Florida Power & Light, 9760 S.W. 344 St. Homestead, FL 33035

FPL.

AUG 25 2011
L-2011-334
10 CFR 50.90

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555-0001

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Response to NRC Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Reactor Systems Issues

References:

(1) M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2010-113), "License Amendment Request No. 205: Extended Power Uprate (EPU)," (TAC Nos. ME4907 and ME4908), Accession No. ML103560169, October 21, 2010.

(2) Email from J. Paige (NRC) to S. Hale (FPL), "Turkey Point EPU - Reactor Systems (SRXB) Request for Additional Information - Round 1.4 (Part 4)," Accession No. ML11202A174, July 21, 2011.

(3) M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2011-233), "Response to NRC Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Reactor Systems Issues," Accession No. ML11221A227, August 5, 2011.

(4) Email from J. Paige (NRC) to S. Hale (FPL), "Turkey Point EPU - Reactor Systems (SRXB) Request for Additional Information - Round 1.4 (Part 4)," Accession No. ML11213A247, July 29, 2011.

(5) Email from J. Paige (NRC) to S. Hale (FPL), "RE: DRAFT: Turkey Point EPU - Reactor Systems (SRXB) Requests for Additional Information - Round 1.3 (Part 3)," June 22, 2011.

(6) M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2011-305), "Response to NRC Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Reactor Systems Issues," August 19, 2011.

By letter L-2010-113 dated October 21, 2010 [Reference 1], Florida Power and Light Company (FPL) requested to amend Renewed Facility Operating Licenses DPR-31 and DPR-41 and revise the Turkey Point Units 3 and 4 Technical Specifications (TS). The proposed amendment will increase each unit's licensed core power level from 2300 megawatts thermal (MWt) to 2644 MWt and revise the Renewed Facility Operating Licenses and TS to support operation at this increased core thermal power level. This represents an approximate increase of 15% and is therefore considered an extended power uprate (EPU).

By email from the NRC Project Manager (PM) dated July 21, 2011 [Reference 2], additional information regarding reactor safety analysis issues was requested by the NRC staff in the Reactor Systems Branch (SRXB) to support the review of the EPU LAR [Reference 1]. The RAI consisted of thirty-nine (39) questions regarding loss-of-coolant accident (LOCA) and non-LOCA analyses. On August 5, 2011, FPL provided its response to RAI questions SRXB-1.3.1-1.3.6 and 1.3.16-1.3.38 via FPL letter L-2011-233 [Reference 3] in which it was stated that the response to RAI questions SRXB-1.3.7-1.3.15 on steam line breaks would follow under a separate correspondence.

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By email from the NRC PM dated July 29, 2011, FPL received three (3) additional RAI question on Anticipated Transients without Scram (ATWS) Events [Reference 4]. At that time, it was noted that the final RAI version had omitted portions of the RAI questions on LR Section 2.8.5.1.2 that had appeared on an earlier draft, dated June 22, 2011 [Reference 5], affecting RAI question SRXB-1.3.8 and an unnumbered question on minor steam line breaks <1.4ft^2. FPL included the omitted portions of RAI question SRXB-1.3.8 in the response provided on August 19, 2011 to RAI questions SRXB-1.3.7-1.3.15 on steam line breaks and 1.4.1-1.4.3 on ATWS via FPL letter L-2011-305 [Reference 6] and indicated the response to the unnumbered RAI question on minor steam line breaks (<1.4 ft^2) would be provided under a separate correspondence.

FPL's response to the unnumbered SRXB RAI question on minor steam line breaks (<1.4 ft^2) is provided in the Attachment to this letter.

In accordance with 10 CFR 50.91(b)(1), a copy of this letter is being forwarded to the State Designee of Florida.

This submittal does not alter the significant hazards consideration or environmental assessment previously submitted by FPL letter L-2010-113 [Reference 1].

This submittal contains no new commitments and no revisions to existing commitments.

Should you have any questions regarding this submittal, please contact Mr. Robert J. Tomonto, Licensing Manager, at (305) 246-7327.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on August 25, 2011.

Very truly yours,

Michael Kiley
Site Vice President
Turkey Point Nuclear Plant

Attachment

cc: USNRC Regional Administrator, Region II
USNRC Project Manager, Turkey Point Nuclear Plant
USNRC Resident Inspector, Turkey Point Nuclear Plant
Mr. W. A. Passetti, Florida Department of Health

Turkey Point Units 3 and 4

RESPONSE TO NRC RAI REGARDING EPU LAR NO. 205
AND SRXB REACTOR SYSTEMS ISSUES

ATTACHMENT

Response to Request for Additional Information

The following information is provided by Florida Power and Light Company (FPL) in response to the U. S. Nuclear Regulatory Commission's (NRC) Request for Additional Information (RAI). This information was requested to support License Amendment Request (LAR) 205, Extended Power Uprate (EPU), for Turkey Point Nuclear Plant (PTN) Units 3 and 4 that was submitted to the NRC by FPL via letter (L-2010-113) dated October 21, 2010 [Reference 1].

By email from the NRC Project Manager (PM) dated July 21, 2011 [Reference 2], additional information regarding reactor safety analysis issues was requested by the NRC staff in the Reactor Systems Branch (SRXB) to support the review of the EPU LAR [Reference 1]. The RAI consisted of thirty-nine (39) questions regarding loss-of-coolant accident (LOCA) and non-LOCA analyses. On August 5, 2011, FPL provided its response to RAI questions SRXB-1.3.1-1.3.6 and 1.3.16-1.3.38 via FPL letter L-2011-233 [Reference 3] in which it was stated that the response to RAI questions SRXB-1.3.7-1.3.15 on steam line breaks inside and outside containment would follow under a separate correspondence.

