Entergy

Entergy Nuclear Vermont Yankee, LLC
Entergy Nuclear Operations, Inc.
185 Old Ferry Road
Brattleboro, VT 05302-0500

April 17, 2003
BVY 03-39

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: **Vermont Yankee Nuclear Power Station**
License No. DPR-28 (Docket No. 50-271)
Technical Specification Proposed Change No. 257
Implementation of ARTS/MELLLA at Vermont Yankee – Supplement No. 1

In response to the April 17, 2003 telephone request from the US NRC Project Manager, Vermont Yankee is providing supporting information for the proposed change. Attachment 1 to this supplemental letter provides a copy of Reference 1, GNF 0000-0006-1823-SRLR, "Supplemental Reload Licensing Report for Vermont Yankee Nuclear Power Station, Reload 22, Cycle 23," Revision 0, dated October 2002.

If you have any questions concerning this transmittal, please contact Ronda Daflucas at (802) 258-4232.

Sincerely,

Michael A. Balduzzi

Michael A. Balduzzi
Vice President, Operations

Attachment

cc: USNRC Region 1 Administrator
USNRC Resident Inspector – VYNPS
USNRC Project Manager – VYNPS
Vermont Department of Public Service

A001

STATE OF VERMONT)
)ss
WINDHAM COUNTY)

Then personally appeared before me, Michael A. Balduzzi, who, being duly sworn, did state that he is Vice President, Operations of Vermont Yankee Nuclear Power Station, that he is duly authorized to execute and file the foregoing document in the name and on the behalf of Vermont Yankee Nuclear Power Station, and that the statements therein are true to the best of his knowledge and belief.

T.B. Silko

Thomas B. Silko, Notary Public
My Commission Expires February 10, 2007

SUMMARY OF VERMONT YANKEE COMMITMENTS

BVY NO.: 03-39, "Technical Specification Proposed Change No. 257 Implementation of ARTS/MELLLA at Vermont Yankee – Supplement No. 1"

The following table identifies commitments made in this document by Vermont Yankee. Any other actions discussed in the submittal represent intended or planned actions by Vermont Yankee. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Licensing Manager of any questions regarding this document or any associated commitments.

COMMITMENT	COMMITTED DATE OR "OUTAGE"
None	N/A

Attachment 1

Vermont Yankee Nuclear Power Station

Technical Specification Proposed Change No. 257

Implementation of ARTS/MELLLA at Vermont Yankee – Supplement No. 1

GNF 0000-0006-1823-SRLR, Rev. 0, October 2002

Global Nuclear Fuel
A Joint Venture of GE, Toshiba, & Hitachi

0000-0006-1823-SRLR
Revision 0
Class I
October 2002

0000-0006-1823-SRLR, Rev. 0
Supplemental Reload Licensing Report
for
Vermont Yankee Nuclear Power Station
Reload 22 Cycle 23

Approved: JGM Andersen for.
G.A. Watford, Manager
Fuel Engineering Services

Approved: Rick Kingston
R.E. Kingston
Customer Account Leader

Important Notice Regarding

Contents of This Report

Please Read Carefully

This report was prepared by Global Nuclear Fuel - Americas, LLC (GNF) solely for Entergy Nuclear Vermont Yankee (ENVY) for ENVY's use with the U.S. Nuclear Regulatory Commission (USNRC) to amend ENVY's operating license for the Vermont Yankee Nuclear Power Station. The information contained in this report is believed by GNF to be an accurate and true representation of the facts known, obtained or provided to GNF at the time this report was prepared.

The only undertakings of GNF respecting information in this document are contained in the contract between ENVY and GNF for fuel bundle fabrication and services for Vermont Yankee Nuclear Power Station and nothing contained in this document shall be construed as changing said contract. The use of this information except as defined by said contract, or for any purpose other than that for which it is intended, is not authorized; and with respect to any such unauthorized use, neither GNF nor any of the contributors to this document makes any representation or warranty (expressed or implied) as to the completeness, accuracy or usefulness of the information contained in this document or that such use of such information may not infringe privately owned rights; nor do they assume any responsibility for liability or damage of any kind which may result from such use of such information.

Acknowledgement

The engineering and reload licensing analyses, which form the technical basis of this Supplemental Reload Licensing Report, were performed by "Fuel Engineering Services" and "Nuclear and Safety Analysis" personnel. The Supplemental Reload Licensing Report was prepared by G.M. Baka. This document has been verified by E.W. Gibbs.

The basis for this report is *General Electric Standard Application for Reactor Fuel*, NEDE-24011-P-A-14, June 2000; and the U.S. Supplement, NEDE-24011-P-A-14-US, June 2000.

1. Plant-unique Items

Appendix A: Analysis Conditions
Appendix B: Decrease in Core Coolant Temperature Events
Appendix C: Introduction of GE14 Fuel
Appendix D: Stability Solution Option I-D Exclusion and Buffer Regions
Appendix E: List of Acronyms

2. Reload Fuel Bundles

Fuel Type	Cycle Loaded	Number
Irradiated:		
GE9B-P8DWB335-10GZ-80M-150-T (GE8x8NB)	18	24
GE13-P9HTB380-12GZ-100T-146-T (GE13)	20	72
GE13-P9HTB379-13GZ-100T-146-T (GE13)	20	40
GE13-P9DTB386-11G4.0/1G3.0-100T-146-T-2425 (GE13)	22	88
GE13-P9DTB225-NOG-100T-146-T-2570 (GE13)	22B	16
New:		
GE14-P10DNAB394-12G5.0-100T-150-T-2590 (GE14C)	23	20
GE14-P10DNAB394-8G5.0/6G4.0-100T-150-T-2589 (GE14C)	23	16
GE14-P10DNAB394-7G5.0/6G4.0-100T-150-T-2538 (GE14C)	23	92
Total		368

3. Reference Core Loading Pattern

Nominal previous cycle core average exposure at end of cycle:	30743 MWd/MT (27890 MWd/ST)
Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations:	30490 MWd/MT (27660 MWd/ST)
Assumed reload cycle core average exposure at beginning of cycle:	18488 MWd/MT (16772 MWd/ST)
Assumed reload cycle core average exposure at end of cycle (hard bottom burn):	31561 MWd/MT (28632 MWd/ST)
Reference core loading pattern:	Figure 1

4. Calculated Core Effective Multiplication and Control System Worth - No Voids, 20°C

Beginning of Cycle, $k_{effective}$	
Uncontrolled	1.106
Fully controlled	0.948
Strongest control rod out	0.988
R, Maximum increase in cold core reactivity with exposure into cycle, Δk	0.000

5. Standby Liquid Control System Shutdown Capability

Boron (ppm) (at 20°C)	Shutdown Margin (Δk) (at 160°C, Xenon Free)
800	0.059

6. Reload Unique GETAB Anticipated Operational Occurrences (AOO) Analysis Initial Condition Parameters [1]

Operating domain: ICF
[2]Exposure range: BOC23 to EOR23-1102 MWd/MT (1000 MWd/ST)

	Peaking Factors						
Fuel Design	**Local**	**Radial**	**Axial**	**R-Factor**	**Bundle Power (MWt)**	**Bundle Flow (1000 lb/hr)**	**Initial MCPR**
GE14C	1.45	1.62	1.38	1.040	6.829	113.3	1.37
GE13	1.45	1.58	1.38	1.020	6.651	105.3	1.32

Operating domain: ICF
Exposure range: EOR23-1102 MWd/MT (1000 MWd/ST) to EEOC23

	Peaking Factors						
Fuel Design	**Local**	**Radial**	**Axial**	**R-Factor**	**Bundle Power (MWt)**	**Bundle Flow (1000 lb/hr)**	**Initial MCPR**
GE14C	1.45	1.62	1.22	1.040	6.827	112.9	1.39
GE13	1.45	1.59	1.22	1.020	6.705	104.9	1.34

[1] The GE8x8NB fuel type was not analyzed for Cycle 23 since it had previously been shown to be bounded by the GE13 fuel type.
[2] End of Rated (EOR) is defined as end-of-cycle all rods out, 100% power/100% flow and normal feedwater temperature.

