SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400/RENEWED LICENSE NO. NPF-63
APPLICATION FOR REVISION TO TECHNICAL SPECIFICATION
CORE OPERATING LIMITS REPORT (COLR) REFERENCES

LOSS OF FORCED REACTOR COOLANT FLOW ANALYSIS FOR HARRIS NUCLEAR PLANT, UNIT 1

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Nature of Changes

Item	Page	Description and Justification
1.	All	Initial release.

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Nomenclature

BOC	Beginning-of-Cycle
CE	Combustion Engineering
CHF	Critical Heat Flux
DNB(R)	Departure from Nucleate Boiling (Ratio)
HTP	High Thermal Performance
HFP	Hot Full Power
LOCF	Loss of Forced Reactor Coolant Flow
MDNBR	Minimum Departure from Nucleate Boiling Ratio
MTC	Moderator Temperature Coefficient
PORV	Power Operated Relief Valve
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RPS	Reactor Protection System
RTP	Rated Thermal Power
SAFDL	Specified Acceptable Fuel Design Limit
TS	Technical Specification
W	Westinghouse

1.0 Introduction

The analysis documented herein describes a LOCF analysis for Harris Nuclear Plant using the S-RELAP5 computer code. This analysis demonstrates the application of the Reference 1 methodology to the Harris Nuclear Plant.

2.0 Conclusion

Based on the results of this analysis, margin exists to the DNB SAFDL. Because the core power does not increase appreciably during this event, the challenge to the fuel centerline melt SAFDL is not limiting. The pressurization transient does not present a severe challenge to the maximum pressure criterion since system temperatures and pressure increase less significantly for a loss of flow event compared to complete loss of load type events. Therefore, the event acceptance criteria are met.

3.0 Analytical Methodology

The analysis is performed using the approved Reference 1 methodology. The S-RELAP5 code is used to model the primary and secondary side systems of the Harris Nuclear Plant and to calculate reactor power, total reactivity and fluid conditions (such as coolant flow rates, core inlet temperatures, pressurizer pressure and level). The MDNBR for the event is calculated using the thermal-hydraulic conditions from the S-RELAP5 calculation as input to the XCOBRA-IIIC code (Reference 2) along with the HTP CHF correlation (Reference 3). The Harris Nuclear Plant core is composed solely of AREVA NP HTP fuel assemblies, thus mixed core considerations do not apply.

3.1 *Nodalization*

The plant configuration is represented by an S-RELAP5 model. The S-RELAP5 model nodalizes the primary and secondary sides into control volumes representing reasonable homogenous regions, interconnected by flow paths, or "junctions". The reactor vessel, RCS piping and steam generator nodalization diagrams are shown in Figures 3.1 to 3.3. The current analysis is based on a Harris Nuclear Plant specific model.

In general, the plant nodalization is defined to be consistent wherever possible for different plant types. Most of the differences existing between the model used for the current Harris Nuclear Plant analysis (W 3-loop plant) and that for the sample problem in Reference 1 (CE 2 x 4 plant) are attributed to plant specific differences. The RCS loop nodalization (including the steam generator primary side components) is different between W and CE plants to accommodate the different loop configurations. Namely, the loop configuration for the Harris Nuclear Plant application described herein consists of three individual loops each with one hot leg, a U-tube steam generator, a cold leg and a RCP. The Reference 1 sample problem, on the other hand, is based on a CE plant design which consists of two coolant loops each with one hot leg, a U-tube steam generator, two cold legs and two RCPs.

The vessel nodalization is very similar, differing most notably in the details of the downcomer and the lower downcomer and lower head flow paths to accommodate the flow skirt in the CE vessel (not present in the W design).

The steam generator secondary and steam line models are nodalized slightly different between the current model for Harris Nuclear Plant and the Reference 1 sample problem model, namely, the steam generator downcomer and boiler regions in the current model each contain one fewer nodes. Although the number of nodes decreased by one in each of these regions, the characteristics of the steam generator, specifically the volume distribution in the downcomer and the heat transfer to the boiler region, are more accurately captured. The overall effect of these changes on the analysis is negligible for this event.

Other plant specific differences include the number and location of the main steam safety valves, the geometry of the pressurizer surgeline and the pressurizer PORV design.

