**Response to**

**Request for Additional Information No. 29, Supplement 2**

**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 15.04.03-1:**

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:**

During the development of a response to this question, it was discovered that the bank drop transient presented in the U.S. EPR FSAR, Tier 2, Section 15.4.3 – Control Rod Misoperation (System Malfunction or Operator Error) does not represent a limiting case regarding the minimum departure from nucleate boiling ratio (DNBR) and the fuel centerline temperature specified acceptable fuel design limits (SAFDL). This response presents a representative limiting bank drop transient. FSAR Section 15.4.3, including Table 15.4-9—Dropped RCCA – Sequence of Events, Figure 15.4-15—Dropped RCCA – Reactor Power, Figure 15.4-16—Dropped RCCA – Reactivity, Figure 15.4-17—Dropped RCCA – Average Core Heat Flux, Figure 15.4-18—Dropped RCCA – Primary System Pressure, and Figure 15.4-19—Dropped RCCA – Primary System Temperature will be revised to correspond to this representative limiting case. In addition, new Figure 15.4-55—Dropped RCCA – Minimum DNBR and Maximum LPD Normalized to SAFDL provides a plot of DNBR and linear power density (LPD) as a function of time.

The scale of the U.S. EPR FSAR, Tier 2, Figure 15.4-15, Figure 15.4-16, Figure 15.4-17, Figure 15.4-18 and Figure 15.4-19 is increased to 200 seconds in order to show reactor stabilization.

The DNBR normalized to the DNBR SAFDL as a function of time is provided in new U.S. EPR FSAR, Tier 2, Figure 15.4-55. The low DNBR reactor trip provides adequate protection of the DNBR SAFDL for the bank drop event.

The LPD SAFDL is a criterion for preventing fuel centerline melting and excessive cladding strain (see U.S. EPR FSAR, Tier 2, Section 15.0.0.3.9). The maximum LPD normalized to the LPD SAFDL as a function of time is provided in new U.S. EPR FSAR, Tier 2, Figure 15.4-55. Because the LPD SAFDL is not exceeded, fuel centerline melting and excessive cladding strain do not occur.

**FSAR Impact:**

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

# U.S. EPR Final Safety Analysis Report Markups

#### 15.4.3.1.3 Results

15.04.03-1

~~The analyses demonstrate that the most limiting RCCA drop event is the one in which the ACT control function is active with automatic turbine valve control.~~As indicated in Table 15.0-62—Transient Analysis Limiting Cases and Table 15.0-63—Transient Analysis Limiting Case Conditions, there is no one specific limiting case. The analyses demonstrate that the RCCA drop event with the ACT control function active and automatic turbine valve control is more limiting because of the more rapid return to power. ~~Table 15.4-9—Dropped RCCA—Sequence of Events presents the sequence of events for a representative, 423 pcm, dropped RCCA bank case. Figure 15.4-15—Dropped RCCA—Reactor Power through Figure 15.4-19—Dropped RCCA—Primary System Temperature present the plant response for this event.~~The representative case selected for presentation is a 468 pcm dropped RCCA bank case with EOC conditions. The results of this case are presented in Table 15.4-9—Dropped RCCA – Sequence of Events, Figure 15.4-15—Dropped RCCA – Reactor Power, Figure 15.4-16—Dropped RCCA – Reactivity, Figure 15.4-17—Dropped RCCA – Average Core Heat Flux, Figure 15.4-18—Dropped RCCA – Primary System Pressure, Figure 15.4-19—Dropped RCCA – Primary System Temperature, and Figure 15.4-55—Dropped RCCA – Minimum DNBR and Maximum LPD Normalized to SAFDL.

The DNB RT setpoints as well as the dynamic compensation built into the low DNB channel algorithm are adequate to protect the DNB SAFDL for conditions that cause the low DNB channel to issue an RT. For conditions where the DNB degradation does not cause an RT, the DNB LCO is adequate to protect the DNB SAFDL.

The peak fuel centerline temperature remains below the fuel melting point and uniform clad strain remains below one percent.