Operating domain: Nominal Exposure Range: BOC23 to EOC23 Event: Inadvertent HPCI Transient							
	Peaking Factors						
Fuel Design	Local	Radial	Axial	R-Factor	Bundle Power (MWt)	Bundle Flow (1000 lb/hr)	Initial MCPR
GE14	1.45	1.72	1.62	1.040	7.225	83.81	1.25
GE13	1.45	1.64	1.62	1.020	6.906	79.35	1.24

7. Selected Margin Improvement Options

Recirculation pump trip:	No
Rod withdrawal limiter:	No
Thermal power monitor:	No
Improved scram time:	Yes (ODYN Option B)
Measured scram time:	No
Exposure dependent limits:	Yes
Exposure points analyzed:	3

8. Operating Flexibility Options

Single-loop operation:	Yes
Load line limit:	No
Extended load line limit:	Yes
Maximum extended load line limit:	No
Increased core flow throughout cycle:	Yes
Flow point analyzed:	107.0 %
Increased Core Flow at EOC	Yes

Feedwater temperature reduction throughout cycle: No

Final feedwater temperature reduction: No

ARTS Program: No

Maximum extended operating domain: No

Moisture separator reheater OOS: No

Turbine bypass system OOS: No

Safety/relief valves OOS: Yes
(credit taken for 5 of 6 valves)

ADS OOS: No

EOC RPT OOS: No

Main steam isolation valves OOS: No

9. Core-wide AOO Analysis Results

Methods used: GEMINI; GEXL-PLUS

Operating domain: ICF
Exposure range: BOC23 to EOR23-1102 MWd/MT (1000 MWd/ST)

			Uncorrected ΔCPR		
Event	**Flux (%NBR)**	**Q/A (%NBR)**	**GE14C**	**GE13**	**Fig.**
FW Controller Failure	355	122	0.25	0.22	2
Load Reject w/o Bypass	369	118	0.26	0.22	3
Turbine Trip w/o Bypass	372	118	0.26	0.22	4

Operating domain: ICF
Exposure range: EOR23-1102 MWd/MT (1000 MWd/ST) to EEOC23

			Uncorrected ΔCPR		
Event	**Flux (%NBR)**	**Q/A (%NBR)**	**GE14C**	**GE13**	**Fig.**
FW Controller Failure	293	119	0.26	0.23	5
Load Reject w/o Bypass	316	117	0.29	0.24	6
Turbine Trip w/o Bypass	323	117	0.28	0.23	7

Operating domain: Nominal Exposure range: BOC23 to EOC23					
			Uncorrected ΔCPR		
Event	**Flux (%NBR)**	**Q/A (%NBR)**	**GE14C**	**GE13**	**Fig.**
Inadvertent HPCI	119	117	0.15	0.14	8

10. Local Rod Withdrawal Error (With Limiting Instrument Failure) AOO Summary

Rod Block Reading, (%)	Rod Position (Feet Withdrawn)	ΔCPR
104	3.5	0.12
105	4.0	0.18
106	4.0	0.18
107	4.5	0.22
108	4.5	0.22
109	5.0	0.26
110	5.5	0.30
Unblocked	N/A	0.48

Setpoint selected: 110%

Limiting rod pattern: Figure 9

At least one RBM channel must be operable when moving rods above 30.0% power.

11. Cycle MCPR Values [3] [4]

Safety limit: 1.10

Single loop operation safety limit: 1.12

[3] Operating Limits for the ELLLA domain are bounded by the ICF Operating Limits.
[4] For single-loop operation, the MCPR operating limit is 0.02 greater than the two-loop value.

Non-pressurization events:

Exposure range: BOC23 to EOC23		
	All Fuel Types	
Rod Withdrawal Error (for RBM setpoint to 110%)	1.40	
Fuel Loading Error (mislocated)	Non-limiting	
Loss of Feedwater Heating (See Appendix B)	1.21	
Inadvertent High Pressure Coolant Injection (See Appendix B)	1.29	
	GE14C	GE13
Fuel Loading Error (misoriented)	1.21	1.23

Pressurization events:

Operating domain: ICF Exposure range: BOC23 to EOR23-1102 MWd/MT (1000 MWd/ST)				
	Option A		Option B	
	GE14C	GE13	GE14C	GE13
FW Controller Failure	1.48	1.37	1.37	1.32
Load Reject w/o Bypass	1.49	1.37	1.38	1.32
Turbine Trip w/o Bypass	1.48	1.37	1.37	1.32

Operating domain: ICF Exposure range: EOR23-1102 MWd/MT (1000 MWd/ST) to EEOC23				
	Option A		Option B	
	GE14C	GE13	GE14C	GE13
FW Controller Failure	1.59	1.46	1.42	1.35
Load Reject w/o Bypass	1.61	1.47	1.44	1.36
Turbine Trip w/o Bypass	1.61	1.46	1.44	1.35

12. Overpressurization Analysis Summary

Event	Psl (psig)	Pdome (psig)	Pv (psig)	Plant Response
MSIV Closure (Flux Scram)	1278	1279	1303	Figure 10

13. Loading Error Results

Variable water gap misoriented bundle analysis: Yes [5]

Misoriented Fuel Bundle	ΔCPR
GE13-P9DTB386-11G4.0/1G3.0-100T-146-T-2425 (GE13)	0.13
GE13-P9DTB225-NOG-100T-146-T-2570 (GE13)	0.04
GE14-P10DNAB394-7G5.0/6G4.0-100T-150-T-2538 (GE14C)	0.11
GE14-P10DNAB394-8G5.0/6G4.0-100T-150-T-2589 (GE14C)	0.11
GE14-P10DNAB394-12G5.0-100T-150-T-2590 (GE14C)	0.11

14. Control Rod Drop Analysis Results

This is a banked position withdrawal sequence plant, therefore, the control rod drop accident analysis is not required. NRC approval is documented in NEDE-24011-P-A-US.

15. Stability Analysis Results

The Vermont Yankee Nuclear Power Station has implemented the Option I-D solution documented in *Application of the "Regional Exclusion with Flow-Biased APRM Neutron Flux Scram" Stability Solution (Option I-D) to the Vermont Yankee Nuclear Power Plant, Licensing Topical Report,* GENE-637-018-0793, July 1993. The NRC approved ODYSY methodology (NEDC-32992P-A, July 2001) has been applied to this reload. ODYSY applications offer the benefit of more accurate simulations of BWR stability events and conditions. Option I-D has (1) "prevention" elements (Exclusion and Buffer Regions) and (2) a "detect & suppress" element (MCPR safety limit protection provided by the flow-biased APRM flux trip for the dominant core wide mode of coupled thermal-hydraulic/neutronic reactor instability). Core and hot channel decay ratio calculations to determine the Exclusion Region and additional bases demonstrate that core wide is the predominant reactor instability mode for Vermont Yankee. The detect and suppress calculation consists of (A) calculation of a 95% probability/95% confidence level statistically-based hot bundle oscillation magnitude for anticipated core-wide mode reactor instability and (B) calculation of the stability-based Operating Limit MCPR (OLMCPR) which provides 95/95 MCPRSL protection.

The detect and suppress calculation requires the use of the DIVOM (which is defined as the Delta CPR over Initial MCPR Versus the Oscillation Magnitude) curve. Recent TRACG evaluations by GE-NE have shown that the generic core-wide DIVOM curve specified in NEDO-32465-A may not be conservative for current plant operating conditions for plants which have implemented Stability Option I-D. Specifically, a non-conservative deficiency has been identified for high power-to-flow ratios in the generic core-wide mode DIVOM curve. The deficiency results in a non-conservative slope of the associated core-wide DIVOM curve so that the APRM flux trip setpoint is too high. GE has made a

[5] Includes a 0.02 penalty due to variable water gap R-factor uncertainty.

Part 21 Notification on this issue. For Option I-D plants, the applicability of the core wide mode DIVOM curve may be determined by comparing the core average power-to-flow ratio following a simulated flow runback on the rated rod line to approximately 30% of rated core flow to a value of 66 MWt/Mlbm/hr (OG01-0228-001). If the core average power-to-flow ratio exceeds this value, then the generic core wide mode DIVOM curve is not applicable and appropriate corrective actions should be taken. For Vermont Yankee, the calculated core average power-to-flow ratio is 58.0 MWt/Mlbm/hr and the existing DIVOM slope is bounding for Cycle 23 operation.

(1) The Exclusion Region and Buffer Region for Cycle 22 were validated for Cycle 23 operation. The regions are provided in Appendix A.

(2A) The hot channel oscillation magnitudes at natural circulation and 45% rated core flow were calculated for Cycle 22. A reload review process has determined that the current hot channel oscillation magnitudes are applicable to Cycle 23. The current values are used in the calculation of the stability-based OLMCPR.

(2B) The stability-based OLMCPR was calculated for Cycle 23. The calculation demonstrated that reactor stability does not produce the limiting OLMCPR for Cycle 23. It is shown that the rated OLMCPR, OLMCPR(100%P/100%F), is greater than the OLMCPR for a two-pump trip, OLMCPR(2PT); and the OLMCPR at 100% rodline and 45% rated core flow, OLMCPR(100%RL/45%F), is greater than the OLMCPR at steady state, OLMCPR(SS). For this analysis, OLMCPR(100%P/100%F) of 1.32 and OLMCPR(100%RL/45%F) of 1.54 are used. For both scenarios considered the criteria are met:

OLMCPR(100%P/100%F) > OLMCPR(2PT) i.e. 1.32 > 1.27

OLMCPR(100%RL/45%F) > OLMCPR(SS) i.e. 1.54 > 1.36

16. Loss-of-Coolant Accident Results

LOCA method used: SAFER/GESTR-LOCA

The ECCS-LOCA analysis results for GE14 fuel are presented in Section 5 of GE-NE-0000-0000-9381-02P, August 2002. For GE13 and GE8x8NB (GE9B), results are presented in NEDC-32814P, March 1998. For GE8x8NB (GE9B) fuel this analysis yielded a Licensing Basis peak cladding temperature of 1730°F, a maximum local oxidation fraction of <0.8% and a core-wide metal-water reaction of <0.1%. For GE13 fuel this analysis yielded a Licensing Basis peak cladding temperature of 1850°F, a maximum local oxidation fraction of <2.6% and a core-wide metal-water reaction of <0.1%. For GE14 fuel this analysis yielded a Licensing Basis peak cladding temperature of 1950°F, a maximum local oxidation fraction of <3.0% and a core-wide metal-water reaction of <0.1%.