3.2 *Chosen Parameters*

The parameters and equipment states are chosen to provide a conservative estimate of the challenge to DNB. The biasing and assumptions for key input parameters are consistent with the approved Reference 1 methodology. The key assumptions are given in Table 3.1 and the biasing of key parameters is provided in Table 3.2. The process of defining the biasing and assumptions for key input parameters is consistent with the Reference 1 sample problem.

3.3 *Sensitivity Studies*

This event is controlled primarily by the primary system flow coast down. The S-RELAP5 code assessments in Reference 1 validate the model relative to this controlling parameter. Thus, no additional model sensitivity studies are needed for this application.

The biasing of input parameters is chosen to produce a conservative estimate of the challenge to DNB for this application. Thus, no additional input parameter sensitivity studies are needed.

3.4 *Definition of Event Analyzed and Bounding Input*

The event is analyzed from full power initial conditions since the margin to the DNB limit is minimized at the beginning of the event. The input parameter biasing and assumptions for this event, shown in Tables 3.1 and 3.2, are consistent with the approved methodology.

Table 3.1 Key Assumptions

Parameter	Assumption
Time of loss-of-offsite power	Offsite power is available
Mitigating systems	
• Low Primary Flow Trip	Available
• RCP Underfrequency Trip	Available
• Pressurizer Spray	Available
• Pressurizer PORVs	Available
Operator Actions	No operator actions credited
Single Failures	No single failure will adversely affect the consequences of this event
Number of Operating Loops	All loops are in operation consistent with HFP operation

Table 3.2 Key Input Parameters Biases

Parameter	Bias
Initial reactor core power (MWt)	Rated thermal power plus calorimetric uncertainty
Initial RCS vessel average temperature (°F)	Maximum TS value [ ] plus measurement and control deadband uncertainties [ ]
Initial RCS pressure (psia)	Nominal value [ ] minus measurement and control deadband uncertainties [ ]
Initial RCS flow rate (Mlbm/hr)	TS minimum accounting for measurement uncertainty
Scram reactivity (pcm)	Minimum HFP worth assuming the most reactive rod is stuck out of the core
Moderator temperature coefficient (pcm/°F)	Most positive TS value at HFP
Doppler reactivity coefficient (pcm/°F)	Nominal BOC [ ]
Pellet-to-clad gap conductance and fuel rod thermal properties (Btu/hr-ft^2-°F)	BOC
RCP Underfrequency RPS trip setpoint (Hz)	Nominal minus uncertainty
RCP Underfrequency RPS trip time delay (sec)	Maximum

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Figure 3.1 S-RELAP5 Reactor Vessel Nodalization

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Figure 3.2 S-RELAP5 Reactor Coolant System Nodalization (Loop 1)

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Figure 3.3 S-RELAP5 Steam Generator Secondary System and Steam Line Nodalization (Loop 1)

4.0 Complete Loss of Forced Reactor Coolant Flow (FSAR Event 15.3.2)

4.1 *Identification of Causes and Event Description*

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical power to all reactor coolant pumps. If the reactor is at power at the time of the event, the immediate effect of loss of forced reactor coolant flow is a rapid increase in the reactor coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not tripped promptly.

A reactor trip on reactor coolant pump underfrequency is provided to trip the reactor for an underfrequency condition, resulting from frequency disturbances on the power grid. If the maximum grid frequency decay rate is less than approximately 5 Hz/sec, this trip function will protect the core from underfrequency events without requiring tripping of the RCP breakers. A reactor trip on reactor coolant pump undervoltage is provided to protect against conditions which can cause a loss of voltage to all reactor coolant pumps, i.e., loss of offsite power.

The event initiated from Mode 1 conditions bounds other modes of operation. A sensitivity analysis that showed that the underfrequency event initiated from HFP with a TS MTC of 0 pcm/°F bounds an underfrequency event initiated from 70% RTP with a TS MTC of +5 pcm/°F. Due to a more rapid flow coastdown, an underfrequency initiating event bounds an undervoltage initiating event.

No single active failure will adversely affect the consequences of the event.