#### 15.4.3.1.4 Radiological Consequences

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

#### 15.4.3.1.5 Conclusions

The analyses show that the plant instrumentation, protection functions, and equipment are sufficient to preclude fuel or cladding damage for the dropped RCCA event. The core remains adequately cooled throughout this event.

### 15.4.3.2 Statically Misaligned RCCA

The effects of a statically misaligned RCCA are described in the setpoint methodology of Reference 2.

## Table 15.4-8—Dropped RCCA – Equipment Status

| Plant Equipment or System | Status |
|---|---|
| RCCA control mode | ACT control function or manual |
| Turbine control valves position control mode | Open consistent with full-power operation, fixed-position, and automatically controlled cases |
| RCPs | Operating until LOOP |
| SPNDs | Cases with low DNBR RT: No signal from SPND in location with lowest DNBR.<br>Cases with HLPD RT: No signal from SPND in location with highest (or alternately, second-highest) LPD. |
| Excore detectors | Cases with no low DNBR or HLPD RT: No signal from excore detector in core quadrant with highest (or alternately, second-highest) power |
| Pressurizer heaters | Disable |
| Pressurizer spray | Available |
| Main feedwater | Automatic |
| Auxiliary feedwater | Available |
| Rod block system | Disable |
| Initial RCCA position | PDIL |
| Rod position control system NI feedback | Sensor-shadowed |
| RPS NI signals | Decalibrated and sensor-shadowed |

15.04.03-1

## Table 15.4-9—Dropped RCCA – Sequence of Events

| Event | Time (s) |
|---|---|
| Dropped control rod initiated | 0 |
| Control bank withdrawal begins | ~~2.2~~0.8 |
| Low DNBR trip signal | 3.36 |
| Rod completely dropped | 3.5 |
| ~~Low DNBR trip signal~~ | ~~13.1~~ |
| ~~Low DNBR trip (with sensor delay)~~ | ~~13.6~~ |
| ~~TT signal~~ | ~~13.7~~ |
| ~~LOOP~~ | ~~13.7~~ |
| Control rods drop for core reactivity (with delays for sensor, breaker opening and gripper release) | ~~13.9~~3.76 |
| TT signal | 4.36 |
| LOOP | 4.36 |

