.5.4B-29(b) considers the reactor vessel mixture inventory, including the downcomer, lower plenum, core fluid channel, upper plenum and upper head, which shows the decrease in the inventory during the injection gap period. Once the pressures in the DVI lines drop below the IRWST injection pressure, flow enters the reactor vessel from the IRWST. The mixture level in the reactor vessel is approximately at the hot leg elevation as

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shown in Figure 15.6.5.4B-30(a) for the majority of the transient; however, the upper plenum mixture level drops during the injection gap period and the core briefly uncovers as the mixture level drops below the top of the active fuel. The 2-inch break cases exhibit minimal heatup as a result of the core uncovery as shown in Figure 15.6.5.4B-30(b).

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15.6.5.4B.3.5 Direct Vessel Injection Line Break

This case models the double-ended rupture of the DVI line at the nozzle into the downcomer. The broken loop injection system (consisting of an accumulator, a core makeup tank, and an IRWST delivery line) is modeled to spill completely out the DVI side of the break. The steady-state reactor coolant system conditions for this transient are shown in Table 15.6.5-9. Design maximum resistances are applied to the inlet and outlet lines of the intact loop core makeup tank to conservatively minimize intact loop core makeup tank delivery through the time of minimum reactor coolant system mass inventory. Minimum resistances are applied to the broken loop IRWST injection line to maximize the spill to containment, thus minimizing the reactor coolant system mass inventory. This case uses a containment backpressure defined to be a constant 20 psia. While not exactly reflecting the containment pressure history that occurs as a result of the DVI line break, it represents a conservatively low estimate of the expected containment pressure response during a DEDVI transient. The containment pressurizes for a DEDVI break as a result of the break mass and energy releases in addition to the ADS-4 discharge paths that vent directly to the containment atmosphere.

The containment pressurization was calculated using the mass and energy releases from the NOTRUMP small-break LOCA code in the WGOTHIC containment model. Mass and energy releases from both sides of the DVI break (both vessel side and DVI side) and ADS-4 valve discharges were provided in a tabular form to the WGOTHIC AP1000 model used to compute containment pressurization for the long-term cooling analysis.

The event times for this transient are shown in Table 15.6.5-13. Transient results are shown in Figures 15.6.5.4B-36 through 15.6.5.4B-55. The break is assumed to open instantaneously at 0 seconds. The accumulator on the broken loop starts to discharge via the DVI line to the containment. Figure 15.6.5.4B-36 shows the subcooled discharge from the downcomer nozzle, which causes a rapid reactor coolant system (RCS) depressurization (Figure 15.6.5.4B-38(a) and Figure 15.6.5.4B-38(b)). A reactor trip signal is generated, followed by generation of the "S" signal. Following a delay, the isolation valves on the core makeup tank and PRHR delivery lines begin to open. The PRHR heat removal and integrated heat removal are shown in Figure 15.6.5.4B-54 and Figure 15.6.5.4B-55. The "S" signal also causes closure of the main feedwater isolation valves after a 2-second delay and trips the reactor coolant pumps after a 7.3-second delay. The opening of the core makeup tank isolation valves allows the broken loop core makeup

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tank to discharge directly to the containment (Figure 15.6.5.4B-39), and a small circulatory flow develops through the intact loop core makeup tank (Figure 15.6.5.4B-40).

As the pressure falls, the reactor coolant system fluid saturates, and a mixture level forms in the upper plenum and then falls to the hot leg elevation (Figure 15.6.5.4B-41). The upper parts of the reactor coolant system start to drain, and a mixture level forms in the downcomer (Figure 15.6.5.4B-42) and falls below the elevation of the break. Two-phase discharge, then vapor flow occurs from the downcomer side of the break (Figure 15.6.5.4B-37).

In the core makeup tank connected to the broken loop, a level forms and starts to fall. The ADS Stage 1 setpoint is reached, and the ADS Stage 1 valves open after the signal delay time elapses. The ensuing steam discharge from the top of the pressurizer (through the ADS valves; Figures 15.6.5.4B-43(a), -43(b) and -43(c)) increases the reactor coolant system depressurization rate. The depressurization rate is also increased due to the steam discharge from the downcomer to the containment (Figure 15.6.5.4B-37) as the downcomer mixture level falls below the DVI nozzle (Figure 15.6.5.4B-42).

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During the initial portion of the DEDVI break, only liquid flows out the top of the core (Figure 15.6.5.4B-45). Soon, steam flow follows (Figure 15.6.5.4B-46) correlating with the void fraction increase in the core (Figure 15.6.5.4B-44). The break in the downcomer stalls fluid flow into the bottom of the core (Figure 15.6.5.4B-47) leaving insufficient liquid in the upper plenum. The mixture level therefore starts to decrease (Figure 15.6.5.4B-41). The mixture level falls early in the transient and then starts to recover, as flow slowly re-enters the core from the downcomer (Figure 15.6.5.4B-41 compared to -47).