4.2 *Acceptance Criteria*

For Harris Nuclear Plant, the Complete Loss of Forced Reactor Coolant Flow event is classified as a Condition III event, but is analyzed against more restrictive Condition II acceptance criteria in order to bound other Condition II events, e.g., partial loss of coolant flow. For this event, the principally challenged Condition II criterion is:

> The fuel cladding integrity should be maintained by ensuring that fuel design limits are not exceeded. This is demonstrated by assuring that the minimum calculated departure from nucleate boiling ratio (DNBR) is not less than the applicable limits of the DNBR correlation being used and that fuel centerline melt does not occur.

The analysis documented herein demonstrates that the DNB SAFDL is met for this event. Fuel centerline melt is not challenged since there is no appreciable increase in core power. System overpressure is bounded by more challenging events.

4.3 *Analysis Results*

The results of the analysis indicate that the predicted MDNBR is greater than the safety limit. The critical heat flux correlation limit ensures that, with 95% probability and 95% confidence, DNB is not expected to occur; therefore, no fuel is expected to fail. The fuel centerline melt threshold is not penetrated during this event. Thus, Condition II acceptance criteria are met for this event.

Sequence of events and results for the underfrequency event are given in Table 4.1. The responses to key system variables are given in Figure 4.1 to Figure 4.7.

Table 4.1 Sequence of Events

Event	Time (sec)
Initiate transient (three-pump coastdown)	0.0
Reactor scram on underfrequency trip (begin rod insertion)	1.2
Compensated PORV opens	2.0
Uncompensated PORV opens	3.1
MDNBR occurs	3.3
Peak power-to-flow ratio	3.6
Peak core average coolant temperature	3.8

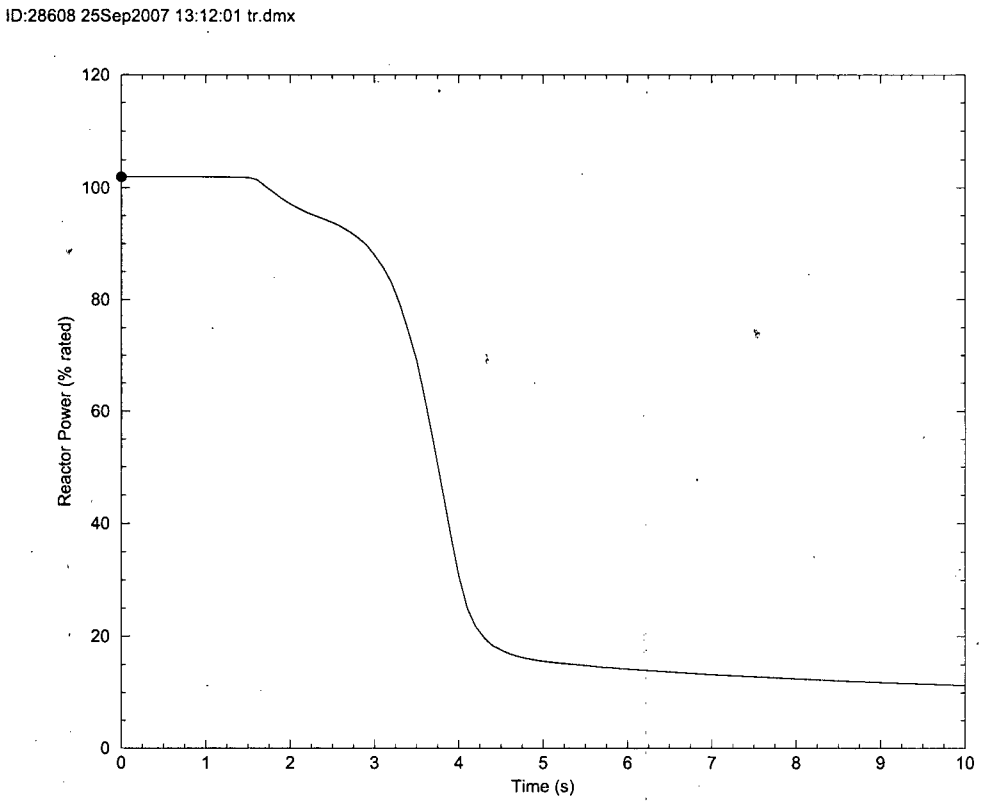

Figure 4.1 Reactor Power for Loss of Forced Reactor Coolant Flow – Underfrequency Case