The ECCS Initial MCPR values are 1.275 for GE14 and GE13, and 1.20 for GE8x8NB (GE9B).

The ECCS MAPLHGR multiplier for Single Loop Operation (SLO) is 0.82 for all GE14, GE13 and GE8x8NB (GE9B) fuels.

A review of the ECCS-LOCA 10 CFR 50.46 reportable errors for Vermont Yankee SAFER/GESTR analysis was conducted. All known errors have been accounted for in the reference analysis for GE14 fuel. However, errors were identified applicable to the GE13 and GE8x8NB (GE9B) analyses. The impact of the applicable errors are summarized as follows:

10CFR50.46 Error Name and/or Reference	**10CFR50.46 Error Description**	**Licensing Basis ΔPCT °F**	
		GE13	**GE8x8NB**
Error 2000-01, Revision 1	Error in the shell outside diameter value of the steam separator in the SAFER basedeck for Vermont Yankee and time step error impact (Error 2000-01 includes error 2000-04)	-12	-12
Error 2001-01	Error in accounting for the steam condensed when subcooled ECCS liquid is injected into the lower plenum	0	0
Error 2001-02	An inconsistency in the vessel pressure rate equation	+10	+10
Error 2002-01	Error in core spray injection elevation	+5	+5
Error 2002-02	Impact of SAFER Bulk Water Level Error on the Peak Clad Temperature (PCT).	+5	+5
Total ΔPCT		+8	+8

The most limiting and least limiting MAPLHGRs for the new GE14 fuel bundles being added this cycle are as follows:

16. Loss-of-Coolant Accident Results (cont.)

Bundle Type: GE14-P10DNAB394-7G5.0/6G4.0-100T-150-T-2538

Average Planar Exposure		MAPLHGR (kW/ft)	
(GWd/ST)	(GWd/MT)	Most Limiting	Least Limiting
0.00	0.00	9.66	9.75
0.20	0.22	9.73	9.82
1.00	1.10	9.87	9.96
2.00	2.20	10.03	10.14
3.00	3.31	10.19	10.34
4.00	4.41	10.37	10.56
5.00	5.51	10.55	10.78
6.00	6.61	10.74	10.98
7.00	7.72	10.93	11.21
8.00	8.82	11.10	11.40
9.00	9.92	11.26	11.58
10.00	11.02	11.42	11.76
11.00	12.13	11.57	11.90
12.00	13.23	11.64	11.97
13.00	14.33	11.62	11.90
14.00	15.43	11.57	11.80
15.00	16.53	11.50	11.69
17.00	18.74	11.31	11.46
20.00	22.05	10.98	11.12
25.00	27.56	10.44	10.56
30.00	33.07	9.92	10.02
35.00	38.58	9.40	9.51
40.00	44.09	8.90	9.01
45.00	49.60	8.38	8.50
50.00	55.12	7.84	7.98
55.00	60.63	5.91	6.24
57.13	62.97	4.85	5.18
57.71	63.61	--	4.89
57.71	63.62	--	4.88

16. Loss-of-Coolant Accident Results (cont.)

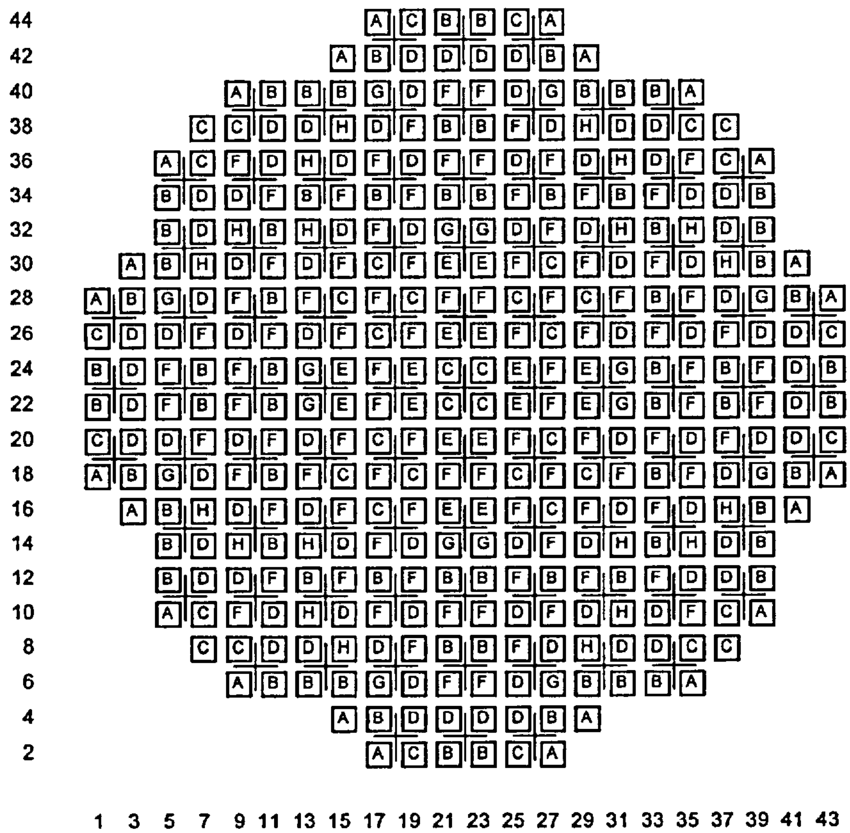
Bundle Type: GE14-P10DNAB394-8G5.0/6G4.0-100T-150-T-2589

Average Planar Exposure		MAPLHGR (kW/ft)	
(GWd/ST)	(GWd/MT)	Most Limiting	Least Limiting
0.00	0.00	9.56	9.65
0.20	0.22	9.63	9.72
1.00	1.10	9.77	9.85
2.00	2.20	9.95	10.04
3.00	3.31	10.11	10.24
4.00	4.41	10.29	10.45
5.00	5.51	10.47	10.67
6.00	6.61	10.66	10.87
7.00	7.72	10.85	11.10
8.00	8.82	11.02	11.29
9.00	9.92	11.18	11.49
10.00	11.02	11.34	11.69
11.00	12.13	11.51	11.85
12.00	13.23	11.58	11.94
13.00	14.33	11.58	11.89
14.00	15.43	11.55	11.80
15.00	16.53	11.49	11.69
17.00	18.74	11.30	11.46
20.00	22.05	10.98	11.12
25.00	27.56	10.44	10.56
30.00	33.07	9.91	10.02
35.00	38.58	9.40	9.51
40.00	44.09	8.89	9.01
45.00	49.60	8.38	8.50
50.00	55.12	7.83	7.97
55.00	60.63	5.89	6.22
57.08	62.92	4.85	5.18
57.66	63.56	--	4.89
57.66	63.56	--	4.88

16. Loss-of-Coolant Accident Results (cont.)

Bundle Type: GE14-P10DNAB394-12G5.0-100T-150-T-2590

Average Planar Exposure		MAPLHGR (kW/ft)	
(GWd/ST)	(GWd/MT)	Most Limiting	Least Limiting
0.00	0.00	9.81	9.89
0.20	0.22	9.87	9.95
1.00	1.10	9.97	10.07
2.00	2.20	10.11	10.24
3.00	3.31	10.26	10.42
4.00	4.41	10.41	10.60
5.00	5.51	10.57	10.79
6.00	6.61	10.74	10.97
7.00	7.72	10.91	11.17
8.00	8.82	11.06	11.34
9.00	9.92	11.19	11.51
10 00	11.02	11.33	11.66
11.00	12.13	11.46	11.80
12.00	13.23	11.53	11.88
13.00	14.33	11.52	11.83
14.00	15.43	11.49	11.75
15.00	16.53	11.44	11.66
17.00	18.74	11.28	11.45
20.00	22.05	10.97	11.11
25.00	27.56	10.44	10.55
30.00	33.07	9.91	10.02
35.00	38.58	9.40	9.51
40.00	44.09	8.89	9.01
45.00	49.60	8.38	8.49
50.00	55.12	7.84	7.98
55.00	60.63	5.92	6.25
57.14	62.98	4.85	5.18
57.72	63.62	--	4.89
57.72	63.63	--	4.88