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The ADS Stage 2 valves open after the appropriate time delay between the actuation of the first two stages of the ADS. The intact loop accumulator starts to inject into the downcomer (Figure 15.6.5.4B-50) causing the mixture level in the downcomer to slowly rise (Figure 15.6.5.4B-42). The mixture level also increases slightly within the upper plenum.

The ADS Stage 3 valves open upon completion of the time delay of 120 seconds between the actuation of Stages 2 and 3 of the ADS. The broken loop core makeup tank level reaches the ADS Stage 4 setpoint, but the ADS Stage 4 valves do not open until the minimum time delay between the actuation of ADS Stages 3 and 4 occurs. Two-phase discharge ensues through three of the four Stage 4 paths (Figures 15.6.5.4B-48(a), -48(b) and -49). During the same timeframe, the broken loop core makeup tank and accumulator empty rapidly.

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The fluid level at the top of the intact loop core makeup tank starts to decrease slowly (Figure 15.6.5.4B-52) because injection from the tank has begun (Figure 15.6.5.4B-40). The intact loop accumulator empties (Figure 15.6.5.4B-50), temporarily interrupting CMT injection, and the reduced pressure in the injection line allows the core makeup tank to inject continuously.

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During the period of accumulator injection, the downcomer mixture level rises slowly (Figure 15.6.5.4B-42). Figure 15.6.5.4B-53(a) presents the RCS mass inventory. Figure 15.6.5.4B-53(b) presents the reactor vessel mixture inventory which includes the downcomer, lower plenum, core fluid channel, upper plenum and upper head. With injection available only from the intact loop core makeup tank for a period of time, the downcomer level remains fairly constant and core boil-off increases the rate of reactor vessel mixture inventory depletion until sufficient CMT/IRWST injection flow can be introduced. The level in the upper plenum is maintained near the hot leg elevation (Figure 15.6.5.4B-41) throughout the remainder of the transient.

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Once the pressure in the broken DVI line falls below that in the IRWST, the water from the tank begins spilling to containment.

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Stable, but decreasing, injection continues from the intact loop core makeup tank as the inventory slowly depletes; the reactor coolant system pressure declines slowly. The reactor coolant system pressure continues to fall until it drops below that of the IRWST and injection begins (Figure 15.6.5.4B-51). With the reduced initial RCS inventory recovery from the accumulators and only a single intact injection path available for the DEDVI line break, the minimum inventory occurs after the initiation of continuous IRWST injection flow. After injection flow greater than the sum of the break and ADS flows exists, a slow rise in the reactor vessel mixture inventory (Figure 15.6.5.4B-53(b)) occurs. Since no core uncovery is predicted for this scenario, no cladding heatup occurs.

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Another DEDVI line break analysis is performed that is the same as the case discussed above except that containment pressure is assumed to be at 14.7 psia. Table 15.6.5-13A provides the time sequence of events for this analysis. Figures 15.6.5.4B-36A through -55A provide the transient results for this analysis. The transient is like the case at 20 psia except that IRWST injection occurs somewhat later due to the lower containment pressure causing a drop in the upper plenum mixture level to the top of the active fuel with brief uncovery periods.

The critical heat flux assessment described in subsection 15.6.5.4B.2.3 addresses core cooling during a time period where the NOTRUMP computer code may not conservatively predict the core average void fraction. The requirements of 10 CFR 50.46 are met during this period since this CHF assessment indicates peak core heat flux is less than critical heat flux. Cladding temperatures will remain near the coolant saturation temperature, well below the 10 CFR 50.46 peak cladding temperature limit.

15.6.5.4B.3.6 10-inch Cold Leg Break

This case models a 10-inch break occurring in the cold leg connected to the balance line of CMT-1. The reactor steady-state initial conditions assumed for this transient are found in Table 15.6.5-9. The event times for this transient are given in Table 15.6.5-14.

Transient results are shown in Figures 15.6.5.4B-56(a) through 15.6.5.4B-78. The break opens at time zero, and the pressurizer pressure begins to fall, as shown in Figures 15.6.5.4B-56(a) and 15.6.5.4B-56(b), as mass is lost out the break. The pressurizer mixture level initially decreases as given in Figure 15.6.5.4B-57. The break fluid flow is shown in Figures 15.6.5.4B-75 and -76 for the liquid and vapor components respectively. The pressurizer pressure falls below the reactor trip set point. This causes the reactor to trip (after the appropriate time delay) and isolation of the steam generator steam lines. The core makeup tank isolation valves on both delivery lines and the PRHR delivery line isolation valve open after an "S" signal occurs (with appropriate delays); the reactor coolant pumps trip after an "S" signal with a 7.3-second delay. The reactor coolant system is cooled by natural circulation with energy being removed by the steam generator safety valves, the core makeup tanks, and the PRHR heat exchanger. The PRHR heat removal rate and integrated heat removal are shown in Figure 15.6.5.4B-77 and Figure 15.6.5.4B-78. Once the core makeup tank isolation valves open, the core makeup tanks begin to inject borated water into the reactor coolant system as shown in Figures 15.6.5.4B-61 and -62.

