Response to

Request for Additional Information No. 29, Revision 0

8/01/2008

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 15.04.01 - Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition
SRP Section: 15.04.02 - Uncontrolled Control Rod Assembly Withdrawal at Power
SRP Section: 15.04.03 - Control Rod Misoperation (System Malfunction or Operator Error)
SRP Section: 15.04.04-15.04.05 - Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate
SRP Section: 15.04.08 - Spectrum of Rod Ejection Accidents (PWR)
SRP Section: 15.05.01-15.05.02 - Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory
Application Section: Ch 15
SRSB Branch

Question 15.04.01-1:

Please provide plots of DNBR and peak fuel centerline temperature as a function of time during this event to demonstrate that these limits are met.

> Regulatory basis: SRP 15.4.1 UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER STARTUP CONDITION REVIEW, III Review Procedures, 9 *states " For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria."*

Response to Question 15.04.01-1:

A plot of the minimum departure from nucleate boiling ratio (DNBR) and peak fuel centerline temperature normalized to their respective SAFDLs (specified acceptable fuel design limits) is provided in the enclosed new U.S. EPR FSAR, Tier 2, Figure 15.4-46—Uncontrolled Control Bank Withdrawal from a Subcritical or Low Power Startup Condition - Normalized Minimum DNBR and FCM to SAFDL. Because this event is initiated from a low power condition where the incore monitoring system is inactive, a deterministic evaluation of DNBR and fuel centerline melt (FCM) is performed with the thermal-hydraulic code LYNXT. This is consistent with ANP-10287P, Revision 0, "Incore Trip Setpoint and Transient Methodology for the U.S. EPR Topical Report." Additionally, a deterministic calculation of cladding strain for this event shows a maximum cladding strain of 0.80 percent (i.e., 80 percent of the one percent uniform cladding strain limit).

FSAR Impact:

U.S. EPR FSAR, Tier 2, Figure 15.4-46 will be added as described in the response and indicated on the enclosed markup. U.S. EPR FSAR, Tier 2, Section 15.4.1.3 will be revised as described in the response and indicated on the enclosed markup to incorporate the new figure.

Question 15.04.01-2:

Please explain Figure 15.4-4—"Uncontrolled Control Bank Withdrawal from a Subcritical or Low Power Startup Condition - Primary System Temperature". The figure shows the temperature excursion in the hot leg of loop 4 following reactor trip during coastdown of the RCPs. Explain the physical basis for the temperature excursion. Describe the S-RELAP model used in these calculations. Explain the consistency of these temperature predictions and the EPR non-LOCA vessel model shown in ANP-10263P-A, Rev 0, "Codes and Methods Applicability Report for the US EPR".

> Regulatory basis: SRP 15.4.1, review procedures item 6 states: *The significant results of the analysis should be presented and should include maximum power levels reached for the reactor and the peak fuel rod, reactor temperatures and pressures, maximum heat flux levels, and the related fuel duty.*

Response to Question 15.04.01-2:

Per discussion with the NRC staff during the May 2008 audit, this question was clarified to address the apparent difference in the temperature behavior of the four loops. To more clearly distinguish the hot leg and cold leg temperatures, U.S. EPR FSAR, Tier 2, Figure 15.4-4 is supported with two additional figures in the U.S. EPR FSAR, Tier 2, Section 15.4. Figure 15.4-44—Uncontrolled Control Bank Withdrawal from a Subcritical or Low Power Startup Condition - Primary Hot Leg Temperature shows the hot leg temperatures, and Figure 15.4-45—Uncontrolled Control Bank Withdrawal from a Subcritical or Low Power Startup Condition - Primary Cold Leg Temperature shows the cold leg temperatures. The figures show the coolant temperature behavior is nearly identical in the four loops.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Figure 15.4-44 and Figure 15.4-45 will be added as described in the response and indicated on the enclosed markup. U.S. EPR FSAR, Tier 2, Section 15.4.1.3 will be revised as indicated on the enclosed markup to incorporate the new figures.

Question 15.04.02-1:

Please provide plots of DNBR and peak fuel centerline temperature as a function of time during this event to demonstrate that these limits are met.

> *Regulatory basis: SRP 15.4.2 UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL AT POWER, III Review Procedure 7 states " For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria."*

Response to Question 15.04.02-1:

A response to this question will be provided by September 26, 2008.

Question 15.04.02-2:

For Table 15.4-4 “Uncontrolled Control Bank Withdrawal at Power” Include the scram reactivity used in the analysis in the Table.

> Regulatory basis: SRP 15.4.2 UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL AT POWER, III Review Procedure, 3 states *” For a PWR, the reviewer ascertains that a full range of Anticipated Operational Occurence conditions are analyzed; the AOO calculation models are adequate; and that scram response of the flux, temperature, or pressure instrumentation is correctly calculated.”*

Response to Question 15.04.02-2:

The scram reactivity values are provided in the revised U.S. EPR FSAR, Tier 2, Table 15.4-4. The new information appears in the last row of the table.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Table 15.4-4 will be revised as described in the response and indicated on the enclosed markup.

Question 15.04.03-1:

Please provide plots of DNBR and peak fuel centerline temperature as a function of time during this event to demonstrate that these limits are met.

> Regulatory basis: SRP 15.4.3 CONTROL ROD MISOPERATION (SYSTEM *MALFUNCTION* OR OPERATOR ERROR) III. REVIEW PROCEDURES 4 states *"For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria".*

Response to Question 15.04.03-1:

A response to this question will be provided by November 26, 2008.

Question 15.04.04-15.04.05-1:

Provide the calculated DNBR as a function of time during the transient.

> Regulatory basis: SRP 15.4.4 – 15.4.5 STARTUP OF AN INACTIVE LOOP OR RECIRCULATION LOOP AT AN INCORRECT TEMPERATURE, AND FLOW CONTROLLER MALFUNCTION CAUSING AN INCREASE IN BWR CORE FLOW *RATE includes 4. PWR without loop isolation valves: Startup of a pump in an inactive loop. III Procedure includes "The results of the analysis are reviewed and compared with the acceptance criteria presented in Subsection II of this SRP section regarding the maximum pressure in the reactor coolant and main steam systems, as well as minimum DNBR (PWR) or MCPR (BWR, if applicable). Time-related variations of the following parameters should be reviewed for consistency"*

Response to Question 15.04.04-15.04.05-1:

A plot of the minimum departure from nucleate boiling ratio (DNBR) and maximum linear power density (LPD) normalized to their respective specified acceptable fuel design limits (SAFDL) is shown on the enclosed markup as U.S. EPR FSAR, Tier 2, Figure 15.4-47—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature - Normalized Minimum DNBR and Maximum LPD to SAFDL. The line indicated as "LPD" is the calculated LPD normalized to either the fuel centerline melt or cladding strain limits (whichever is more limiting), including applicable uncertainties.

For anticipated operational occurrence (AOO) events, the limiting LPD of cladding strain is more restrictive and is therefore used to provide the normalized LPD. The LPD SAFDL is designed to protect both the fuel centerline melt (FCM) and cladding strain limits as discussed in ANP-10287P, Revision 0, "Incore Trip Setpoint and Transient Methodology for the U.S. EPR Topical Report." A deterministic evaluation of DNBR is performed with the thermal-hydraulic code LYNXT. Additionally, a deterministic evaluation of the maximum LPD shows that the cladding strain and peak fuel centerline temperature SAFDLs are protected.

U.S. EPR FSAR, Tier 2, Section 15.4.4.2 will be revised to clarify the approach used for this analysis. U.S. EPR FSAR, Tier 2, Section 15.4.4.3 will incorporate the new figure.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Section 15.4.4.2 and Section 15.4.4.3 will be revised as described in the response and indicated on the enclosed markup. U.S. EPR FSAR, Tier 2, Figure 15.4-47 will be added as described in the response and indicated on the enclosed markup.

Question 15.04.08-1:

It appears that the ejected rod accident has been performed assuming the center control rod is ejected. The ejection of an asymmetric control rod may result in a higher radial peaking factor. Justify that the analyzed cases are limiting with respect to local peaking factors.

Response to Question 15.04.08-1:

The center control rod is not the only rod considered for the rod ejection accident. The neutronics model is a full core nodal model. Each of the inserted control rods for the quarter core symmetry (in the full core model) is ejected to find the maximum rod worth locations in the NEMO-K analysis for hot full power (HFP) and hot zero power (HZP) at beginning of cycle (BOC) and end of cycle (EOC). The ejected rod transient is simulated in the neutronics computer code NEMO-K at conditions that are projected to bound the calculated values for ejected rod worth. The worth is increased by its uncertainty and other reactivity considerations as outlined in ANP-10286P, "U.S. EPR Rod Ejection Accident Methodology Topical Report."

The thermal-hydraulic code LYNXT is used to calculate the thermal effects of the ejected rod transient. Power peaking uncertainty and allowance factors are applied to the fuel rod powers supplied to LYNXT. Additional factors, such as hot channel and flow maldistribution factors, are applied in the LYNXT model input for off-nominal considerations.

The key parameters in this base analysis would be compared to a current cycle-specific set of values to verify that the ejected rod transients bound the cycle of interest as defined in ANP-10286P Section 9.0. Peaking values for F_Q (peak local power) and $F_{\Delta H}$ (peak rod power) are included as key parameters in the comparisons.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.04.08-2:

Please explain the differences in the ejected rod worths shown in Table 15.4-15 and those shown in Tables 15.-17 and -18.

