BNL-91311-2010-IR

Trace Assessment for BWR ATWS Analysis

Lap-Yan Cheng, David Diamond, Arantxa Cuadra
Gilad Raitses and Arnold Aronson

April 22, 2010

ES&T Department/NEIS Division

Brookhaven National Laboratory

P.O. Box 5000

Upton, NY 11973-5000

www.bnl.gov

Notice: This manuscript has been authored by employees of Brookhaven Science Associates, LLC under Contract No. DE-AC02-98CH10886 with the U.S. Department of Energy. The publisher by accepting the manuscript for publication acknowledges that the United States Government retains a non-exclusive, paid-up, irrevocable, world-wide license to publish or reproduce the published form of this manuscript, or allow others to do so, for United States Government purposes.

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or any third party's use or the results of such use of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof or its contractors or subcontractors. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

Printed on recycled paper

BNL 91311– 2010 IR

TRACE ASSESSMENT

FOR BWR ATWS ANALYSIS

Manuscript Completed: April 22, 2010

Prepared by:

Lap-Yan Cheng, David Diamond, Arantxa Cuadra, Gilad Raitses, and Arnold Aronson

Energy Sciences and Technology Department
Brookhaven National Laboratory

Prepared for:

Istvan Frankl, NRC Project Manager

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
NRC Job Code N6939

TABLE OF CONTENTS

ABSTRACT

A TRACE/PARCS input model has been developed in order to be able to analyze anticipated transients without scram (ATWS) in a boiling water reactor. The model is based on one developed previously for the Browns Ferry reactor for doing loss-of-coolant accident analysis. This model was updated by adding the control systems needed for ATWS and a core model using PARCS. The control systems were based on models previously developed for the TRAC-B code. The PARCS model is based on information (e.g., exposure and moderator density (void) history distributions) obtained from General Electric Hitachi and cross sections for GE14 fuel obtained from an independent source. The model is able to calculate an ATWS, initiated by the closure of main steam isolation valves, with recirculation pump trip, water level control, injection of borated water from the standby liquid control system and actuation of the automatic depressurization system. The model is not considered complete and recommendations are made on how it should be improved.

ACKNOWLEDGEMENTS

The authors would like to thank Prof. Thomas Downar and his staff at the University of Michigan for their technical support. In particular, the project depended greatly on the advice of Dr. Yunlin Xu at Michigan. The authors appreciate the technical advice of Dr. Anthony Ulses at the U.S. Nuclear Regulatory Commission (NRC) and the advice and support of Mr. Istvan Frankl, the Project Manager. A conversation with staff at Information Systems Laboratory was very helpful. Lastly, the authors appreciate the help of Ms. Lynda Fitz with the administrative requirements of the project.

1. INTRODUCTION

1.1 Background

TRACE [1] is a thermal-hydraulic code designed to perform best-estimate analyses of loss-of-coolant accidents, operational transients, and other accident scenarios in boiling and pressurized water reactors. It can also model phenomena occurring in experimental facilities designed to simulate transients in reactor systems. Models used include multidimensional two-phase flow, non-equilibrium thermo-dynamics, generalized heat transfer, reflood, level tracking and point reactor kinetics.

PARCS [2] is a neutron kinetics code designed to perform analyses of the neutron and power distribution throughout a reactor core for steady state (the eigenvalue problem) and as a function of time. A nodal method is used to obtain the solution in three dimensions and a pin power reconstruction method is used to obtain more detailed results. Two neutron energy groups are used along with six neutron precursor groups. Other solution algorithms are also available.

Predictions of boiling water reactor (BWR) behavior made using the TRACE/PARCS coupled code package (also sometimes referred to by only "TRACE") must be assessed by comparing code predictions to available data and/or analyses. These comparisons help to quantify the conservatism of licensing calculations and the ability of TRACE/PARCS to model and simulate various accidents and transients. The events of interest in the current study are anticipated transients without scram (ATWSs). This would expand the existing assessment database for the coupled code. This is obviously advantageous and also as specifically recommended by the Advisory Committee on Reactor Safeguards [3, 4].

1.2 Objectives

The primary objective of this project was to develop the capability of TRACE/PARCS to calculate the thermal-hydraulic and neutronic phenomena associated with BWR ATWS events. This was to be done by developing the requisite TRACE/PARCS input model for an ATWS initiated by the closure of the main steam isolation valves (MSIVs). An ancillary objective was to report on BNL's experience as a new user of TRACE, i.e., to identify problems that were encountered and to make recommendations so that future use of the code by other users would be more efficient. The constraints on the project were the version of the codes (TRACE 5.184 / PARCS v2.7) to be used and the information available at the time to develop the model.

1.3 Organization of Report

Section 2 summarizes the description of the model developed for BWR ATWS analysis. The details of this model are given in calculational notebooks that have been sent separately to the NRC as they contain information proprietary to General Electric Hitachi (GEH). Section 3 describes the ATWS event calculated to demonstrate the ability of the model as it exists. Section 4 contains the recommendations for improving the model and TRACE/PARCS. Lastly, references are found in Section 5.

2. MODEL FOR BWR ATWS

The BWR plant model for the ATWS analysis was developed from two existing Browns Ferry input decks: a TRACE deck for loss-of-coolant accident (LOCA) analysis [5] (labeled ISL-2009) and a TRAC-B deck for ATWS analysis [6] (labeled ISL-2005). The TRACE deck named BF_Final_FN_SS.inp (SNAP model editor file BF_Final_FN_SS.med) is an updated version of the Browns Ferry model implemented in the TRAC-B deck named BFGE14-SS-P100F80_R01.inp[a]. The major differences between the two input decks are:

1) Modeling of emergency core cooling systems (ECCS, hydraulic components and controls) in the LOCA deck
2) Modeling of containment components (drywell and wetwell) in the LOCA deck
3) Use of POWER component table to define reactor power in the LOCA deck
4) Use of PARCS coupled to TRAC-B to calculate reactor power in the ATWS deck
5) Controlling water level by regulating feedwater flow in the ATWS deck
6) Modeling of standby liquid control system (SLCS) in the ATWS deck

The relevant features from the two existing Browns Ferry models were combined to create the plant model for the current ATWS analysis. This was done by first deleting from the LOCA deck items 1) through 3) on the above list and then duplicating items 4) through 6) from the ATWS deck. The end result is a Browns Ferry TRACE model that resembles the TRAC-B model and is coupled to a modified PARCS model for the power calculation. The new TRACE model updated the modeling of the GE14 fuel assembly using the information from a GEH document [8]. This modification in the geometric model of the core required a new PARCS model which also incorporated control rod bank pattern, moderator density (void) history and exposure based on the GEH reference.

A detailed description of the original Browns Ferry TRACE model is available in a TRACE calculation notebook [9]. Similarly the models for the control of feedwater and standby liquid control system (SLCS) have been described in a technical report [6]. The discussion of the modifications to the TRACE LOCA deck and its adaptation for the current ATWS analysis by incorporating control systems based on the TRAC-B ATWS deck is described in another calculation notebook [10].

Examples of the work done to develop the new model are:

- Changes were made to the TRACE built-in flow controller to adjust the recirculation pump torque to provide the desired jet pump mass flow rate.
- Changes were made to the feedwater control system to adjust mass flow to maintain a desired downcomer water level.
- Changes were made to the SLCS flow control system to simulate mixing of SLCS in the lower plenum as a function of jet pump mass flow rate. The SLCS pipe connections to

---

[a] It is noted that the conversion of the TRAC-B ATWS deck [6] to the TRACE format (using TRACE to run the TRAC-B input file and then importing the 'tpr' file to SNAP) failed to convert the SVs and CBs. It turned out that was due to an error in the later versions of TRACE (in this case TRACE ver. 5.0 Patch 1). An input deck in TRACE format that was based on the TRAC-B ATWS deck was obtained from the NRC staff [7]. It was the input statements from this TRACE deck that were incorporated in the current model.

the VESSEL were modified. The junction connections were made the same as in the NRC TRACE deck (similar to the ISL-2005 model).

- A control system to set the total SRV/ADS (safety and relief valves and automatic depressurization system) valve flow area based on pressure setpoints and ADS actuation signal was developed. A PIPE component and a BREAK component for steam flow from the SRV/ADS valves were added.
- The flow area of the pressure control valve was adjusted to attain the desired steam dome pressure.
- The number of rod groups in CHAN 2 was reduced from four to three making it similar to the other 23 CHAN components.
- All emergency core cooling system (ECCS) components and other components representing the containment were deleted.
- Signal variables, control blocks, and trips from the ISL-2005 ATWS model were implemented (based on the TRACE input deck from the NRC [7]).
- The VESSEL was defined as the component surrounding the CHAN (this omission in the ISL-2009 LOCA deck resulted in unrealistic heat structure temperatures in a TRACE/PARCS coupled run).
- The power density distribution in the fuel pellet was corrected (error in the ISL-2009 LOCA deck; no power in the peripheral node of the fuel pellet).
- GE14 fuel rod geometry was implemented using information from a GEH document [8] (changes in the number of grey rods and their position, changes in length of partial rods).
- A few SNAP input check errors including changes in the GRAV card for Valve 58 (ADS) and PIPE 28 and 29 (SLCS injection) were corrected.
- Signal variables and control blocks to create data for plots were added.

A new PARCS model was developed using the ISL-2005 ATWS input deck as the starting point, to complement the TRACE model and allow for a coupled thermal-hydraulics/neutronics analysis of the ATWS transient. The fuel assemblies modeled in PARCS and the CHAN components of TRACE were based on a GE14 fuel assembly. A full core of 764 assemblies was modeled in PARCS with thermal conditions provided by a 24-channel representation of the core in the TRACE thermal-hydraulic model. The channel assignments and their placement within the four radial zones of the core are shown in Figure 1.

