AREVA

October 30, 2013
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U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Response to Eighth Request for Additional Information Regarding ANP-10285P, "Fuel Assembly Mechanical Design Topical Report"

Ref. 1: Letter, Ronnie L. Gardner (AREVA NP Inc.) to Document Control Desk (NRC), "Request for Review and Approval of ANP-10285P, 'U.S. EPR Fuel Assembly Mechanical Design Topical Report'," NRC:07:051, October 2, 2007.

Ref. 2: Letter, Amy M. Synder (NRC) to Pedro Salas (AREVA NP Inc.), "Eighth Request for Additional Information Regarding ANP-10285P, 'Fuel Assembly Mechanical Design Topical Report', (TAC No. RN1224)" July 29, 2013.

AREVA NP Inc. (AREVA NP) requested the NRC's review and approval of the Topical Report ANP-10285P, "U.S. EPR Fuel Assembly Mechanical Design Topical Report" in Reference 1. The NRC provided a request for additional information (RAI) regarding this topical report in Reference 2.

The enclosure to this letter provides a complete response to the eighth RAI regarding ANP-10285P. AREVA NP considers some of the material contained in the enclosed response to be proprietary. As required by 10 CFR 2.390(b), an affidavit is enclosed to support the withholding of the information from public disclosure. A proprietary and a non-proprietary version of this response is enclosed.

The following table indicates the respective pages that contain AREVA NP's final response to the eighth RAI regarding ANP-10285P.

RAI #	Start Page	End Page
72	2	16

This concludes the formal AREVA NP response to the eighth RAI regarding ANP-10285P, and there are no questions from this RAI for which AREVA NP has not provided a response.

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NRO

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If you have any questions related to this information, please contact Len Gucwa by telephone at 434-832-3466, or by e-mail at Len.Gucwa.ext@areva.com.

Sincerely,

Pedro Salas, Director
Regulatory Affairs
AREVA NP Inc.

Enclosures:

1. Proprietary Version of "Response to Eighth Request for Additional Information Regarding ANP-10285P, 'Fuel Assembly Mechanical Design Topical Report'."
2. Non-Proprietary Version of "Response to Eighth Request for Additional Information Regarding ANP-10285P, 'Fuel Assembly Mechanical Design Topical Report'."
3. Notarized Affidavit.

cc: M. J. Miernicki
Docket 52-020

AFFIDAVIT

STATE OF NORTH CAROLINA)
) ss.
COUNTY OF MECKLENBURG)

1. My name is Thomas E Ryan. I am Manager, Product Licensing, for AREVA NP Inc. (AREVA NP) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in the document titled "Response to Eighth Request for Additional Information Regarding ANP-10285P, Fuel Assembly Mechanical Design Topical Report" and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information":

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(c) and 6(d) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

Thomas E. Ryan

SUBSCRIBED before me this 30

day of October 2013.

Amber L. McReynolds
NOTARY PUBLIC, STATE OF NORTH CAROLINA, COUNTY OF CABARRUS
MY COMMISSION EXPIRES: 29 January 2017
Reg. # 201203900154

AMBER L MCREYNOLDS
Notary Public
Cabarrus County
North Carolina
My Commission Expires Jan 29, 2017

Response to Eighth Request for Additional Information – ANP-10285P "U.S. EPR Fuel Assembly Mechanical Design Topical Report"

RAI 72: *Addressing Damping Assumptions for U.S. EPR Fuel Seismic Response Analysis with Detailed Regulatory Basis*

Background

Recent operating experience at nuclear power plants has shown that full reactor coolant system (RCS) flow is not likely to be maintained following a seismic event due to a loss of offsite power (LOOP). Maintaining full RCS flow requires several reactor coolant pumps (RCPs) to be operating at full speed, and these RCPs are not connected to safety-related, seismically qualified, electrical buses. During a LOOP, all RCPs would coast down in a relatively short period of time. Loss of other non-seismically qualified equipment during a seismic event, such as the turbine, could also cause RCPs to coast down, resulting in decreased core flow. The staff's concern is that before the reactor is shutdown (i.e. operating at greater than hot zero power), maximum ground acceleration could occur in conjunction with decreased core flow, reducing the flow rate dependent critical damping ratio, which could cause larger spacer grid impact loads than assumed in the existing analysis.