> Regulatory basis: SRP 15.4.8 SPECTRUM OF ROD EJECTION ACCIDENTS (PWR) III *Review* Procedures 1. E. states "*The applicant's analytical methods are reviewed*".

Response to Question 15.04.08-2:

U.S. EPR FSAR, Tier 2, Tables 15.4-15—Rod Ejection Accident Overpressurization Analysis – Key Input Parameters, 15.4-17—Rod Ejection Accident DNBR Analysis – Ejected Rod Analysis Results for BOC, and 15.4-18—Rod Ejection Accident DNBR Analysis – Ejected Rod Analysis Results for EOC include the rod worths for the rod ejection accidents that are applicable to the different design criteria. U.S. EPR FSAR, Tier 2, Table 15.4-15 includes inputs for the plant system overpressurization criterion, while the other two tables include inputs and results regarding the departure from nucleate boiling ratio (DNBR) and enthalpy rise criteria.

The overpressurization analysis uses different codes and methods than the analysis used to calculate enthalpy rise and DNBR performance. U.S. EPR FSAR, Tier 2, Table 15.4-15 has the input parameters for the overpressurization analysis. Biasing is performed to maximize the pressure excursion using the S-RELAP5 point kinetics model for the power transient. Biased values are selected to be bounding for all times in the operating cycle. High rod worths will maximize the energy deposition and primary system pressure excursion. The point kinetics reactivity model in the S-RELAP5 power calculations is not used to determine the enthalpy rise or DNBR performance, so having higher worth rods is conservative for the overpressurization. The delayed neutron fraction is selected to be the largest beginning of cycle (BOC) value, and the fuel Doppler reactivity coefficient is selected to be the least negative value accounting for uncertainties.

U.S. EPR FSAR, Tier 2, Table 15.4-17 and Table 15.4-18 have the DNBR analysis values of the ejected rod worths for the BOC and end of cycle (EOC) conditions, respectively. For the enthalpy and DNBR evaluations, the rod worths are selected to bound the core cycle depletion maximum rod worths at the point of cycle burnup selected. An uncertainty of at least 15 percent is added in accordance with the methodology in ANP-10286P, "U.S. EPR Rod Ejection Accident Methodology Topical Report." In the rod ejection analyses, the delayed neutron fraction and fuel Doppler coefficient are adjusted to have lower magnitudes in order to enhance the period and peak value of the power pulse.

Table 15.04.08-2-1 summarizes the rod worths and kinetics parameters from the U.S. EPR FSAR, Tier 2, Table 15.4-15, U.S. EPR FSAR, Tier 2, Table 15.4-17, and U.S. EPR FSAR, Tier 2, Table 15.4-18.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Table 15.04.08-2-1— Reactivity Parameters for Rod Ejection Analyses

Parameter	U.S. EPR FSAR, Tier 2, Table 15.4-15	U.S. EPR FSAR, Tier 2, Table 15.4-17 (BOC[1])	U.S. EPR FSAR, Tier 2, Table 15.4-18 (EOC[2])
HFP[3] Rod Worth (pcm[4])	300	64	97
60% FOP[5] Worth (pcm)	500	286	389
HZP[6] Rod Worth (pcm)	700	433	634
Delayed Neutron Fraction	0.007358	0.0055	0.0047
Doppler Coefficient (pcm/°F)			
HZP	-1.17	-1.22	-1.52
HFP	-1.17	-0.96	-1.28

Notes:

1) BOC = beginning of cycle
2) EOC = end of cycle
3) HFP = hot full power
4) pcm = percent millirho
5) FOP = fraction of power
6) HZP = hot zero power

Question 15.04.08-3:

Verify that the 110 cal/gm criterion shown in Table 15.4-14—“Rod Ejection Accident DNBR Analysis – Ejected Rod Analysis Limits for U.S. EPR” is the PCMI limit of ANP-10286P, Revision 0, “U.S. EPR Rod Ejection Accident Methodology Topical Report,” AREVA NP Inc., November 2007.

> Regulatory *basis*: SRP 15.4.8 SPECTRUM OF ROD EJECTION ACCIDENTS (PWR) II Acceptance Criteria.

Response to Question 15.04.08-3:

ANP-10286P, “U.S. EPR Rod Ejection Accident Methodology Topical Report” Section 2.1.1 addresses the enthalpy (cal/g) limit for M5^{TM} clad fuel. The maximum corrosion for M5^{TM} cladding at end of life burnups is 0.035 mm. From the geometry of the U.S. EPR fuel rod (see U.S. EPR FSAR, Tier 2, Chapter 4), the corresponding oxide-to-wall thickness ratio is 0.061. Using the pellet/cladding mechanical interaction (PCMI) fuel cladding failure criteria of Standard Review Plan 4.2, the PCMI failure limit of is 110 cal/gm. For the U.S. EPR FSAR, Tier 2 analysis, a single cal/g limit for all fuel is used to keep the analysis independent of burnup.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.04.08-4:

Explain the physical basis of the loop 4 temperature excursion shown in Figure 15.4-35—HFP Rod Ejection Accident Overpressurization Analysis –Primary System Temperature, Figure 15.4-39—60% NP Rod Ejection Accident Overpressurization Analysis – Primary System Temperature, and Figure 15.4-43—HZP Rod Ejection Accident Overpressurization Analysis – Primary System Temperature. Consider the time delays in the nuclear power, heat transfer from the fuel to the coolant, and fluid mixing in the reactor pressure vessel.

> Regulatory basis: SRP 15.4.8 SPECTRUM OF ROD EJECTION ACCIDENTS (PWR) III Review Procedures 1. E. states "*The applicant's analytical methods are reviewed*".

Response to Question 15.04.08-4:

Per discussion with the NRC staff during the May 2008 audit, this question was clarified to address the apparent difference in the temperature behavior of the four loops. To more clearly distinguish the hot leg and cold leg temperatures in U.S. EPR FSAR, Tier 2, Figure 15.4-35, Figure 15.4 39, and Figure 15.4-43 are supported with six additional figures in U.S. EPR FSAR, Tier 2, Section 15.4. Three figures show the hot leg temperatures, while the other three show the cold leg temperatures.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Figure 15.4-48, Figure 15.4-49, Figure 15.4-50, Figure 15.4-51, Figure 15.4-52, and Figure 15.4-53 will be added as described in the response and indicated on the enclosed markup. U.S. EPR FSAR, Tier 2, Section 15.4.8.3.2 will be revised as indicated on the enclosed markup to incorporate the new figures.

Question 15.04.08-5:

Please identify the computer codes used in the analysis. Explicitly or by reference provide a description of each code. Please provide the nodalization used in these numerical models.

Response to Question 15.04.08-5:

The computer codes used are described in ANP-10263P-A, “Codes and Methods Applicability Report for the U.S. EPR.” Each code has supporting topical reports approved for use in pressurized water reactor (PWR) applications. The computer codes used for the rod ejection analysis were S-RELAP5, COPERNIC, LYNXT, and NEMO-K. The specific application of these codes to the rod ejection accident analysis is described in ANP-10286P, “U.S. EPR Rod Ejection Accident Methodology Topical Report” and in the following U.S. EPR FSAR, Tier 2, Sections:

- S-RELAP5 (U.S. EPR FSAR, Tier 2, Section 15.0.2.4)
  For cases where the reactor does not trip or for overpressurization considerations, the non-LOCA baseline model is used.
- COPERNIC (U.S. EPR FSAR, Tier 2, Section 4.2.3)
  BAW-10231PA, Rev.1, “COPERNIC Fuel Rod Design Computer Code,” Framatome ANP, Jan. 2004.
- NEMO-K (U.S. EPR FSAR, Tier 2, Section 4.3.3)
  BAW-10221PA, “NEMO-K A Kinetics Solution in NEMO,” Sept. 1998.
- LYNXT (U.S. EPR FSAR, Tier 2, Section 4.4.4.5)
  BAW-10156A, Revision 1, “LYNXT Core Transient Thermal Hydraulic Program,” Aug. 1993.

For each computer code, a nodalization appropriate to the analysis is used. The computer codes are not fully coupled; instead, the analysis parameters are passed to each code in the format required for each code. The axial and radial nodalization is briefly described below.

Axial Nodalization

Radial Nodalization

[ ] U.S. EPR FSAR, Tier 2, Figure 4.3-3—Typical Initial Core Loading Map provides the full core fuel assembly location map.

ANP-10286P provides additional information on the code interfaces and analysis process.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question15.04.08-6:

Identify the core location of the limiting rod in both the initial and equilibrium core.

Response to Question 15.04.08-6:

It is understood that the "limiting rod" is the "limiting control rod." As noted in ANP-10286P, "U.S. EPR Rod Ejection Accident Methodology Topical Report," the rod cluster control assembly (RCCA) in the N05 fuel location is the limiting rod ejection location for the equilibrium core design. All the cases in the reference NEMO-K analysis assume the rod is ejected from this location (See U.S. EPR FSAR, Tier 2 Figure 4.3-34—Rod Cluster Control Assembly Pattern for the RCCA location map). The analysis parameters are set to bound all cycles. The results presented in ANP-10286P are shown not to be sensitive to the ejected rod core location when the other key parameters remain the same.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.04.08-7:

Provide the physical basis to show that the least negative Doppler temperature coefficient has been used in the calculations.

Response to Question 15.04.08-7:

The Doppler temperature coefficients (DTC) listed in U.S. EPR FSAR, Tier 2, Table 15.4-17—Rod Ejection Accident DNBR Analysis – Ejected Rod Analysis Results for BOC and Table 15.4-18—Rod Ejection Accident DNBR Analysis – Ejected Rod Analysis Results for EOC are calculated from the NEMO-K models used for each of the ejected rod transient simulations. The NEMO-K model used for the safety analysis simulations used a conservative fuel temperature response of the cross sections to bound the uncertainties on the DTC and to bound cycle-to-cycle variations.

