ArevaEPRDCPEm Resource

From: Pederson Ronda M (AREVA NP INC) [Ronda.Pederson@areva.com]
Sent: Wednesday, September 24, 2008 5:53 PM
To: Getachew Tesfaye
Cc: WELLS Russell D (AREVA NP INC); John Rycyna; DELANO Karen V (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 29, Supplement 1
Attachments: RAI 29 Supplement 1 Response US EPR DC.pdf

Getachew,

By letter dated August 29, 2008, AREVA NP Inc. provided responses to 15 of the 17 questions of the subject request for additional information (RAI). The attached file, "RAI 29 Supplement 1 Response US EPR DC.pdf" provides technically correct and complete responses to 1 of the remaining 2 questions, per the commitment AREVA NP Inc.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which support the response to RAI 29 Question 15.04.02-1.

The following table provides the page(s) in the response document, "RAI 29 Supplement 1 Response US EPR DC.pdf" containing the response to the question.

Question #	Start Page	End Page
RAI 29 — 15.04.02-1	2	3

The schedule for a technically correct and complete response to the remaining question was provided previously and remains as indicated below.

Question #	Response Date
RAI 29 — 15.04.03-1	November 26, 2008

Sincerely,

Ronda Pederson
ronda.pederson@areva.com
Licensing Manager, U.S. EPR Design Certification
New Plants Deployment
AREVA NP, Inc.
An AREVA and Siemens company
3315 Old Forest Road
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From: WELLS Russell D (AREVA NP INC)
Sent: Thursday, August 28, 2008 5:33 PM
To: 'Getachew Tesfaye'
Cc: 'John Rycyna'; Pederson Ronda M (AREVA NP INC); DUNCAN Leslie E (AREVA NP INC); DELANO Karen V (AREVA NP INC)
Subject: AREVA NP's Response to U.S. EPR Design Certification Application RAI No. 29

Getachew,

The proprietary and non-proprietary versions of the technically correct and complete responses to 15 of the 17 questions of RAI 29, are submitted via AREVA NP Inc. letter, "Response to U.S. EPR Design Certification Application RAI 29," NRC 08:058, dated August 28, 2008. An affidavit to support withholding of information from public disclosure, per 10CFR2.390(b), is provided as an enclosure to that letter.

Complete answers are not provided for 2 of the questions. The schedule for technically correct and complete response for these questions is provided below.

Question #	Response Date
RAI 29—15.04.02-1	September 26, 2008
RAI 29—15.04.03-1	November 26, 2008

Sincerely,

(Russ Wells on behalf of)

Ronda Pederson
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From: Getachew Tesfaye [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Friday, August 01, 2008 6:12 PM
To: ZZ-DL-A-USEPR-DL
Cc: Christopher VanWert; Shanlai Lu; Peter Hearn; Joseph Donoghue; John McKirgan; Joseph Colaccino; John Rycyna
Subject: U.S. EPR Design Certification Application RAI No. 29, FSAR Ch 15

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on July 11, 2008, and discussed with your staff on July 16, 2008. Draft RAI Questions 15.04.08-10 and 11 were deleted as a result of that discussion. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. For any RAIs that cannot be answered within 30 days, it is expected that a date for receipt of this information will be provided to the staff within the 30 day period so that the staff can assess how this information will impact the published schedule.

Thanks,
Getachew Tesfaye
Sr. Project Manager
NRO/DNRL/NARP
(301) 415-3361

Hearing Identifier: AREVA_EPR_DC_RAIs
Email Number: 182

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Subject: Response to U.S. EPR Design Certification Application RAI No. 29, Supplement 1
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From: Pederson Ronda M (AREVA NP INC)

Created By: Ronda.Pederson@areva.com

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Tracking Status: None

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RAI 29 Supplement 1 Response US EPR DC.pdf		167864

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Return Notification: No
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Response to

Request for Additional Information No. 29 Supplement 1, Revision 0

8/01/2008

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020

SRP Section: 15.04.02 - Uncontrolled Control Rod Assembly Withdrawal at Power
Application Section: Ch 15
SRSB Branch

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 representative 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 attached new U.S. EPR FSAR, Tier 2, Figure 15.4-54—Uncontrolled Control Bank Withdrawal at Power – Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL. This event results in both low DNBR channel and high LPD channel incore trips.