As time proceeds, the loops drain to the reactor vessel. The mixture level in the downcomer begins to drop as seen in Figure 15.6.5.4B-60, and the core remains completely covered with the exception of a few short oscillatory time intervals in which the mixture level drops below the active fuel (Figure 15.6.5.4B-69). Due to the size and location of the break involved, the accumulator setpoint is reached prior to the core makeup tanks transitioning from recirculation to injection mode. The flows from the core makeup tanks are shown in Figures 15.6.5.4B-61 and -62, and from the accumulators, in Figures 15.6.5.4B-63 and -64. Core makeup tank 2 reaches the 67.5-percent level first, and after an appropriate delay, the ADS Stage 1 valves open. When the ADS is actuated, the mixture level increases in the pressurizer (Figure 15.6.5.4B-57) because an opening has been created at the top of the pressurizer. After these valves open, a more rapid depressurization occurs as seen in Figure 15.6.5.4B-56(a).

During the initial portion of the 10-inch break, both liquid and steam flow out the top of the core (Figures 15.6.5.4B-71 and -72) as the void fraction in the core increases (Figure 15.6.5.4B-73). The break in the cold leg draws fluid from the bottom of the core, leaving insufficient liquid in the upper plenum. The mixture level, therefore, starts to decrease (Figure 15.6.5.4B-69). The mixture level falls until accumulator flows enter the downcomer (Figures 15.6.5.4B-63 and -64).

As Figures 15.6.5.4B-61 and -62 indicate, when the accumulators begin to inject, the flow from both core makeup tanks is reduced and the flow is nearly reduced to zero due to the

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pressurization of the injection lines of the core makeup tanks by the accumulators. The opening of ADS Stage 2 valves maintains the depressurization rate as shown in Figure 15.6.5.4B-56(a). ADS Stage 3 valves subsequently open. This increases the system venting capability. Figures 15.6.5.4B-70(a), -70(b) and -70(c) indicate the instantaneous liquid, instantaneous vapor and integrated total mass discharged from the ADS Stage 1-3 valves, respectively. The ADS Stage 4 valves open when the core makeup tank water level is reduced to 20 percent. Figures 15.6.5.4B-67(a), -67(b) and -74 indicate the instantaneous liquid, instantaneous vapor and integrated total mass discharged from the ADS Stage 4 valves. After the ADS Stage 4 path opens, the pressurizer begins to drain mixture into the hot legs as seen in Figure 15.6.5.4B-57. The Figure 15.6.5.4B-68(a) mass inventory plot considers the primary inventory to be the reactor coolant system proper, including the pressurizer; the mass present in the passive safety system components is not included. The Figure 15.6.5.4B-68(b) mass inventory plot considers the reactor vessel mixture inventory, including the downcomer, lower plenum, core fluid channels, upper plenum and upper head. Once the downcomer pressure drops below the IRWST injection pressure, flow enters the reactor vessel from the IRWST. The mixture level in the reactor vessel is approximately at the hot leg elevation as shown in Figure 15.6.5.4B-69 throughout this transient; core uncovery does not occur for any prolonged period of time and may be deemed negligible. The 10-inch break case exhibits large margins to the 10 CFR 50.46 Appendix-K limit of 2200°F (1204.44°C).

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15.6.5.4B.3.7 Direct Vessel Injection Line Break (Entrainment Sensitivity)

In order to assess the potential impact of higher than expected entrainment in the upper plenum and hot legs on the overall system response and core cooling, an AP1000 plant sensitivity run was performed. The simulation utilizes the same initial conditions as the base DEDVI line simulation with a single failure of an ADS-4 valve on the pressurizer side, previously presented in subsection 15.6.5.4B.3.5. The DEDVI line simulation currently presented in subsection 15.6.5.4B.3.5 has been updated to address the limiting single failure of an ADS-4 line on the non-pressurizer side. The DEDVI line break entrainment sensitivity did not need to be updated to reflect this condition. The results of the 20 psia containment backpressure DEDVI line break transient response presented in subsection 15.6.5.4B.3.5 are not significantly different as a result of the single failure assumption change. While some transient timing differences exist between the results, the overall behavior is very similar. In addition, the change in the single failure assumption impacts the transient results after the ADS-4 valves actuate. For the sensitivity case, the upper plenum and hot legs are transitioned to homogeneous conditions at this time and the results will be very similar regardless of the single failure assumption, since the entire inventory in the upper plenum will be very rapidly discharged to containment. As such, the entrainment sensitivity study and results presented herein represents a valid entrainment sensitivity for the 20 psia containment backpressure DEDVI line transient. The sensitivity case presented herein was performed with the DEDVI line break simulation as described in the following.