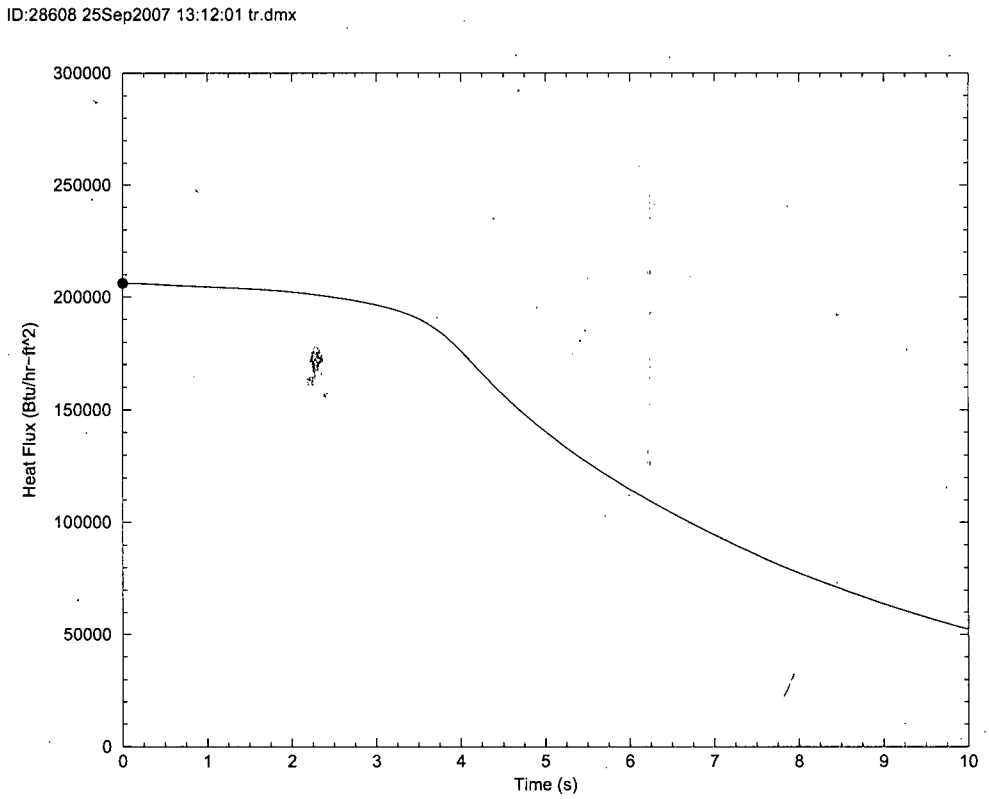

Figure 4.2 Core Average Heat Flux for Loss of Forced Reactor Coolant Flow – Underfrequency Case