The guidance used to evaluate external forces on fuel assemblies is NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) Section 4.2, "Fuel System Design," Revision 3, March 2007, Appendix A, "Evaluation Of Fuel Assembly Structural Response To Externally Applied Forces." SRP Section 4.2, Appendix A, Section II.2, states, "analytical methods used in performing structural response analyses should be reviewed." This includes the bases for the various input assumptions, such as the critical damping ratio, that have a direct impact on the spacer grid impact loads. SRP Section 4.2, Appendix A, Section IV.2, which gives the safe-shutdown earthquake (SSE) acceptance criteria, further states, "control rod insertability must be assured," and it must be assured for "SSE loads alone if [SRP 4.2, Appendix A] Subsection IV.1 does not require an analysis for combined loads." This means that control rod insertability still needs to be demonstrated for SSE-only loads even if the combined loads analysis does not exceed P(crit) – the spacer grid crushing load.

Analyses typically compare maximum spacer grid loads to P(crit) to show that control rod insertability will be maintained since there is a presumption that significant permanent grid deformation does not occur for loads less than P(crit), and that only buckling could prevent control rod insertion. However, for the U.S. EPR design, significant permanent grid deformation is predicted under maximum spacer grid loads without spacer grid buckling, which could challenge control rod insertability. Therefore, if the maximum spacer grid impact loads are being under-predicted due to the use of a non-conservative critical damping ratio, then spacer grid impact loads would need to be updated accordingly, and control rod insertability may need to be re-evaluated.

Regulatory Basis and Acceptance Criteria

General Design Criterion (GDC) 2 states:

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

The applicable component is the fuel, with the safety functions being maintenance of fuel integrity and control rod insertability. The applicable natural phenomenon for the fuel seismic response analysis is the most severe earthquake, which is the SSE. Fuel integrity is typically demonstrated by showing that coolability is always maintained – specifically by demonstrating that the departure from nucleate boiling ratio (DNBR) is always maintained above an appropriate lower limit under all normal conditions of operation in accordance with SRP Section 4.4, and for all anticipated operational occurrences in accordance with SRP Chapter 15.

The current AREVA fuel seismic response analysis does not justify the appropriateness of the assumed critical damping ratio corresponding to full reactor coolant system flow based on the above considerations in Topical Report ANP-10285P, "U.S. EPR Fuel Assembly Mechanical Design Topical Report." Therefore, the staff cannot conclude that the U.S. EPR design meets the requirements of GDC 2.

Request for Additional Information

SRP Section 4.2, Appendix A discusses fuel coolability criteria related to a loss-of-coolant accident (LOCA), which is based on the assumption of combined loads, and addresses the impact on the ECCS analysis. The U.S. EPR design has shown that it may be necessary to perform additional coolability and control rod insertability evaluations above decay heat power levels, including full power operation, in order to address permanent spacer grid deformation caused by spacer grid impact forces that do not exceed P(crit) under SSE-only loads.

Since the assumed critical damping ratio has a direct impact on the predicted spacer grid impact loads, it will also have a direct impact on the amount of permanent grid deformation that is predicted. Therefore, the predicted permanent grid deformation resulting from a reduced critical damping ratio during a LOOP following a seismic event should be evaluated for the U.S. EPR design with respect to control rod insertability and fuel rod coolability.