To verify that a particular cycle design is bounded by the ejected rod analysis, the unmodified cycle specific values for the DTC are reduced by the uncertainty and compared to the analyzed values at beginning of cycle (BOC), end of cycle (EOC), hot full power (HFP), and hot zero power (HZP). This means the DTC for the designed cycle will be larger in magnitude by the uncertainty than the value analyzed. An example comparison is shown in ANP-10286P, "U.S. EPR Rod Ejection Accident Methodology Topical Report."

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.04.08-8:

The FSAR states that “The plant simulation computer code S-RELAP5 (Reference 1) is used to determine the peak pressure response of the primary system to the RCCA ejection event.” Please identify the model and nodalization used.

Response to Question 15.04.08-8:

These calculations use the non-LOCA S-RELAP5 model and nodalization described in ANP-10263P-A, “Codes and Methods Applicability Report for the U.S. EPR.”

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.04.08-9:

Please describe the xenon condition at the start of the REA transient analysis. Justify that the assumed xenon condition conservative.

Response to Question 15.04.08-9:

ANP-10286P, "U.S. EPR Rod Ejection Accident Methodology Topical Report" Section 4.1.1 states: "The initial conditions are required to be a reasonable representation of the limiting conditions allowed by Technical Specifications that maximize the worth." The hot full power (HFP) xenon concentration is skewed so that the power offset represents the maximum offset achievable at HFP. The skew of the power shape to the top of the core increases the worth of any partially inserted rods (rods positioned at their respective power dependent insertion limit) (see U.S. EPR FSAR, Tier 2, Figure 4.3-2—U.S. EPR Rod Group Insertion Limits Versus Thermal Power). The power shape is more outlet-peaked, which results in lower minimum departure from nucleate boiling ratio (MDNBR) predictions.

The ejected rod worth in the model and the peak local power (F_Q) are increased by the respective uncertainties and by a design allowance. All the cases are run from this condition.

The ejected rod worths and post-ejected F_Q at the nominal HFP xenon are compared to the values at skewed xenon in Table15.04.08-9-1 and Table 15.04.08-9-2, respectively, for both Cycle 1 and the equilibrium cycle models without any adjustments for uncertainties or design allowances. The skewed xenon produces higher ejected rod worth and F_Q values than nominal HFP xenon.

Table 15.04.08-9-1—Ejected Rod Worths for Various Xenon Conditions

	Ejected Rod Worth, pcm[1]	
	Nominal HFP[2] Xenon	**Skewed Xenon**
Cycle 1, BOC[3] HZP[4]	[ ]	[ ]
BOC HFP	[ ]	[ ]
EOC[5] HZP	[ ]	[ ]
EOC HFP	[ ]	[ ]
Equilibrium Cycle, BOC HZP	[ ]	[ ]
BOC HFP	[ ]	[ ]
EOC HZP	[ ]	[ ]
EOC HFP	[ ]	[ ]

Notes:

1) pcm = percent millirho
2) HFP = hot full power
3) BOC = beginning of cycle
4) HZP = hot zero power
5) EOC = end of cycle

Table 15.04.08-9-2— Post Ejected F_Q for Various Xenon Conditions

	Post-Ejected F_Q[1]	
	Nominal HFP[2] Xenon	**Skewed Xenon**
Cycle 1, BOC[3] HZP[4]	[ ]	[ ]
BOC HFP	[ ]	[ ]
EOC[5] HZP	[ ]	[ ]
EOC HFP	[ ]	[ ]
Equilibrium Cycle, BOC HZP	[ ]	[ ]
BOC HFP	[ ]	[ ]
EOC HZP	[ ]	[ ]
EOC HFP	[ ]	[ ]

Notes:

1) F_Q = peak local power
2) HFP = hot full power
3) BOC = beginning of cycle
4) HZP = hot zero power
5) EOC = end of cycle

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.05.01-15.05.02-1:

Table 6.8-1—Extra Borating System Design and Operating Parameters provides a nominal flow rate of 52 gpm / pump and a maximum allowable flow rate of 110.8 gpm (for the system). This corresponds to the flow rate of 15.5 lbm/sec shown in Figure 15.5-9—Inadvertent Operation of the EBS. Table 9.3.4-1—Major CVCS Component Design Data Sheet 1 of 5 shows 2 Centrifugal Charging Pumps with a normal flow rate of 176 gpm, and a maximum flow rate of 285 gpm. This does not correspond to the flow rate of 13 to 16 lbs/sec shown in Figure 15.5-18—CVCS Malfunction that Increases RCS Inventory. Please explain.

Response to Question 15.05.01-15.05.02-1:

The evaluation of the extra borating system (EBS) malfunction uses the maximum flow rate because the pumps could potentially deliver this flow rate to the reactor coolant system (RCS). This represents a bounding assumption for the inadvertent operation of the EBS event, which increases reactor coolant system (RCS) inventory.

The U.S. EPR FSAR, Tier 2, Table 9.3.4-1—Major CVCS Component Design Data provides the normal and maximum flow rates for each chemical and volume control system (CVCS) charging pump. Because the CVCS charging pumps serve multiple functions, some of the flow is diverted, and the actual flow delivered to the RCS depends on the configuration of the CVCS system. Several factors affect the delivered flow, such as the reactor coolant pump (RCP) seal flow, losses through the pipes and components, number of pumps operating, the pump flow curve, water temperature, and the position setting of the charging line control valve 30KBA34 AA101 (see U.S. EPR FSAR, Tier 2, Figure 9.3.4-1—Chemical and Volume Control System).

The setting of the control valve 30KBA34 AA101 depends on the plant operating conditions and is normally set to maintain a constant flow. During normal operation, the control valve is set such that the total flow rate to RCS loops 2 and 4, with one charging pump operating, is about 140 gpm. Flow is also provided to the RCP seals and the water jet pump. The one- and two-pump charging flow rates used in the CVCS malfunction analysis are based on this valve position and other factors mentioned above. Conservative water temperatures (59 ˚F) are used to maximize the mass flow rate into the RCS.

Because the RCS pressure varies during the course of the event, the charging flow is a function of RCS pressure. The charging flow rates that occur during the event (for two pump operation) are shown in the U.S. EPR FSAR, Tier 2, Figure 15.5-18—CVCS Malfunction that Increases RCS Inventory – Charging Flow.

Because the event is postulated to occur from normal operating conditions, the use of the normal operating position of the control valve 30KBA34 AA101 and the corresponding flow rates used in the analysis of the event are appropriate.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.05.01-15.05.02-2:

Explain the charging flow shown in Figure 15.5-18—CVCS Malfunction that Increases RCS Inventory –Charging Flow. Explain why the flow rate to loop 2 and 4 are different prior to RT. It is recognized that the analysis assumes that 2 pumps are de-energized at RT; one pump is immediately started and tripped again at about 1000 sec. Explain the rapid changes in flow shown in the figure between reactor trip and isolation of the changing pumps at 1019 seconds.

Response to Question 15.05.01-15.05.02-2:

The actual delivered flow to the two loops is calculated as explained in the response to Question 15.05.01-15.05.02-1. The flow rates are not the same due to the difference in losses through the two loops based on actual line-loss calculations.

The flow rate depends on the back-pressure from the reactor coolant system (RCS) to the charging flow. Before reactor trip (RT), the pressure is constant (due to pressure control) and the charging flow rates are constant. After RT, the pressure decreases due to RCS shrinkage. Then, the pressure increases due to continued increase in pressurizer (PZR) level until terminated by the opening of pressurizer safety relief valves (PSRV). The PSRVs subsequently cycle to control RCS pressure. The charging flow increases as the RCS pressure decreases, and the flow decreases as the RCS pressure increases. The relationship can be seen by comparing U.S. EPR FSAR, Tier 2, Figures 15.5-14—CVCS Malfunction that Increases RCS Inventory – PZR Pressure and 15.5-18—CVCS Malfunction that Increases RCS Inventory – Charging Flow.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

U.S. EPR Final Safety Analysis Report Markups

consequences. The uncontrolled RCCA bank withdrawals from HZP are simulated by withdrawing either the control banks or the shutdown banks with maximum RCCA velocity.

15.4.1.3 Results

15.04.01-2

Table 15.4-3—Uncontrolled Control Bank Withdrawal from a Subcritical or Low-Power Startup Condition – Sequence of Events presents the sequence of events for the limiting case. Figure 15.4-1—Uncontrolled Control Bank Withdrawal from a Subcritical or Low Power Startup Condition - Reactor Power through Figure 15.4-5—Uncontrolled Control Bank Withdrawal from a Subcritical or Low Power Startup Condition - Cold Leg Mass Flow, Figure 15.4-44—Uncontrolled Control Bank Withdrawal from a Subcritical or Low Power Startup Condition - Primary Hot Leg Temperature, and Figure 15.4-45—Uncontrolled Control Bank Withdrawal from a Subcritical or Low Power Startup Condition - Primary Cold Leg Temperature present the response of the most important system parameters.

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The minimum DNBR remains above the design limit value (refer to Section 4.4). The peak fuel centerline temperature remains below the fuel melting point. The fuel temperatures do not increase high enough to cause enough fuel expansion to exceed one percent uniform clad strain. Figure 15.4-46—Uncontrolled Control Bank Withdrawal from a Subcritical or Low Power Startup Condition - Normalized Minimum DNBR and FCM to SAFDL presents the DNB and fuel centerline melt (FCM) normalized to their respective SAFDLs.