**Figure 15.4-15—Dropped RCCA - Reactor Power**

15.04.03-1

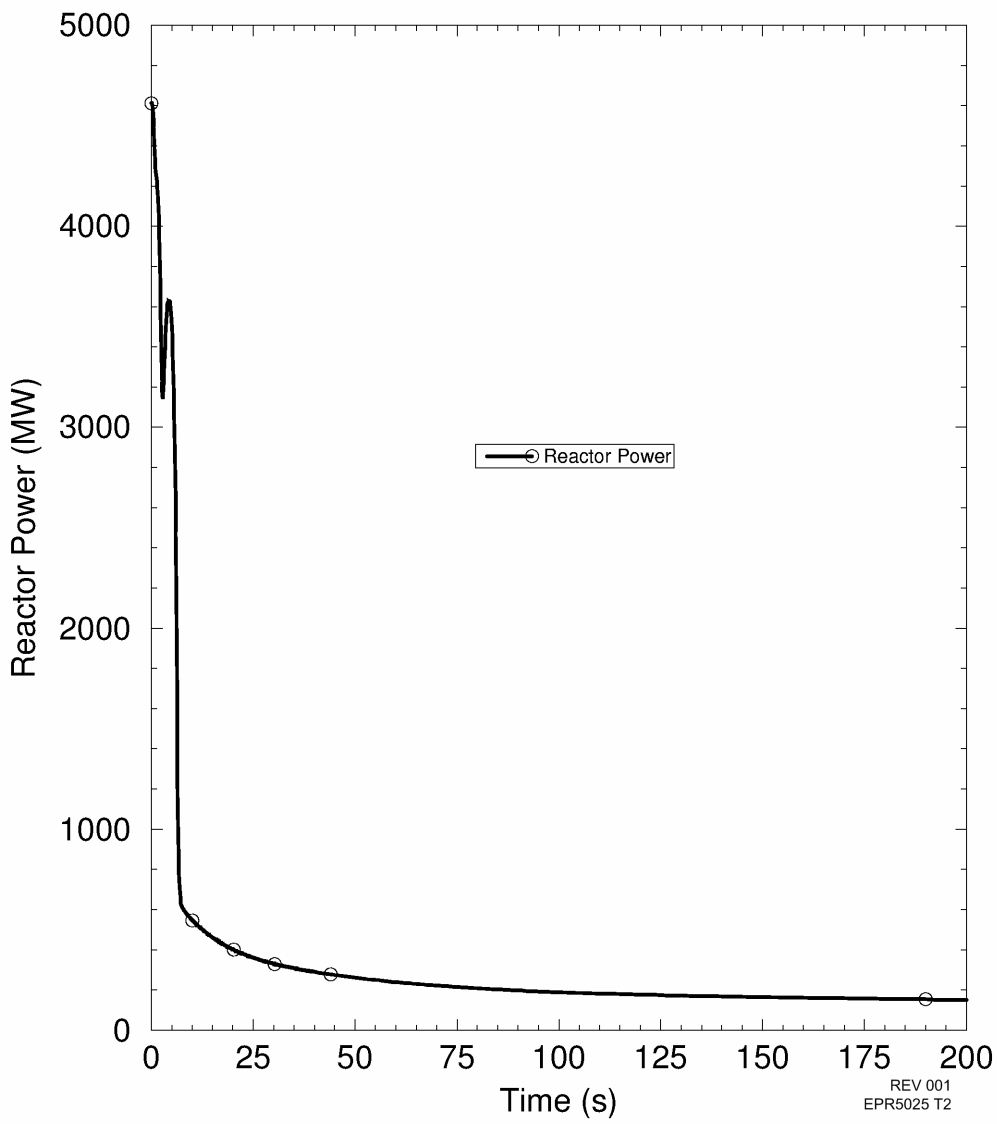