The PARCS model was put together from available data with the acknowledgement that improvements would have to be made in the future. The cross sections were generated for another project (for EPU and MELLA+ analyses) and used in this project because they were the best representation of GE14 fuel available. However, the description of the core, specifically the exposure and moderator density (void) history and control rod pattern, were for a core containing different GE14 fuel [8]. This is explained in more detail in the calculation notebook for PARCS. [11]

								17	17	17	17	17	17	17	25	25	25	25	25	25	25								
							17	14	16	16	16	16	16	14	22	24	24	24	24	24	22	25							
					17	17	14	16	14	15	14	15	14	15	23	22	23	22	23	22	24	22	25	25					
					17	14	16	14	15	14	15	14	15	14	22	23	22	23	22	23	22	24	22	25					
				17	14	15	14	15	14	15	14	13	12	13	21	20	21	22	23	22	23	22	23	22	25				
		17	17	14	15	14	15	14	13	12	13	12	13	12	20	21	20	21	20	21	22	23	22	23	22	25	25		
		17	14	15	14	15	14	13	12	13	12	13	12	10	18	20	21	20	21	20	21	22	23	22	23	22	25		
	17	14	16	14	15	14	13	12	13	12	13	12	10	11	19	18	20	21	20	21	20	21	22	23	22	24	22	25	
17	14	16	14	15	14	13	12	13	12	13	12	10	11	10	18	19	18	20	21	20	21	20	21	22	23	22	24	22	25
17	16	14	15	14	13	12	13	12	13	12	10	11	10	11	19	18	19	18	20	21	20	21	20	21	22	23	22	24	25
17	16	15	14	15	12	13	12	13	12	10	11	10	11	10	18	19	18	19	18	20	21	20	21	20	23	22	23	24	25
17	16	14	15	14	13	12	13	12	10	11	10	11	10	11	19	18	19	18	19	18	20	21	20	21	22	23	22	24	25
17	16	15	14	13	12	13	12	10	11	10	11	10	11	10	18	19	18	19	18	19	18	20	21	20	21	22	23	24	25
17	16	14	15	12	13	12	10	11	10	11	10	11	10	5	6	18	19	18	19	18	19	18	20	21	20	23	22	24	25
17	14	15	14	13	12	10	11	10	11	10	11	10	4	11	19	7	18	19	18	19	18	19	18	20	21	22	23	22	25
17	14	15	14	13	12	10	11	10	11	10	11	10	3	11	19	8	18	19	18	19	18	19	18	20	21	22	23	22	25
17	16	14	15	12	13	12	10	11	10	11	10	11	10	2	9	18	19	18	19	18	19	18	20	21	20	23	22	24	25
17	16	15	14	13	12	13	12	10	11	10	11	10	11	10	18	19	18	19	18	19	18	20	21	20	21	22	23	24	25
17	16	14	15	14	13	12	13	12	10	11	10	11	10	11	19	18	19	18	19	18	20	21	20	21	22	23	22	24	25
17	16	15	14	15	12	13	12	13	12	10	11	10	11	10	18	19	18	19	18	20	21	20	21	20	23	22	23	24	25
17	16	14	15	14	13	12	13	12	13	12	10	11	10	11	19	18	19	18	20	21	20	21	20	21	22	23	22	24	25
17	14	16	14	15	14	13	12	13	12	13	12	10	11	10	18	19	18	20	21	20	21	20	21	22	23	22	24	22	25
	17	14	16	14	15	14	13	12	13	12	13	12	10	11	19	18	20	21	20	21	20	21	22	23	22	24	22	25	
		17	14	15	14	15	14	13	12	13	12	13	12	10	18	20	21	20	21	20	21	22	23	22	23	22	25		
		17	17	14	15	14	15	14	13	12	13	12	13	12	20	21	20	21	20	21	22	23	22	23	22	25	25		
				17	14	15	14	15	14	15	14	13	12	13	21	20	21	22	23	22	23	22	23	22	25				
					17	14	16	14	15	14	15	14	15	14	22	23	22	23	22	23	22	24	22	25					
					17	17	14	16	14	15	14	15	14	15	23	22	23	22	23	22	24	22	25	25					
							17	14	16	16	16	16	16	14	22	24	24	24	24	24	22	25							
								17	17	17	17	17	17	17	25	25	25	25	25	25	25								

Figure 1. Channel Layout in the Core

3. ANALYSIS OF A BWR ATWS

The MSIV closure ATWS was simulated by a sequence of TRACE V5.184 runs:

- TRACE stand-alone steady-state to initialize the BWR model.
- TRACE/PARCS coupled case to initialize an integrated run.
- TRACE/PARCS coupled transient run for the MSIV closure ATWS.

The steady state was run with both power and flow at 100% of nominal and with the control rod pattern corresponding to GEH's analysis of beginning-of-cycle 20 (the same statepoint for which the (E, VH) vectors were available). The corresponding value of k_{eff} was 1.014. Table 1 presents the results of the TRACE/PARCS coupled steady-state calculation by examing some key thermal-hydraulic parameters. In general the agreement with the Browns Ferry FSAR [12] and the previous TRACE calculation [6] is excellent.

Table 1 Comparison of Steady-State Thermal-Hydraulic Parameters

Parameter	Units	TRACE Value	Reference Value
Core Power	MWt	3293	3293 [12]
Steam Dome Pressure	MPa	7.173	7.170 [6]
Main Steam Line Flow	kg/s	1688	1686 [12]
Total Core Flow	kg/s	12920	12920 [12]
Feedwater Flow	kg/s	1688	1686 [12]
Feedwater Temperature	K	464.8	464.8 [12]
Downcomer Level	m	11.13	11.13 [6]

The steady-state core-average axial power profile calculated by PARCS is shown in Figure 2 and the core-average radial power distribution is shown in Figure 4. The results are seen to be reasonable in spite of the fact that the model was not based on consistent cross section data (see Section 2).

The axial power distribution is influenced primarily by the fuel bundle design, the (E, VH) distribution, and the void distribution. The latter is shown in Fig. 3. The radial power distribution, shown in Fig. 4, is for the full core. Since the core actually has one-eighth core symmetry these results also demonstrate the degree of convergence of the fuel assembly power, which is excellent. For the NW quadrant the control rod bank that is only partially (~28%) withdrawn is highlighted in yellow and the bank that is almost fully (~76%) withdrawn is marked in green. All other banks in the quadrant are fully withdrawn.

The transient was initiated by the closure of the MSIVs and the calculation was carried out for almost 800 seconds. At that time the boron concentration in the core is greater than the concentration used for branch points in generating the cross sections (the cross section data did not have a branch for boron concentration above 400 ppm). Hence, the calculation has limited validity at that point in time. The sequence of events is given in Table 2 and graphs of time dependent quantities are given in Figures 5-27.

After MSIV closure the short-term (up to approximately two minutes) response of the reactor is driven by three effects: recirculation pump trip (RPT), safety and relief valve (SRV) operation, and loss of feedwater (FW) heating. The RPT is required under ATWS conditions in order to reduce flowrate, increase void fraction and reduce power. It is initiated by high steam dome pressure for an MSIV closure event. The opening and closing of the SRVs to relieve the pressure built up in the steamline changes the void fraction in the core and affects flowrates. The loss of FW heating results from the isolation of the system.

Table 2 Accident Scenario for Isolation ATWS (MSIV Closure)

Event	Action	Timing /Trip Setpoint
MSIV closure	Initiated by trip	0.0 s (input)
Recirculation pump trip (RPT)	RPT initiated by high steam dome pressure	P = 8.101 MPa (automatic)
Loss of FW heating	FW temperature ramped down to 321.7 K	Following MSIV closure (input)
SRV actuation	SRV lifting	13 valves in 3 banks (automatic)
Water level (WL) to TAF	WL control strategy switched from TAF +16.11' (normal operating condition) to TAF	124 s after RPT (input)
Boron injection[1]	SLC initiated	214 s (input)
Emergency blowdown	ADS actuation	600 s (input)
[1] GEH assumed boron injection initiated at 124 s but the analysis additionally assumed 30 s SLC transit delay and 60 s mixing delay (124+30+60 = 214)		