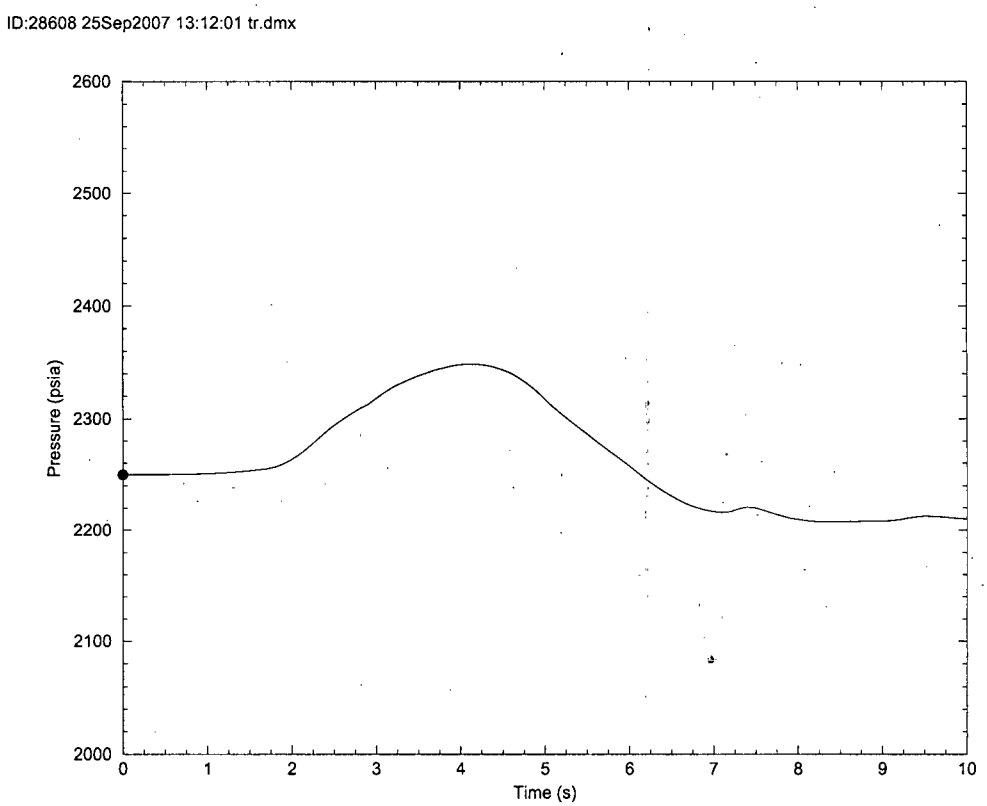

Figure 4.3 Pressurizer Pressure for Loss of Forced Reactor Coolant Flow – Underfrequency Case

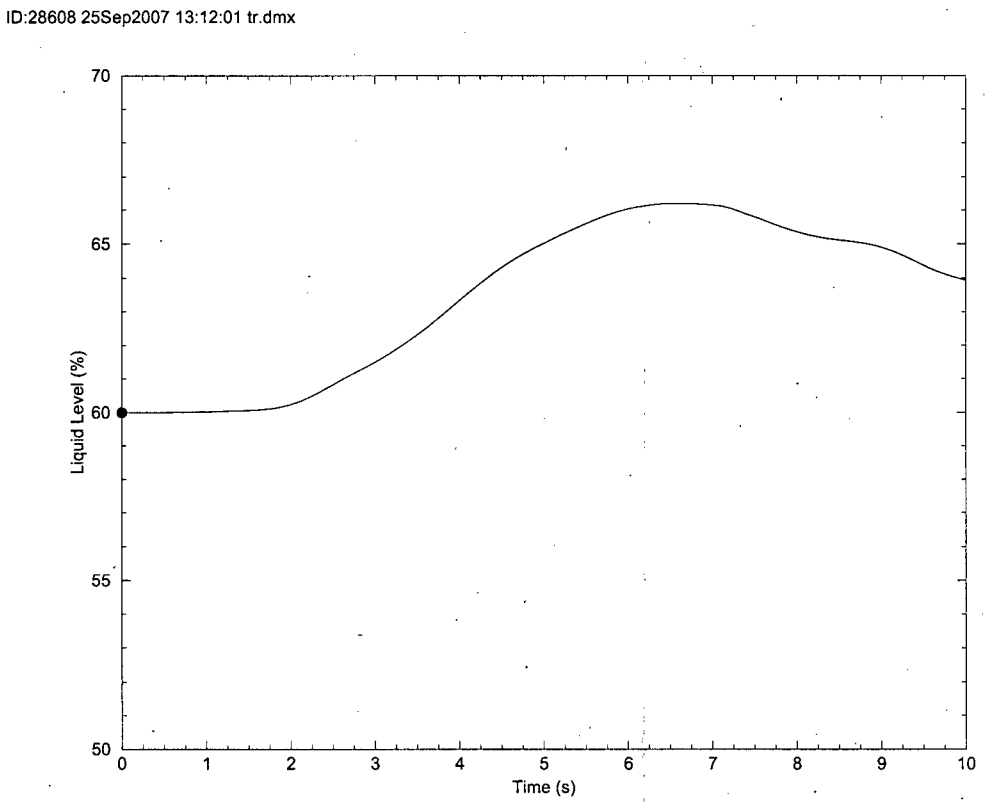

Figure 4.4 Pressurizer Level for Loss of Forced Reactor Coolant Flow – Underfrequency Case