Justify the critical damping ratio used in the fuel assembly structural response analysis for the U.S. EPR. Address the following points in your response:

a. *Quantify any change to the critical damping ratio assumed in the analysis based on RCP coastdown considerations.*
b. *Include considerations for both the unirradiated and irradiated cases. Additionally, provide the Rayleigh damping coefficients being used for the irradiated fuel assembly cases in the fuel assembly structural response analysis for the U.S. EPR.*
c. *Quantify the damping ratio margin (i.e. the difference between the critical damping ratio derived from test data and that credited in the analysis) change for both the unirradiated and irradiated fuel assembly cases.*

d. *Address both SSE-only and combined SSE and LOCA loads analyses.*

Response to RAI 72:

Response Overview

The issue stated in RAI 72 is that for a Safe Shutdown Earthquake (SSE) + Loss of Offsite Power (LOOP) event, prior to reactor shutdown, the maximum ground acceleration could occur in conjunction with the decreased flow caused by the LOOP, reducing the flow rate dependent damping ratio, which could cause a larger spacer grid impact load than predicted in the existing analysis. This response will demonstrate that, up to the time of reactor shutdown, the reduced flow rate caused by an SSE+LOOP will not result in an increase in loads above those predicted in the current analysis. The current analysis is described in Reference 2.

RAI 72 defines the regulatory basis for the SSE + LOOP event as being 10 CFR 50 Appendix A General Design Criterion 2. The regulation which implements General Design Criterion 2 with regards to earthquakes is 10 CFR 50 Appendix S. Consideration of this regulation is necessary since it defines the criteria to be satisfied by a SSE.

The important elements of 10 CFR 50 Appendix S relevant to this response are summarized below:

- SSE is the vibratory ground motion for which certain structures, systems, and components must be designed to remain functional. Functionality is based on meeting the SSE criteria defined in 10 CFR 50 Appendix S.

- The loads to be considered in the design of these systems, structures, and components, are the seismic loads in combination with the normal operating, functional, and accident-induced loads.

  If an SSE causes a LOOP, the impact of this combined event on the fuel assembly should be addressed. It will be conservatively assumed for this response that a LOOP can be caused by an SSE, and that the LOOP can occur at any time during the SSE.

- The criterion stated in 10 CFR 50 Appendix S relative to control rods is reactor shutdown, and not necessarily control rod insertability.

  It will be conservatively assumed for the purpose of this response that reactor shutdown occurs at the time that all control rods insert completely during a LOOP + SSE event.

- The criterion stated in 10 CFR 50 Appendix S relative to fuel rods is radiological dose limits per 10 CFR 50.34(a)(1), and not necessarily fuel rod cladding integrity.

  It will be conservatively demonstrated in this response that the DNBR criterion is met for the SSE+LOOP event and therefore the radiological criteria in 10 CFR 50.34(a)(1) are met.

- 10 CFR 50 Appendix S identifies the operating basis earthquake (OBE) as the earthquake for which the plant must continue to operate. The regulation requires shutdown if the severity of the earthquake exceeds the OBE level based on a prompt evaluation, as defined in Regulatory Guide 1.166 (reference 1). In contrast, 10 CFR 50 Appendix S defines the SSE as the earthquake for which certain structures, systems and components must remain functional.

As previously noted, the conservative assumption will be made that an SSE can trigger a LOOP, and the response will demonstrate that up to the time of shutdown, the criteria are met as set forth in 10 CFR 50 Appendix S, which apply to SSE + anticipated operational occurrence (AOO) scenarios.

The key issue is the impact of a reduction in RCS flow, due to a LOOP occurring during an SSE, on the evaluation of the maximum impact load on the fuel assembly spacer grid. The purpose of this response is to show that the fuel assembly does not respond in such a manner as to challenge the SSE + AOO acceptance criteria: 1) reactor pressure boundary integrity, 2) reactor shutdown, and 3) 10 CFR 50.34(a)(1) radiological limits.

The postulated event (SSE + LOOP) is, in essence, a superposition of two events:

1. The Complete Loss of Forced Reactor Coolant Flow, which is an AOO analyzed in U.S. EPR FSAR, Tier 2, Section 15.3.2, and
2. An SSE as defined in 10 CFR 50 Appendix S with the potential for spacer grid deformation

The first event has been analyzed in U.S. EPR FSAR Tier 2, Section 15.3.2.and was shown to meet all SRP criteria. The two criteria with direct relevance for this evaluation are:

1. Pressure boundary integrity: the maximum pressure during the loss of flow event does not exceed the design pressure limits, and
2. The DNBR is above the 95 percent probability with 95 percent confidence DNBR limits for the entire transient.