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The transient response is essentially identical until ADS-4 actuation, at which time the bounding entrainment conditions are included in the analysis by assuming homogenous conditions in the regions downstream of the core (upper plenum, hot leg, and pressurizer inlet). In addition, since homogenous treatment of these regions will eliminate the pressure drop effect of the accumulated mass stored in the upper plenum, the NOTRUMP model was conservatively adjusted to account for this effect following the transition of the ADS-4 flow paths to noncritical conditions.

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The event times for this transient are shown in Table 15.6.5-15. Transient results are shown in Figures 15.6.5.4B-79(a) through 15.6.5.4B-90. Figures 15.6.5.4B-79(a) and 15.6.5.4B-79(b) present comparison of the pressure in the upper portion of the downcomer between the base and sensitivity cases. The sensitivity case results in higher pressure in the upper portion of the downcomer and subsequently results in delayed IRWST injection (Figure 15.6.5.4B-80). This can also be observed in the intact DVI line flow, which comprises all intact injection flow components (accumulator, CMT, and IRWST) per Figure 15.6.5.4B-81, and the pressurizer mixture level response (Figure 15.6.5.4B-90), which follows the change in pressure response. As expected, the initial ADS-4 liquid discharge is much higher (Figure 15.6.5.4B-82) until the inventory, which resided in the upper plenum and hot leg regions, depletes (Figure 15.6.5.4B-83). The net effect is a decrease in the ADS-4 vapor discharge rate (Figure 15.6.5.4B-84) and subsequently higher RCS pressures.

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Due to the elimination of the inventory stored in the upper plenum, the downcomer mass is also reduced (Figure 15.6.5.4B-85). Since the static head that existed in the upper plenum is eliminated when the model is made homogenous, the downcomer mixture is subsequently driven into the core as the static heads equilibrate. This results in the core region mass increasing initially due to the introduction of cold downcomer fluid to the core region (Figure 15.6.5.4B-86). The net effect of the sensitivity case is that the vessel inventory is substantially decreased over the base model simulation (Figure 15.6.5.4B-87); however, this inventory is sufficient to provide adequate core cooling because the ADS-4 continually draws liquid flow through the core (Figure 15.6.5.4B-82). Even though there is no liquid storage in the upper plenum for the homogenous case (Figure 15.6.5.4B-88), the core collapsed liquid level (Figure 15.6.5.4B-89) is not impacted significantly.

This sensitivity demonstrates that the AP1000 plant response is relatively insensitive to upper plenum and hot leg entrainment. Even with the assumption of homogenous fluid nodes above the core, adequate core cooling is demonstrated. No significant core uncovery/heatup is predicted for this scenario.

15.6.5.4B.4 Conclusions

The small-break LOCA analyses performed show that the performance of the AP1000 plant design to small-break LOCA scenarios is excellent and that the passive safeguards systems in the AP1000 are sufficient to mitigate small-break LOCAs. Specifically, it is concluded that:

- The primary side can be depressurized by the ADS to allow stable injection into the core.
- Injection from the core makeup tanks, accumulators, and IRWST prevents excessive cladding heatup for small-break LOCAs analyzed, including double-ended ruptures in the passive safeguards system lines. The peak AP1000 heat flux during the accumulator injection period is below the predicted critical heat flux.
- The effect of increasing upper plenum/hot leg entrainment does not significantly affect plant safety margins.

The analyses performed demonstrate that the 10 CFR 50.46 Acceptance Criteria are met by the AP1000. Summarizing the small-break LOCA spectrum:

Break Location/Diameter	AP1000 Plant Minimum Reactor Vessel Mixture Inventory (lbm)	Peak Cladding Temperature(°F)
Inadvertent ADS	56,080	654.7
2-inch cold leg break	55,644	663.5
10-inch cold leg break	71,849	(1)
DEDVI (20.0 psia)	72,879	(1)
DEDVI (Entrainment Study)	57,364	(1)

The 2-inch cold leg break exhibits the limiting minimum reactor vessel mixture inventory conditions and the limiting peak cladding temperature. The AP1000 design is such that the minimum reactor vessel mixture inventory occurs around the time of IRWST injection for most breaks. All breaks simulated in the break spectrum produce results that demonstrate significant margin to peak cladding temperature regulatory limits.

15.6.5.4C Post-LOCA Long-Term Cooling

15.6.5.4C.1 Long-Term Cooling Analysis Methodology

The AP1000 safety-related systems are designed to provide adequate cooling of the reactor indefinitely. Initially, this is achieved by discharging water from the IRWST into the vessel.

(1) There is no core heatup as a result of this transient. PCT occurs at transient initiation.

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When the low-3 level setpoint is reached in the IRWST, the containment recirculation subsystem isolation valves open and water from the containment reactor coolant system (RCS) compartment can flow into the vessel through the PXS piping. The water in containment rises in temperature toward the saturation temperature. Long-term heat removal from the reactor and containment is by heat transfer through the containment shell to atmosphere.