By email from the NRC PM dated July 29, 2011, FPL received three (3) additional RAI question on Anticipated Transients without Scram (ATWS) Events [Reference 4]. It was noted at that time that the final RAI version had omitted portions of the RAI questions on LR Section 2.8.5.1.2 that had appeared on an earlier draft, dated June 22, 2011 [Reference 5 / Attachment 2], affecting RAI question SRXB-1.3.8 and an unnumbered question on minor steam line breaks < $1.4 ft^2$. FPL included the omitted portions of RAI question SRXB-1.3.8 in the response provided on August 19, 2011, to RAI questions SRXB-1.3.7-1.3.15 on steam line breaks and 1.4.1-1.4.3 on ATWS via FPL letter L-2011-305 [Reference 6] and indicated that the response to the unnumbered question on minor steam line breaks (< $1.4 ft^2$) would be provided under a separate correspondence.

FPL's response to the unnumbered SRXB RAI question on minor steam line breaks (< $1.4 ft^2$) is provided below.

2.8.5.1.2 Steam System Piping Failures Inside and Outside Containment

2.8.5.1.2J Minor steam line breaks (< 1.4 ft2) are said to be bounded by the major steam line break. Show that there is no minor break, larger than a credible break; but too small to cause steam line isolation, that is not bounded by the major steam line break.

A spectrum break sizes between a main steamline depressurization (i.e., credible steamline break) and a major rupture of a steam pipe (i.e., double-ended rupture) were analyzed to determine the maximum break size that did not result in a steamline isolation signal. The results of these analyses showed that a 0.9 ft^2 split break on the main steamline was the maximum break size that did not result in a steamline isolation signal.

In the analysis of a major rupture of a steam pipe presented in LR §2.8.5.1.2.2.1.2, no credit was taken for the addition of boron to the core from the accumulators. This conservative assumption was modeled to assure a more limiting and bounding plant response following a double-ended rupture. However, for the analyses of the intermediate break sizes, the effect of accumulator boron should also be considered. Specifically, with no steamline isolation, the primary-side

depressurization will be more symmetric and slightly more severe, such that the accumulators would always be expected to actuate in these cases. As such, an additional analysis of the spectrum of the intermediate break sizes with minimum accumulator boron credited was performed. The results of these analyses again showed that a 0.9 ft^2 split break on the main steamline was the maximum break size that resulted in no steamline isolation.

A detailed calculation of the minimum DNBR was performed for each of the 0.9 ft^2 split break analysis cases of without and with accumulator boron credited. For comparison purposes, a detailed calculation of the minimum DNBR was also performed for a double-ended rupture case, similar to that presented in LR §2.8.5.1.2.2.1.2, but with accumulator boron credited. As shown in Table SRXB-LR 2.8.5.1.2J-1, it was determined that the minimum DNBR values calculated for the cases with the intermediate break size were less limiting than the minimum DNBR value calculated in the analysis of a major rupture of a steam pipe presented in LR §2.8.5.1.2.2.1.2 (i.e., Case 1: double-ended rupture, no accumulator boron credited).

Table SRXB-LR 2.8.5.1.2J-1: Comparison of Minimum DNBR Values to Major Rupture of a Steam Pipe Presented in LR §2.8.5.1.2.2.1.2

Case	Break Type (Size)	Accumulator Boron Credited	Change in Minimum DNBR [(1)]
1	Double-Ended Rupture	No	N/A
2	Double-Ended Rupture	Yes	+6.9%
3	Split Break (0.9 ft^2)	No	+9.0%
4	Split Break (0.9 ft^2)	Yes	+78.2%

(1) The change in the minimum DNBR from the case presented in LR §2.8.5.1.2.2.1.2 is provided for each of the analysis cases examined.

A comparison of the sequence of events for each of the analysis cases examined is provided in Table SRXB-LR 2.8.5.1.2J-2; the transient plots for the analysis of a 0.9 ft^2 split break on the main steamline with minimum accumulator boron credited are provided in Figures SRXB-LR 2.8.5.1.2J-1 through SRXB-LR 2.8.5.1.2J-8.

In summary, based on the results of the analyses performed with the intermediate break sizes, it was shown that the plant response to a major rupture of a steam pipe (i.e., double-ended rupture), as presented in LR §2.8.5.1.2.2.1.2, continues to be the bounding overall plant response for any postulated steamline break size.

Table SRXB-LR 2.8.5.1.2J-2: Sequence of Events
Steam System Piping Failures at Hot Zero Power with Offsite Power Available

	Time (sec)			
Event	**Case 1**	**Case 2**	**Case 3**	**Case 4**
Main steam line ruptures in loop 1 (double-ended rupture or split break)	0.0	0.0	0.0	0.0
High steam line flow setpoint reached in loop 1	0.01	0.01	N/A	N/A
High steam line flow setpoint reached in loop 2	0.36	0.36	N/A	N/A
High steam line flow setpoint reached in loop 3	0.37	0.37	N/A	N/A
Low SG pressure SI setpoint reached in loop 1 (1)	0.37	0.37	2.65	2.65
Low SG pressure SI setpoint reached in loops 2 and 3 (1)	1.05	1.05	3.24	3.24
High steam line flow/Low SG pressure SI setpoint is reached	1.06	1.06	N/A	N/A
Low pressurizer pressure SI setpoint is reached	N/A	N/A	23.69	23.69
SI actuation occurs	3.06	3.06	25.69	25.69
Main feedwater isolation completed in both intact loops	12.06	12.06	34.69	34.69
Steam line isolation completed in all three loops	18.96	18.96	N/A	N/A
Accumulators begin to inject	See below		45.25	45.25
Borated water from the accumulators reaches the core	N/A	Note (2)	N/A	45.75
SI pumps achieve full speed	24.06	24.06	46.69	46.69
SI flow injection begins (cold leg pressure falls below SI pump shutoff pressure)	24.25	24.25	46.75	46.75
Main feedwater isolation completed in the faulted loop	33.06	33.06	55.69	55.69
Criticality attained	39.25	39.25	60.0	62.0
Borated water from the SIS reaches the core	45.25	45.25	66.75	Note (2)

Accumulators begin to inject	100.0	100.0	See above	
Peak core heat flux occurs	149.5	103.75	164.0	115.5
Reactor becomes subcritical	186.25	134.0	204.5	136.75
Peak core heat flux (fraction of nominal)	0.134	0.114	0.129	0.0595

(1) This function operates on a lead/lagged steam pressure signal consisting of a 50-second lead and a 5-second lag. The lead/lag steam pressure signal responds quickly such that the Low SG Pressure setpoint is reached much sooner than the actual steam generator pressure.