The LPD line on the figure is the calculated linear power density normalized to either the fuel centerline melt or cladding strain limits (whichever is more limiting), including the applicable uncertainties. For anticipated operational occurrence (AOO) events, the SAFDL for cladding strain is more limiting and is therefore used to provide the normalized LPD. The LPD SAFDL is designed to protect both the fuel centerline melt and cladding strain limits, as discussed in ANP-10287P, Revision 0, "Incore Trip Setpoint and Transient Methodology for the U.S. EPR Topical Report."

The DNB reactor trip (RT) and high LPD RT setpoints, as well as the dynamic compensation built into the low DNB channel algorithm and the high LPD channel algorithm, are adequate to protect the SAFDL for the conditions that cause the low DNB channel or high LPD channel to issue an RT signal. For conditions where the DNB and LPD degradation do not cause an RT, both the DNB limiting condition for operation (LCO) and the LPD LCO are adequate to protect the SAFDL. This demonstrates that both the fuel cladding integrity is maintained and the peak centerline temperatures remain below the fuel centerline melt limit.

U.S. EPR FSAR, Tier 2, Section 15.4.2.will be revised to clarify the event analysis. In response to RAI-34, Question 15-8, the description of the low DNBR algorithm and incore transient methodology was included in U.S. EPR FSAR, Tier 2, Section 15.0.0.3.9.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Section 15.4.2 will be revised as described in the response and indicated on the enclosed markup.

U.S. EPR FSAR, Tier 2, Figure 15.4-54 will be added as described in the response and indicated on the enclosed markup.

U.S. EPR Final Safety Analysis Report Markups

A. The thermal margin limits as specified in SRP Section 4.4 (Reference 3) are met.

- Response: The results in Section 15.4.1.3 demonstrate that this requirement is met. The minimum DNBR remains above the design limit value

B. Fuel centerline temperatures for (PWRs) as specified in SRP Section 4.2 (Reference 3) do not exceed the melting point.

- Response: The results in Section 15.4.1.3 demonstrate that this requirement is met. The peak fuel centerline temperature remains below the fuel melting point.

C. Uniform cladding strain (for BWRs) as specified in SRP Section 4.2 (Reference 3) does not exceed one percent.

- Response: This SRP requirement is for BWRs. The results in Section 15.4.1.3 demonstrate that the U.S. EPR PWR also meets this requirement.

15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

15.4.2.1 Identification of Causes and Event Description

The uncontrolled control rod assembly withdrawal at power event is defined as an uncontrolled addition of reactivity due to the withdrawal of RCCA banks (described in Sections 15.4.1.1, 4.2, and 4.3) during power operation. This event is postulated to result from either a failure in the RCCA position control system or an operator error that results in an uncontrolled withdrawal of a group of RCCA banks or sub-banks. This transient causes an increase in core power with a corresponding increase in heat flux. Due to the time lag in the response of the secondary system, the heat removal from the steam generators (SG) follows the heat increase in the primary system. Simultaneously, a net increase in the reactor coolant temperature and pressure occurs. Depending on the power level and point in the fuel cycle, the transient terminates from an RT signal due to either a high neutron flux rate of change, high pressurizer level, low DNBR, high LPD, high core power level, or high SG pressure. This event is an AOO of moderate frequency as described in Section 15.0.0.1.

15.04.02-1

The uncontrolled withdrawal of an RCCA bank at power results in either a slow or a fast increase in reactivity. In a slow reactivity increase, the increase in coolant temperature follows the increase in reactor power. In a fast reactivity increase, the reactor power increases at a much faster rate than the coolant temperature. The slow reactivity transient is terminated by a high pressurizer level, low DNBR, high LPD, high core power level, or high SG pressure trip. The fast reactivity transient is terminated by the high neutron flux rate of change RT.