The purpose of the long-term cooling analysis is to demonstrate that the passive systems provide adequate emergency core cooling system performance during the IRWST injection/containment recirculation time scale. The long-term cooling analysis is performed using the WCOBRA/TRAC computer code to verify that the passive injection system is providing sufficient flow to the reactor vessel to cool the core and to preclude boron precipitation.

The AP1000 long-term cooling analysis is supported by the series of tests at the Oregon State University AP600 APEX Test Facility. This test facility is designed to represent the AP600 reactor safety-related systems and nonsafety-related systems at quarter-scale during long-term cooling. The data obtained during testing at this facility has been shown to apply to the AP1000 (Reference 25). These tests were modeled using WCOBRA/TRAC with an equivalent noding scheme to that used for AP600 (Reference 17) in order to validate the code for long-term cooling analysis.

Reference 24 provides details of the AP1000 WCOBRA/TRAC modeling. The coarse reactor vessel modeling used for AP600 has been replaced with a detailed noding like that applied in the large-break LOCA analyses described in subsection 15.6.5.4A. The reactor vessel noding used in the AP1000 long-term cooling analyses in core and upper plenum regions is equivalent to that used in full-scale test simulations (see Reference 24).

A DEDVI line break is analyzed because it is the most limiting long-term cooling case in the relationship between decay power and available liquid driving head. Because the IRWST spills directly onto the containment floor in a DEDVI break, this event has the highest core decay power when the transfer to sump injection is initiated. In postulated DEDVI break cases, the compartment water level exceeds the elevation at which the DVI line enters the reactor vessel, so water can flow from the containment into the reactor vessel through the broken DVI line; this in-flow of water through the broken DVI line assists in the heat removal from the core. The steam produced by boiling in the core vents to the containment through the ADS valves and condenses on the inner surface of the steel containment vessel. The condensate is collected and drains to the IRWST to become available for injection into the reactor coolant system. The WCOBRA/TRAC analysis presented analyzes the DEDVI small-break LOCA event from a time (3000 seconds) at which IRWST injection is fully established to beyond the time of containment recirculation. During this time, the head of water to drive the flow into the vessel for IRWST injection decreases from the initial level to its lowest value at the containment recirculation

switchover time. PXS Room B is the location of the break in the DVI line. At this break location, liquid level in containment at the time of recirculation is a minimum.

A continuous analysis of the post-LOCA long term cooling is provided from the time of stable IRWST injection through the time of sump recirculation for the DEDVI break. Maximum design resistances are applied in WCOBRA/TRAC for both the ADS Stage 4 flow paths and the IRWST injection and containment recirculation flow paths.

The break modeled is a double-ended guillotine rupture of one of the direct vessel injection lines. The long-term cooling phase begins after the simultaneous opening of the isolation valves in the IRWST DVI lines and the opening of ADS Stage 4 squib valves, when flow injection from the IRWST has been fully established. Initial conditions are consistent with the NOTRUMP DEDVI case at 20 psia containment pressure reported in subsection 15.6.5.4B.

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15.6.5.4C.2 DEDVI Line Break with ADS Stage 4 Single Failure, Passive Core Cooling System Only Case; Continuous Case

This subsection presents the results of a DEDVI line break analysis during IRWST injection phase continuing into sump recirculation. Initial conditions at the start of the case are prescribed based on the NOTRUMP DEDVI break results to allow a calculation to begin shortly after IRWST injection begins in the small break long-term cooling transient. The WCOBRA/TRAC calculation is then allowed to proceed until a quasi-steady-state is achieved. At this time, the predicted results are independent of the assumed initial conditions. This calculation uses boundary conditions taken from a WGOTHIC analysis of this event. During the calculation, which is carried out for 10,000 seconds until a quasi-steady-state sump recirculation condition has been established, the IRWST water level is decreased continuously until the sump recirculation setpoint is reached.

In the analysis, one of the two ADS Stage 4 valves in the PRHR loop is assumed to have failed. The initial reactor coolant system liquid inventory and temperatures are determined from the NOTRUMP calculation. The core makeup tanks do not contribute to the DVI injection during this phase of the transient. Steam generator secondary side conditions are taken from the NOTRUMP calculation (at the beginning of long-term cooling). The reactor coolant pumps are tripped and not rotating.

The temperatures of the liquid in the containment sump and the containment pressure are based on WGOTHIC calculations of the conservative minimum pressure during this long-term cooling transient, including operation of the containment fan coolers. Small changes in the RCS compartment level do not have a major effect on the predicted core collapsed liquid level or on the predicted flow rate through the core. The minimum compartment floodup level for this break scenario is 107.8 feet or greater.