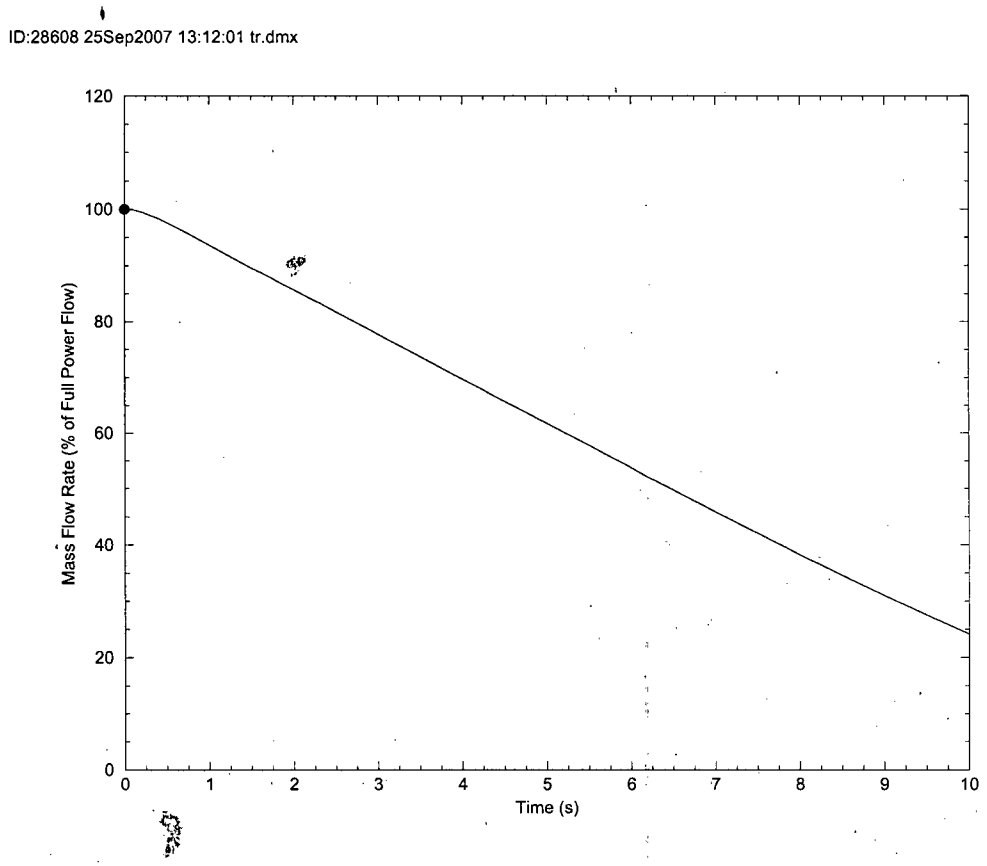

Figure 4.5 Reactor Coolant System Mass Flow Rate for Loss of Forced Reactor Coolant Flow – Underfrequency Case