15.04.02-1

In addition to the rate of reactivity increase, the limiting transients for the RCCA bank withdrawal at power depend on the starting reactor power level and the point in the fuel cycle burnup. The analyses considered power levels of 25 percent, 60 percent, and HFP to determine the limiting case. These power levels were selected because each of these power levels represents a breakpoint in the primary average temperature versus power curve for the U.S. EPR. The highest power level for each plateau of primary average temperature is analyzed. The limiting points in the fuel cycle burnup correspond to beginning-of-cycle (BOC) and end-of-cycle (EOC) conditions. Therefore, the following matrix of cases is analyzed for this event:

- HFP at BOC and EOC conditions, with reactivity insertion rates spanning full range.
- 60 percent power at BOC and EOC conditions, with reactivity insertion rates spanning full range.
- 25 percent power at BOC and EOC conditions, with reactivity insertion rates spanning full range.

As determined from the transient events, the reactor system is protected during the RCCA bank withdrawal at power with the following RT setpoints:

15.04.02-1

- Low DNBR.
- High LPD.
- Excore high neutron flux rate-of-change protection.
- High core power level protection.
- High pressurizer level protection.
- High SG pressure.

For fast reactivity transients, the excore neutron flux measurement provides a short response time to protect the core. The remaining trips provide core protection for all but the fastest reactivity transients. Only those trips that are credited in terminating the event are listed above.

The applicable acceptance criteria for the uncontrolled RCCA bank withdrawal at power event are as follows:

- The DNBR thermal margin limit is met.
- Fuel centerline temperatures do not exceed the melting point.
- Uniform cladding strain does not exceed one percent.

15.04.02-1

This event primarily challenges the SAFDL. The DNB SAFDL is satisfied by the combination of the low DNB limiting condition for operation (LCO) and RT setpoint, as described in Reference 2. The dynamic compensation of the low DNB channel algorithm is shown to be adequate to protect the SAFDL when the RT setpoint is reached. The fuel centerline melt and cladding strain SAFDLs are satisfied by the combination of the high LPD LCO and RT setpoint as described in Reference 2. The dynamic compensation of the high LPD channel algorithm is shown to be adequate to protect the SAFDL when the RT setpoint is reached. The high linear power density (HLPD) limits are not exceeded, which demonstrates that fuel centerline melt and one percent uniform clad strain is prevented.

15.4.2.2 Method of Analysis and Assumptions

15.04.02-1

This event uses the S-RELAP5 code and associated methodology described in Reference 1 to simulate the responses of the primary and secondary coolant systems, reactor, protective equipment and systems, and automatic controllers. ~~The core thermal-hydraulic computer code LYNXT is used to calculate the core flow, enthalpy distributions, DNBR, and peak fuel centerline temperatures using the RCS response from S-RELAP5 as a boundary condition.~~ The ~~low DNB channel~~ algorithm, described in Section 15.0.0.3.9, is simulated to predict RT and adequacy of the dynamic compensation of the ~~algorithm~~ incore monitoring system in a manner consistent with Reference 2.

~~The low DNB channel algorithm uses the following measurements:~~

- ~~The reactor power distribution is derived from the self-powered neutron detectors that are part of the nuclear incore instrumentation.~~
- ~~The primary system pressure is derived from the primary pressure sensors.~~
- ~~The core flow is derived from the reactor coolant pump (RCP) speed sensors.~~
- ~~The reactor inlet temperature is derived from the cold-leg temperature sensors.~~

~~Cold-leg temperature measurements are treated in the S-RELAP5 model with a filtering module and a lead-lag module. The cold-leg temperature measurement is used for the DNBR calculations, while the core inlet temperature is the relevant parameter for the physical DNBR. A time delay occurs when the reactor coolant travels from the cold-leg measurement location to the core inlet. The lead-lag module compensates for this time delay.~~

Neither a single failure nor equipment taken out of service for maintenance makes this event more severe. The protection system (PS) is the only system or equipment that mitigates this event and its redundant design makes it single-failure proof. Following RT, it assumed that the highest worth rod is stuck above the core. It is also assumed that a loss of offsite power (LOOP) occurs coincident with turbine trip (TT). The opening and closing setpoints of the pressurizer safety relief valves (PSRVs) are biased conservatively low.