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In this transient, the IRWST provides a hydraulic head sufficient to drive water into the downcomer through the intact DVI nozzle. Also, water flows into the downcomer from the broken DVI line once the liquid level in the compartment with the broken line is adequate to support flow. The water flows down the downcomer and up through the core, into the upper plenum. Steam produced in the core and liquid flow out of the reactor coolant system via the ADS Stage 4 valves. There is little flow out of ADS Stages 1, 2, and 3 even when the IRWST liquid level falls below the sparger elevation, so they are not modeled in this calculation. The venting provided by the ADS-4 paths enables the liquid flow through the core to maintain core cooling.

Approximately 500 seconds of WCOBRA/TRAC calculation are required to establish the quasi-steady-state condition associated with IRWST injection at the start of long-term cooling and so are ignored in the following discussion. The hot leg levels are such that during the IRWST injection phase the quality of the ADS Stage 4 mass flows varies as water is carried out of the hot legs. This periodically increases the pressure drop across the ADS Stage 4 valves and the upper plenum pressure. The higher pressure in the upper plenum reduces the injection flow. This cycle of pressure variations due to changing void fractions in the flow through ADS Stage 4 is consistent with test observations and is expected to recur often during long-term cooling.

The head of water in the IRWST causes a flow of subcooled water into the downcomer at an approximate rate of 180 lbm/s through the intact DVI nozzle at the start of long-term cooling. The downcomer level at the end of the code initiation (the start of long-term cooling) is about 18.0 feet (Figure 15.6.5.4C-1). Note that the time scale of this and other figures in subsection 15.6.5.4C.2 is offset by 2500 seconds; that is, a time of 500 seconds on the Figure 15.6.5.4C-1 axis equals 3000 seconds transient time for the DEDVI break. All of the injection water flows down the downcomer and up through the core. The accumulators have been fully discharged before the start of the time window and do not contribute to the DVI flow.

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Boiling in the core produces steam and a two-phase mixture, which flows into the upper plenum. The core is 14 feet high, and the core average collapsed liquid level (Figure 15.6.5.4C-2) is shown from the start of long-term cooling. The boiling process causes a variable rate of steam production and resulting pressure changes, which in turn causes oscillations in the liquid flow rate at the bottom of the core and also variations in the core collapsed level and the flow rates of liquid and vapor out of the top of the core. In the WCOBRA/TRAC noding, the core is divided both axially and radially as described in Reference 24. The void fractions in the top two cells of the hot assembly are shown as Figures 15.6.5.4C-3 and -4. The average void fraction of these upper core cells is about 0.8 during long-term cooling, during IRWST injection, and into the containment recirculation period. **There is a continuous flow of two-phase fluid into the hot legs, and mainly vapor flow toward the ADS Stage 4 valve occurs at the top of the pipe. The collapsed liquid level in the hot leg averages around 1.5 feet (Figure 15.6.5.4C-5). The hot legs on average**

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are more than 50-percent full. Vapor and liquid flows at the top of the core are shown in Figures 15.6.5.4C-6 and 15.6.5.4C-7, the upper plenum collapsed liquid level in Figure 15.6.5.4C-8. Figures 15.6.5.4C-9 and 15.6.5.4C-10 are ADS stage 4 mass flowrates.

The pressure in the upper plenum is shown in Figure 15.6.5.4C-11. The upper plenum pressure fluctuation that occurs is due to the ADS Stage 4 water discharge. The PCT of the hot rod follows saturation temperature (Figure 15.6.5.4C-12), which demonstrates that no uncovery and no cladding temperature excursion occurs. A small pressure drop is calculated across the reactor vessel, and injection rates through the DVI lines into the vessel are presented in Figures 15.6.5.4C-13 and -14. Figure 15.6.5.4C-14 shows the broken DVI line flow during the start of the long-term cooling period increases to about 75 lbm/s after the compartment water level has increased above the nozzle elevation to permit liquid injection into the reactor vessel. In contrast, the intact DVI line flow falls from 180 lbm/s with a full IRWST to about 77 lbm/s flow from the containment at the end of the calculation. The recirculation core liquid throughput is more than adequate to preclude any boron buildup on the fuel.

15.6.5.4C.3 DEDVI Break and Wall-to-Wall Floodup; Containment Recirculation

This subsection presents a DEDVI line break analysis with wall-to-wall flooding due to leakage between compartments, using the window mode methodology. All containment free volume beneath the level of the liquid is assumed filled in this calculation to generate the minimum water level condition during containment recirculation. The time identified for this calculation is 14 days into the event, and the core power is calculated accordingly. The initial conditions at the start of the window are consistent with the analysis described in subsection 15.6.5.4C.2. Containment recirculation is simulated during the time window. The calculation is carried out over a time period long enough to establish a quasi-steady-state solution; after 500 seconds of problem time, the flow dynamics are quasi-steady-state and the predicted results are independent of the assumed initial conditions. The liquid level is simulated constant at 28.2 feet above the bottom inside surface of the reactor vessel (refer to Figure 15.0.3-2 for AP1000 reference plant elevations) during the time window, and the liquid temperatures in the containment sump and the PXS "B" room are 196°F and 182°F, respectively. The containment pressure is conservatively assumed to be 14.7 psia. The single failure of an ADS Stage 4 flow path is assumed as in the subsection 15.6.5.4C.2 case.