15.4.1.4 Radiological Conclusions

Radiological consequences are not calculated for this event because no fuel or cladding damage occurs and no radioactive materials are released to the environment.

15.4.1.5 Conclusions

The analyses presented evaluate an uncontrolled control rod assembly withdrawal from a subcritical or low-power startup condition. During this event, the plant instrumentation, protection functions, and equipment provide an RT sufficiently early to preclude fuel or cladding damage. The core remains adequately cooled throughout this event.

15.4.1.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.4.1 events included in NUREG-0800, Section 15.4.1, (Reference 3) and descriptions of how these criteria are met are listed below:

1. The requirements of GDC 10, 20, and 25 concerning the SAFDL are assumed to be met for this event when:

15.4.4.2 Method of Analysis and Assumptions

The S-RELAP5 computer code calculates the thermal hydraulic response of the primary and secondary systems using the methodology of Reference 1. Section 15.0.2 provides a description of the S-RELAP5 analysis methodology.

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The core thermal-hydraulic computer code LYNXT, as described in Section 4.4.4.5.2, uses the RCS response from S-RELAPS ~~is used~~ to calculate the core flow, enthalpy distributions, and DNBR. A deterministic evaluation is made of the maximum linear power density (LPD). ~~and peak fuel centerline temperatures. It uses the RCS response from S-RELAPS as a boundary condition as described in Reference 2.~~ Administrative controls limit the power level at which the fourth RCP can be started. The event is analyzed from a 60 percent power EOC initial condition. The event is initiated by the startup of the idle fourth RCP. If the moderator temperature coefficient of reactivity is negative, an insertion of positive reactivity and an increase in reactor power occurs. It is assumed conservatively that rod control is in manual mode. The analysis of this event uses a conservatively large negative moderator temperature coefficient associated with the EOC.

Table 15.4-10—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Key Input Parameters presents key input parameters and Table 15.4-11—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Equipment Status presents the status of key equipment available to mitigate this event.

There is no single failure that makes this event more severe. The only plant system that affects the response in this event is the non-safety-related pressurizer sprays, which are assumed active. This condition reduces the increase in RCS pressure and, therefore, reduces DNBR margin.

15.4.4.3 Results

Table 15.4-12—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Sequence of Events presents the sequence of events for this scenario. Figure 15.4-20—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature - Total RCS Loop Flow through Figure 15.4-27—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – RCS Bottom Pressure present the plant response to this event.

RCS flow increases rapidly to full flow following the start of the fourth RCP (see Figure 15.4-20—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature - Total RCS Loop Flow). This condition causes a decrease in the core temperatures (see Figure 15.4-23—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Core Temperatures). Reactor power increases to a maximum value of 75.7 percent during this time period due to the positive reactivity insertion

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associated with the decrease in the core average temperature (see Figure 15.4-24—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Indicated Reactor Power). No RT setpoints are reached. The combination of Doppler feedback (see Figure 15.4-25—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Reactivity) and increasing RCS cold leg temperatures (see Figure 15.4-26—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Cold Leg Temperatures) stop the power excursion once the pump reaches full speed. The DNB LCO is set sufficiently high that the startup of an inactive reactor coolant loop does not challenge the DNB SAFDL limits. Figure 15.4-47—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature - Normalized Minimum DNBR and Maximum LPD to SAFDL presents the DNB and LPD normalized to their respective SAFDLs.

15.4.4.4 Radiological Consequences

No radiological consequences are calculated for this event because no fuel or cladding damage is predicted and there are no radiological releases to the environment.

15.4.4.5 Conclusions

The analysis of the startup of an inactive reactor coolant loop event demonstrates that minimum DNBR remains above the limit. The RCS pressure transient does not approach the peak RCS pressure limit.

15.4.4.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.4.4 events included in NUREG-0800, Section 15.4.4–15.4.5, (Reference 3) and descriptions of how these criteria are met are listed below:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

   – Response: The analysis described in Sections 15.4.4.2 and 15.4.4.3 indicates that there is acceptable design margin with respect to RCS pressure.

2. Fuel-cladding integrity is maintained by keeping the minimum DNBR above the 95 percent probability/95 percent confidence DNBR limit.

   – Response: The results in Section 15.4.4.3 demonstrate that this requirement is met. The DNB LCO is set sufficiently high so that the startup of an inactive RCP at an incorrect temperature event does not challenge the DNB SAFDL limits.

3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

fuel temperatures (at BOC HFP). Figure 15.4-30—Rod Ejection Accident DNBR Analysis – EOC HZP Transient Power Fraction and Figure 15.4-31—Rod Ejection Accident DNBR Analysis – EOC HZP Transient Peak Fuel and Cladding Temperatures show the same parameters for the limiting case in terms of enthalpy rise (at EOC HZP).

15.4.8.3.2 Overpressurization

The overpressure analysis is performed in a separate set of analyses using the point kinetics reactivity simulations in S-RELAP5 (Reference 1). The rod ejection event is modeled conservatively as ejecting the highest worth RCCA within 0.1 s. The result is a rapid reactivity insertion with large local power peaking. The power peak is limited by the fuel temperature reactivity feedback due to increased fuel temperatures while the transient is eventually terminated by the PS. In the cases analyzed, the primary system pressure does not exceed 120 percent of the design pressure (3056 psia, or the Service Limit C as defined in the ASME Code (Reference 6).

The following cases are analyzed using conservatively bounding ejected RCCA worths at each power level:

- Rod ejection at HFP, ejected worth: 300 pcm.
- Rod ejection at 60 percent NP, ejected worth: 500 pcm.
- Rod ejection at HZP, ejected worth: 700 pcm.

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Table 15.4-19—Rod Ejection Accident Overpressurization Analysis – Sequence of Events presents the sequence of events for the HFP, 60 percent NP and HZP cases. Figure 15.4-32—HFP Rod Ejection Accident Overpressurization Analysis – Percent Reactor Power through Figure 15.4-43—HZP Rod Ejection Accident Overpressurization Analysis – Primary System Temperature and Figure 15.4-48—HFP Rod Ejection Accident Overpressurization Analysis - Primary Hot Leg Temperature through Figure 15.4-53—HZP Rod Ejection Accident Overpressurization Analysis - Primary Cold Leg Temperature present the transient response.

15.4.8.4 Radiological Consequences

The radiological consequences of the rod ejection accident are evaluated in Section 15.0.3.9.

15.4.8.5 Conclusions

For the spectrum of rod ejection accidents evaluated, none of the power excursions caused the fuel temperatures to reach either the limiting fuel melt temperature or the fuel enthalpy limits. For the events which exceeded the DNBR limit, the number of fuel failures was less than the value allowed for the radiological release limit. The

Table 15.4-4—Uncontrolled Control Bank Withdrawal at Power – Key Input Parameters

Parameter	Analysis Value
Initial reactor power	4612, 2767, 1153 MW cases
Average RCS temperature	594°F at 100% 594°F at 60% 587°F at 25%
Initial PZR pressure	2250 psia
Initial RCS loop flow rate	119,692 gpm per loop
Maximum possible RCCA bank differential worth	5.59 pcm/sec at BOC 7.09 pcm/sec at EOC
Moderator temperature coefficient	0.0 pcm/°F to +5.73 pcm/°F at BOC -42.0 pcm/°F to -50 pcm/°F at EOC
Doppler temperature coefficient	-1.17 pcm/°F at BOC -1.85 pcm/°F at EOC
Bounding value for fraction of delayed neutrons (β)	0.007358 at BOC 0.005151 at EOC
^{238}U~~-238~~ capture to fission ratio	0.85
Time when LOOP is assumed	At TT
Initial PZR Level	54.3% of span at 100% 54.3% of span at 60% 44.3% of span at 25%
Scram Reactivity	6161 pcm at 100% 5964 pcm at 60% 5698 pcm at 25%

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Table 15.4-5—Uncontrolled Control Bank Withdrawal at Power – Equipment Status

Plant Equipment or System	Status
RCCA position control mode	Manual
Turbine control valves position control mode	Automatic
PZR spray	Available
RCPs	Operating until LOOP

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Figure 15.4-44—Uncontrolled Control Bank Withdrawal from a Subcritical or Low Power Startup Condition - Primary Hot Leg Temperature