The direct consequence of item 1 above is that the combined event, SSE + LOOP, is also acceptable from the point of view of pressure boundary integrity, given that the maximum pressure during the LOOP does not challenge the design pressure limits, and that the SSE loads are factored in the pressure boundary analysis of U.S. EPR FSAR Tier 2, Section 3.9.3.

For the other two criteria, reactor shutdown, and radiological criteria, this response will show that reactor shutdown is assured (by demonstrating conservatively that the control rods insert) and the radiological criteria are met (by demonstrating the positive DNBR margins established for the Complete Loss of Forced Reactor Coolant Flow AOO are not affected by the SSE+LOOP up to the time of reactor shutdown (the time the control rods have reached the bottom of the core)). The minimum DNBR in the Complete Loss of Forced Reactor Coolant Flow transient has positive margins and occurs before the time of complete control rod insertion.

It will be demonstrated that the seismic analysis in ANP-10325P (reference 2) uses damping values for both the un-irradiated and irradiated conditions that are conservative with respect to the flow rates experienced in the time interval between the beginning of the RCP coast-down and up to the time of reactor shutdown (the time that the control rods reach the bottom of the core).

Detailed Response

SSE+LOOP Event Characteristics

The SSE+LOOP event is equivalent to the Complete Loss of Forced Reactor Coolant Flow, which is an AOO analyzed in U.S. EPR FSAR Tier 2, Chapter 15, Section 15.3.2. The plots of the normalized RCS flow rate, and pressurizer pressure for this transient are shown in Figure 72-1 and Figure 72-2. The sequence of events is as follows:

1. The LOOP [

]

The key characteristics of the Complete Loss of Forced Reactor Coolant Flow transient are summarized as follows:

- The maximum pressure during the Complete Loss of Flow event (Figure 72-2) is only 2300 psi, which is less than the design pressure of the U.S. EPR RCS of 2535 psi. This observation supports the conclusion that the SSE+LOOP event does not challenge the pressure boundary integrity.
- The Complete Loss of Flow event has positive DNBR margins. The minimum DNBR values occur during the early part of the transient, [

]

- The RCP flow fractions during the key time points of the LOOP event are shown in Table 72-1. The time corresponding to complete control insertion will be used to define the flow for the purpose of demonstrating the applicability of the damping model used in the fuel assembly spacer grid load analysis.

U.S. EPR Fuel Assembly Damping Model Applicability During an SSE + LOOP Event

This section discusses the damping model used for the U.S. EPR fuel assembly, with a focus on the assessment of the applicability of this model for the reduced flow condition during the SSE + LOOP event. The damping model is presented in technical report ANP-10325P (reference 2), and discussed in detail are the damping test base, the scaling methodology from test temperature to in-reactor conditions, and the analytical model.

U.S. EPR Damping Model Test Base

The un-irradiated condition tests were performed on a [ ] fuel assembly and consisted of [ ] at various flow rates. The hydraulic damping ratio information from the tests is representative of the U.S. EPR as addressed in reference 2.

The average test loop flow velocities used for the test were: [ ] ft/sec. This range of flow velocities is less than the hot full power full flow axial coolant flow velocities of the U.S. EPR and therefore making the measured damping results conservative.

The simulated irradiated condition tests were performed on a [ ] fuel assembly and consisted of [ ] at various flow rates. The hydraulic damping ratio information from the tests is representative of the U.S. EPR as addressed in reference 2.

The average test loop flow velocities used for the test were: [ ] ft/sec. Like in the case of the un-irradiated condition tests, these velocities are less than the U.S. EPR hot full power full flow bundle flow rate.

U.S. EPR Damping Model Applicability to the Reduced Flow Rate Condition

The un-irradiated condition test results are summarized in Figure 72-3, and indicate the contribution of [ ] to the overall fuel assembly lateral damping. The physical interpretation of these results has been discussed in reference 2. The discussion concentrates on the applicability of the damping model to the reduced flow condition.