Focusing on the post 400-second time interval of this case, the containment liquid provides a hydraulic head sufficient to drive water into the downcomer through the DVI nozzles. The water introduced into the downcomer flows down the downcomer and up through the core, into the upper plenum. Steam produced in the core entrains liquid and flows out of the reactor coolant system via the ADS Stage 4 valves. The DVI flow and the venting provided by the ADS paths provide a liquid flow through the core that enables the core to remain cool.

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The downcomer collapsed liquid level (Figure 15.6.5.4C-15) varies between 24 and 25 feet during the analysis. Pressure spikes produced by boiling in the core can cause the mass flow of the DVI flow rates shown in Figures 15.6.5.4C-27 and -28 into the vessel to fluctuate upward and downward.

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Boiling in the core produces steam and a two-phase mixture, which flows out of the core into the upper plenum. The core is 14 feet high, and the core collapsed liquid level (Figure 15.6.5.4C-16) maintains a mean level close to the top of the core. The boiling process causes pressure variations, which in turn, cause variations in the core collapsed level and the flow rates of liquid and vapor out of the top of the core. In the WCOBRA/TRAC analysis, the core is nodalized as described in Reference 24. The void fraction in the top cell is shown in Figure 15.6.5.4C-17 for the core hot assembly, and Figure 15.6.5.4C-18 shows the void fraction that exists one cell further down in the hot assembly. The PCT does not rise appreciably above the saturation temperature (Figure 15.6.5.4C.3-26). The flow through the core and out of the reactor coolant system is more than sufficient to provide adequate flushing to preclude concentration of the boric acid solution. Liquid collects above the upper core plate in the upper plenum, where the average collapsed liquid level is about 3.6 feet (Figure 15.6.5.4C-22). There is no significant flow through the cold legs into either the broken or the intact loops, and there is no significant quantity of liquid residing in any of the cold legs.

The pressure in the upper plenum is shown in Figure 15.6.5.4C-25. The upper plenum pressurization, which occurs periodically, is due to the ADS Stage 4 water discharge. The collapsed liquid level in the hot leg of the pressurizer loop varies between 1.3 feet and 2.0 feet, as shown in Figure 15.6.5.4C-19. Injection rates through the DVI lines into the vessel are presented in Figures 15.6.5.4C-27 and -28.

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15.6.5.4C.4 Post Accident Core Boron Concentration

An evaluation has been performed of the potential for the boron concentration to build up in the core following a cold leg LOCA. The evaluation methodology, simplified calculations, and their results are discussed in Reference 24. This evaluation considers both short-term operations, before ADS is actuated, and long-term operations, after ADS is actuated. These evaluations and their results are discussed in the follow paragraphs.

Short-Term – Prior to ADS actuation, it is not likely for boron to build up significantly in the core. Normally, water circulation mixes boron in the RCS and prevents buildup in the core. In order for boron to start to build up in the core region, water circulation through the steam generators and PRHR HX has to stop. In addition, significant injection of borated water is needed from the CMTs and the CVS. For this situation to happen, the hot legs need to void sufficiently to allow the steam generator tubes to drain. Once the steam generator tubes void, the cold legs will also void since they are located higher than the hot legs. When the top of the cold legs void,

the CMTs will begin to drain. When the CMTs drain to the ADS stage 1 setpoint, ADS is actuated.

Short-Term Results – As shown in subsection 15.6.5.4B.3.4, a 2-inch LOCA requires less than 16 minutes from the time that the hot legs void significantly until ADS is actuated. For larger LOCAs, this time difference is shorter, as seen for the 10-inch cold leg LOCA (subsection 15.6.5.4B.3.6). The core boron concentration will not build up significantly in this short time. If the break is smaller than 2 inches, voiding of the hot legs will occur at a later time. With maximum operation of CVS makeup, it takes more than 3 hours for the core boron concentration to build up significantly. In addition, the volume of the boric acid tank limits the maximum buildup of boron in the core.

Following a small LOCA where ADS is not actuated, the operators are guided to sample the RCS boron concentration and to initiate a post-LOCA cooldown and depressurization. The cooldown and depressurization of the RCS reduces the leak rate and facilitates recovery of the pressurizer level. Recovery of the pressurizer level allows for re-establishment of water flow through the RCS loops, which mixes the boron. The operators are guided to take an RCS boron sample within 3 hours of the accident and several more during the plant cooldown. The purpose of the boron samples is to assess that there is adequate shutdown margin and that the RCS boron concentration has not built up to excessive levels. The maximum calculated core boron concentration 3 hours after a LOCA without ADS actuation is less than 16,000 ppm. Operator action within 3 hours maintains the maximum core boron concentration well below the boron solubility limit for the core inlet temperatures during the cooldown.