(2) In Case 2, the accumulators begin to inject and deliver boron after borated water from the SIS begins to reach the core; as such, the borated water from the SIS reaches the core first in this case. In Case 4, the accumulators begin to inject and deliver boron to the core prior to SI flow injection; as such, the borated water from the accumulators reaches the core first in this case.

Figure SRXB-LR 2.8.5.1.2J-1
Steam System Piping Failure at Hot Zero Power – 0.9 ft^2 Split Break with Offsite Power Available and with Accumulator Boron Credited
Nuclear Power and Core Heat Flux vs. Time

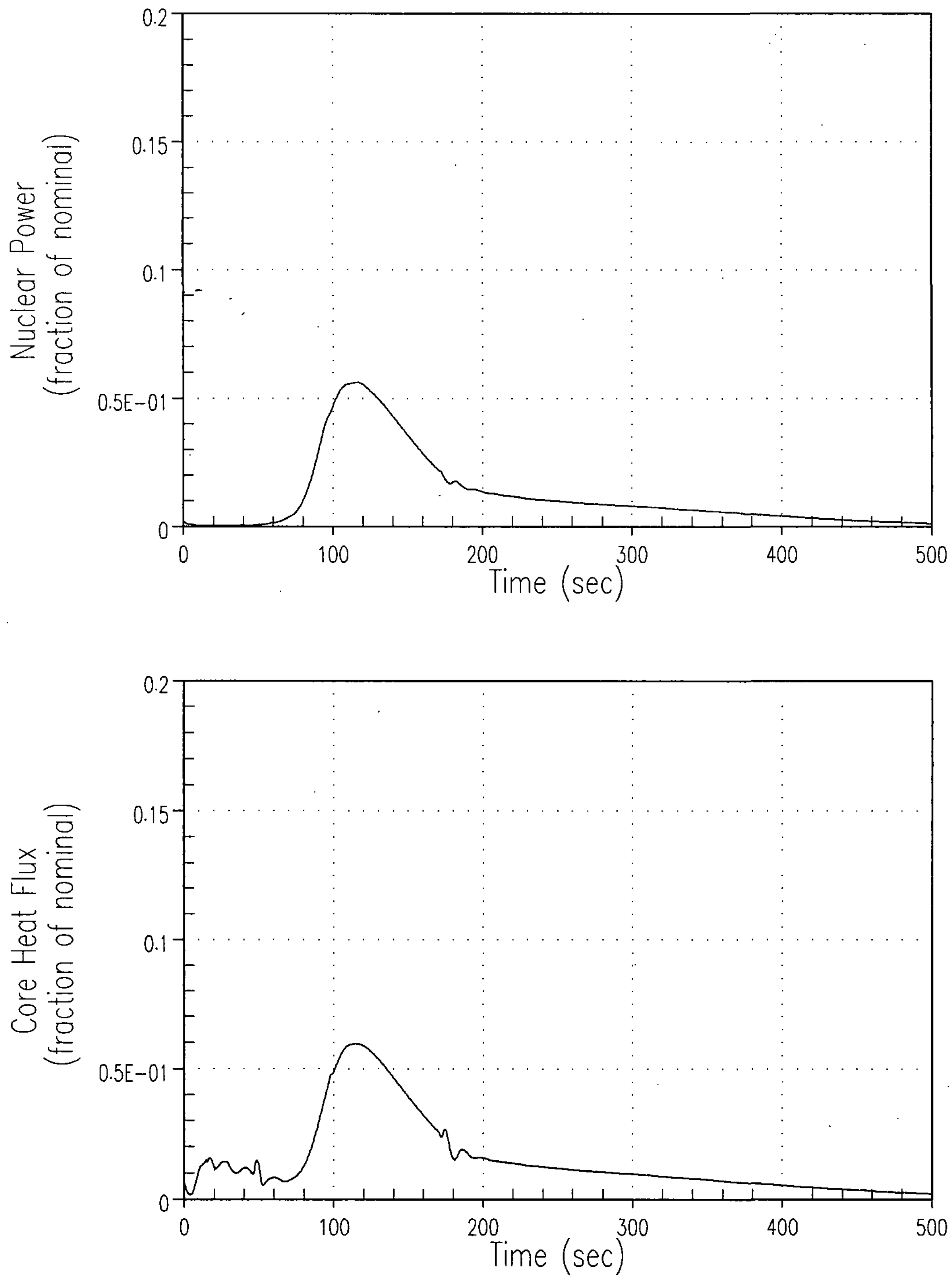
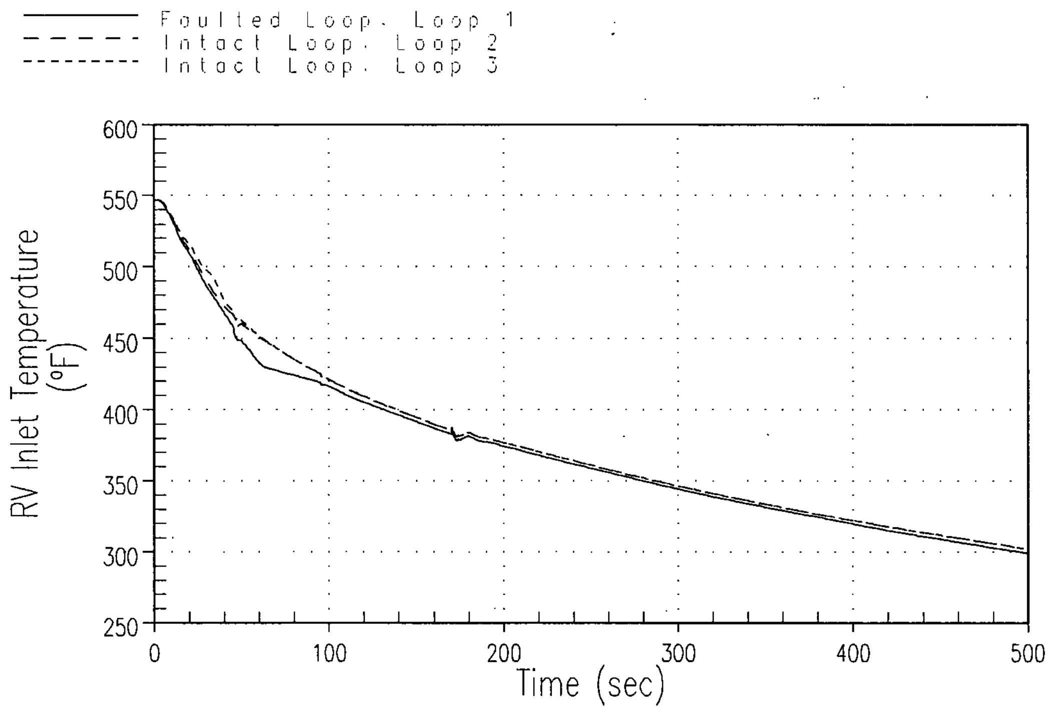