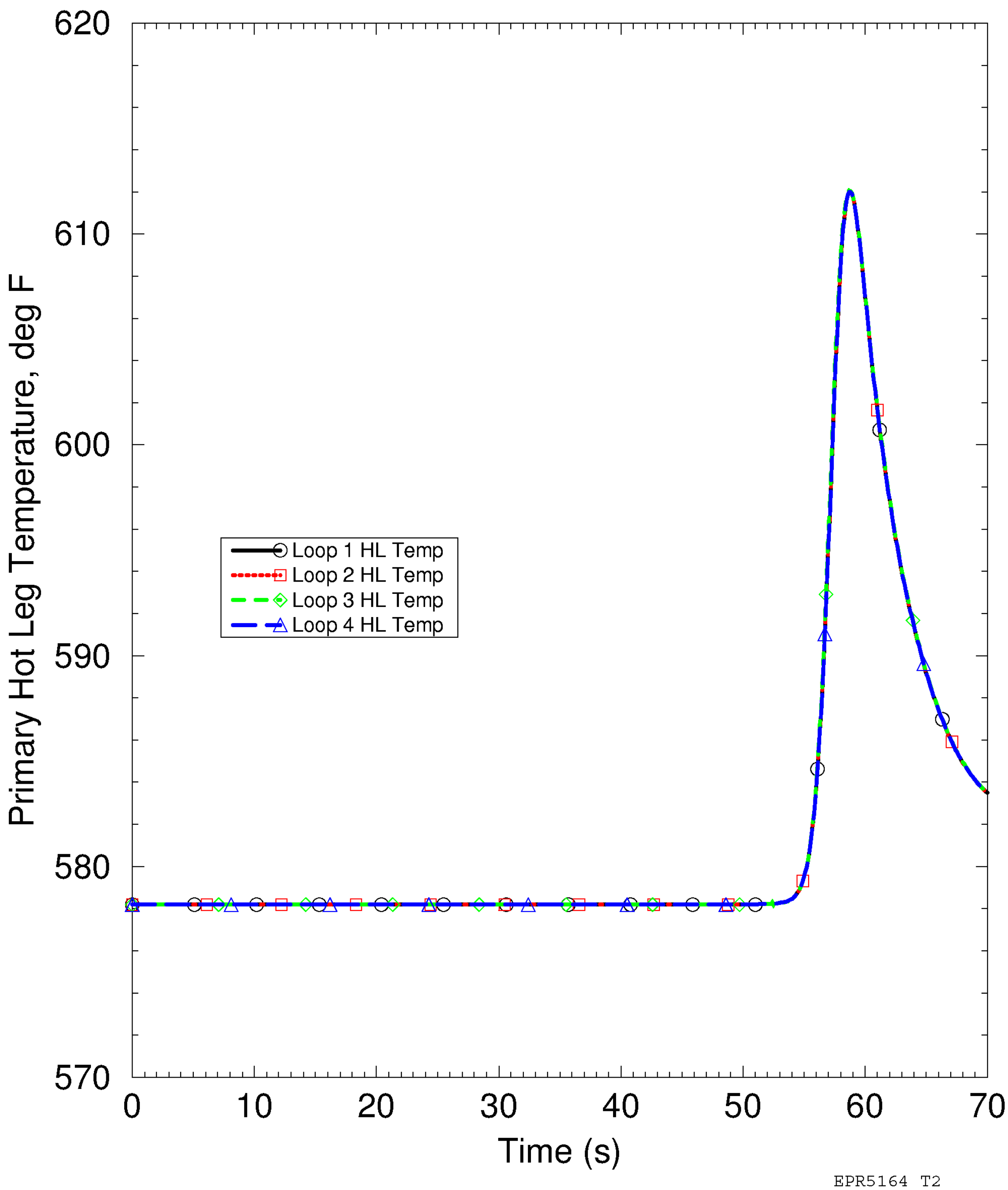

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Figure 15.4-45—Uncontrolled Control Bank Withdrawal from a Subcritical or Low Power Startup Condition - Primary Cold Leg Temperature

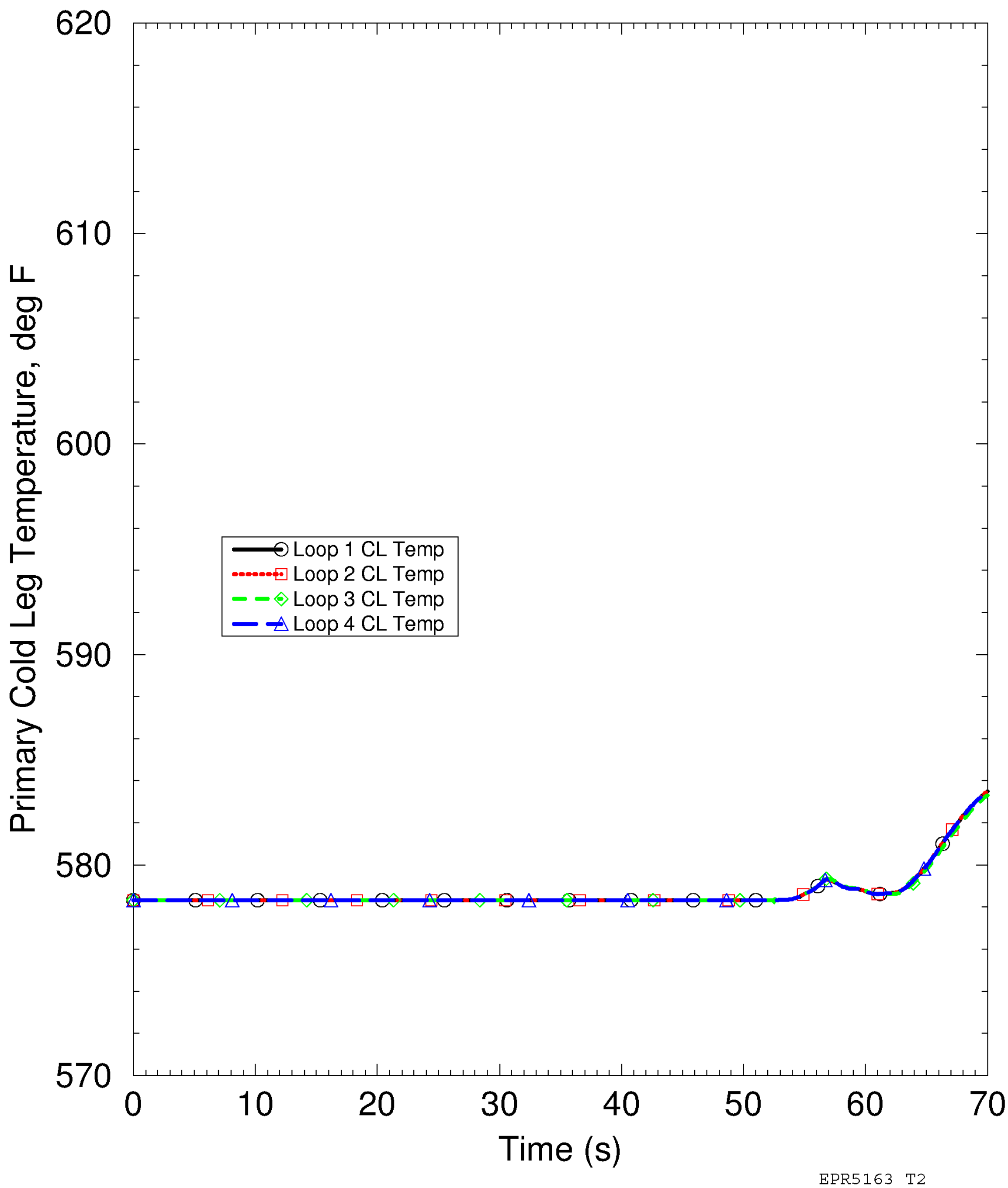

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Figure 15.4-46—Uncontrolled Control Bank Withdrawal from a Subcritical or Low Power Startup Condition - Normalized Minimum DNBR and FCM to SAFDL

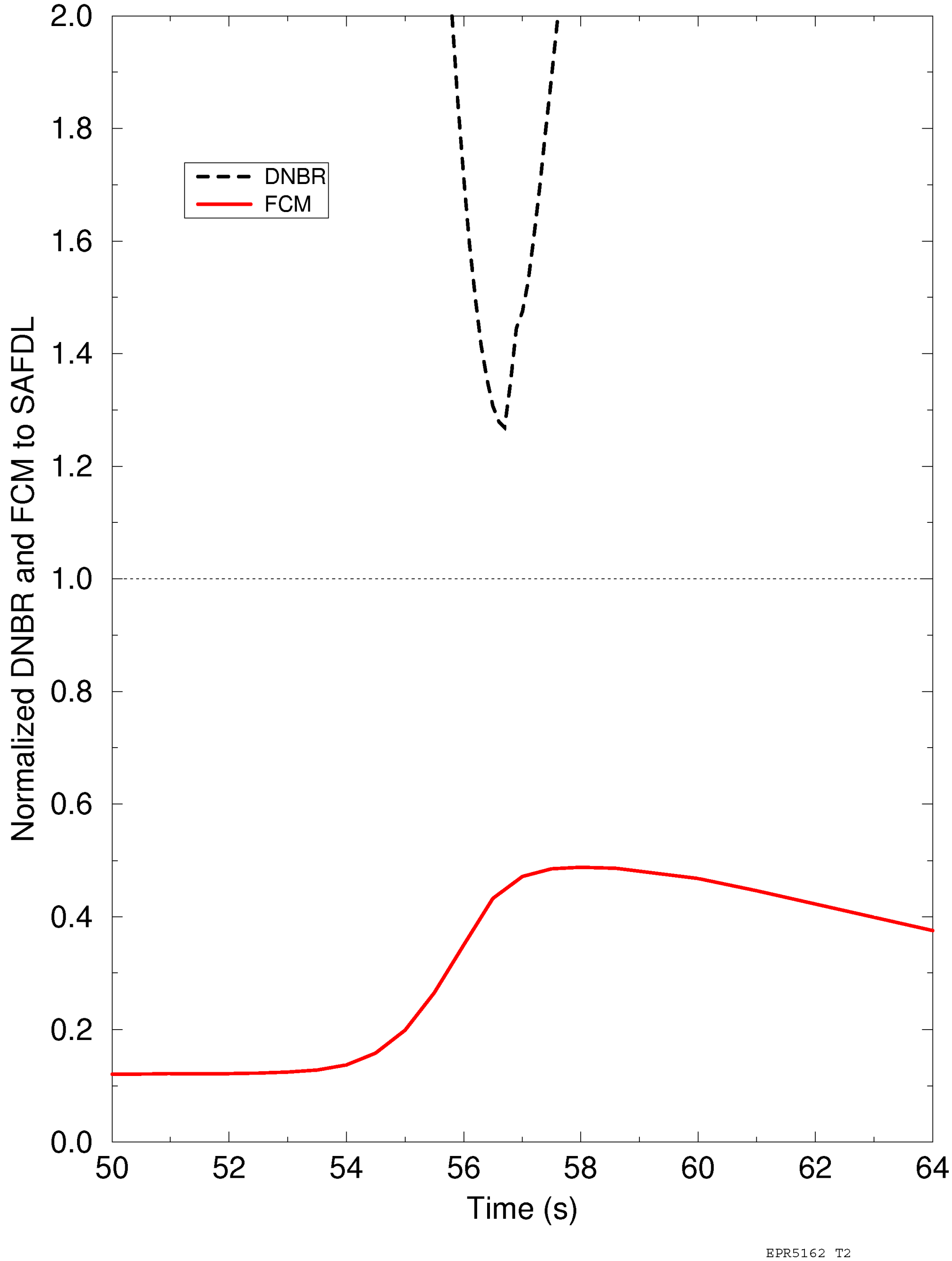

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Figure 15.4-47—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature - Normalized Minimum DNBR and Maximum LPD to SAFDL