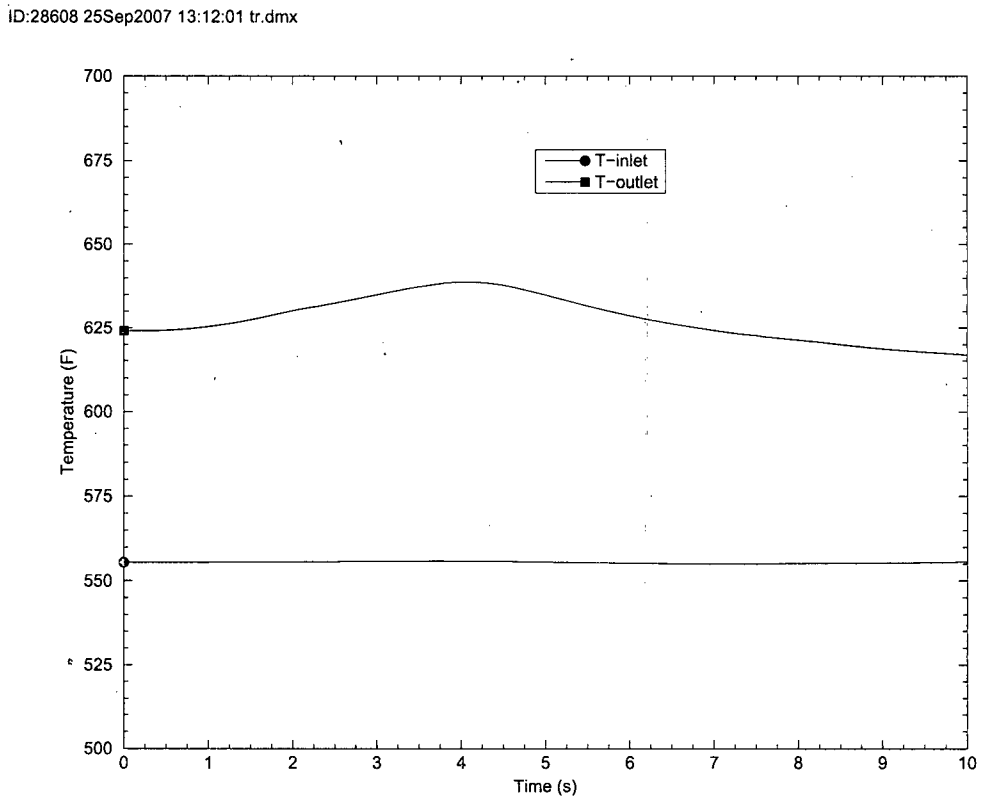

Figure 4.6 Core Inlet and Outlet Temperatures for Loss of Forced Reactor Coolant Flow – Underfrequency Case

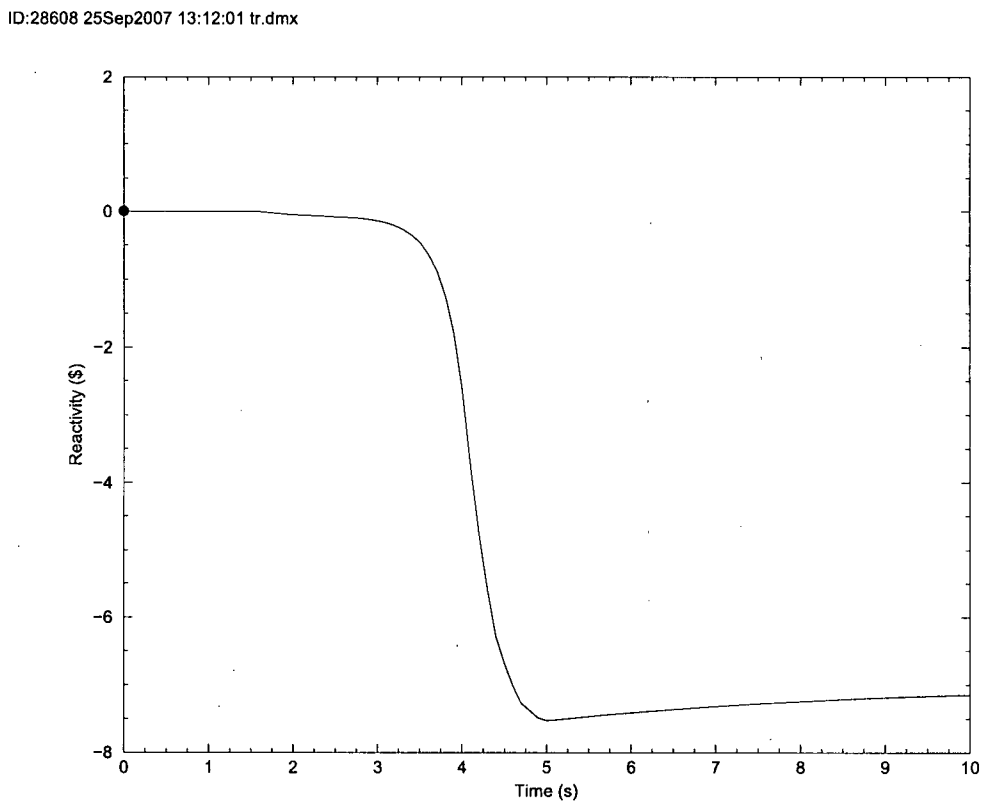

Figure 4.7 Total Core Reactivity for Loss of Forced Reactor Coolant Flow – Underfrequency Case

5.0 Reference

1. EMF-2310(P)(A) Revision 1, *SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors*, Framatome ANP, May 2004.

2. XN-NF-82-21(P)(A) Revision 1, *Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations*, Exxon Nuclear Company, September 1983.

3. EMF-92-153(P)(A) Revision 1, *HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel*, Siemens Power Corporation, January 2005.