Figure SRXB-LR 2.8.5.1.2J-2
Steam System Piping Failure at Hot Zero Power – 0.9 ft^2 Split Break with Offsite Power Available and with Accumulator Boron Credited Reactor Vessel Inlet and Core Average Temperatures vs. Time



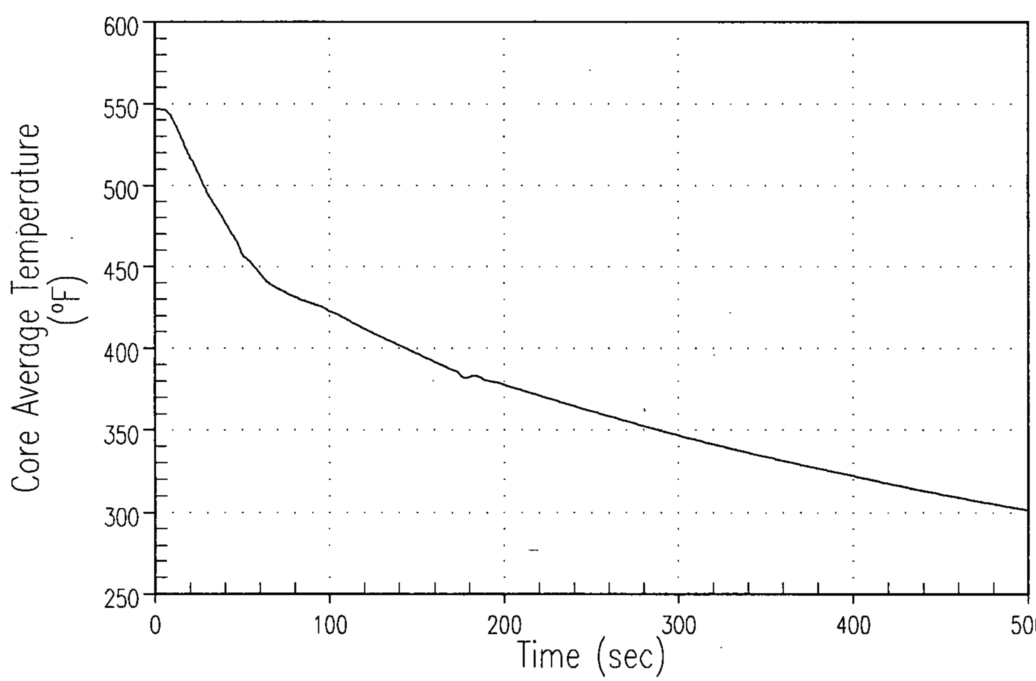
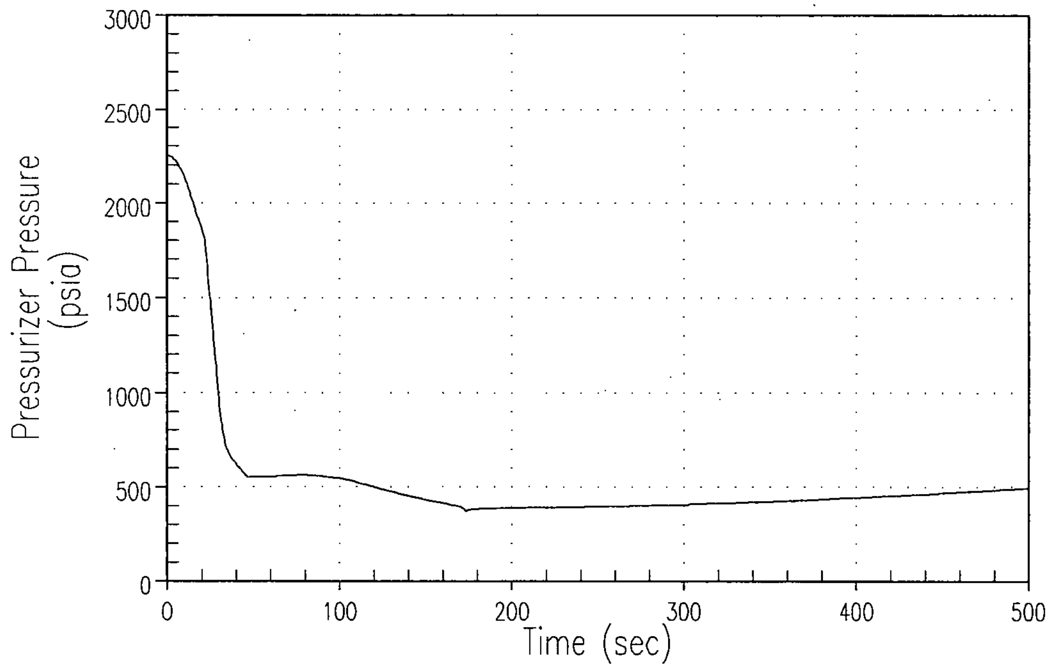