**Figure 15.4-16—Dropped RCCA - Reactivity**

15.04.03-1

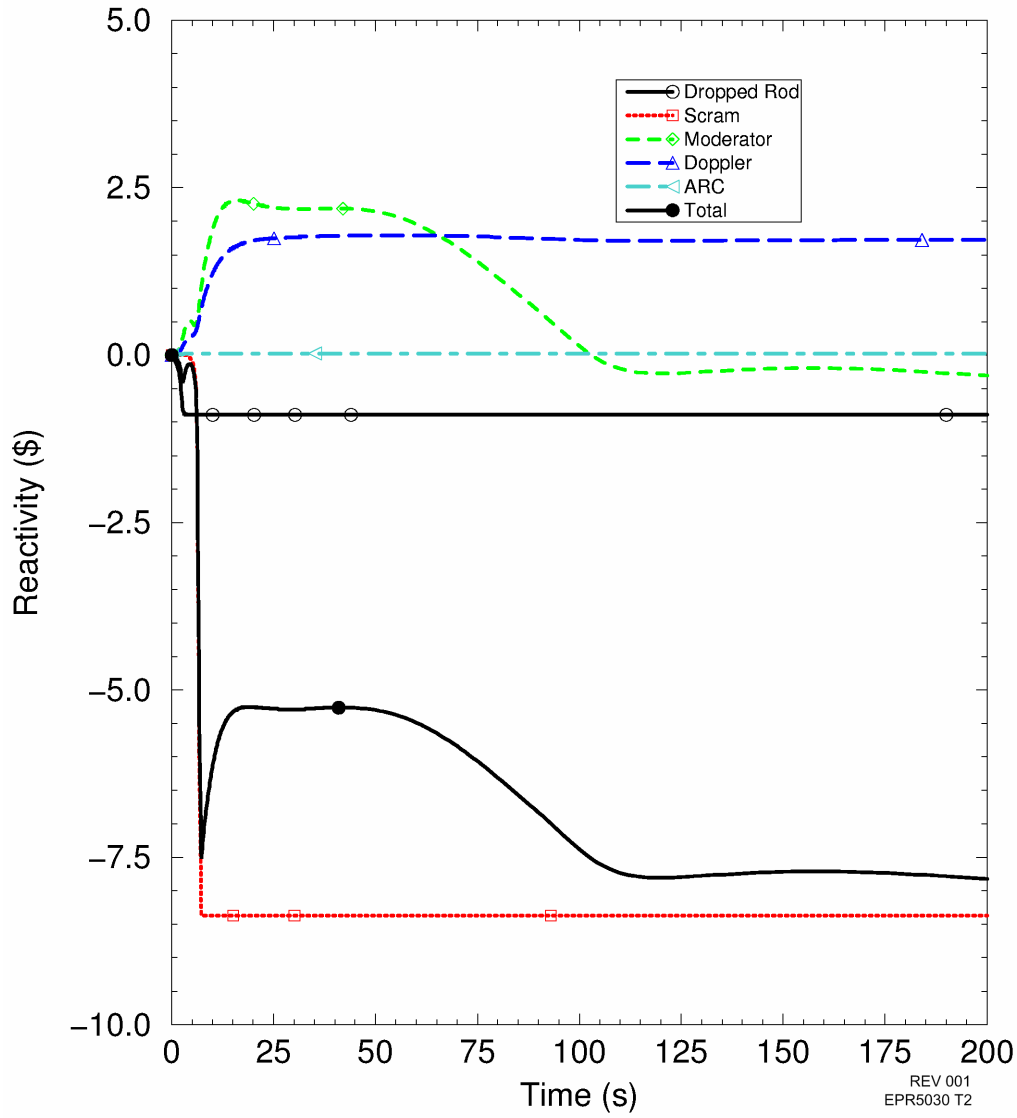

**Figure 15.4-17—Dropped RCCA - Average Core Heat Flux**

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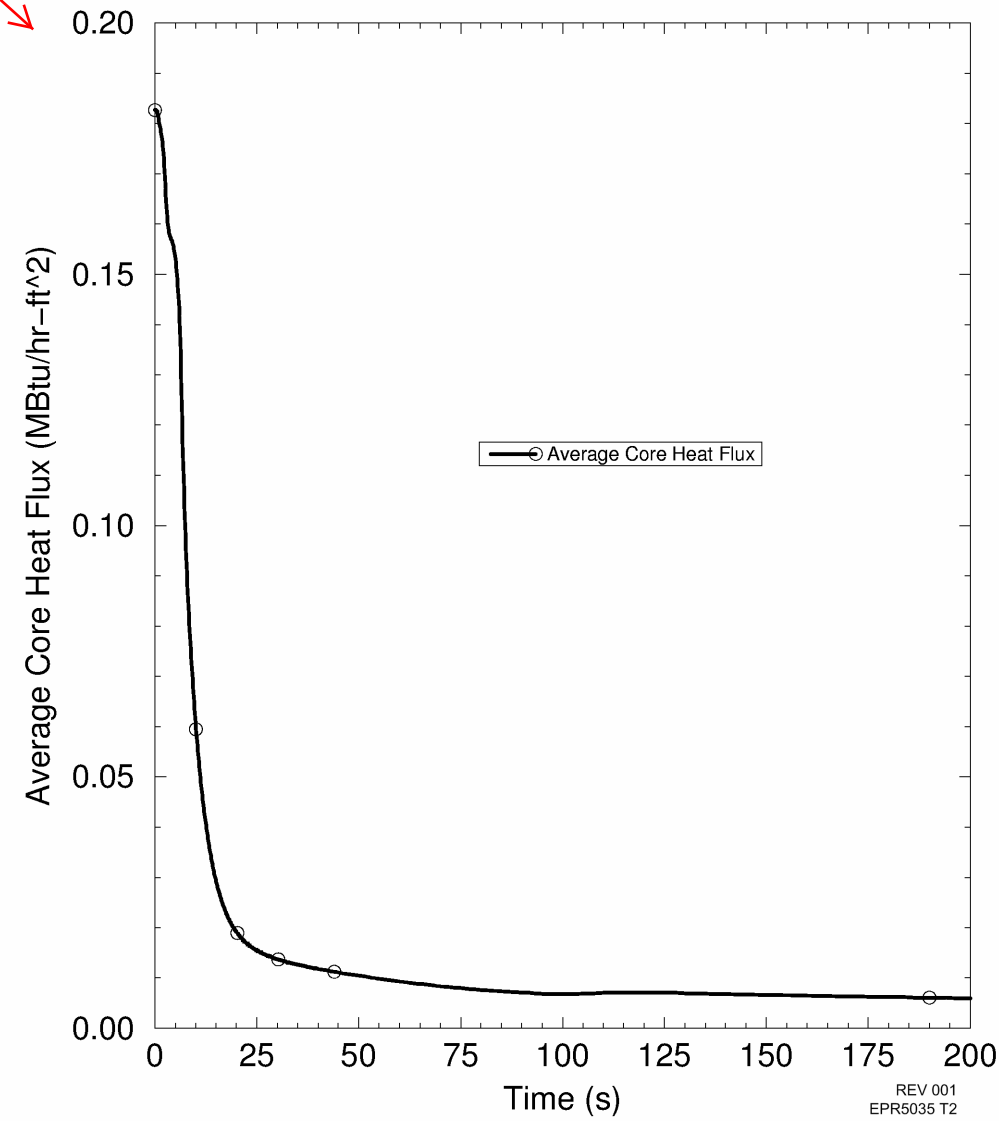