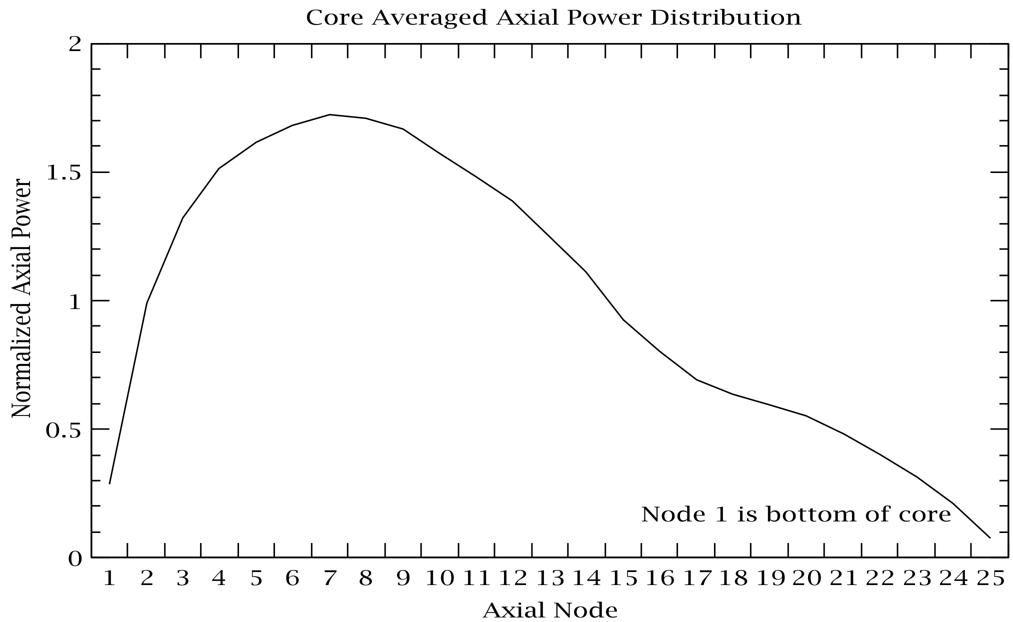

Figure 2. Core-Average Axial Power

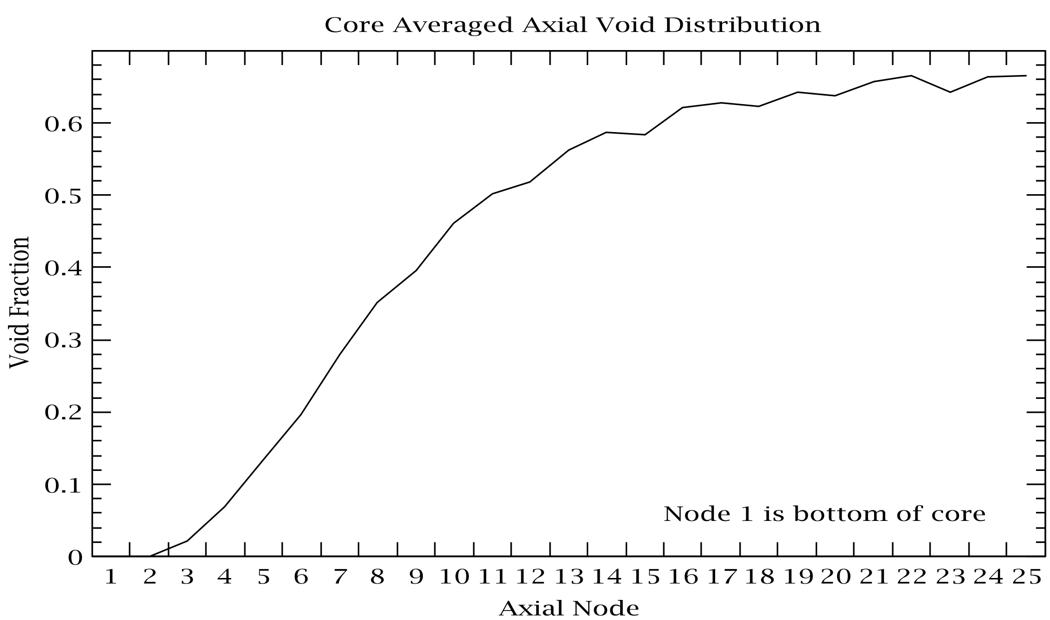

Figure 3. Core-Average Void Profile

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1									0.2944	0.3963	0.4667	0.6373	0.6112	0.7746	0.8094
2								0.3421	0.5630	0.7134	0.7961	0.9924	1.0477	1.1665	1.1170
3						0.2275	0.4574	0.5685	0.7384	0.8625	0.9799	1.1798	1.3191	1.3729	1.3458
4						0.3967	0.5827	0.7324	0.8581	0.9884	1.1951	1.3322	1.4808	1.5434	1.5686
5					0.3894	0.6499	0.7352	0.8632	0.9100	1.0099	1.2886	1.4516	1.5155	1.5438	1.5475
6			0.2570	0.4008	0.6506	0.8057	0.9319	0.9845	0.9933	1.0664	1.2674	1.4012	1.4703	1.5197	1.5590
7			0.4624	0.5851	0.7362	0.9323	1.0579	1.1385	1.1488	1.1813	1.2639	1.3296	1.3730	1.4290	1.6067
8		0.3429	0.5696	0.7336	0.8639	0.9848	1.1386	1.1627	1.1475	1.1630	1.2021	1.2059	1.1598	1.2840	1.4009
9	0.2947	0.5636	0.7391	0.8589	0.9105	0.9935	1.1487	1.1475	1.0354	1.0094	1.0989	1.0398	0.7066	0.6875	1.1018
10	0.3963	0.7136	0.8629	0.9897	1.0101	1.0663	1.1812	1.1629	1.0093	0.9898	1.0666	1.0614	0.6228	0.6205	0.9728
11	0.4667	0.7961	0.9798	1.1950	1.2882	1.2669	1.2634	1.2018	1.0988	1.0666	1.1423	1.0817	0.9403	0.9001	0.9443
12	0.6374	0.9918	1.1790	1.3310	1.4505	1.4003	1.3287	1.2053	1.0395	1.0615	1.0817	1.0443	0.8968	0.8521	0.8928
13	0.6099	1.0464	1.3177	1.4792	1.5138	1.4688	1.3714	1.1588	0.7061	0.6225	0.9401	0.8967	0.5462	0.5049	0.7893
14	0.7724	1.1645	1.3711	1.5415	1.5419	1.5176	1.4272	1.2824	0.6869	0.6198	0.8998	0.8511	0.5048	0.4987	0.7836
15	0.8081	1.1152	1.3438	1.5665	1.5453	1.5568	1.6039	1.3990	1.1007	0.9721	0.9438	0.8926	0.7895	0.7836	0.8071
16	0.8080	1.1152	1.3437	1.5664	1.5453	1.5567	1.6039	1.3990	1.1006	0.9721	0.9438	0.8926	0.7895	0.7836	0.8071
17	0.7724	1.1644	1.3708	1.5412	1.5416	1.5174	1.4270	1.2823	0.6868	0.6198	0.8998	0.8511	0.5048	0.4987	0.7836
18	0.6097	1.0462	1.3173	1.4788	1.5135	1.4684	1.3712	1.1586	0.7060	0.6224	0.9401	0.8967	0.5461	0.5049	0.7894
19	0.6372	0.9915	1.1786	1.3305	1.4500	1.3999	1.3283	1.2050	1.0393	1.0613	1.0816	1.0442	0.8968	0.8521	0.8928
20	0.4664	0.7956	0.9793	1.1943	1.2876	1.2664	1.2630	1.2014	1.0985	1.0664	1.1422	1.0817	0.9402	0.9002	0.9443
21	0.3960	0.7130	0.8622	0.9889	1.0094	1.0657	1.1806	1.1624	1.0090	0.9896	1.0665	1.0613	0.6227	0.6205	0.9729
22	0.2942	0.5627	0.7381	0.8578	0.9095	0.9926	1.1481	1.1470	1.0351	1.0092	1.0988	1.0398	0.7066	0.6876	1.1020
23		0.3420	0.5683	0.7321	0.8627	0.9838	1.1378	1.1621	1.1472	1.1628	1.2020	1.2059	1.1600	1.2842	1.4012
24			0.4572	0.5824	0.7347	0.9311	1.0570	1.1379	1.1484	1.1811	1.2638	1.3297	1.3731	1.4294	1.6072
25			0.2273	0.3966	0.6493	0.8046	0.9312	0.9839	0.9930	1.0662	1.2674	1.4013	1.4706	1.5201	1.5596
26					0.3885	0.6491	0.7347	0.8628	0.9098	1.0098	1.2886	1.4517	1.5159	1.5443	1.5480
27						0.3963	0.5824	0.7321	0.8580	0.9884	1.1952	1.3324	1.4812	1.5440	1.5692
28						0.2273	0.4572	0.5683	0.7383	0.8625	0.9800	1.1801	1.3195	1.3734	1.3464
29								0.3421	0.5631	0.7134	0.7963	0.9926	1.0480	1.1669	1.1175
30									0.2945	0.3962	0.4668	0.6373	0.6114	0.7747	0.8098
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15

Figure 4. Core-Average Assembly Power

16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	
0.8092	0.7742	0.6110	0.6369	0.4665	0.3960	0.2943									1
1.1168	1.1661	1.0474	0.9920	0.7958	0.7130	0.5628	0.3420								2
1.3457	1.3727	1.3188	1.1794	0.9796	0.8621	0.7379	0.5682	0.4571	1 0.2272						3
1.5684	1.5432	1.4805	1.3318	1.1947	0.9880	0.8577	0.7320	0.5823	3 0.3963						4
1.5474	1.5437	1.5153	1.4512	1.2882	1.0095	0.9096	0.8627	0.7347	7 0.6492	0.3886					5
1.5590	1.5195	1.4701	1.4009	1.2671	1.0660	0.9929	0.9840	0.9313	3 0.8049	0.6497	0.3971	0.2278			6
1.6066	1.4289	1.3728	1.3294	1.2636	1.1810	1.1484	1.1380	1.0573	3 0.9316	0.7354	0.5833	0.4587			7
1.4008	1.2839	1.1597	1.2057	1.2019	1.1627	1.1473	1.1623	1.1382	2 0.9844	0.8636	0.7334	0.5706	0.3456		8
1.1018	0.6875	0.7065	1.0397	1.0988	1.0092	1.0352	1.1473	1.1486	6 0.9932	0.9105	0.8594	0.7406	0.5689	0.3143	9
0.9728	0.6205	0.6227	1.0613	1.0665	0.9897	1.0092	1.1627	1.1811	1 1.0663	1.0103	0.9904	0.8644	0.7162	0.3983	10
0.9442	0.9001	0.9402	1.0817	1.1422	1.0665	1.0987	1.2017	1.2635	5 1.2671	1.2885	1.1955	0.9807	0.7972	0.4677	11
0.8928	0.8520	0.8968	1.0442	1.0816	1.0614	1.0395	1.2053	1.3287	7 1.4004	1.4507	1.3313	1.1795	0.9925	0.6380	12
0.7893	0.5049	0.5461	0.8967	0.9402	0.6225	0.7060	1.1588	1.3715	5 1.4688	1.5139	1.4793	1.3179	1.0466	0.6100	13
0.7836	0.4987	0.5048	0.8511	0.8998	0.6198	0.6869	1.2824	1.4272	2 1.5177	1.5419	1.5415	1.3711	1.1645	0.7725	14
0.8071	0.7836	0.7895	0.8926	0.9438	0.9722	1.1007	1.3991	1.6040	0 1.5568	1.5453	1.5664	1.3437	1.1151	0.8079	15
0.8071	0.7836	0.7895	0.8926	0.9438	0.9721	1.1007	1.3990	1.6038	8 1.5567	1.5452	1.5662	1.3436	1.1150	0.8079	16
0.7836	0.4987	0.5048	0.8511	0.8998	0.6198	0.6868	1.2823	1.4270	0 1.5174	1.5416	1.5411	1.3707	1.1640	0.7720	17
0.7894	0.5049	0.5462	0.8967	0.9401	0.6225	0.7060	1.1587	1.3712	2 1.4684	1.5134	1.4787	1.3172	1.0459	0.6096	18
0.8929	0.8521	0.8969	1.0443	1.0817	1.0615	1.0394	1.2051	1.3284	4 1.3999	1.4499	1.3303	1.1783	0.9912	0.6369	19
0.9444	0.9002	0.9404	1.0818	1.1423	1.0665	1.0986	1.2015	1.2630	0 1.2664	1.2875	1.1942	0.9791	0.7954	0.4663	20
0.9730	0.6206	0.6228	1.0615	1.0667	0.9898	1.0092	1.1626	1.1807	7 1.0657	1.0093	0.9888	0.8621	0.7128	0.3959	21
1.1020	0.6877	0.7067	1.0400	1.0991	1.0094	1.0354	1.1473	1.1483	3 0.9928	0.9096	0.8578	0.7380	0.5627	0.2942	22
1.4013	1.2844	1.1602	1.2062	1.2024	1.1632	1.1476	1.1625	1.1381	1 0.9840	0.8627	0.7320	0.5682	0.3419		23
1.6073	1.4295	1.3735	1.3301	1.2643	1.1816	1.1489	1.1384	1.0575	5 0.9314	0.7349	0.5824	0.4572			24
1.5597	1.5203	1.4710	1.4019	1.2681	1.0669	0.9936	0.9846	0.9317	7 0.8050	0.6494	0.3966	0.2273			25
1.5481	1.5446	1.5164	1.4524	1.2895	1.0108	0.9108	0.8638	0.7354	4 0.6497	0.3887					26
1.5693	1.5442	1.4817	1.3333	1.1963	0.9899	0.8595	0.7336	0.5833	3 0.3969						27
1.3465	1.3736	1.3201	1.1810	0.9815	0.8647	0.7407	0.5707	0.4588	8 0.2278						28
1.1175	1.1671	1.0485	0.9936	0.7978	0.7166	0.5691	0.3457								29
0.8097	0.7750	0.6117	0.6381	0.4680	0.3985	0.3144									30
16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	

Figure 4 (cont'd). Core-Average Assembly Power

At longer times (>2 min) the behavior of the reactor is also influenced by the reduction in water level (WL), the injection of boron, and the actuation of the automatic depressurization system (ADS). The reduction in WL is done on a time signal and represents an operator action consistent with emergency operating procedures. The reduction, which can be anywhere from top of active fuel (TAF) plus five feet to TAF minus two feet, in this case it is to TAF. The boron injection from the standby liquid control system (SLC) does not begin to mix with water entering the core until 214 s and again reflects an action that would be initiated by an operator. Lastly, the calculation assumed ADS would start at 10 minutes into the event in order to understand how TRACE would calculate the effect ADS. Normally, ADS would be the result of the suppression pool temperature reaching a trip setpoint.

Figure 5 shows power vs. time. The initial power surge is due to the pressure surge (Fig. 6) after MSIV closure. This in turn causes the collapse of steam voids (Fig. 15) and the addition of reactivity (Figs. 24 and 27)—almost to the point of prompt-criticality. The resulting fuel temperature (Fig. 22) and negative Doppler reactivity (Fig. 26) cause the power to peak and then decline. The behavior during the next ~200 seconds is oscillatory due to the opening and closing of the SRVs (Fig. 10) and the changes in void fraction (Figs. 15 and 27). There is also an effect from the decrease in FW temperature (Fig. 9) which lasts until FW stops after 124 seconds (Fig. 8) when the control of water level is switched from TAF+16.11' to TAF.