Table 15.4-4—Uncontrolled Control Bank Withdrawal at Power – Key Input Parameters presents the key input parameters used in the analyses. The control rod having the greatest worth is assumed to be stuck above the core. In addition, the most conservative negative reactivity insertion curve as a function of time is used in the transient analyses.

The power range considered for this event ranges from 25 percent power up to HFP. The uncontrolled withdrawal of a control bank below 10 percent power is described in Section 15.4.1. For each reactor power and burnup condition considered, the RCCA bank withdrawal rates that are analyzed range from a single RCCA to the maximum bank value for the particular burnup condition. The BOC condition is a minimum reactivity feedback case, while the EOC condition is a maximum reactivity feedback case. For the minimum reactivity feedback case, the moderator temperature coefficient is zero or larger depending on reactor power. For conservatism, the corresponding Doppler coefficient is minimized in absolute value. For the maximum reactivity feedback case, the most negative moderator temperature and Doppler coefficients are assumed.

The analyses minimize DNB margin by conservatively assuming that the pressure control system is operational, pressurizer spray flow is at its maximum value, and the PSRV opening and closing setpoints are biased low. Non-safety-related equipment is considered when it causes a more limiting transient. Table 15.4-5—Uncontrolled Control Bank Withdrawal at Power – Equipment Status lists the plant systems and equipment that are assumed available to mitigate this event.

15.4.2.3 Results

15.04.02-1

Table 15.4-6—Uncontrolled Control Bank Withdrawal at Power – Sequence of Events presents the sequence of events for a representative case. The analysis of this event considered a spectrum of reactivity insertion rates at power levels up to HFP. Table 15.4-6 provides the sequence of events for the maximum RCCA worth withdrawal at power and BOC conditions. Figure 15.4-6—Uncontrolled Control Bank Withdrawal at Power - Reactor Power through Figure 15.4-14—Uncontrolled Control Bank Withdrawal at Power - Pressurizer Spray show the response of the most important system parameters. Figure 15.4-54—Uncontrolled Control Bank Withdrawal at Power - Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL presents a representative case of DNB and LPD normalized to their respective SAFDLs.

~~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~~

15.04.02-1

~~the low DNB channel to issue an RT. For conditions in which 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.~~The DNB RT and high LPD RT setpoints, as well as the dynamic compensation built into the low DNB channel algorithm and the high LPD channel algorithm, are adequate to protect the SAFDL for the conditions that cause the low DNB channel or high LPD channel to issue an RT. For conditions where the DNB and LPD degradation do not cause an RT, both the DNB LCO and LPD LCO are adequate to protect the SAFDL. This demonstrates that both the fuel cladding integrity is maintained and peak centerline temperatures remain below the fuel centerline melt limit.

15.4.2.4 Radiological Consequences

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

15.4.2.5 Conclusions

The analyses show that the plant instrumentation, protection functions, and equipment are sufficient to preclude fuel or cladding damage for the uncontrolled control bank withdrawal at power event. The core remains adequately cooled throughout this event.