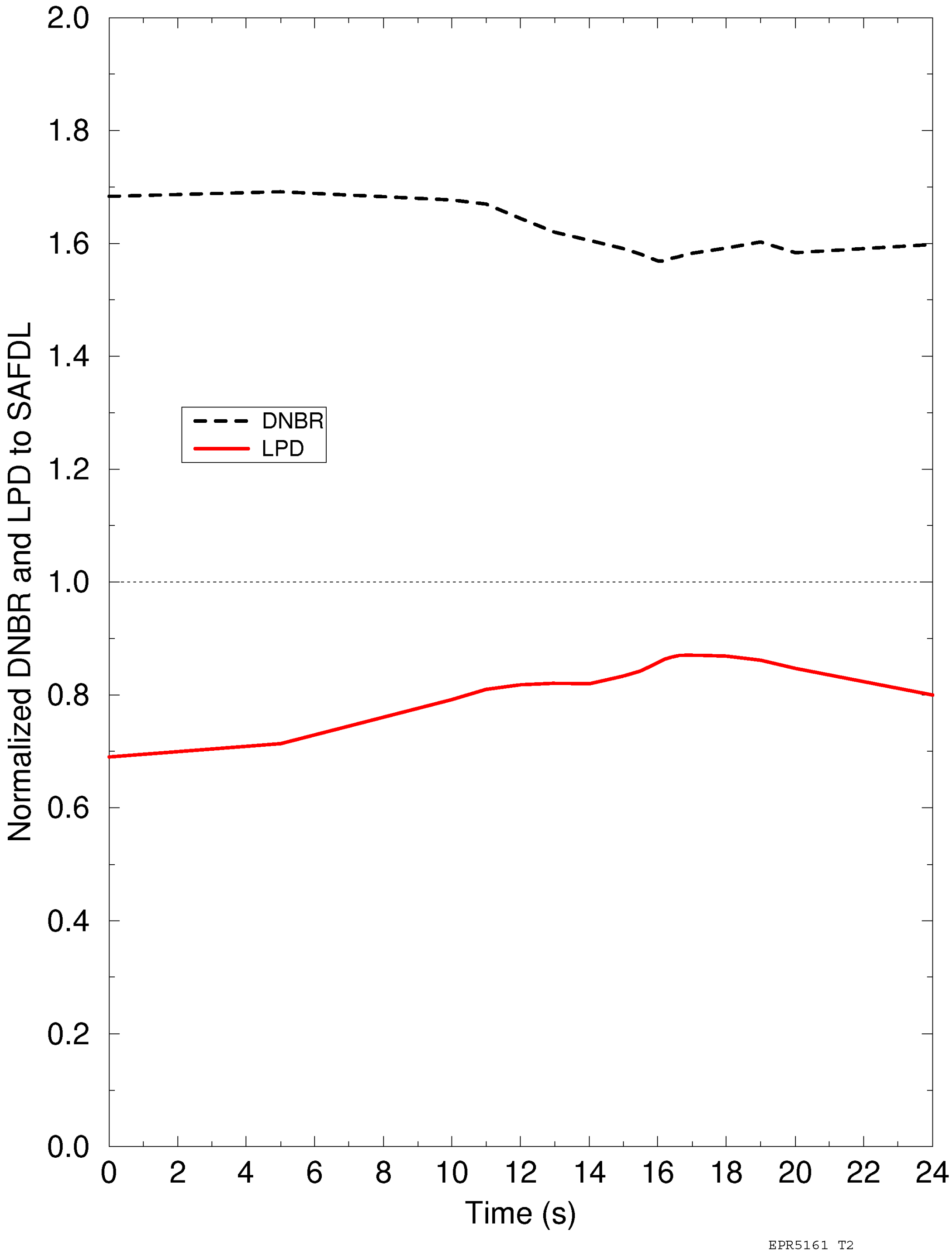

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Figure 15.4-48—HFP Rod Ejection Accident Overpressurization Analysis - Primary Hot Leg Temperature

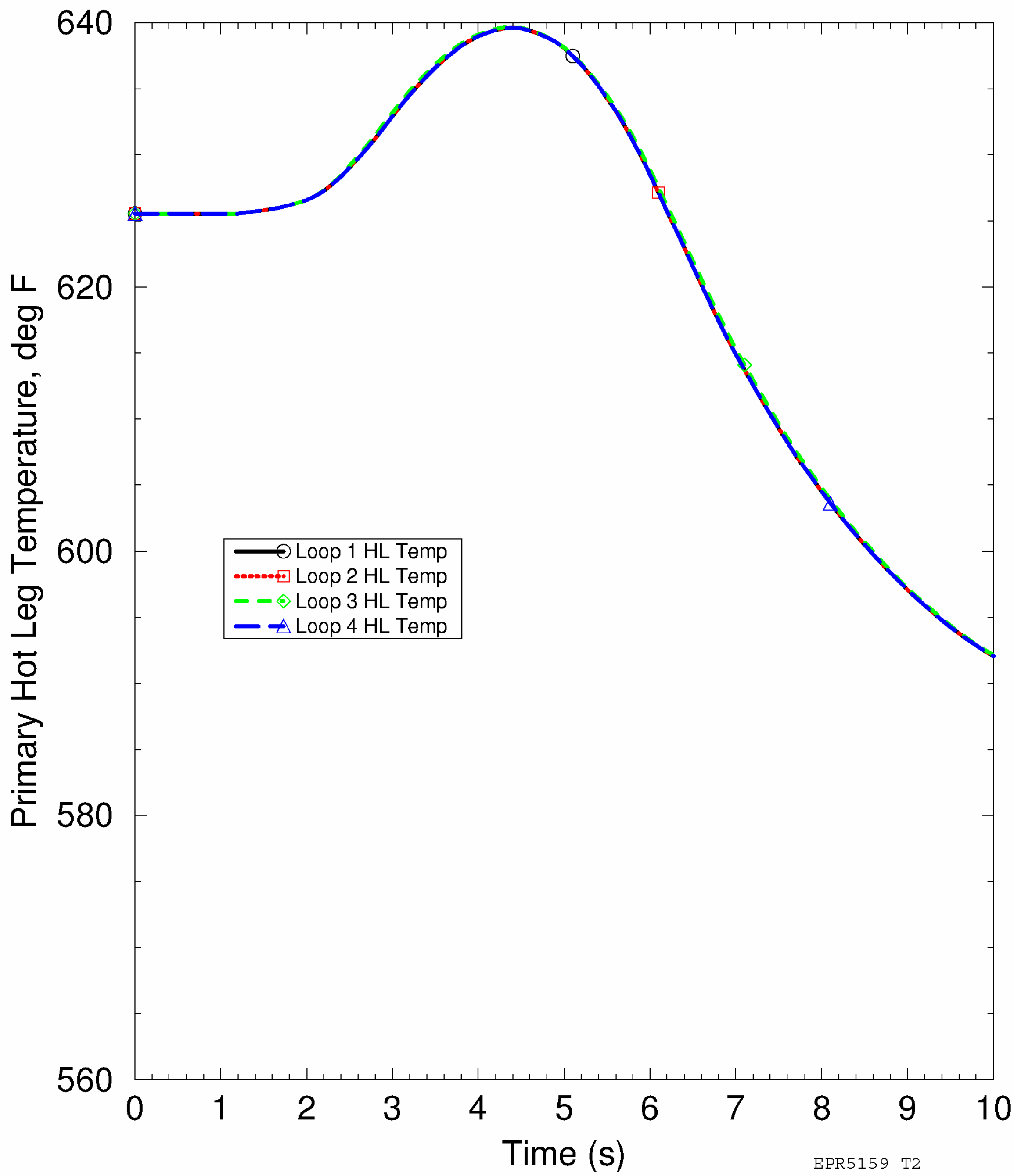

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Figure 15.4-49—HFP Rod Ejection Accident Overpressurization Analysis - Primary Cold Leg Temperature