Fuel Type	
A=GE9B-P8DWB335-10GZ-80M-150-T (Cycle 18)	F=GE14-P10DNAB394-7G5.0/6G4.0-100T-150-T-2538 (Cycle 23)
B=GE13-P9HTB380-12GZ-100T-146-T (Cycle 20)	
C=GE13-P9HTB379-13GZ-100T-146-T (Cycle 20)	G=GE14-P10DNAB394-8G5.0/6G4.0-100T-150-T-2589 (Cycle 23)
D=GE13-P9DTB386-11G4.0/1G3.0-100T-146-T-2425 (Cycle 22)	H=GE14-P10DNAB394-12G5.0-100T-150-T-2590 (Cycle 23)
E=GE13-P9DTB225-NOG-100T-146-T-2570 (Cycle 22B)	

Figure 1 Reference Core Loading Pattern

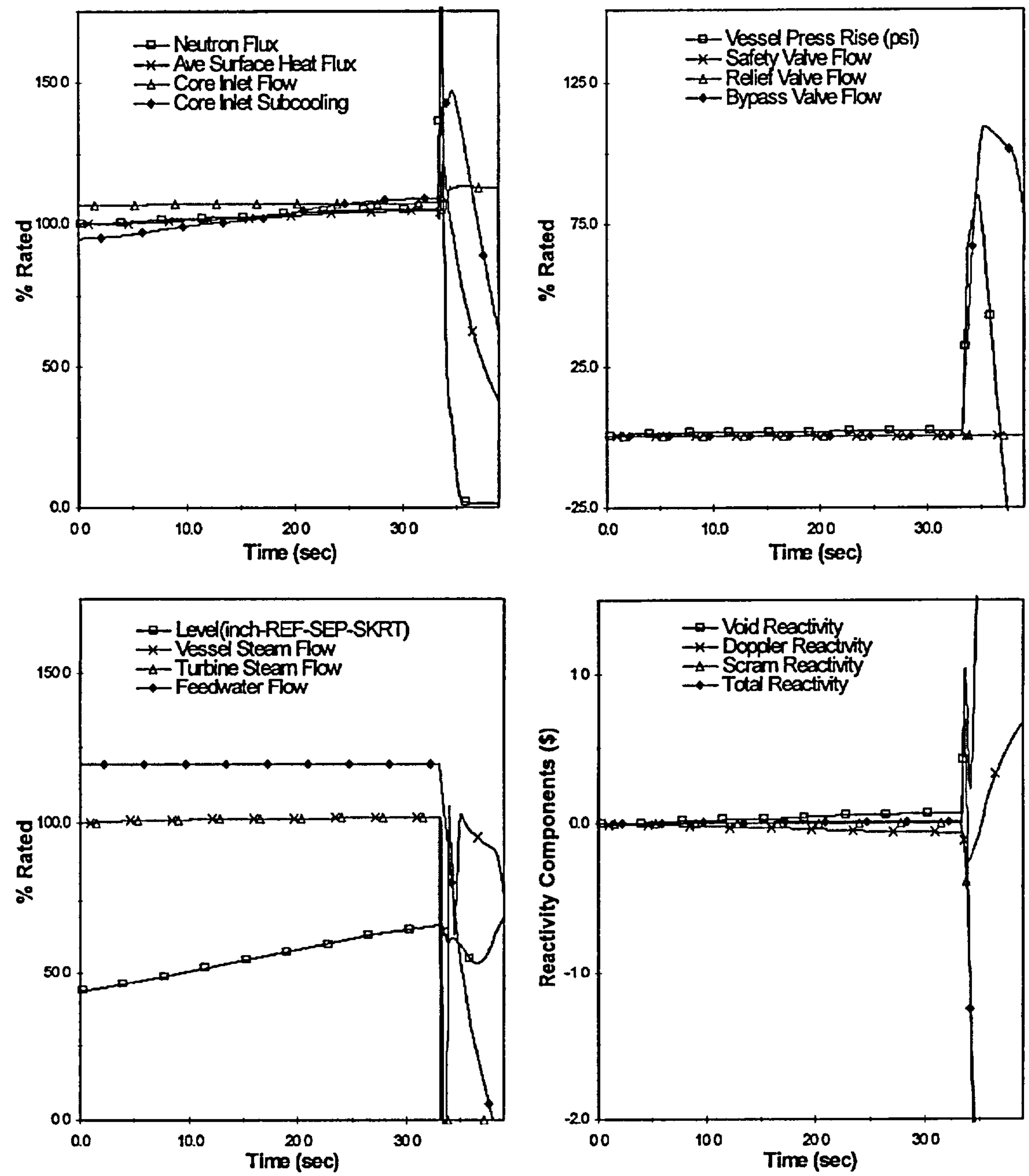


Figure 2 Plant Response to FW Controller Failure
BOC23 to EOR23-1102 MWd/MT (1000 MWd/ST) ICF

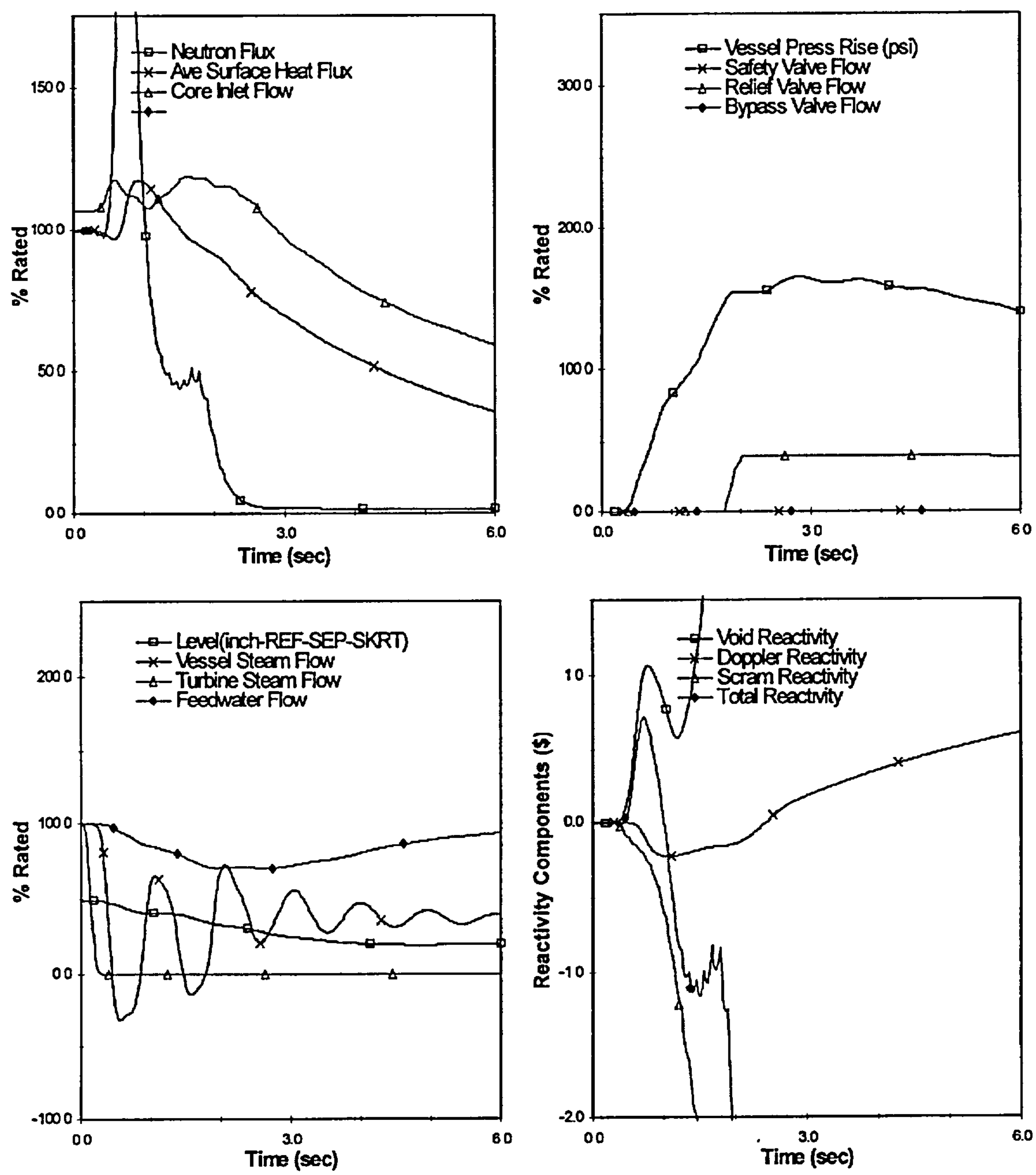


Figure 3 Plant Response to Load Reject w/o Bypass
BOC23 to EOR23-1102 MWd/MT (1000 MWd/ST) ICF