15.4.2.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.4.2 events included in NUREG-0800, Section 15.4.2, (Reference 3) and descriptions of how these criteria are met are listed below:

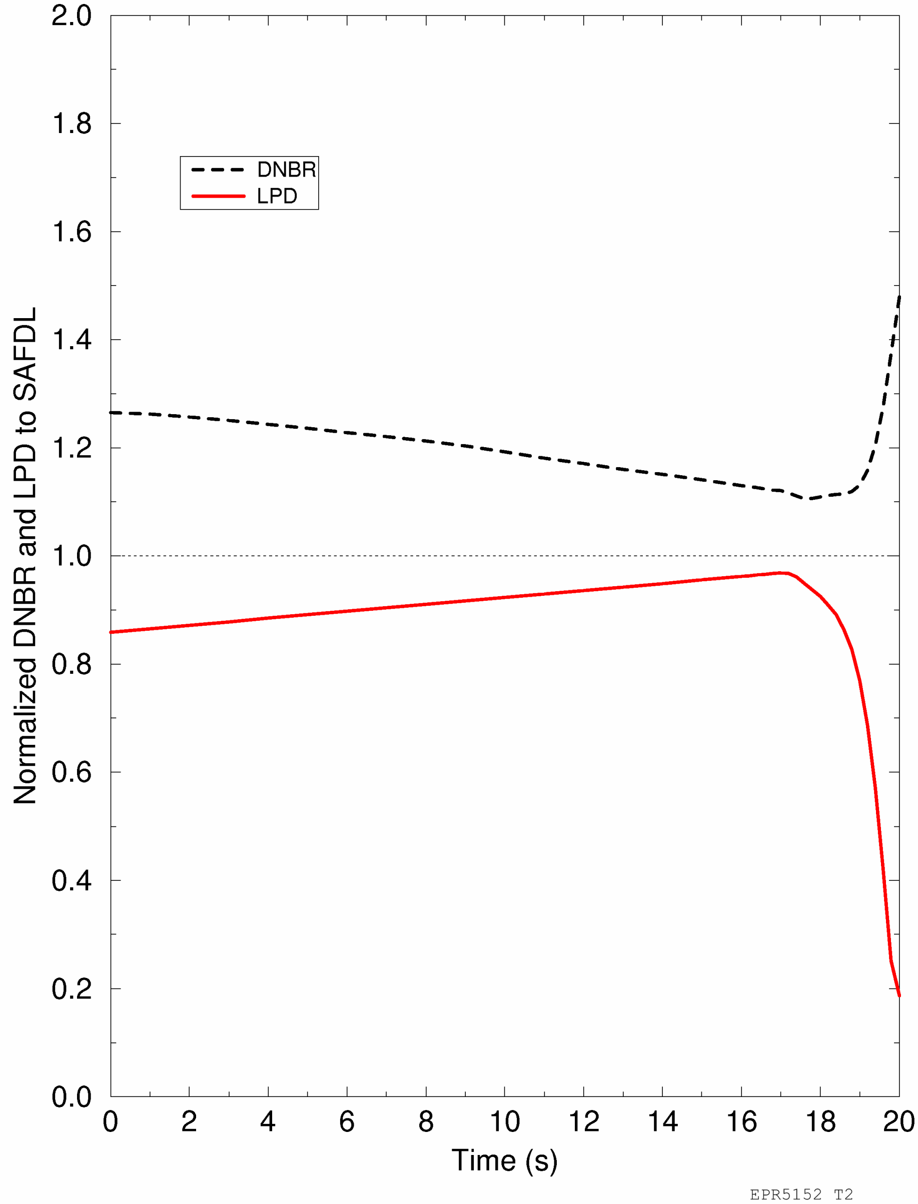
1. The requirements of GDC 10, 17, 20, and 25 concerning the SAFDL are assumed to be met for this event when:

   A. The thermal margin limits as specified in SRP Section 4.4 (Reference 3) are met.

      - Response: The results in Section 15.4.2.3 demonstrate that this requirement is met. 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 in which the DNB degradation does not cause an RT, the DNB LCO is adequate to protect the DNB SAFDL.

   B. Fuel centerline temperatures (for PWRs) as specified in SRP Section 4.2 (Reference 3) do not exceed the melting point.

Figure 15.4-54—Uncontrolled Control Bank Withdrawal at Power - Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL



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