Figure SRXB-LR 2.8.5.1.2J-3
Steam System Piping Failure at Hot Zero Power – 0.9 ft^2 Split Break with Offsite Power Available and with Accumulator Boron Credited Pressurizer Pressure and Pressurizer Water Volume vs. Time



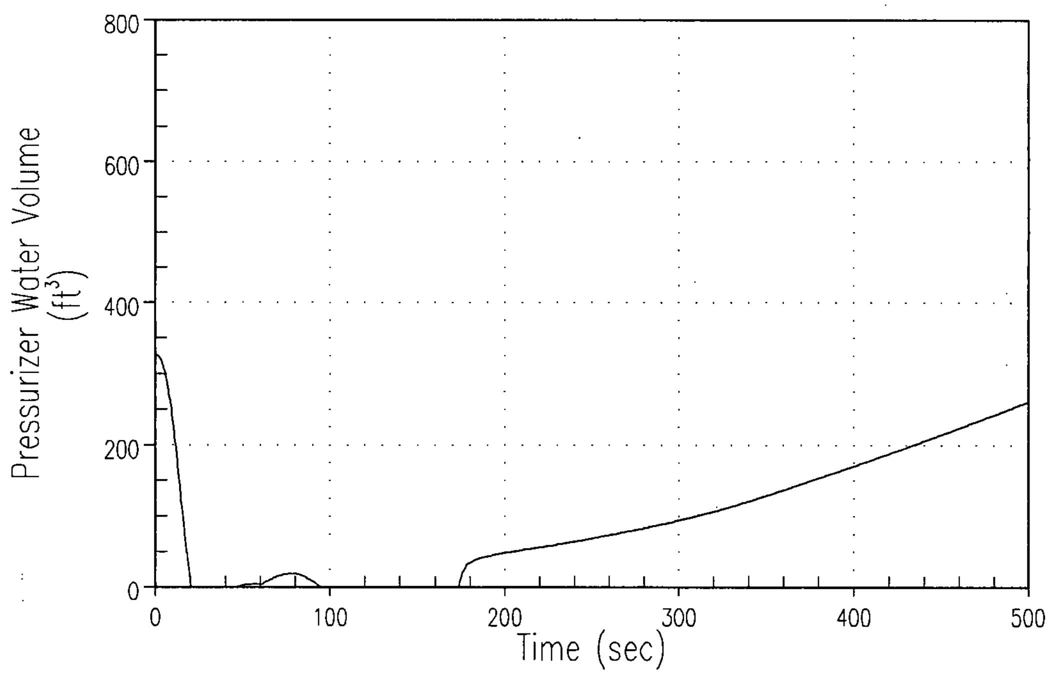


Figure SRXB-LR 2.8.5.1.2J-4
Steam System Piping Failure at Hot Zero Power – 0.9 ft^2 Split Break with Offsite Power Available and with Accumulator Boron Credited
Core Boron Concentration and Reactivity vs. Time