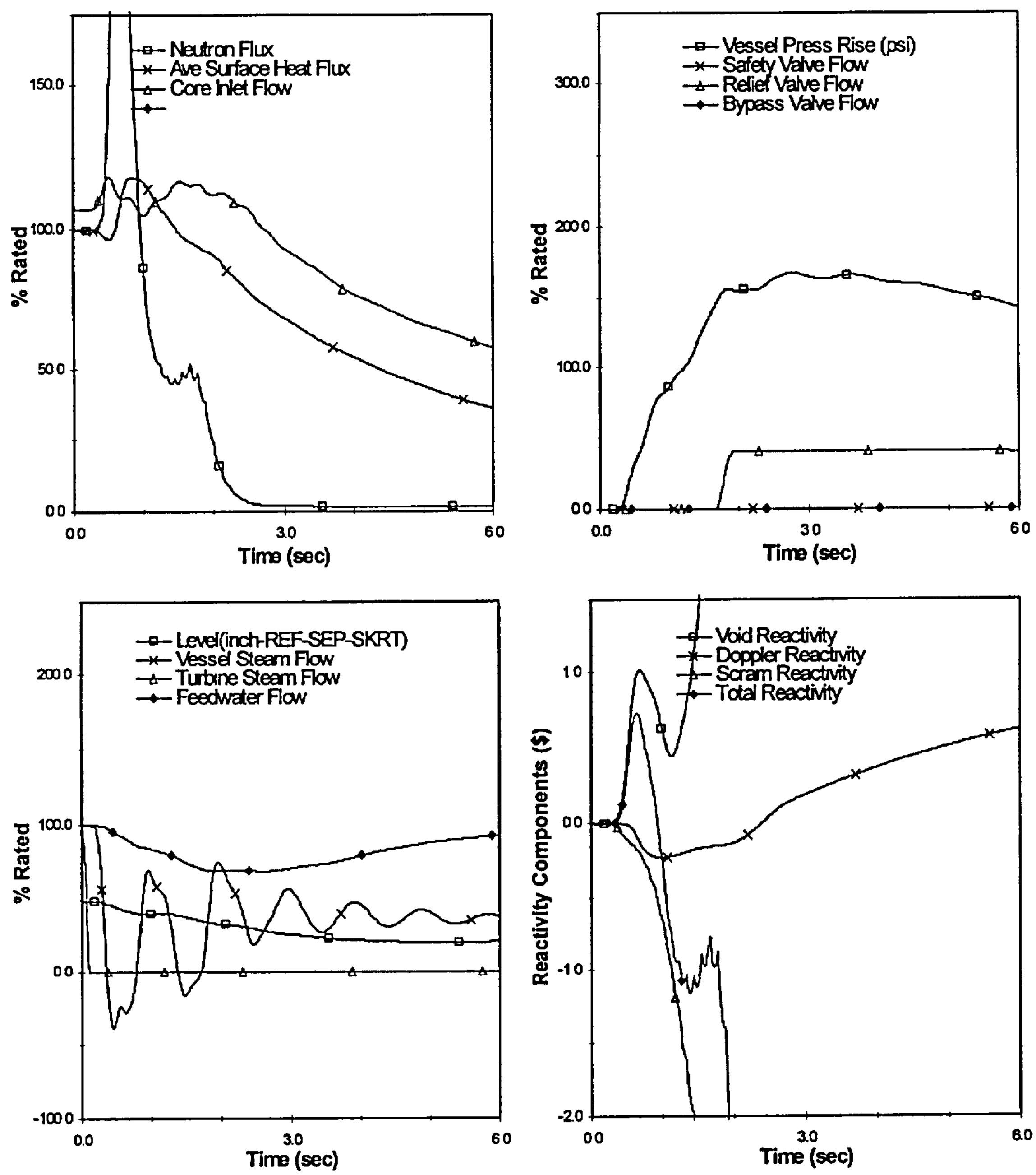


**Figure 4 Plant Response to Turbine Trip w/o Bypass
BOC23 to EOR23-1102 MWd/MT (1000 MWd/ST) ICF**

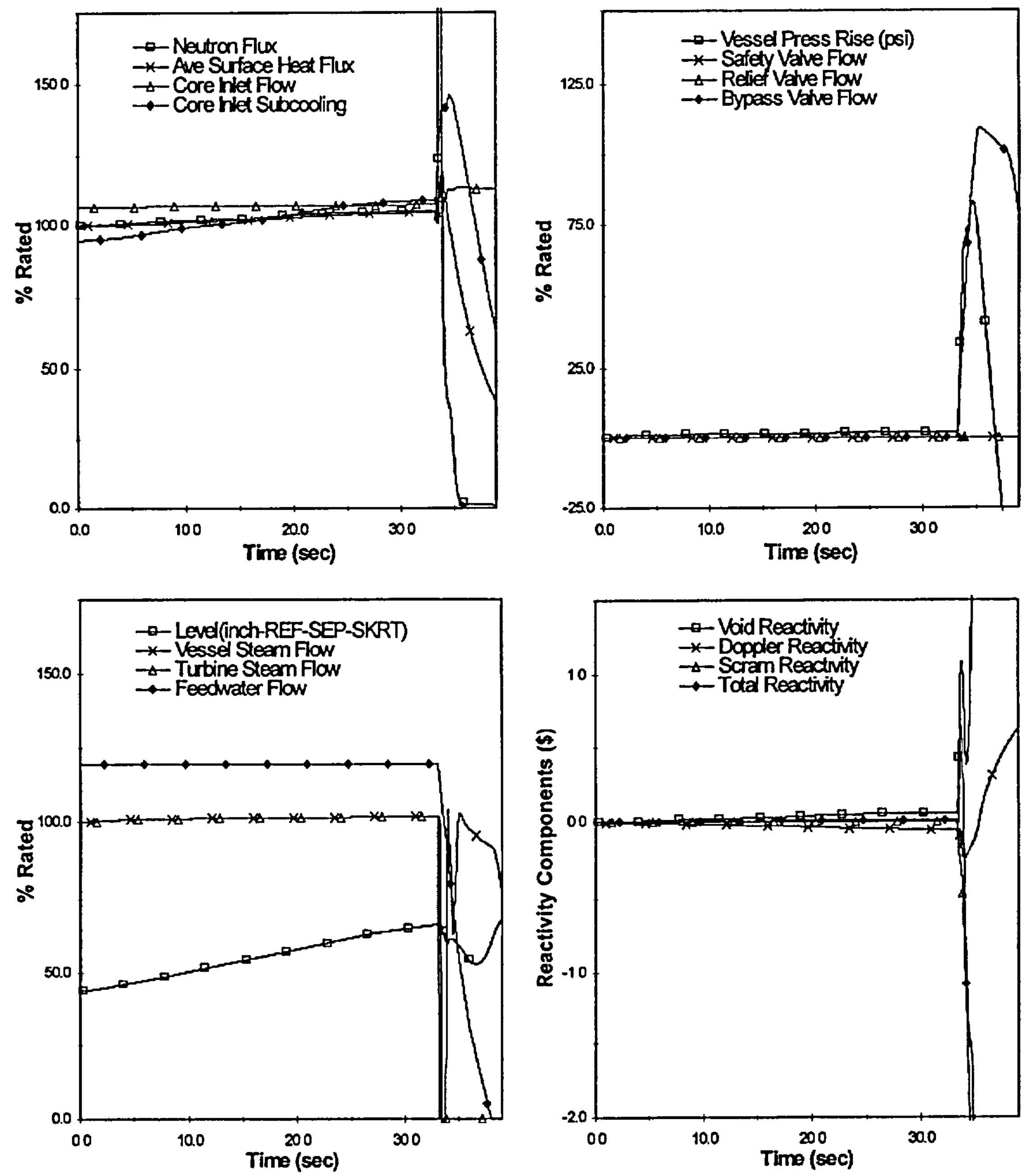


Figure 5 Plant Response to FW Controller Failure
EOR23-1102 MWd/MT (1000 MWd/ST) to EEOC23 ICF