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**Figure 15.4-18—Dropped RCCA - Primary System Pressure**

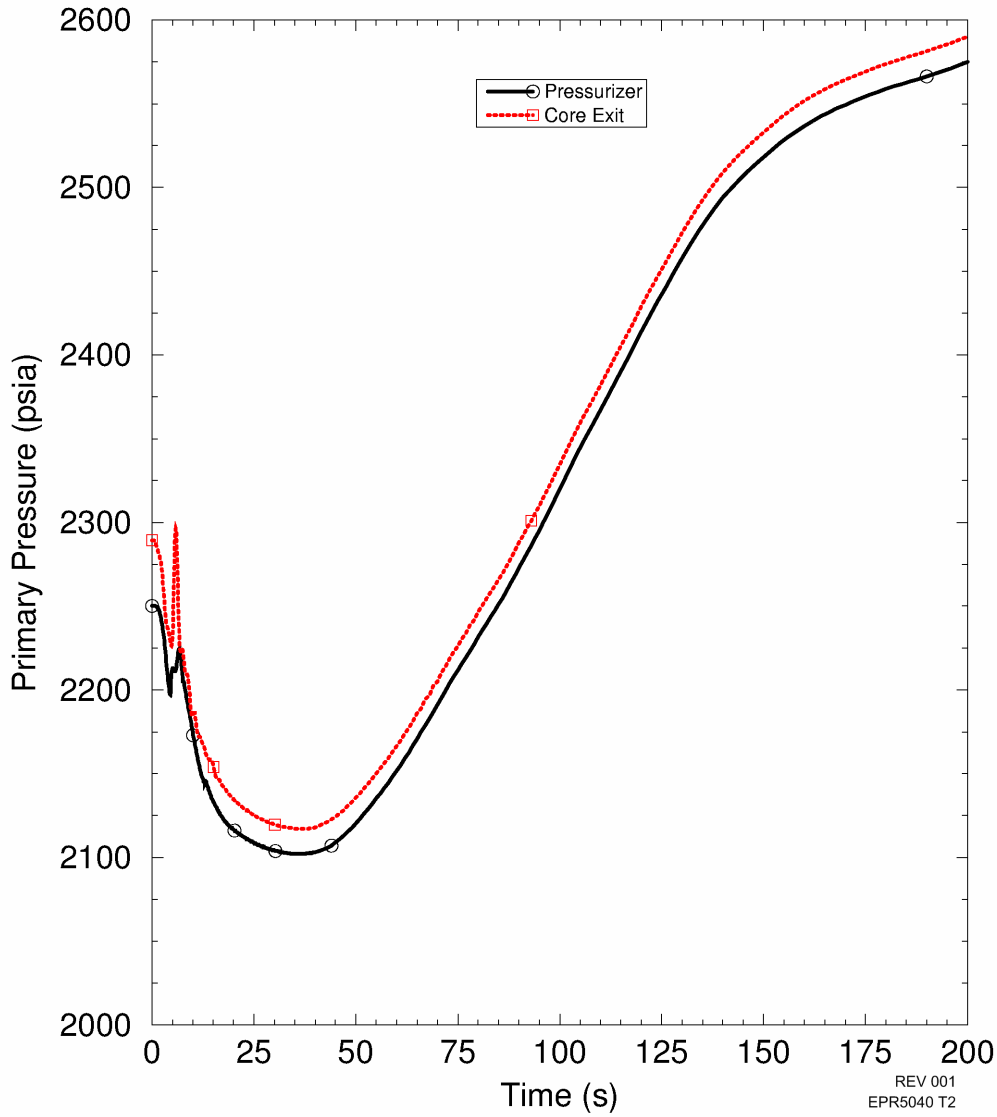

**Figure 15.4-19—Dropped RCCA - Primary System Temperature**

15.04.03-1

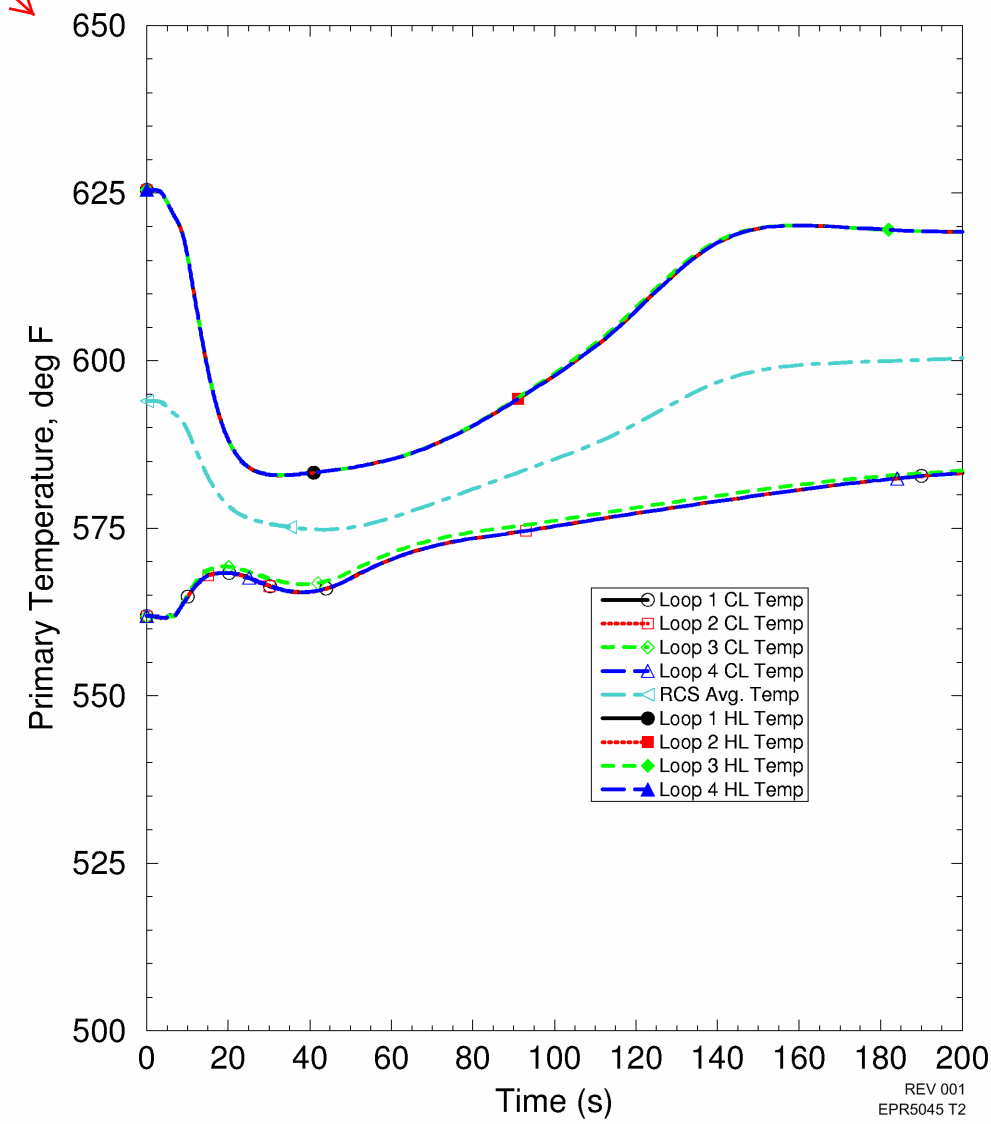