Long-Term – Once ADS is actuated, water carryover out the ADS Stage 4 lines limits the potential core boron concentration buildup following a cold leg LOCA. The design of the AP1000 facilitates water discharge from the hot legs as follows:

- PXS recirculation flow capability tends to fill the hot legs and bring the water level up to the ADS Stage 4 inlet.
- ADS Stage 4 lines discharge at an elevation 3 to 4 feet above the containment water level.

With water carried out ADS Stage 4, the core boron concentration increases until the boron added to the core in the safety injection flow equals the boron removed in the water leaving the RCS through the ADS Stage 4 flow. The lower the ADS Stage 4 vent quality, the lower the core boron concentration buildup.

Long-Term Results – Analyses have been performed (Reference 24) to bound the maximum core boron concentration buildup. These analyses demonstrate that highest ADS Stage 4 vent qualities result from the following:

- Highest decay heat levels
- Lowest PXS injection/ADS 4 vent flows, including high line resistances and low containment water levels

The long-term cooling analysis discussed in subsection 15.6.5.4C.2 is consistent with these assumptions. The ADS Stage 4 vent quality resulting from this analysis is less than 40 percent at the beginning of IRWST injection and reaches a maximum of less than 50 percent around the initiation of recirculation. It decreases after this peak, dropping to a value less than 8 percent at 14 days.

With the maximum ADS Stage 4 vent qualities, the maximum core boron concentration peaks at a value of about 7400 ppm at the time of recirculation initiation. After this time, the core boron concentration decreases as the ADS Stage 4 vent quality decreases, reaching 5000 ppm about 9 hours after the accident. The core boron solubility temperature reaches a maximum of 58°F (at 7400 ppm) and quickly drops to 40°F (at 5000 ppm). With these low core boron solubility temperatures, there is no concern with cold PXS injection water causing boron precipitation in the core. With the IRWST located inside containment, its water temperature is normally expected to be above these solubility temperatures. The minimum core inlet temperature is greater than the solubility temperature considering heatup of the injection by steam condensation in the downcomer and pickup of sensible heat from the reactor vessel, core barrel, and lower support plate.

The boron concentration water in the containment is initially about 2980 ppm. As the core boron concentration increases, the containment concentration decreases slightly. The minimum boron concentration in containment is greater than 2950 ppm. The solubility temperature of the containment water at its maximum boron concentration is 32°F.

With high decay heat values, the ADS Stage 4 vent flows and velocities are high. These high vent velocities result in flow regimes that are annular for more than 30 days. The annular flow regime moves water up and out the ADS Stage 4 lines. This flow regime is based on the Taitel-Dukler vertical flow regime map. Lower decay heat levels can be postulated later in time or just after a refueling outage. Significantly lower decay heat levels result in lower ADS Stage 4 vent qualities. They also result in ADS Stage 4 vent flows/velocities that are lower. Even with low ADS Stage 4 vent flow velocities, the AP1000 plant will move water out the ADS Stage 4 operating as a manometer. Small amounts of steam generated in the core reduce the density of the steam/water mixture in the core, upper plenum, and ADS Stage 4 line as it bubbles up through the water. As a result, the injection head is sufficient to push the less dense, bubbly steam/water mix out the ADS Stage 4 line.

At the time recirculation begins, the containment level will be about 109.3 feet (for a non-DVI LOCA) and will be about 108.0 feet (for a DVI LOCA). Over a period of weeks after a LOCA, water may slowly leak from the flooded areas in containment to other areas inside containment that did not initially flood. As a result, the minimum containment water could decrease to 103.5 feet. During recirculation operation following a LOCA and ADS actuation, the operators are guided to maintain the containment water level above the 107-foot elevation by adding borated water to the containment. In addition, if the plant continues to operate in the recirculation mode, the operators are guided to increase the level to 109 feet within 30 days of the accident. These actions provide additional margin in water flow through the ADS Stage 4 line. The operators are also guided to sample the hot leg boron concentration prior to initiating recovery actions that might introduce low temperature water to the reactor.

15.6.5.4C.5 Conclusions

Calculations of AP1000 long-term cooling performance have been performed using the WCOBRA/TRAC model developed for AP1000 and described in Reference 24. The DEDVI case was chosen because it reaches sump recirculation at the earliest time (and highest decay heat). A window mode case at the minimum containment water level postulated to occur 2 weeks into long-term cooling was also performed.

The DEDVI small-break LOCA exhibits no core uncovery due to its adequate reactor coolant system mass inventory condition during the long-term cooling phase from initiation into containment recirculation. Adequate flow through the core is provided to maintain a low cladding temperature and to prevent any buildup of boric acid on the fuel rods. The wall-to-wall floodup case using the window mode technique demonstrates that effective core cooling is also provided at the minimum containment water level. The results of these cases demonstrate the capability of the AP1000 passive systems to provide long-term cooling for a limiting LOCA event.

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