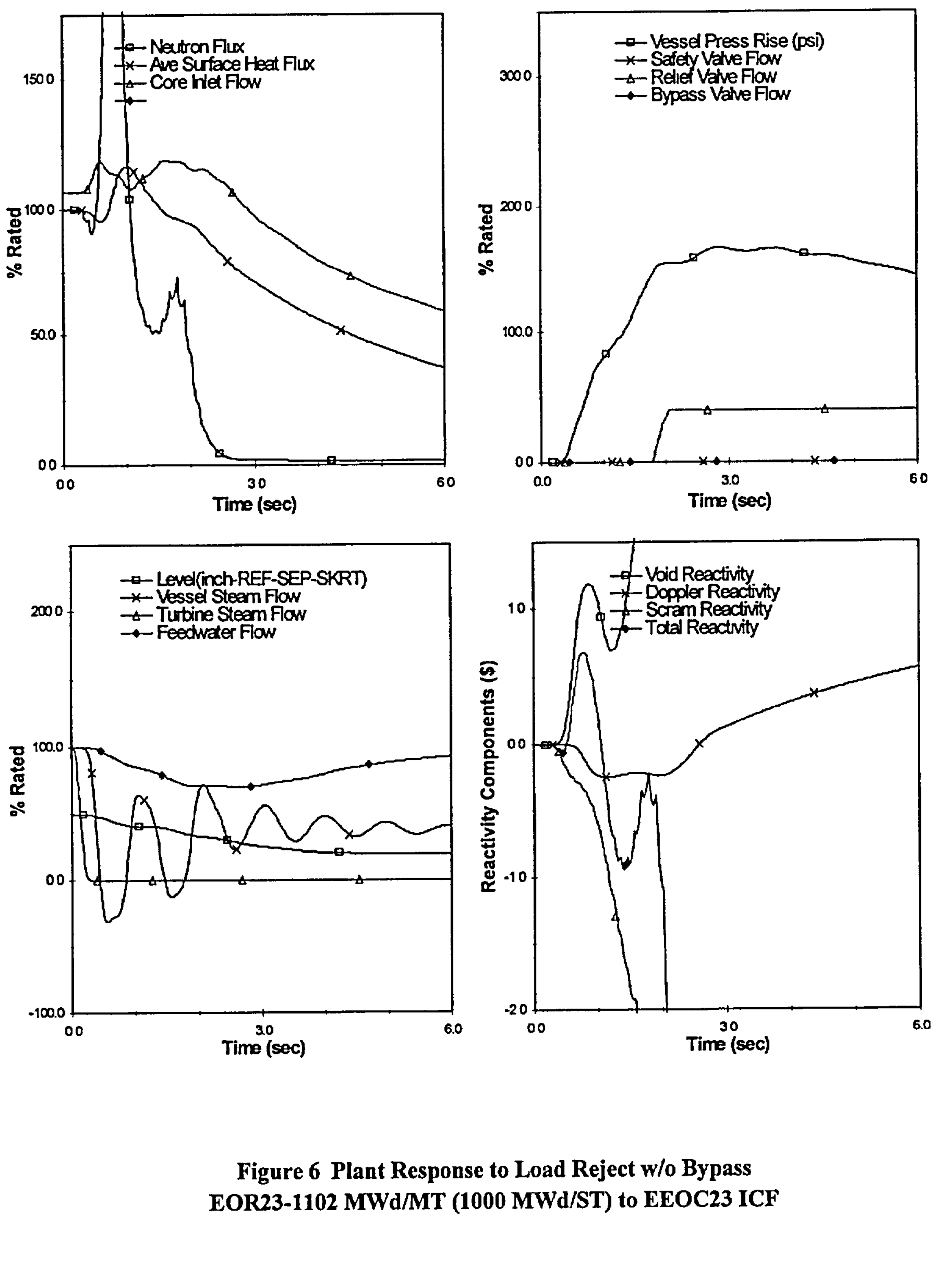


Figure 6 Plant Response to Load Reject w/o Bypass
EOR23-1102 MWd/MT (1000 MWd/ST) to EEOC23 ICF

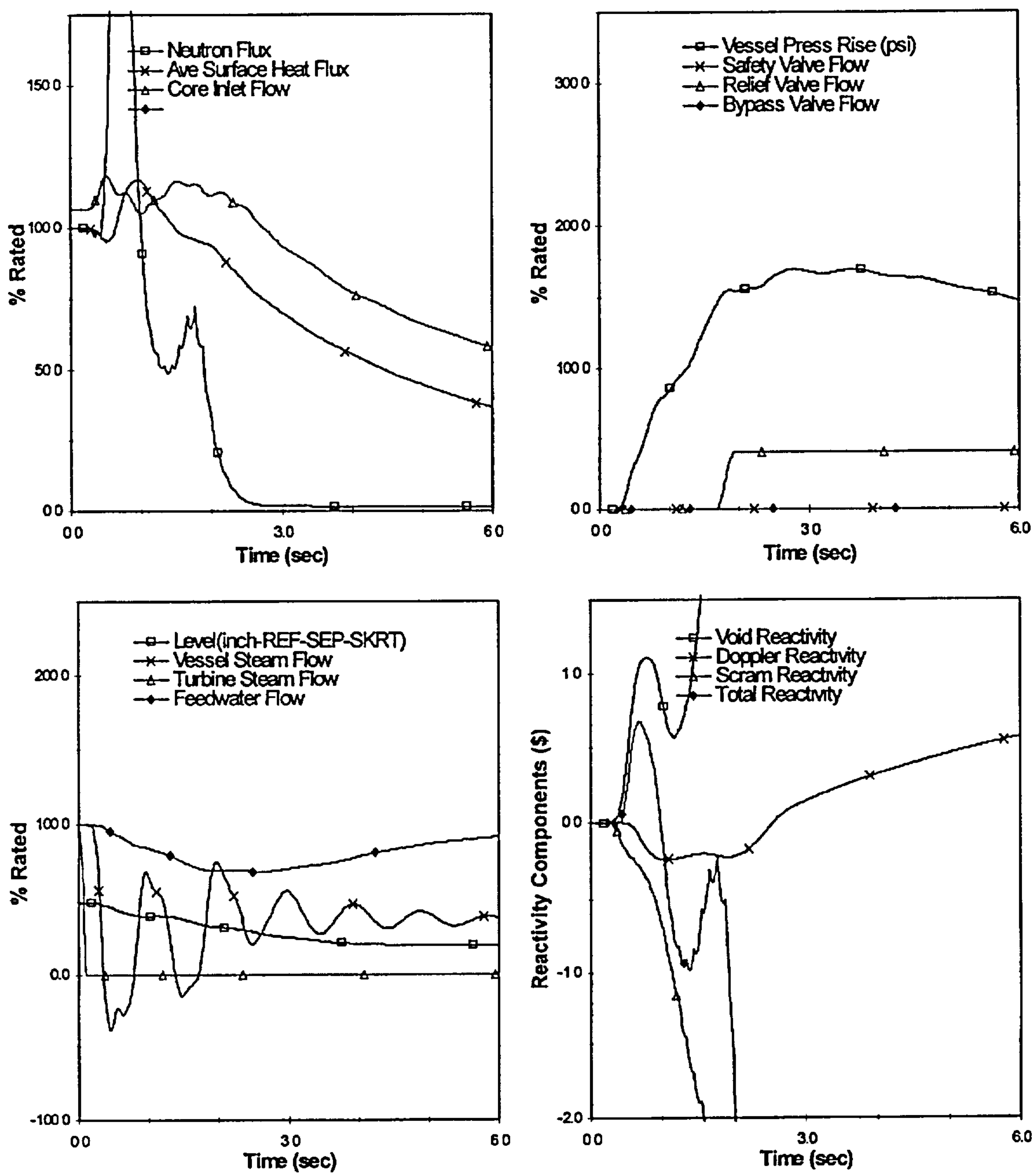


Figure 7 Plant Response to Turbine Trip w/o Bypass
EOR23-1102 MWd/MT (1000 MWd/ST) to EEOC23 ICF