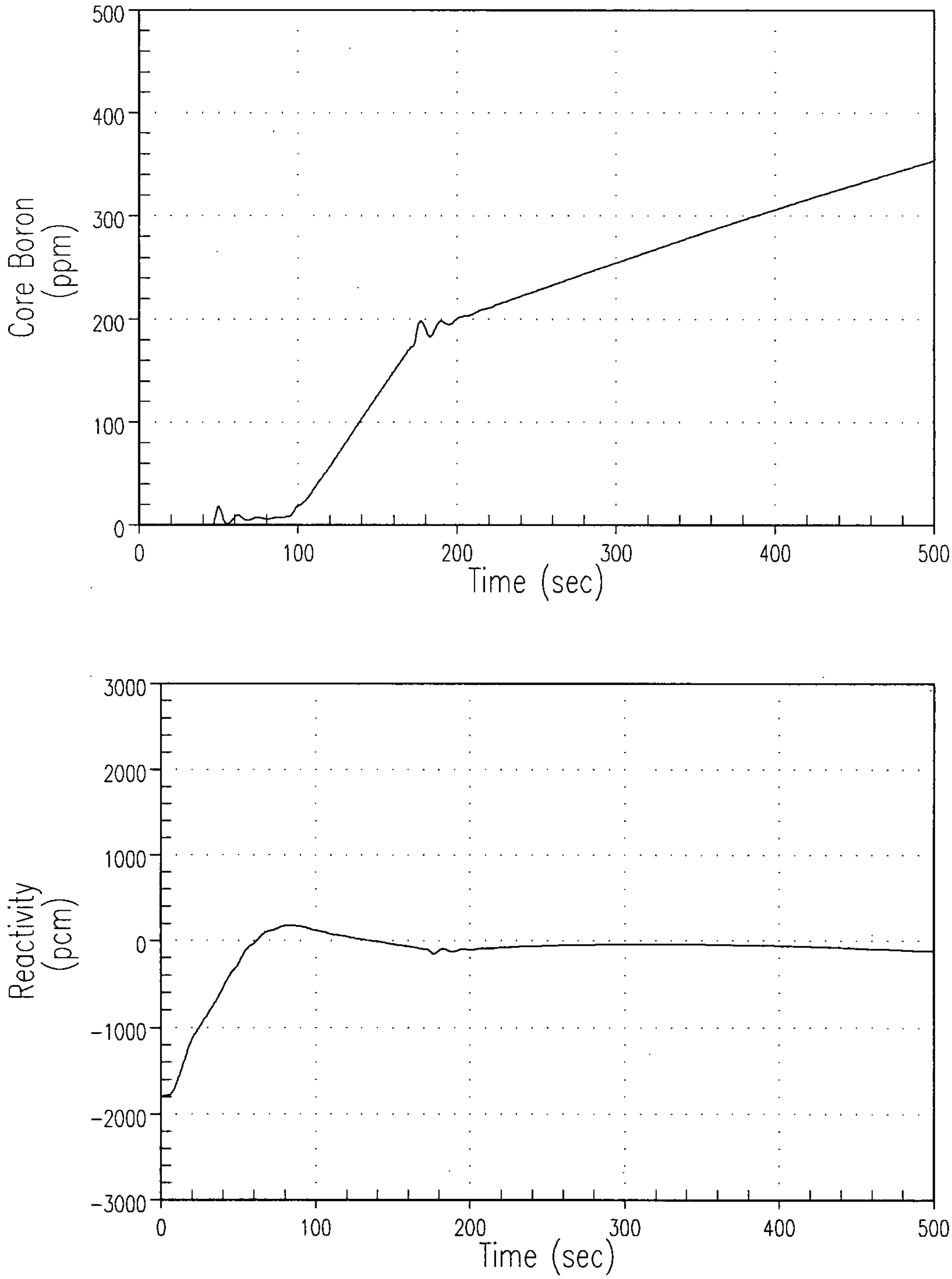


Figure SRXB-LR 2.8.5.1.2J-5
Steam System Piping Failure at Hot Zero Power – 0.9 ft^2 Split Break with Offsite Power Available and with Accumulator Boron Credited Steam Pressure and Steamline Break Flow vs. Time

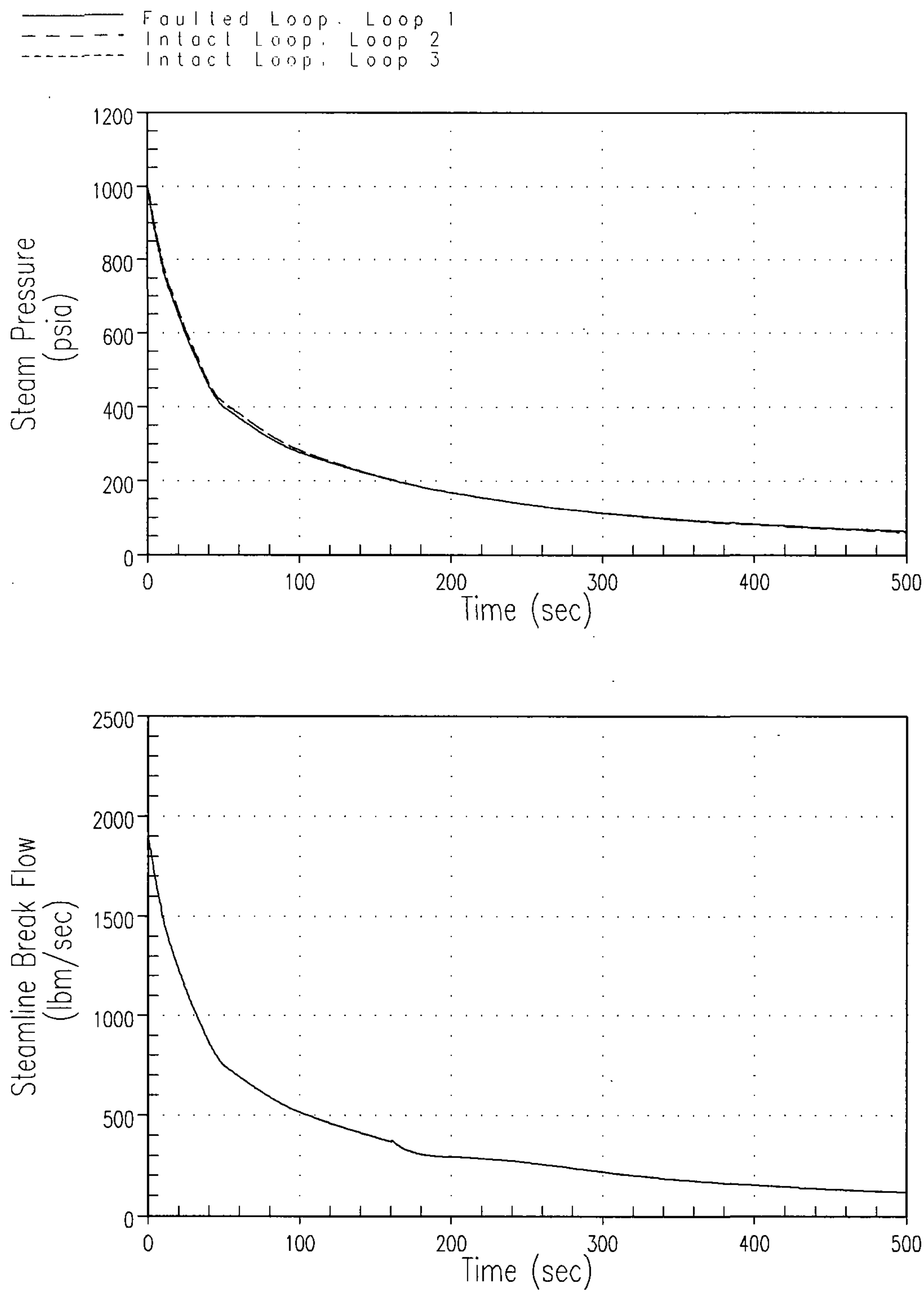


Figure SRXB-LR 2.8.5.1.2J-6
Steam System Piping Failure at Hot Zero Power – 0.9 ft^2 Split Break with Offsite Power Available and with Accumulator Boron Credited Feedwater Flow and Core Flow vs. Time