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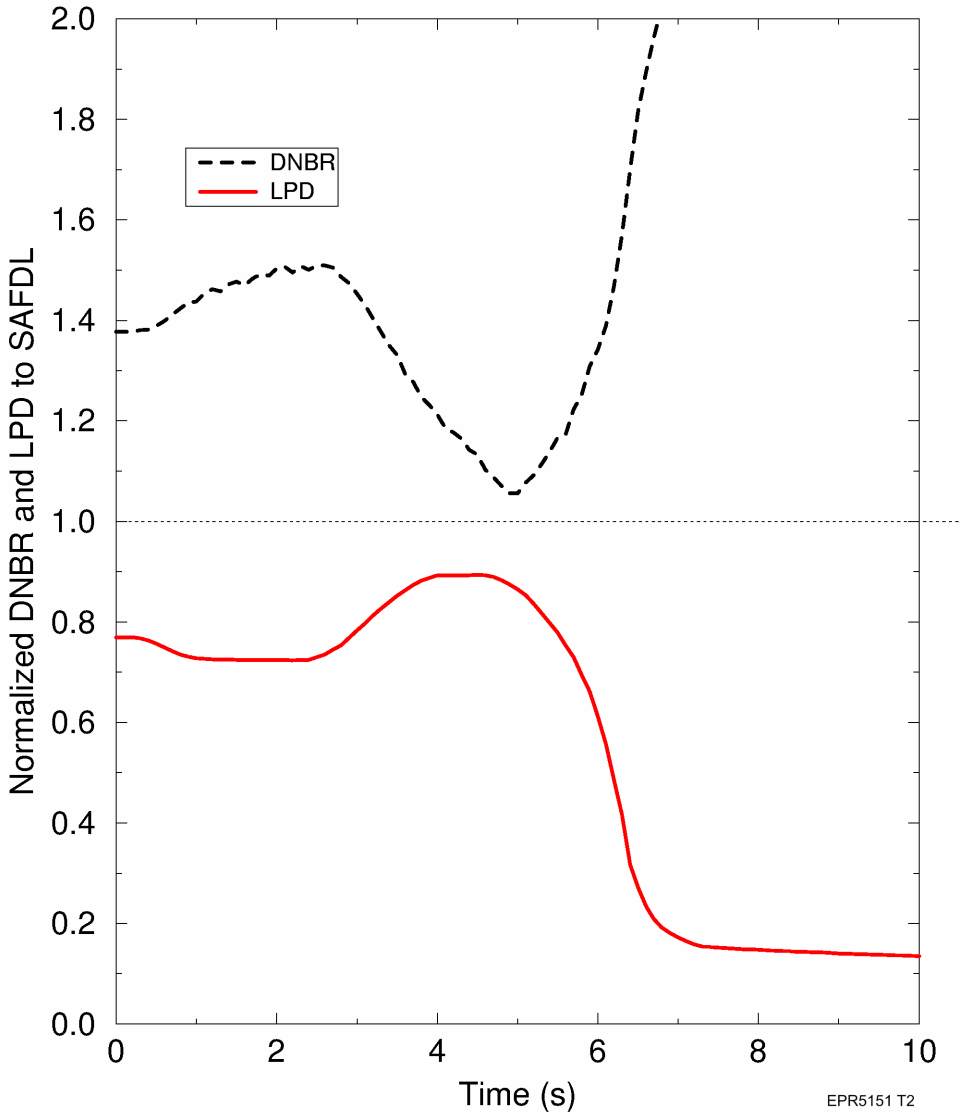
**Figure 15.4-55—Dropped RCCA – Minimum DNBR and Maximum LPD Normalized to SAFDL**

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