The effect of the control system which simulates the operator action to bring the WL down to TAF is seen in Fig. 7. This becomes important after ~150 seconds and is one of the reasons that the power drops off after that time (Fig. 5). Another reason is the actuation of the SLC at 214 seconds (Figs. 17-19, 21 and 25). The final effect is the actuation of the ADS at 600 seconds (Fig. 10). This causes the pressure to fall (Fig. 6) and additional voiding throughout the system (Figs. 13-16).

The TRACE calculation was done on a PC with an Intel Core 2 Duo CPU (T8300) at 2.4 GHz and 4 GB of ram. The TRACE executable was run in a DOS window under the Microsoft Vista operating system (32-bit). The CPU time and time step size for the ATWS transient are shown in Figs. 28 and 29 respectively. An informal testing has shown a factor of 1.5 increase in computational speed when the same calculation was done on a system with an Intel i-7 CPU (930 at 2.93 GHz).

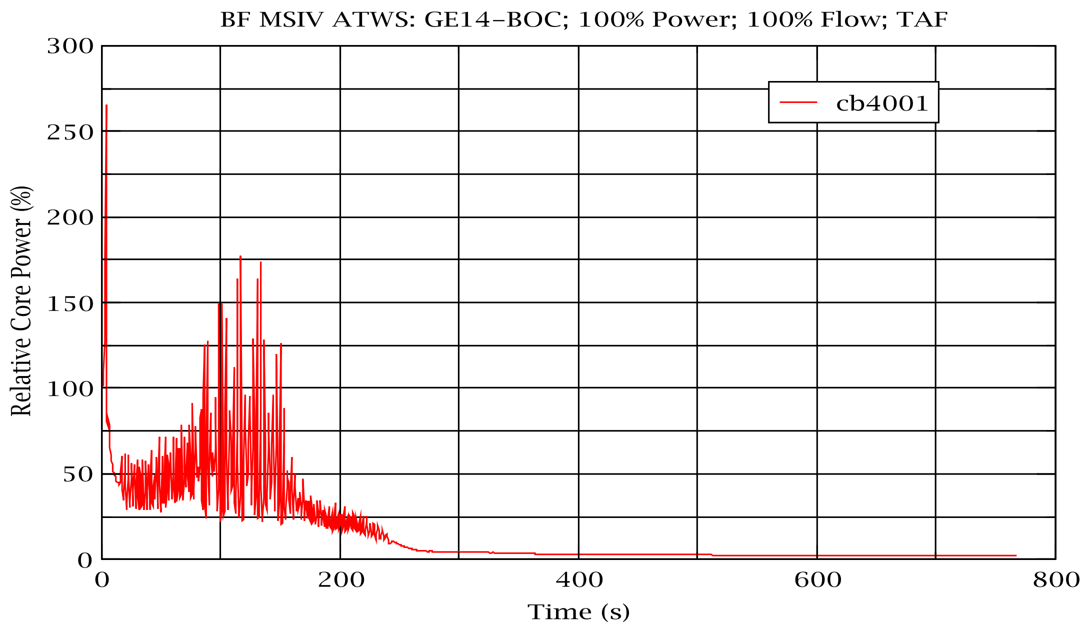

Figure 5. Relative Core Power

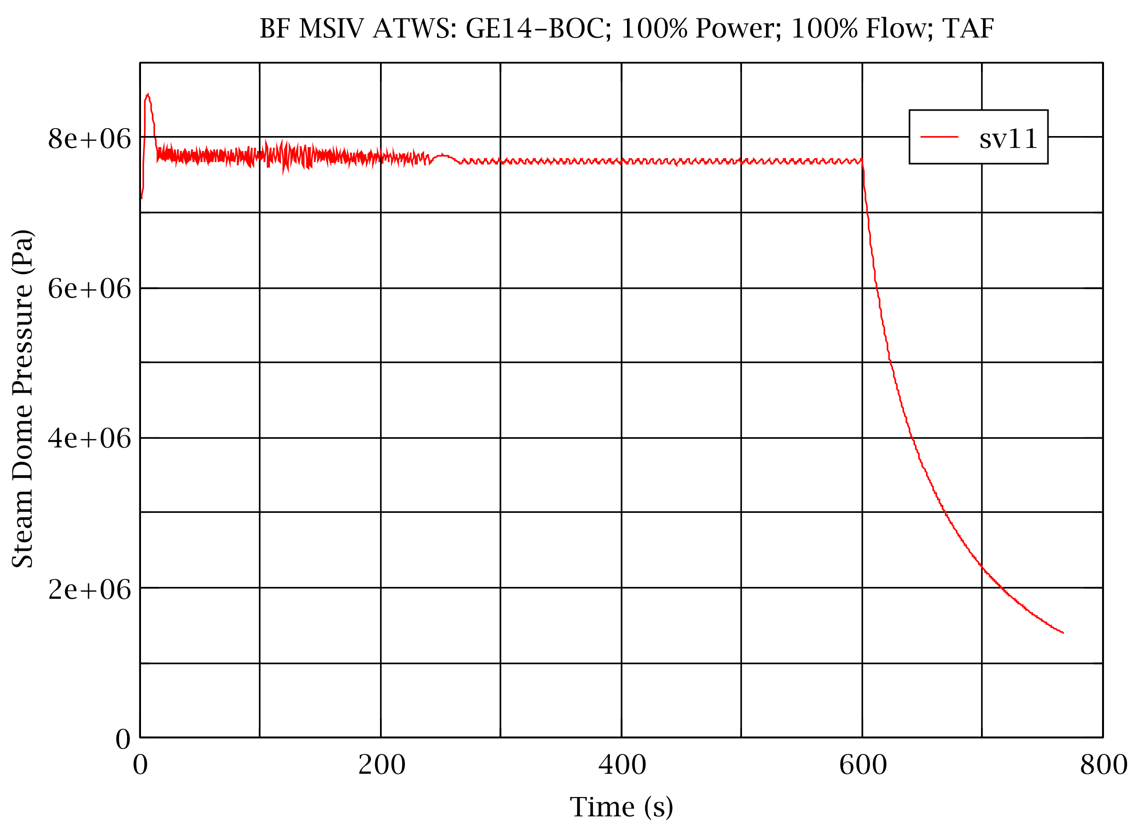

Figure 6. Steam Dome Pressure

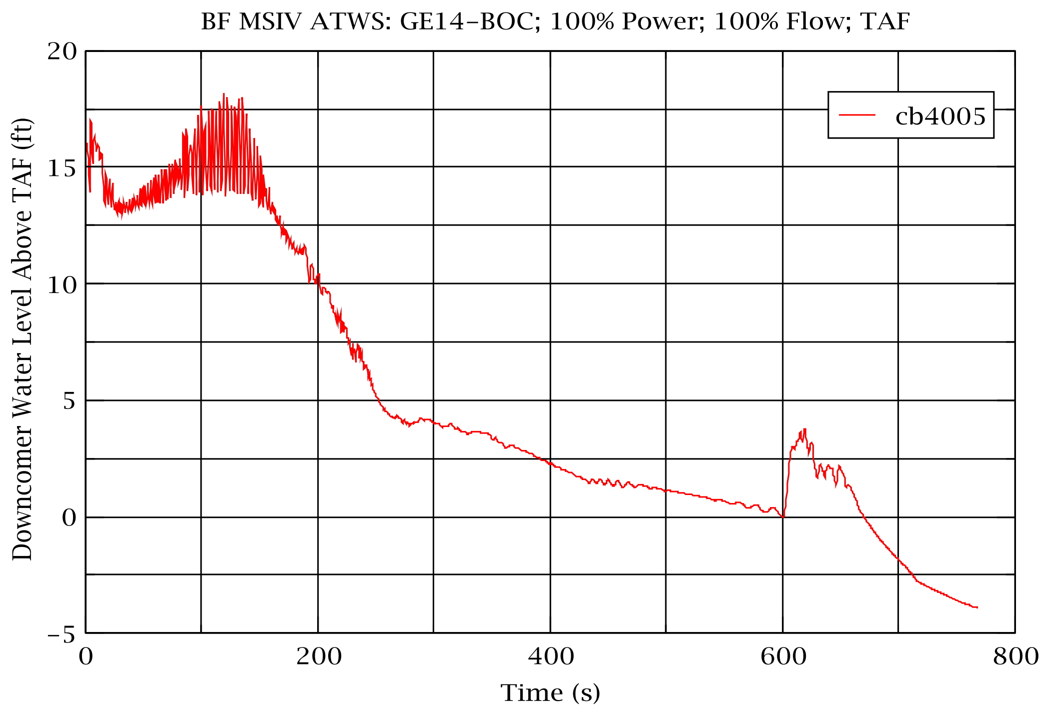

Figure 7. Collapsed Downcomer Water Level

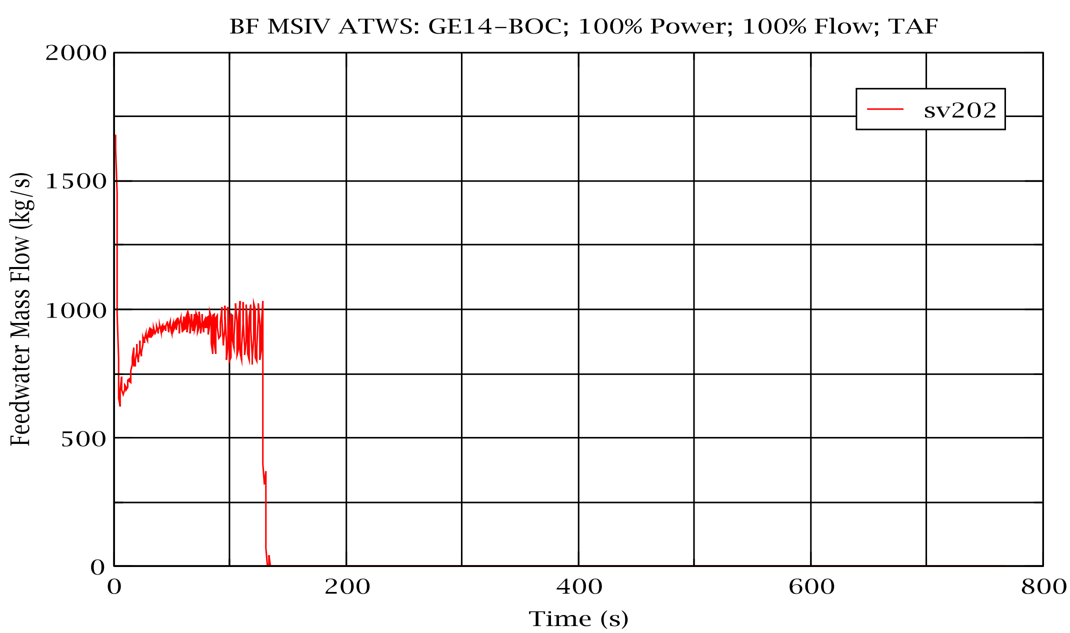

Figure 8. Feedwater Mass Flow

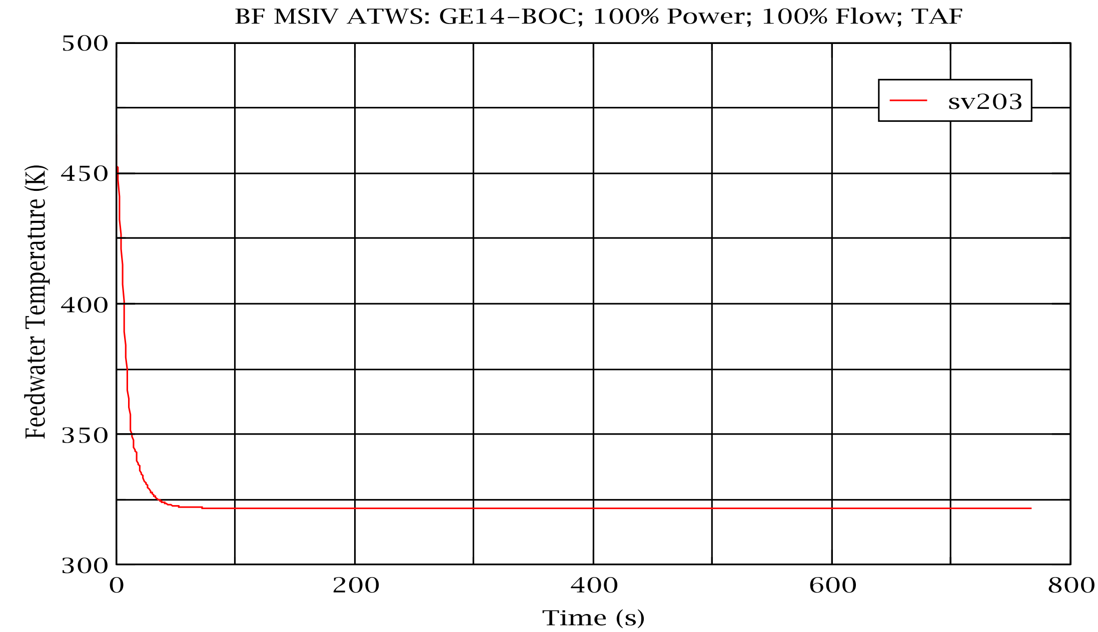

Figure 9. Feedwater temperature

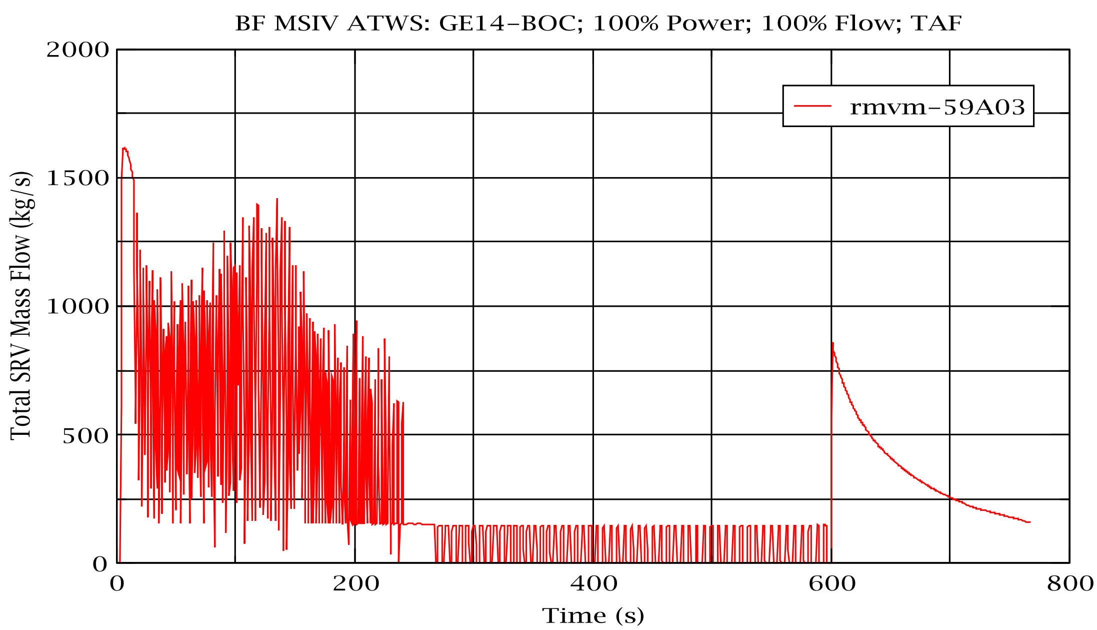

Figure 10. SRV Mass Flow

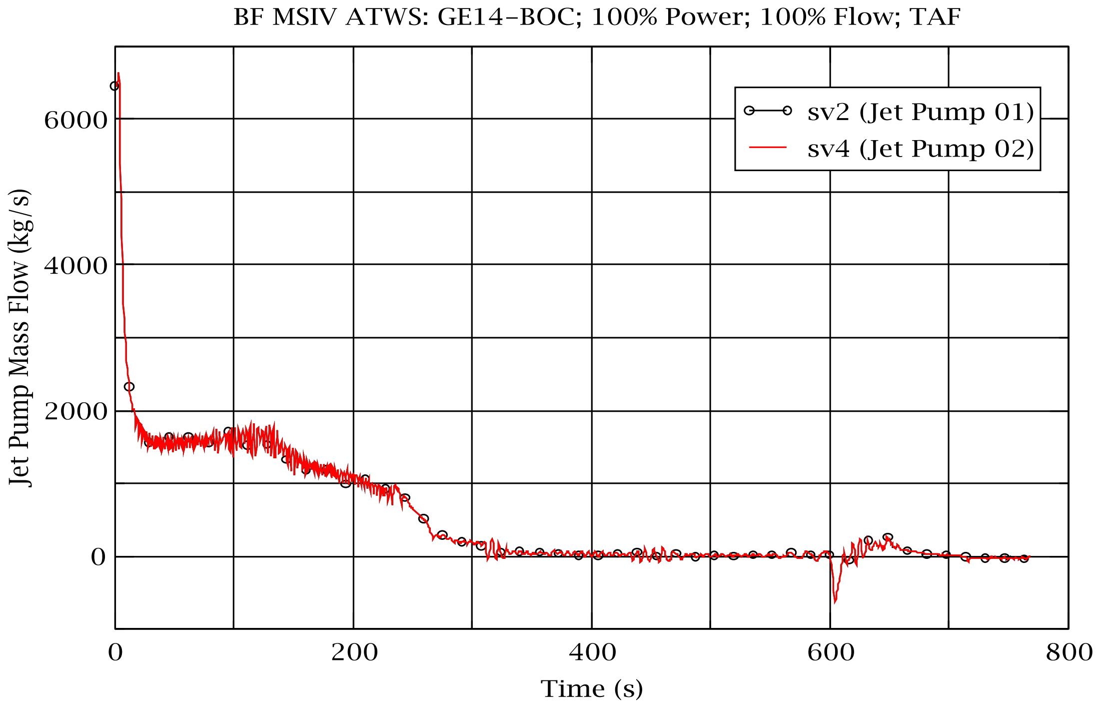

Figure 11. Jet Pump Mass Flow

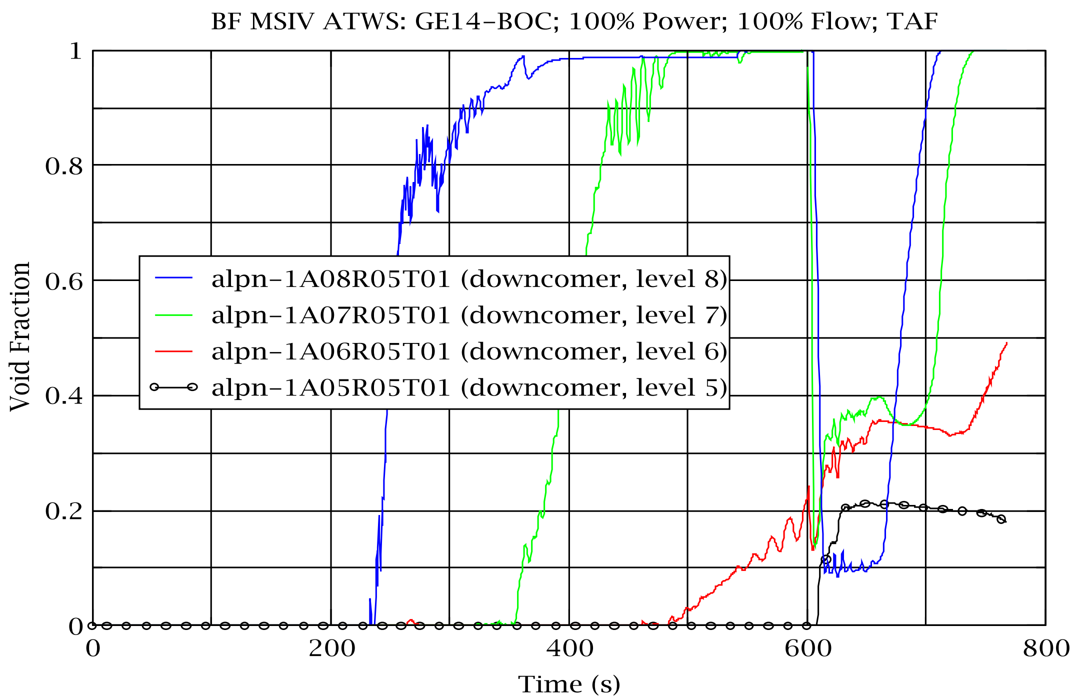

Figure 12. Void Fraction (Downcomer)

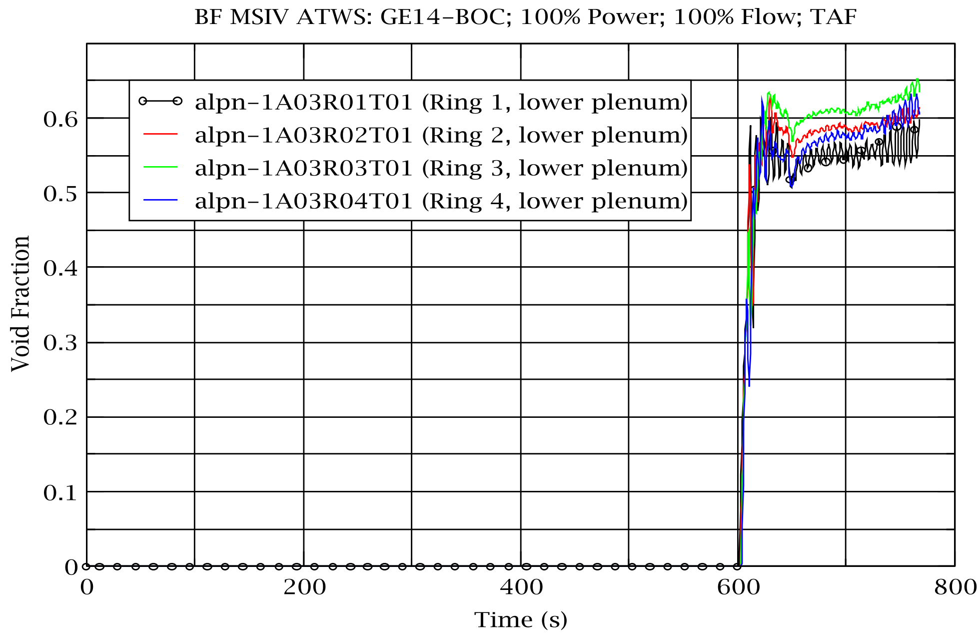

Figure 13. Void Fraction (Lower Plenum)