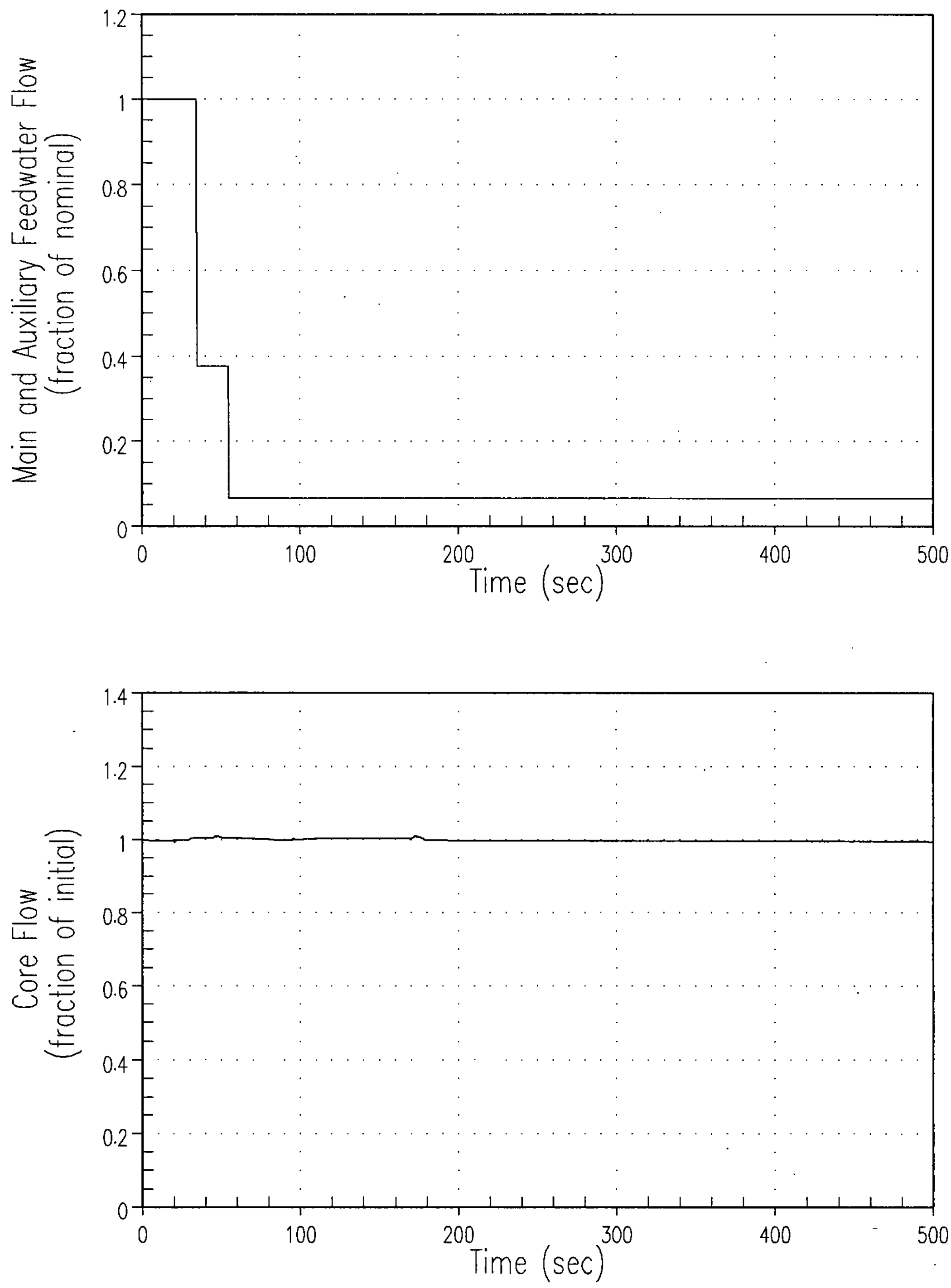
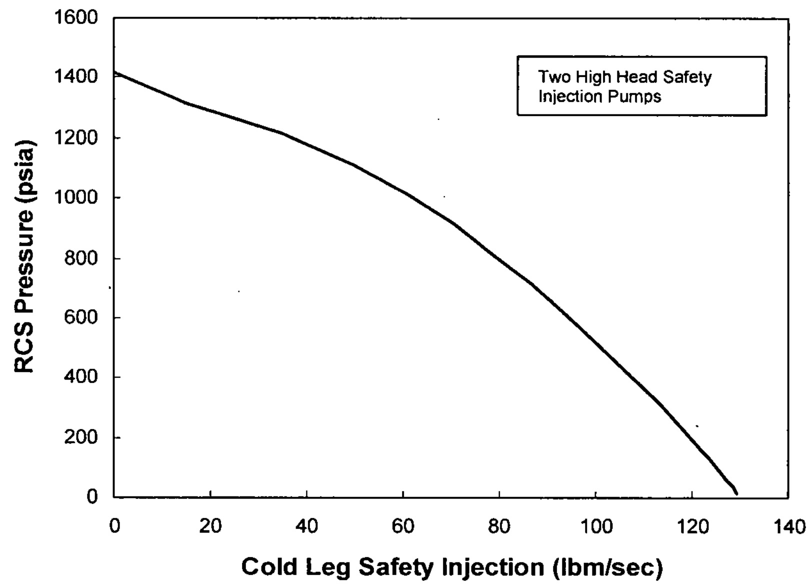
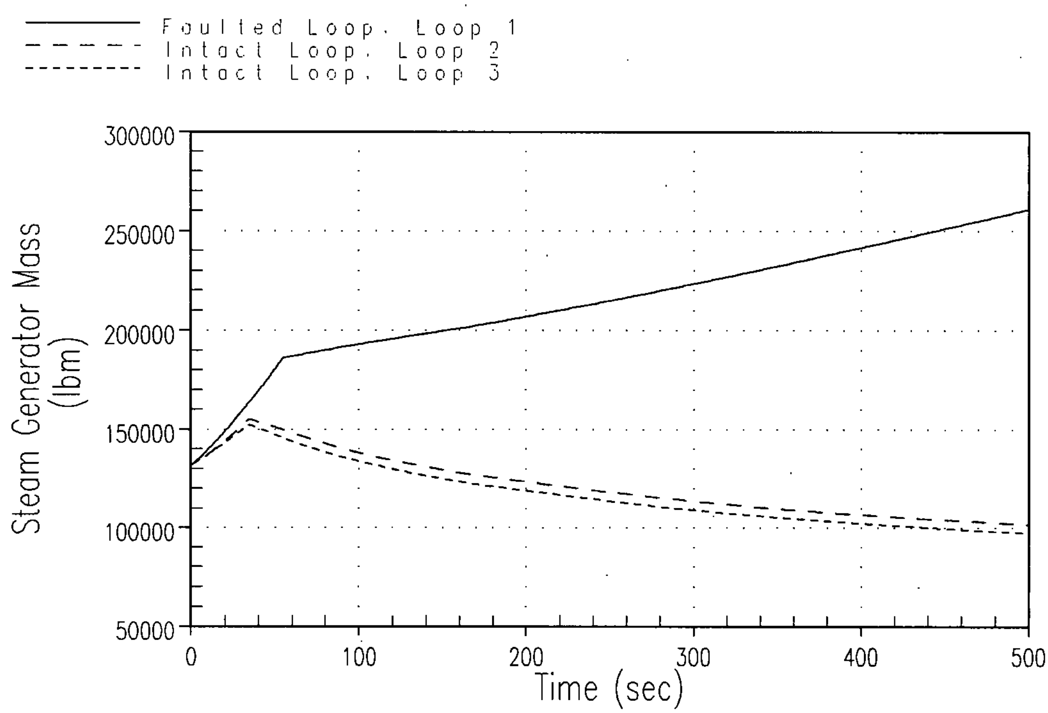