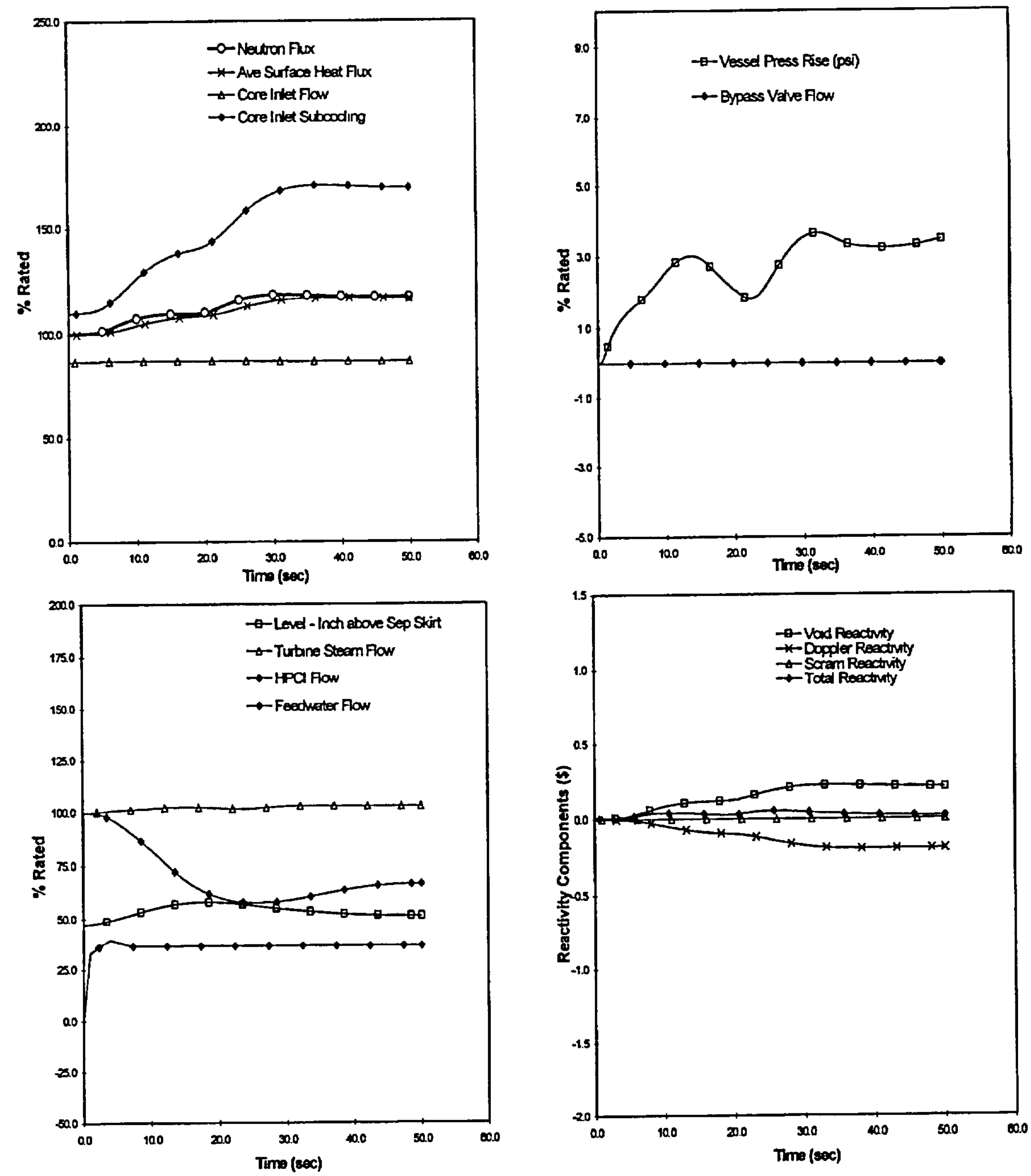


**Figure 8 Plant Response to Inadvertent HPCI
BOC23 to EOC23**

	2	6	10	14	18	22	26	30	34	38	42
43											
39				32		0		32			
35											
31		0		0		0		0		0	
27											
23		10		14				14		10	
19						0					
15		0		0		0		0		0	
11											
7				32		0		32			
3											

Notes: 1. Number indicates number of notches withdrawn out of 48.
Blank is a fully withdrawn rod.
2. Error rod is (22,31).

Figure 9 Limiting Rod Pattern

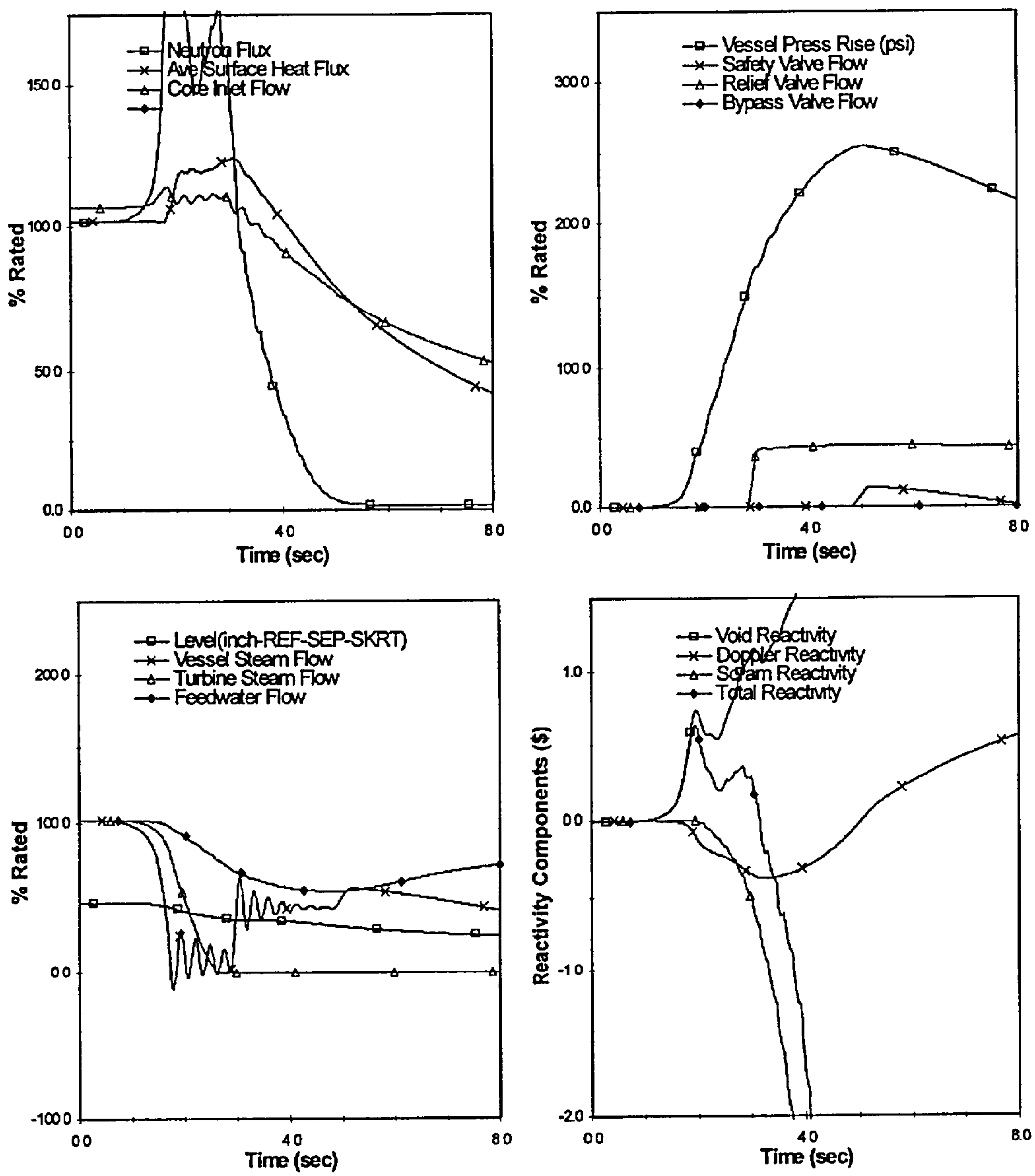


Figure 10 Plant Response to MSIV Closure (Flux Scram)

Appendix A
Analysis Conditions

To reflect actual plant parameters accurately, the values shown in Table A-1 were used this cycle. [6]

Table A-1

	Analysis Value
Parameter	ICF
Thermal power, MWt	1593.0
Core flow, Mlb/hr	51.36
Reactor pressure, psia	1041.4
Inlet enthalpy, BTU/lb	522.9
Non-fuel power fraction	0.036
Steam flow, Mlb/hr	6.46
Dome pressure, psig	1010.0
Turbine pressure, psig	968.3
Number of Safety/Relief Valves	3
Number of Single Spring Safety Valves	2
Relief mode lowest setpoint, psig	1113.0
Safety mode lowest setpoint, psig	1277.0

[6] Transient analyses assumed 1 of 4 SR/Vs out of service.

Appendix B
Decrease in Core Coolant Temperature Events

The Loss of Feedwater Heating (LFWH) event and Inadvertent High Pressure Coolant Injection (HPCI) start-up event are the only cold water injection AOOs checked on a cycle-by-cycle basis.

The LFWH event was analyzed for Vermont Yankee Cycle 23 at 100% rated power using the BWR Simulator Code. The use of this code is permitted in GESTAR II. The transient plots, neutron flux and heat flux values normally reported in Section 9 are not an output of the BWR Simulator Code; therefore, those items are not included in this document. The OLMCPR result is shown in Section 11.

For Cycle 23, it was shown that the inadvertent HPCI start-up event was not bounded by the LFWH event in accordance with *Determination of Limiting Cold Water Event*, NEDC-32358P-A. Therefore, further analysis was required to calculate ΔCPR for HPCI. The ΔCPR for the HPCI event (0.19) was calculated using ODYN analysis per Reference B-1. Figure 8 represents the plant response to inadvertent HPCI at BOC-23, which is the limiting exposure for this transient.

Reference

B-1. *Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors*, NEDO-24154-P-A (Supplement 1 – Volume 4), Revision 1, February 2002.

Appendix C
Introduction of GE14 Fuel

Implementation of GE14 is documented in Reference C-1.

Reference

C-1. *GE14 Fuel Design Cycle-Independent Analyses For Entergy Nuclear - Vermont Yankee,* GE-NE-0000-0000-7896-01P, Revision 1, September 2002.

Appendix D
Stability Solution Option I-D Exclusion and Buffer Regions

The stability Option I-D Exclusion Region and Buffer Regions for Vermont Yankee Cycle 22 were validated for Cycle 23 operation. The endpoints of the regions are defined in Table D-1. The region boundaries are defined using the generic shape function, Equation D-1. The regions are shown on the Vermont Yankee Cycle 23 power/flow map in Figure D-1.