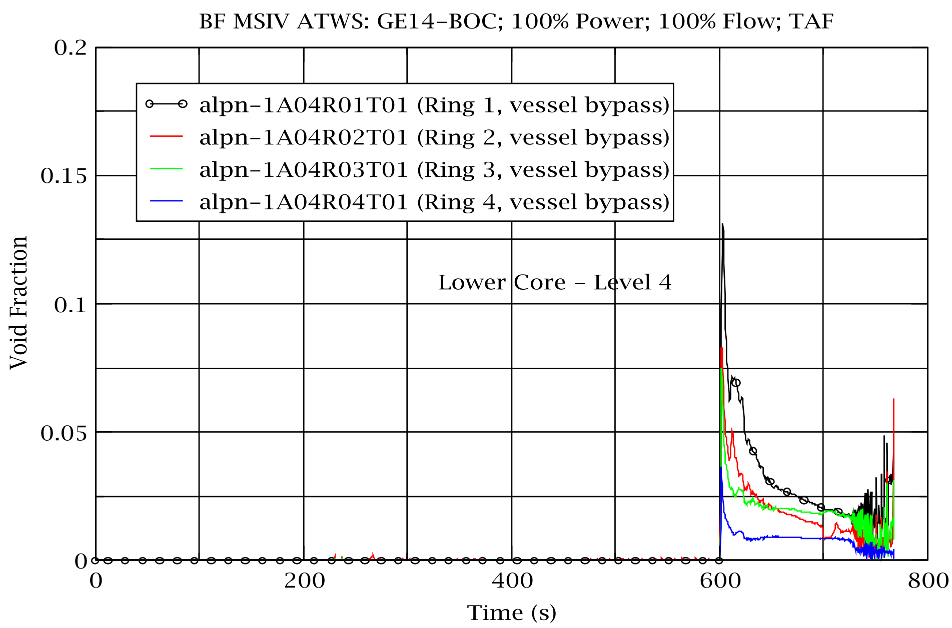

Figure 14. Void Fraction (Bypass)

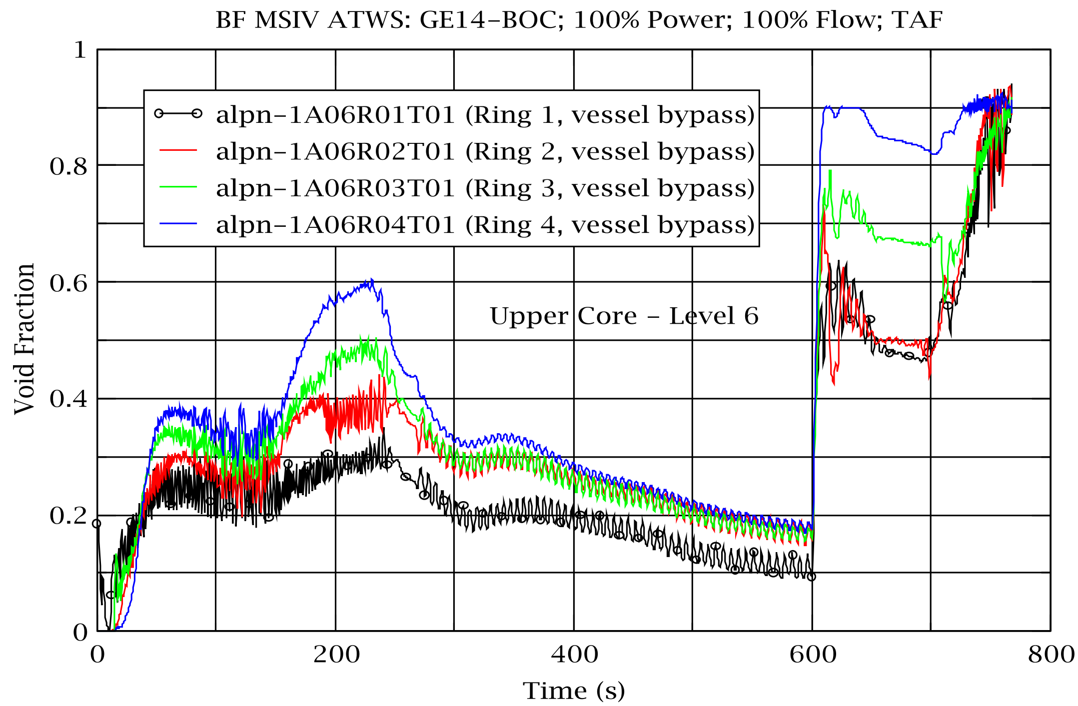

Figure 15. Void Fraction (Bypass)

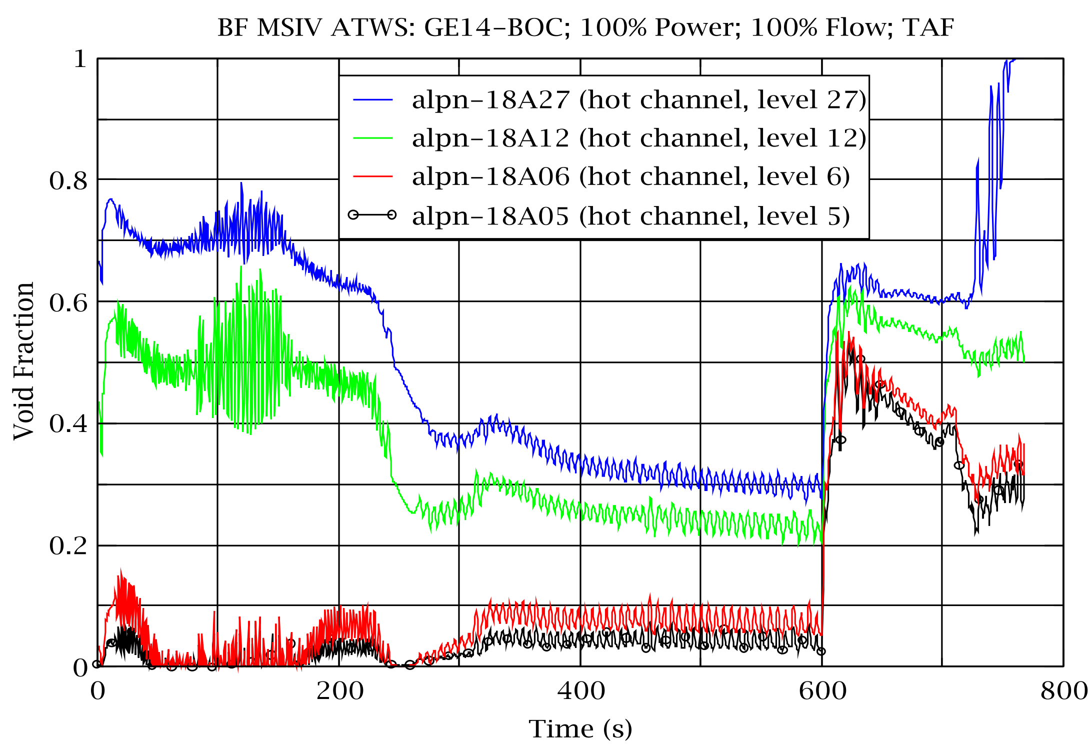

Figure 16. Void Fraction (Hot Channel)

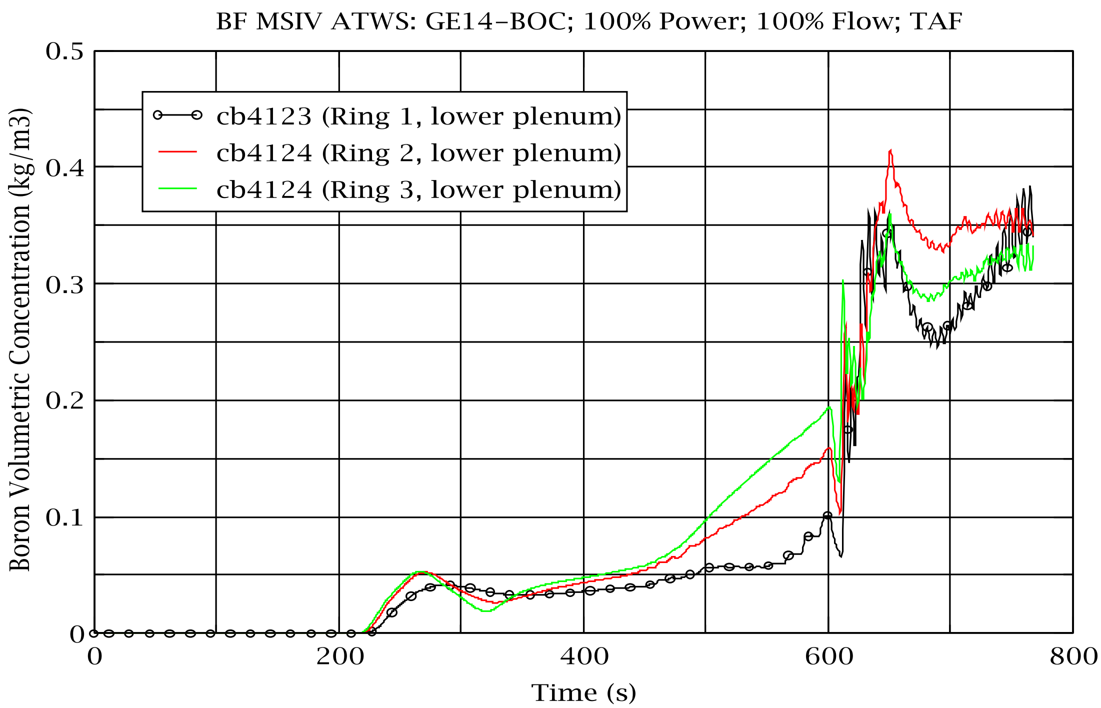

Figure 17. Boron Concentration (Lower Plenum)