Figure SRXB-LR 2.8.5.1.2J-7
Steam System Piping Failure at Hot Zero Power – 0.9 ft² Split Break
with Offsite Power Available and with Accumulator Boron Credited
Safety Injection Flow vs. RCS Pressure [1] and Steam Generator Mass vs. Time





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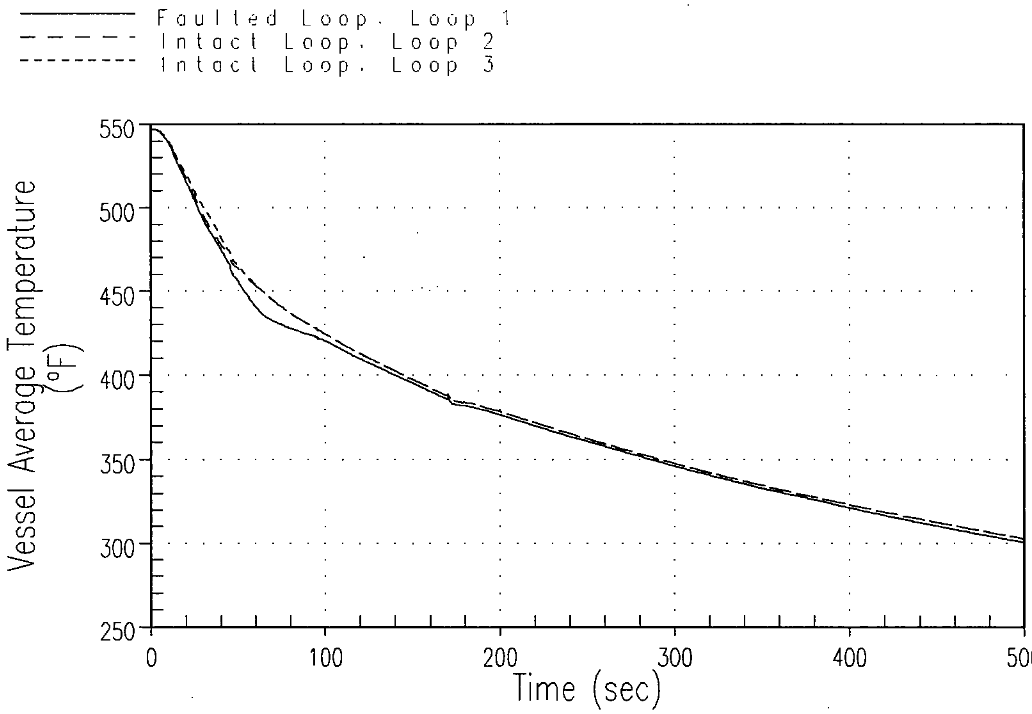
[1] This safety injection flow versus RCS pressure curve is the same as that presented in LR §2.8.5.1.2.2.1.2.

Figure SRXB-LR 2.8.5.1.2J-8
Steam System Piping Failure at Hot Zero Power – 0.9 ft^2 Split Break with Offsite Power Available and with Accumulator Boron Credited
Reactor Vessel Average Temperature vs. Time



References

1. M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2010-113), "License Amendment Request No. 205: Extended Power Uprate (EPU)," (TAC Nos. ME4907 and ME4908), Accession No. ML103560169, October 21, 2010.
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6. M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2011-305), "Response to NRC Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Reactor Systems Issues," August 19, 2011.