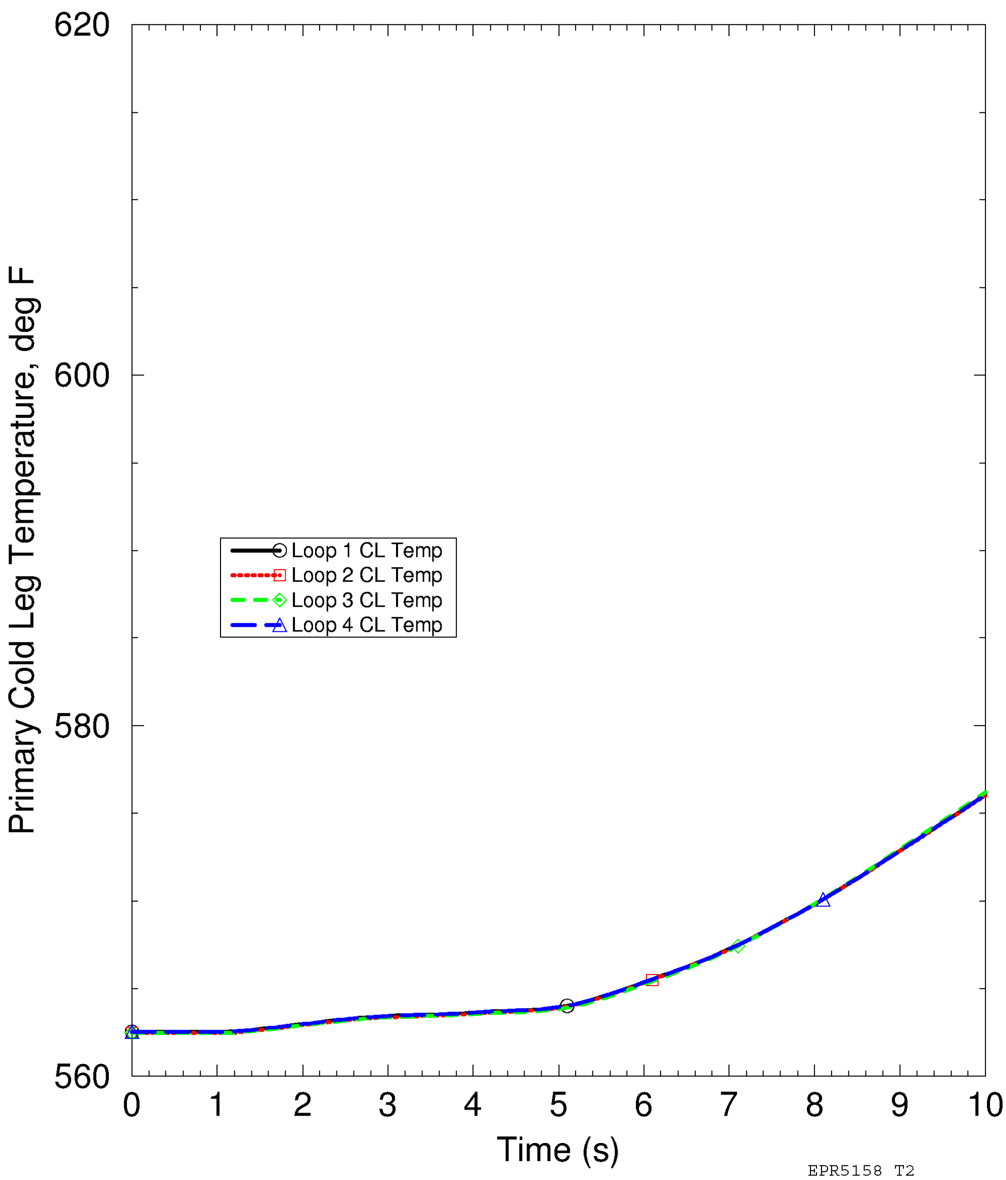

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Figure 15.4-50—60% NP Rod Ejection Accident Overpressurization Analysis - Primary Hot Leg Temperature

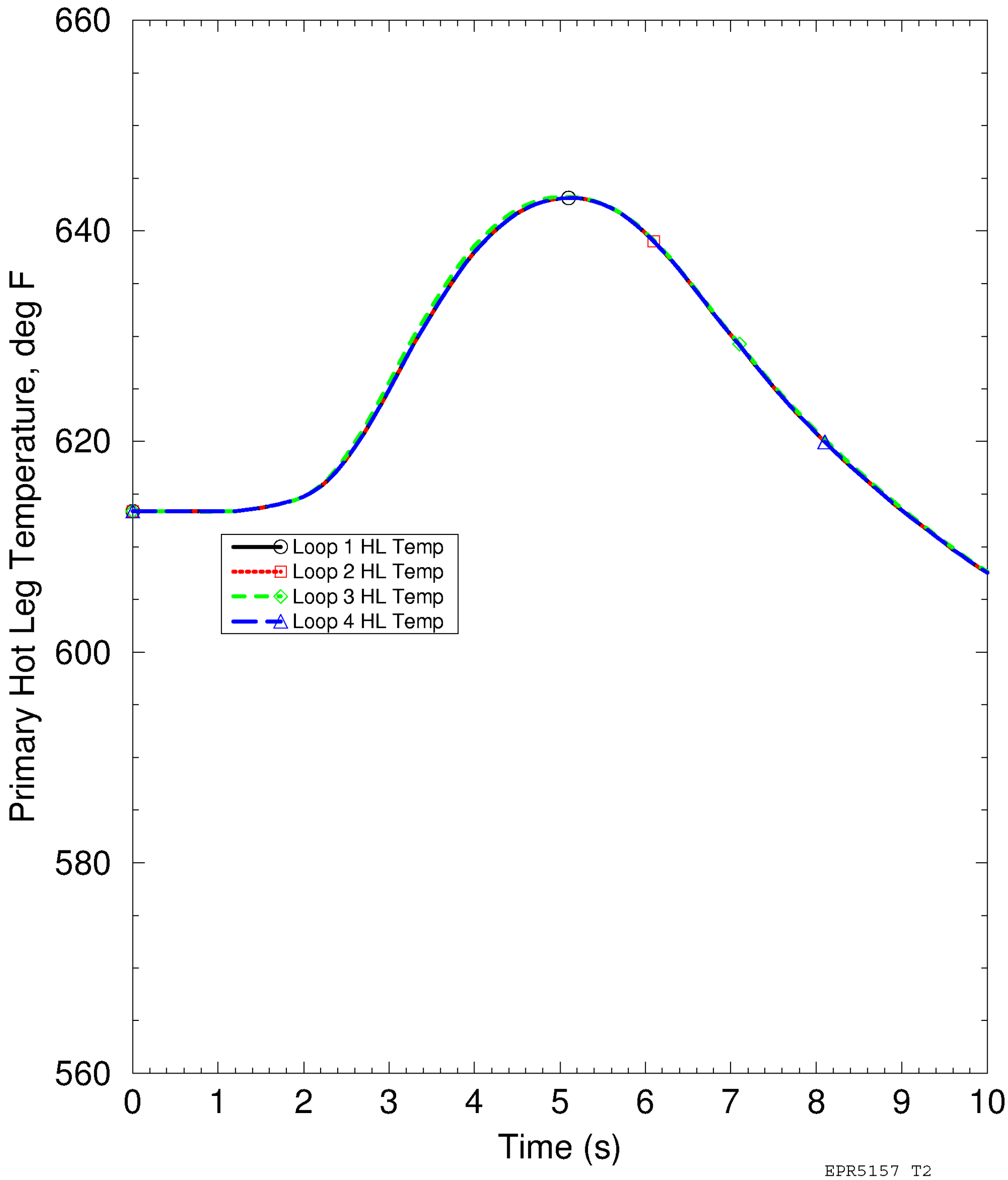

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Figure 15.4-51—60% NP Rod Ejection Accident Overpressurization Analysis - Primary Cold Leg Temperature