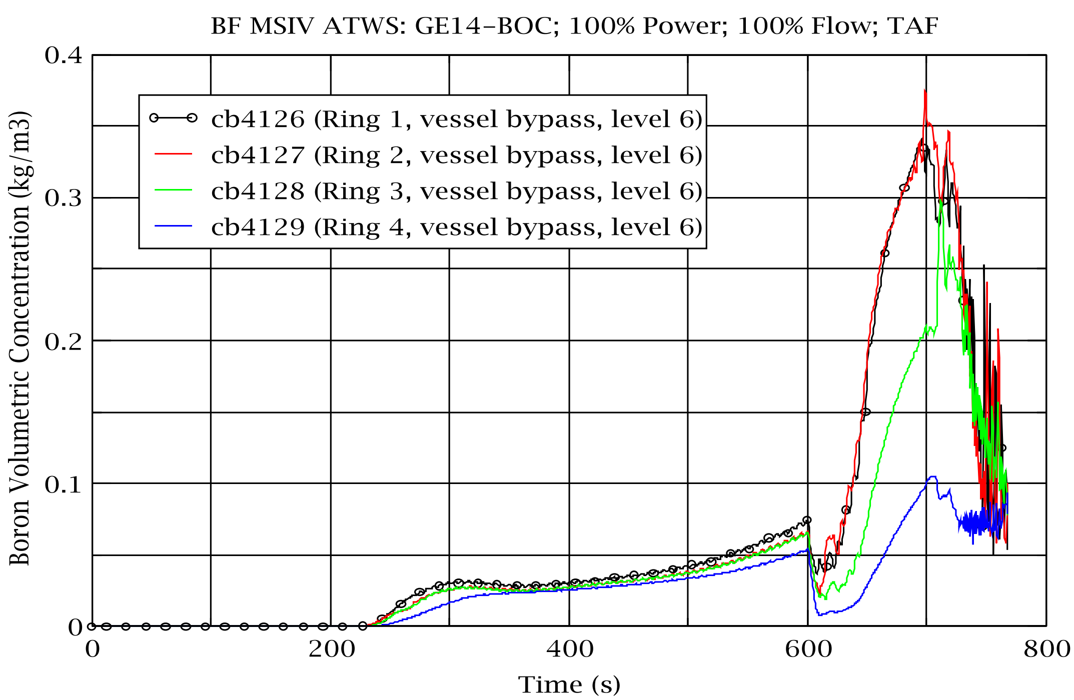

Figure 18. Boron Concentration (Bypass)

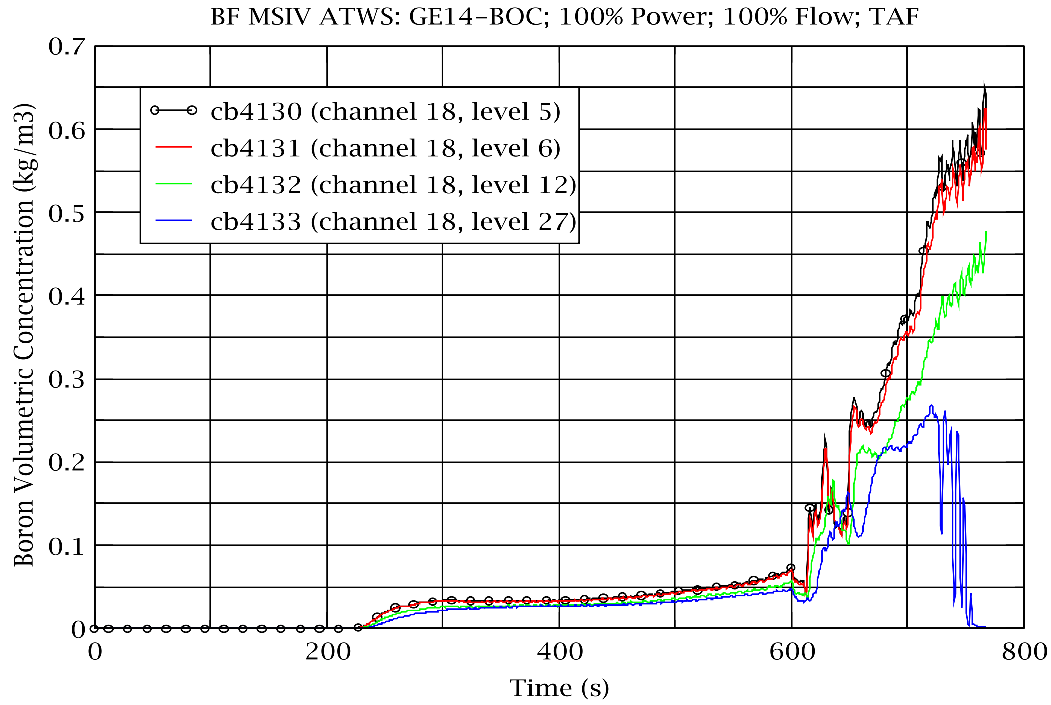

Figure 19. Boron Concentration (Core)