The [ ] flow rate is the key flow rate for this evaluation. Using the scaling methodology discussed in reference 2, the in-reactor damping ratio for this case, is [ ], calculated as:

[ ] **Equation 72-1**

As discussed in reference 2, the [ ] used in the load analysis in reference 2.

The core flow rate at the time of complete control rod insertion is calculated using the core flow fraction in Table 72-1 [ ], and a conservative thermo-hydraulic design core flow rate of [ ] (this value is a conservative lower bound for all operation scenarios). The value of [ ] is derived from the Thermal Design Flow of 119, 692 gpm per loop shown in Table 15.0.5 of the US EPR Tier 2 FSAR. The average core flow rate at [ ]

The irradiated condition test results are summarized in Figure 72-4. The physical interpretation of these results has been presented in reference 2. [ ]

In conclusion, at the time that the control rods reach the bottom of the core, the core flow rate of [

] This makes the analytical results for fuel assembly loads in reference 2 bounding for the SSE + LOOP event at least up to the point of full control rod insertion. Taking the core flow rate at the time that the control rods reach the bottom of the core as base, this represents an [

]

Additional Considerations Regarding the Damping Model Applicability for the Reduced Flow Condition

Some additional considerations supporting the conservatism of this evaluation are listed below:

- [

  ]

- The U.S. EPR switchyard design is such that the plant can operate in "island" mode during a LOOP. The switchyard contains two normal auxiliary transformers (NATs). Only one NAT is necessary to provide house load power to the plant (with the exception of some non-safety loads). In the event of a LOOP, plant equipment such as the reactor coolant pumps and main feedwater pumps will therefore continue to operate allowing for a controlled shutdown of the plant utilizing normally available systems and equipment. The switchyard is not seismically qualified, but the probability the both NATs being damaged is very low.

Responses to the specific questions

The responses to the specific questions are provided below.

a) *Quantify any change to the critical damping ratio assumed in the analysis based on RCP coastdown considerations.*

   Response: No change is necessary. This response demonstrates that the analytic damping used in the faulted load analysis is conservative at least up to the time of full control rod insertion.

b) *Include considerations for both the unirradiated and irradiated cases. Additionally, provide the Rayleigh damping coefficients being used for the irradiated fuel assembly cases in the fuel assembly structural response analysis for the U.S. EPR.*

<u>Response:</u> The irradiated and un-irradiated conditions are addressed in the response. The analytic Rayleigh damping coefficients are [ ], as presented in reference 2. The values are calculated as:

[ ] **Equation 72-2**

The value of the first natural frequency, f_1, in un-irradiated condition [ ]

c) *Quantify the damping ratio margin (i.e. the difference between the critical damping ratio derived from test data and that credited in the analysis) change for both the unirradiated and irradiated fuel assembly cases.*

<u>Response:</u> The validity range of the damping model is assessed via a margin on flow to the fully inserted control rod state. For both the un-irradiated and irradiated condition, the analytic damping model is conservatively adequate for flow levels [ ] below the core flow at the time that the control rods reach the bottom of the core. Beyond this, the damping model retains additional margin compared to the test supported values.

d) *Address both SSE-only and combined SSE and LOCA loads analyses.*

<u>Response:</u> The SSE only and SSE+LOCA are addressed in ANP-10325P (reference 2). As demonstrated, both analyses have adequate conservatism in the selection of damping parameters to accommodate the effects of a reactor coolant pump coast-down.

References:

1. NRC, Regulatory Guide 1.166, Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-Earthquake Action, March 1997.
2. ANP-10325P, "U.S. EPR Fuel Assembly - Faulted Condition Analysis", May 2013.