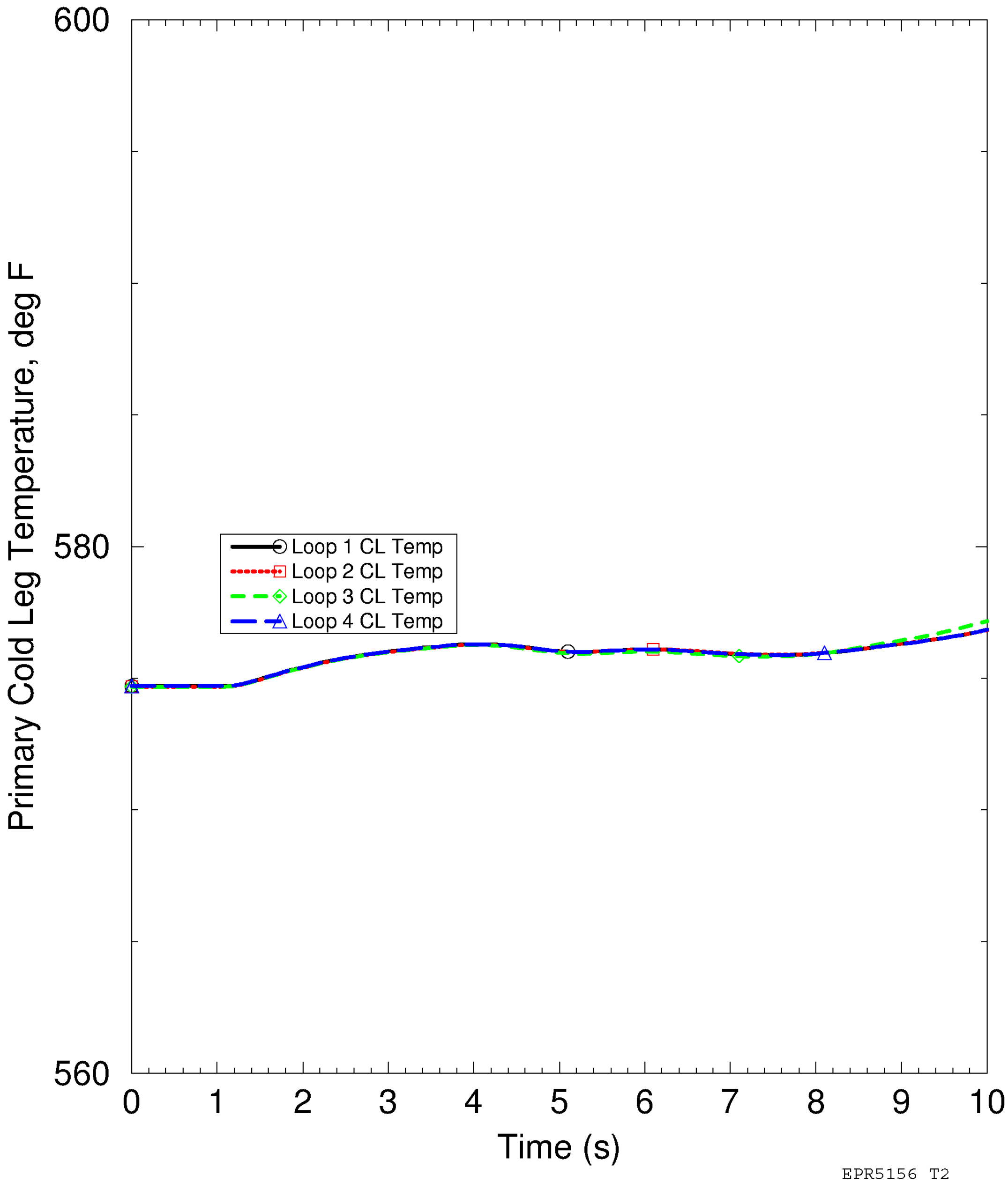

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Figure 15.4-52—HZP Rod Ejection Accident Overpressurization Analysis - Primary Hot Leg Temperature

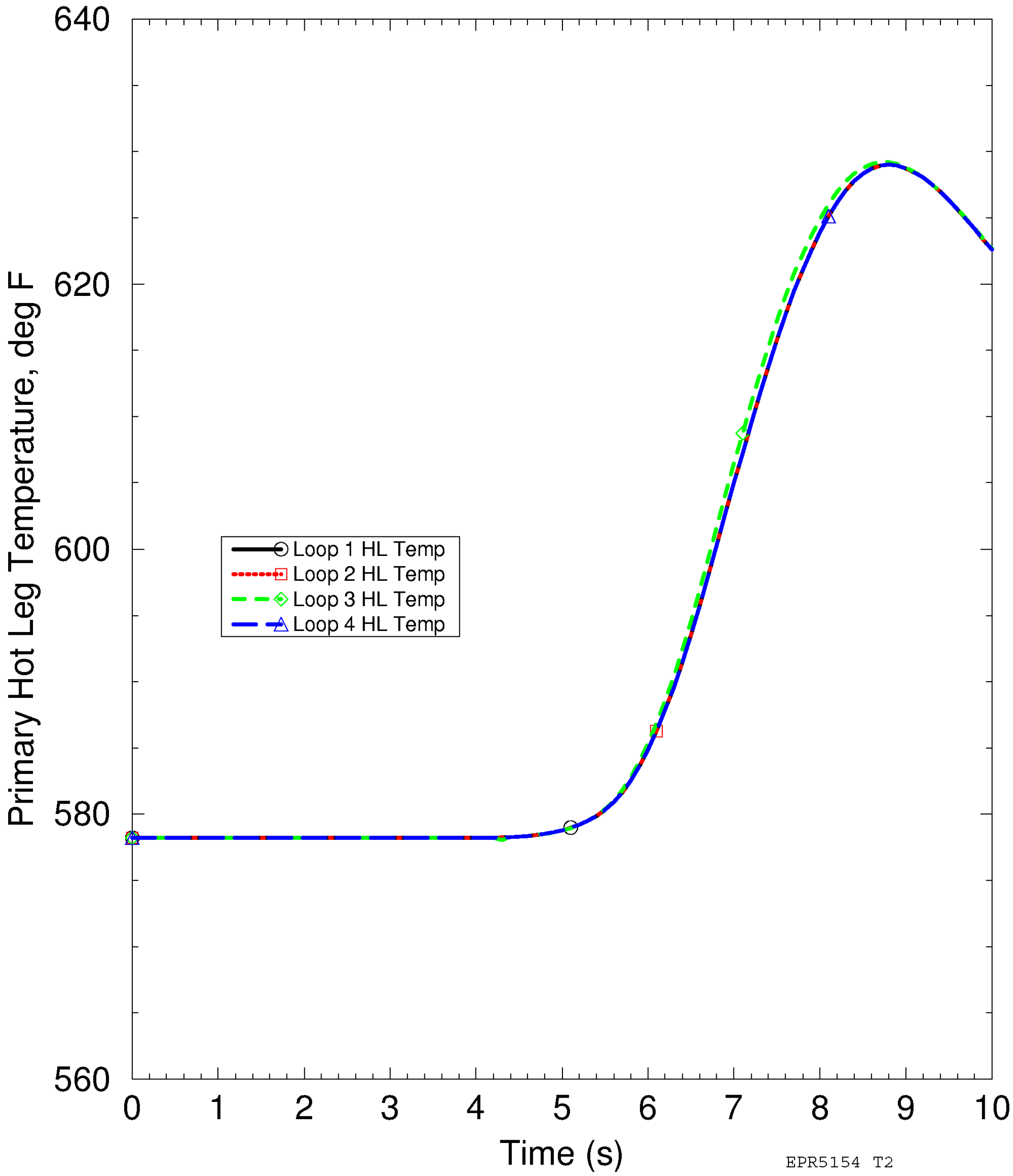

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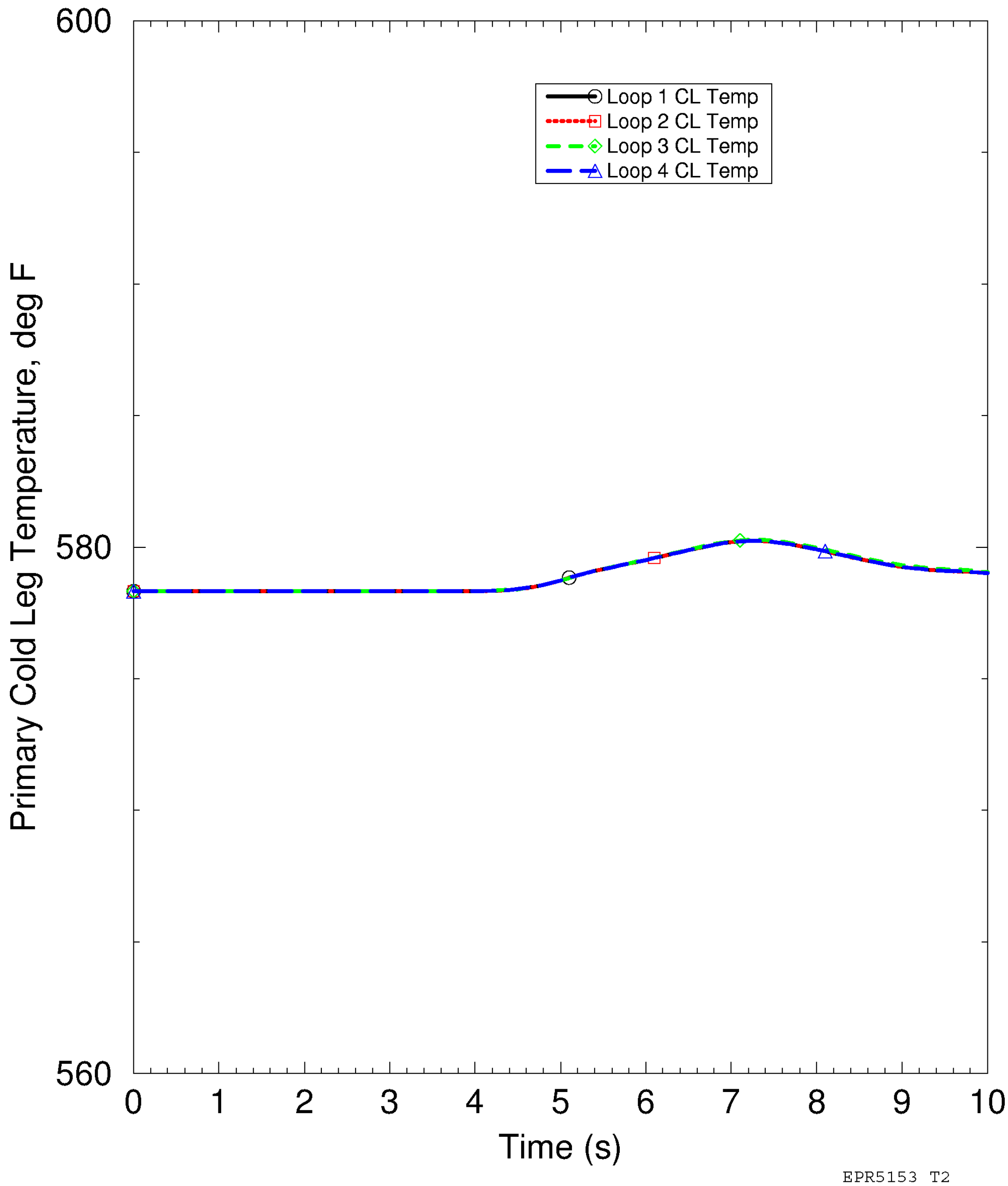
Figure 15.4-53—HZP Rod Ejection Accident Overpressurization Analysis - Primary Cold Leg Temperature



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