Table D-1. Exclusion Buffer Region Endpoints[1]

Exclusion Region	Power (% rated)[2]	Flow (% rated)[3]
A	73.5	50.3
B	31.8	30.6
Buffer Region	**Power (% rated)[2]**	**Flow (% rated)[3]**
A	77.6	55.3
B	26.8	29.6

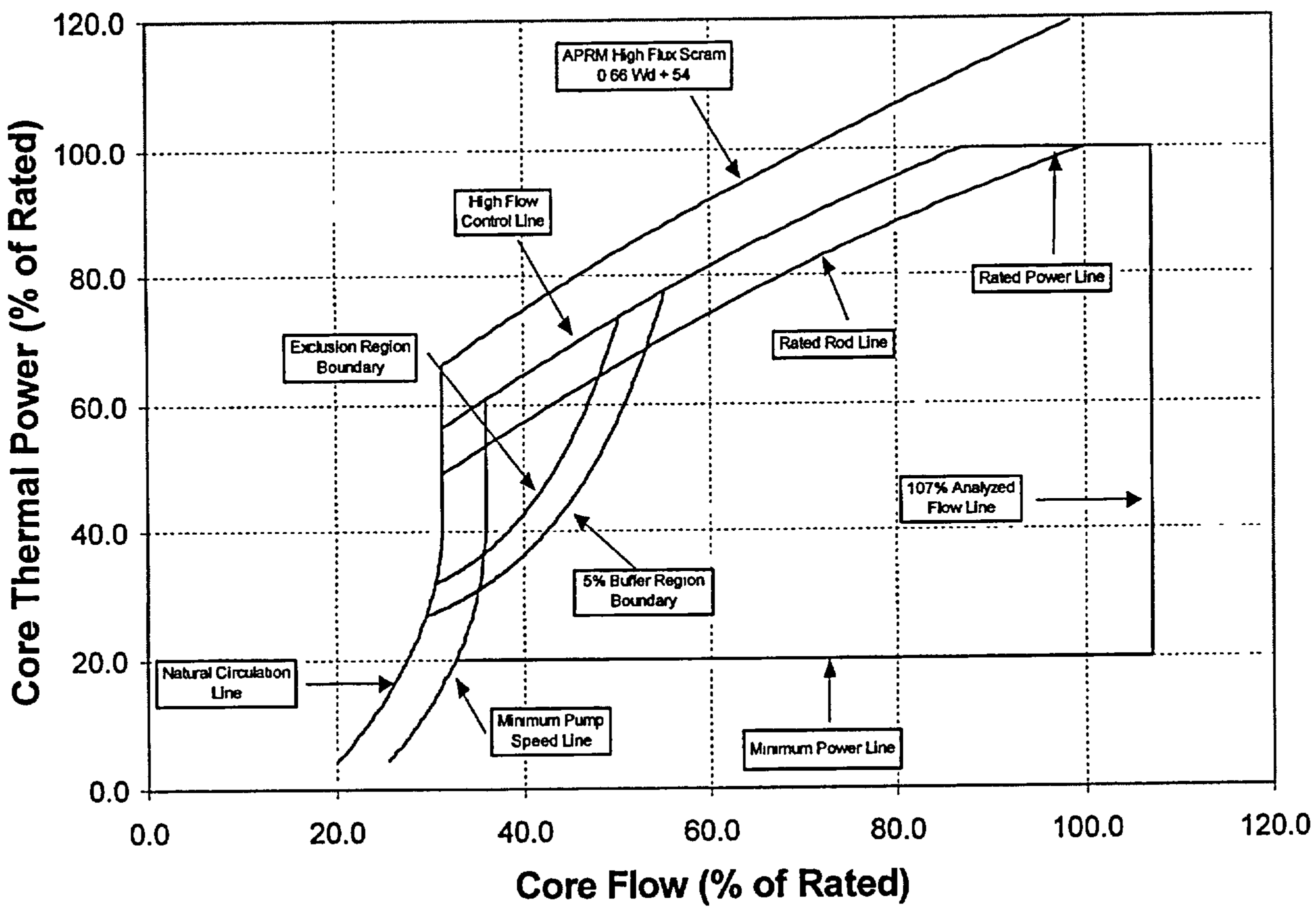
1. Point "A" is on the highest flow control line, point "B" is on the natural circulation line.
2. Rated power is 1593 MWt
3. Rated flow is 48 Mlb/hr

Equation D-1. Generic Shape Function

$$P = P_B \left(\frac{P_A}{P_B} \right)^{\frac{1}{2}\left[\frac{W-W_B}{W_A-W_B} + \left(\frac{W-W_B}{W_A-W_B} \right)^2 \right]}$$

where,
P = a core thermal power value on the region boundary (% of rated),
W = the core flow rate corresponding to power, P, on the region boundary (% of rated),
P_A = core thermal power at point A (% of rated),
P_B = core thermal power at point B (% of rated),
W_A = core flow rate at point A (% of rated), and
WB = core flow rate at point B (% of rated).

Figure D-1. Exclusion and Buffer Region Boundaries on Power/Flow Map



References

D-1. *Application of the "Regional Exclusion with Flow-Biased APRM Neutron Flux Scram" Stability Solution (Option I-D) to the Vermont Yankee Nuclear Power Plant, Licensing Topical Report,* GENE-637-018-0793, July 1993.

D-2. *Vermont Yankee Cycle 23 Option I-D Stability ODYSY Analysis,* GE-NE-0000-0007-1914-01 Revision 2, October 2002.

D-3. *Vermont Yankee Cycle 23 Option I-D Stability Detect and Suppress Analysis,* GE-NE-0000-0007-1914-02, Revision 0, October 2002.

Appendix E
List of Acronyms

Acronym	Description
ΔCPR	Delta Critical Power Ratio
Δk	Delta k-effective
NBR	Percent Nuclear Boiler Rated
2RPT	Two Recirculation Pump Trip
ADS	Automation Depressurization System
ADSOOS	Automation Depressurization System Out of Service
AOO	Anticipated Operational Occurrence
APRM	Average Power Range Monitor
ARTS	APRM, Rod Block and Technical Specification Improvement Program
BOC	Beginning of Cycle
BSP	Backup Stability Protection
BWROG	Boiling Water Reactor Owners Group
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
DIVOM	Delta CPR over Initial MCPR vs. Oscillation Magnitude
DR	Decay Ratio
ECCS	Emergency Core Cooling System
EEOC	Extended End of Cycle
ELLLA	Extended Load Line Limit Analysis
EOC	End of Cycle
EOR	End of Rated (All Rods Out 100%Power / 100%Flow)
ER	Exclusion Region
FFWTR	Final Feedwater Temperature Reduction
FMCPR	Final MCPR
FOM	Figure of Merit
FWTR	Feedwater Temperature Reduction
GDC	General Design Criterion
GETAB	General Electric Thermal Analysis Basis
GSF	General Shape Function
HAL	Haling Burn
HBB	Hard Bottom Burn
HBOM	Hot Bundle Oscillation Magnitude
HCOM	Hot Channel Oscillation Magnitude
HFCL	High Flow Control Line
ICA	Interim Corrective Action
ICF	Increased Core Flow
IMCPR	Initial MCPR
IVM	Initial Validation Matrix
LHGR	Linear Heat Generation Rate
LOCA	Loss of Coolant Accident
LPRM	Local Power Range Monitor
LTR	Licensing Topical Report

MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCPR	Minimum Critical Power Ratio
MELLLA	Maximum Extended Load Line Limit Analysis
MELLLA+	MELLLA Plus
MOC	Middle of Cycle
MRB	Maximal Region Boundaries
MSIV	Main Steam Isolation Valve
MTU	Metric Ton Uranium
MWd	Megawatt day
MWd/ST	Megawatt days per Standard Ton
MWd/MT	Megawatt days per Metric Ton
MWt	Megawatt Thermal
NBP	No Bypass
NCL	Natural Circulation Line
NOM	Nominal Burn
NTR	Normal Trip Reference
OLMCPR	Operating Limit MCPR
OOS	Out of Service
OPRM	Oscillation Power Range Monitor
PCT	Peak Clad Temperature
Pdome	Peak Dome Pressure
PHE	Peak Hot Excess
PLHGR	Peak Linear Heat Generation Rate
PLUOOS	Power Load Unbalance Out of Service
Psl	Peak Steam Line Pressure
Pv	Peak Vessel Pressure
Q/A	Heat Flux
RBM	Rod Block Monitor
RC	Reference Cycle
RPS	Reactor Protection System
RVM	Reload Validation Matrix
RWE	Row Withdrawal Error
SC	Standard Cycle
SLMCPR	Safety Limit Minimum Critical Power Ratio
SLO	Single Loop Operation
SRLR	Supplemental Reload Licensing Report
SS	Steady State
STU	Standard Ton Uranium
TBVOOS	Turbine Bypass Valves Out of Service
TCVOOS	Turbine Control Valve Out of Service
TCVSC	Turbine Control Valve Slow Closure
TLO	Two Loop Operation
TRF	Trip Reference Function
UB	Under Burn