Table 72-1— RCP Flow at Control Rod Insertion Position

[ ]

Figure 72-1— Complete Loss of Forced Reactor Coolant Flow Event - RCS Flow Rates

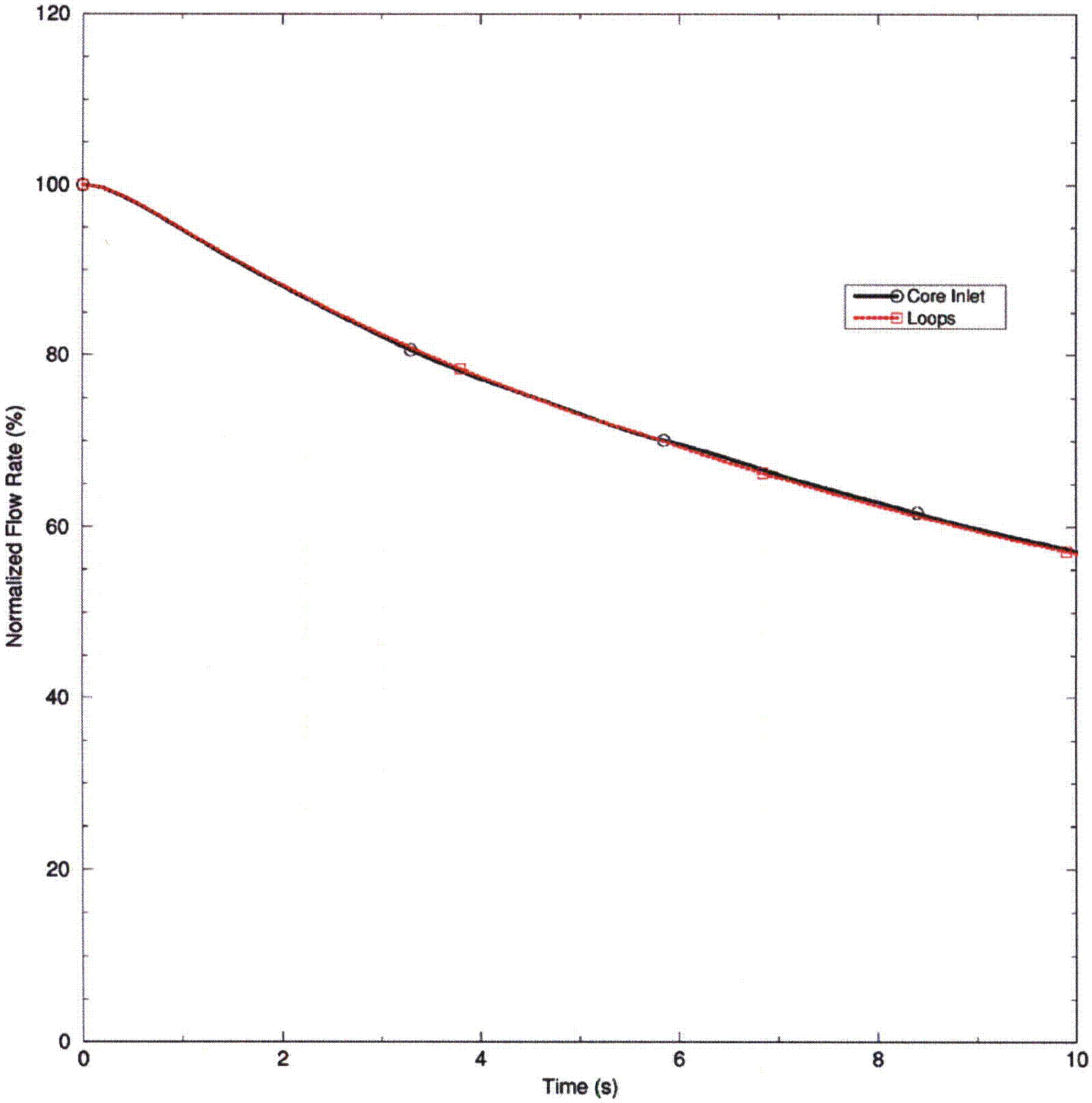

Figure 72-2: Complete Loss of Forced Reactor Coolant Flow Event - Pressurizer Pressure

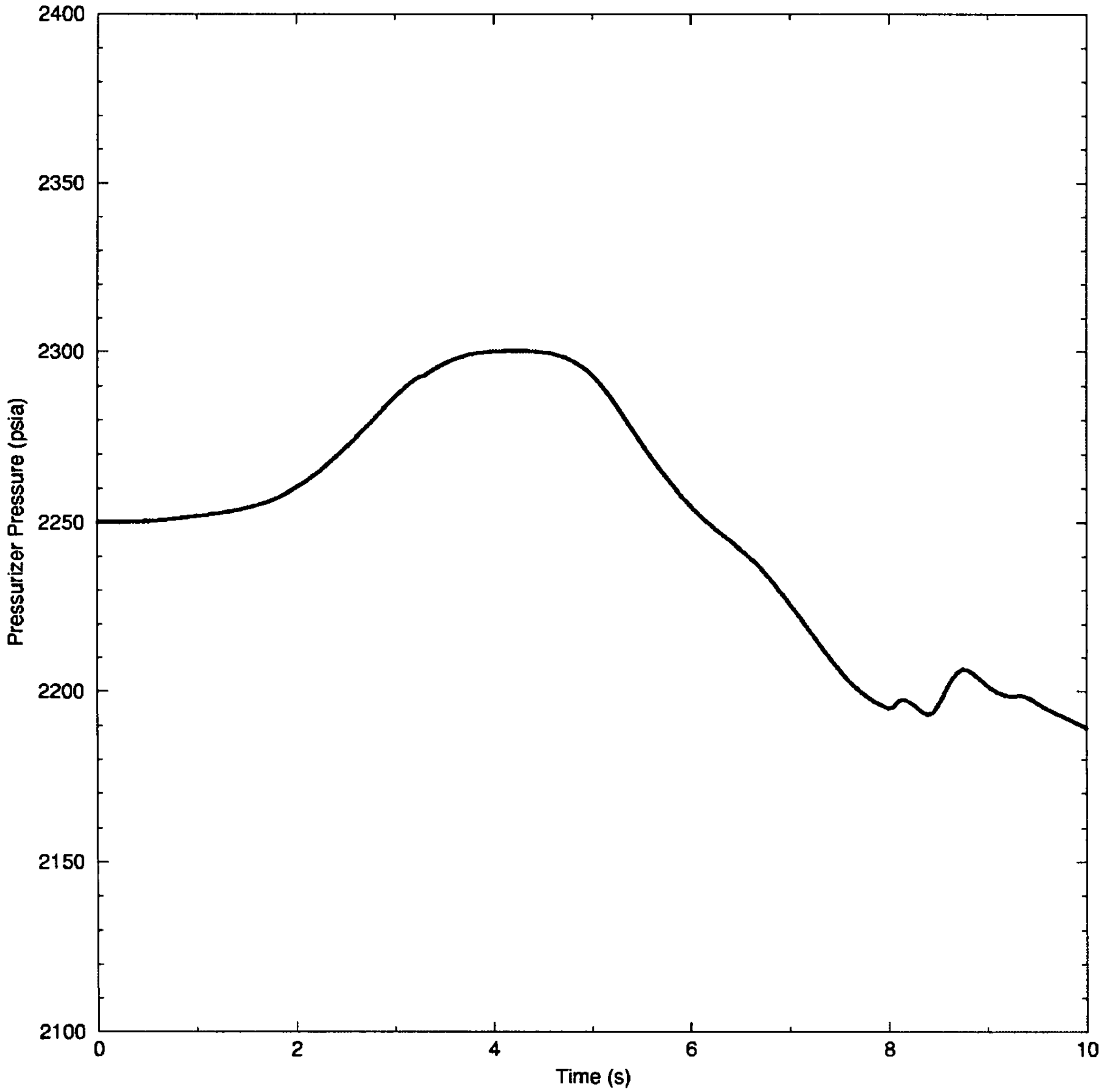

Figure 72-3: Un-Irradiated Damping Test Results

Figure 72-4: Simulated Irradiated Damping Test Results