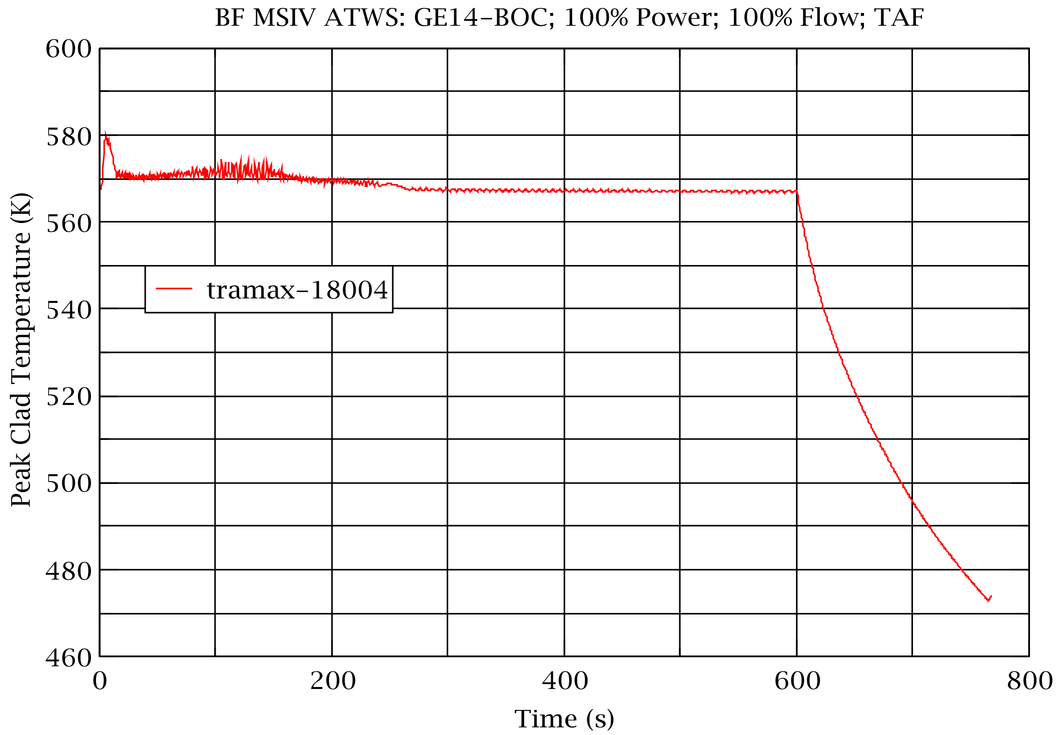

Figure 20. Peak Clad Temperature

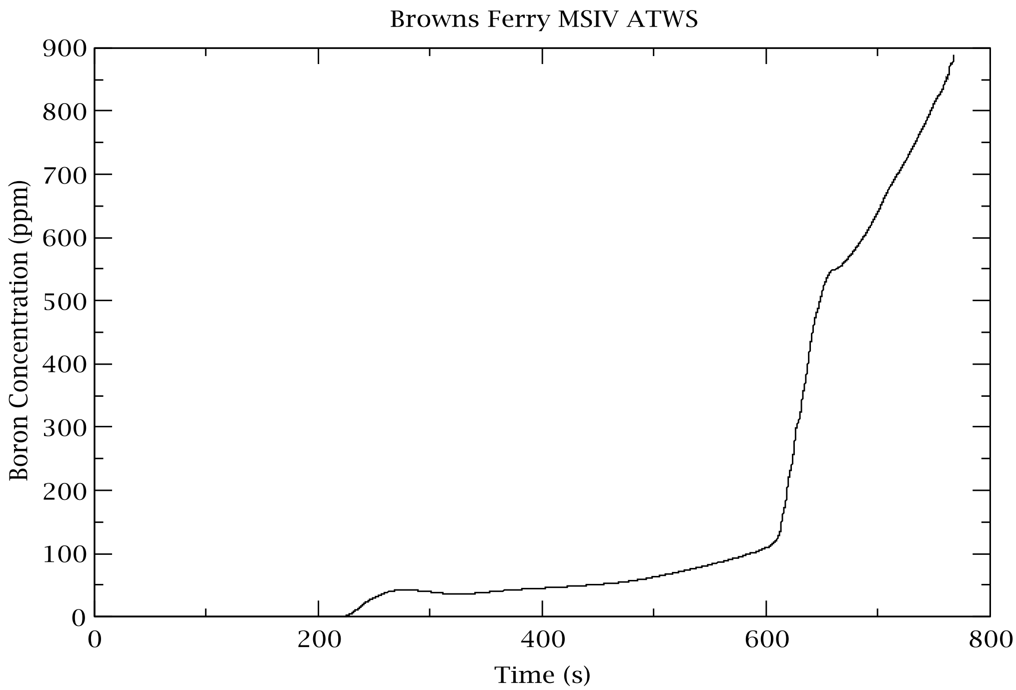

Figure 21. Core-Average Boron Concentration

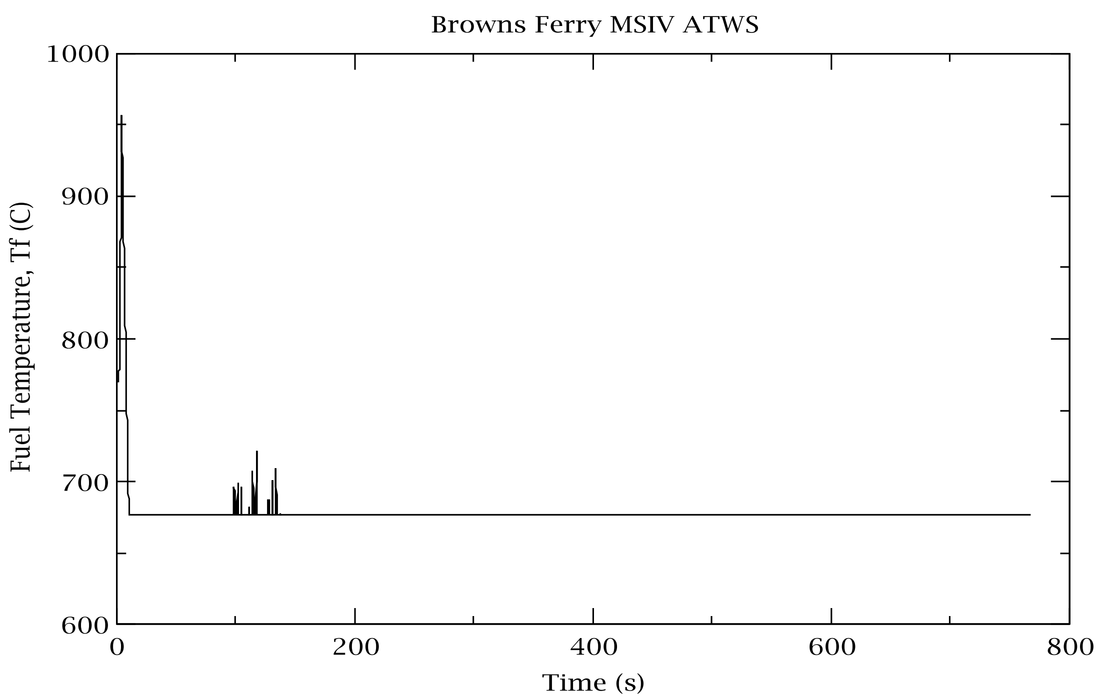

Figure 22. Core-Average Fuel Temperature

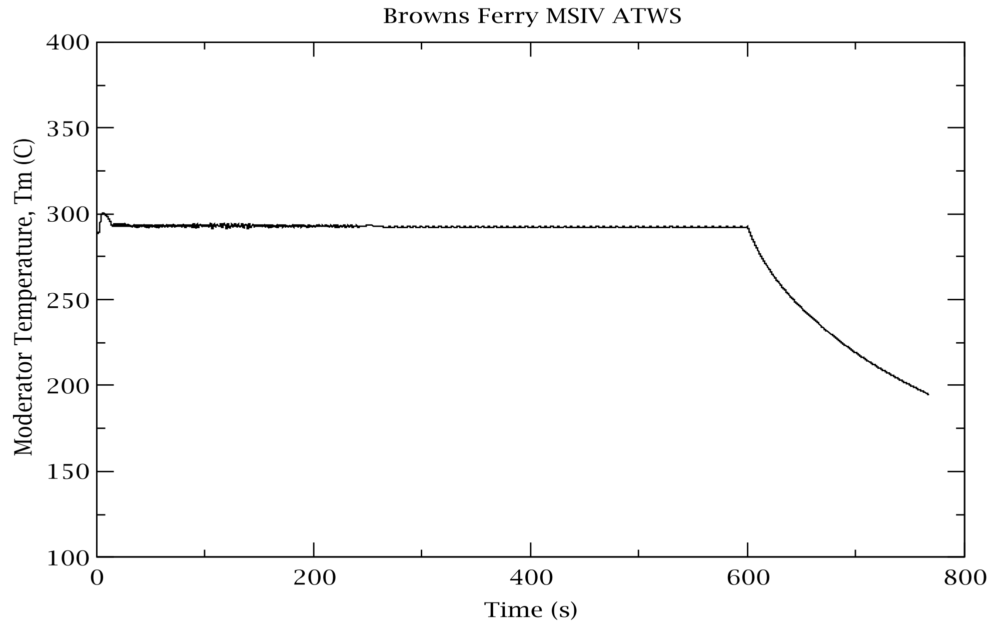

Figure 23. Core-Average Moderator Temperature

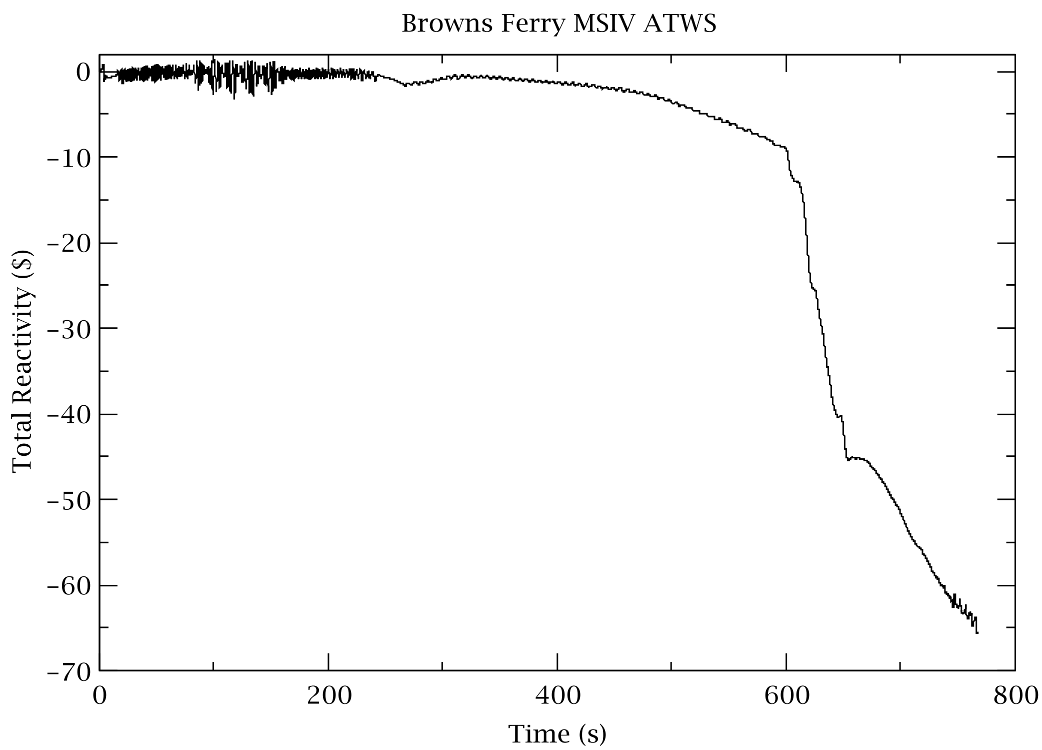

Figure 24. Core Reactivity

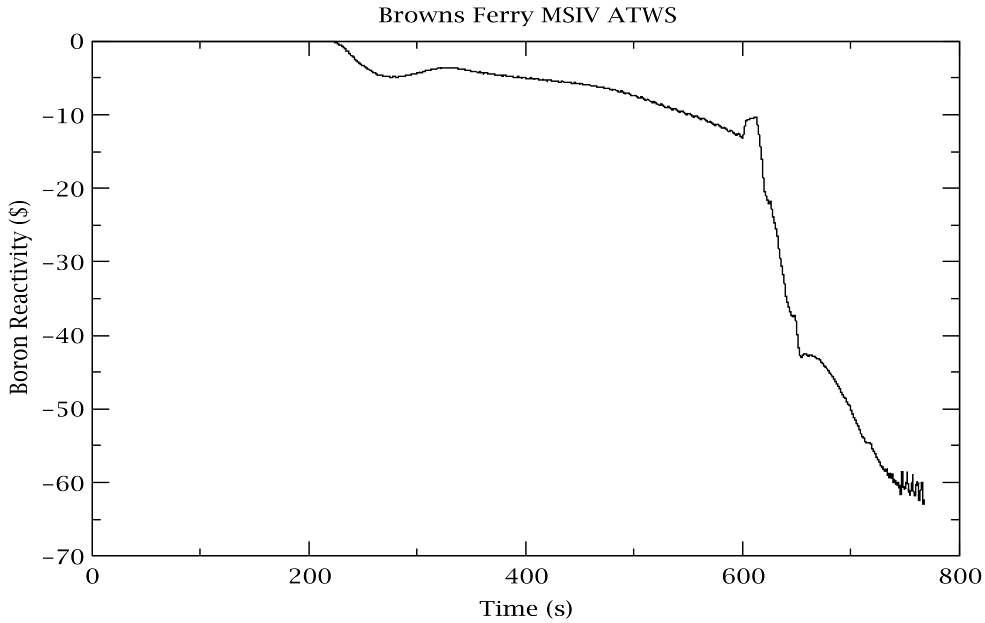

Figure 25. Boron Reactivity

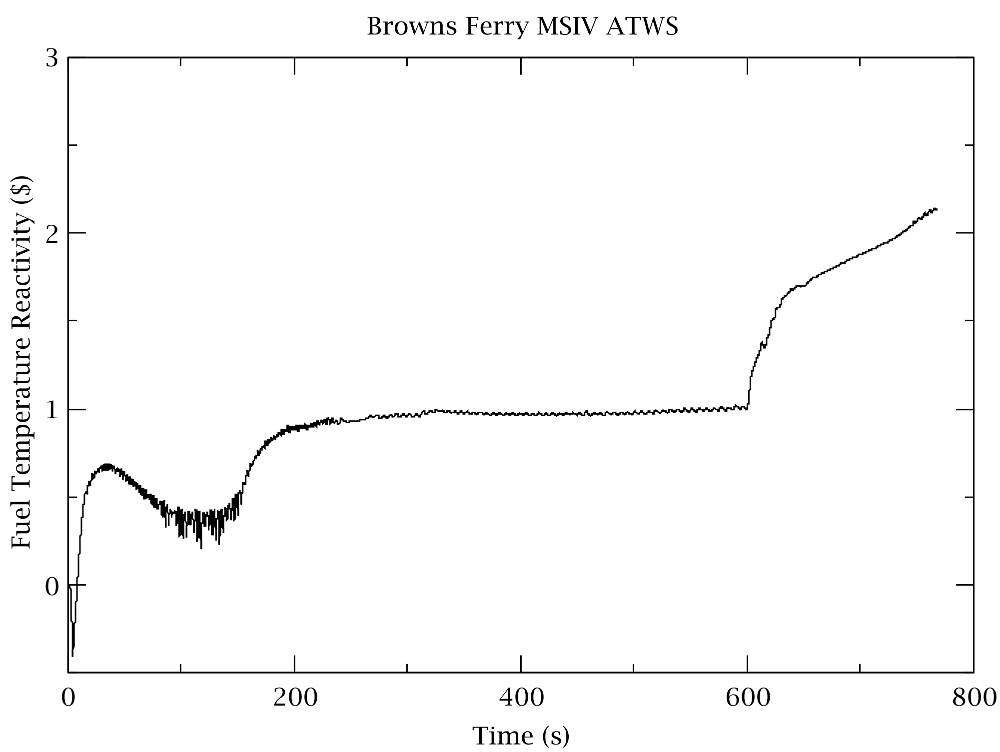

Figure 26. Fuel Temperature Reactivity

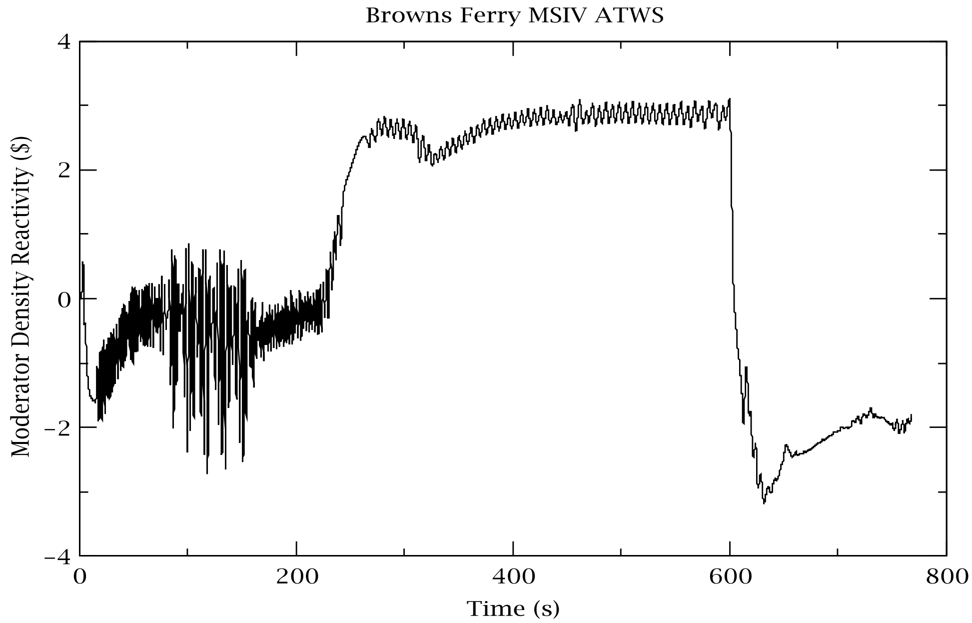

Figure 27. Moderator Density Reactivity

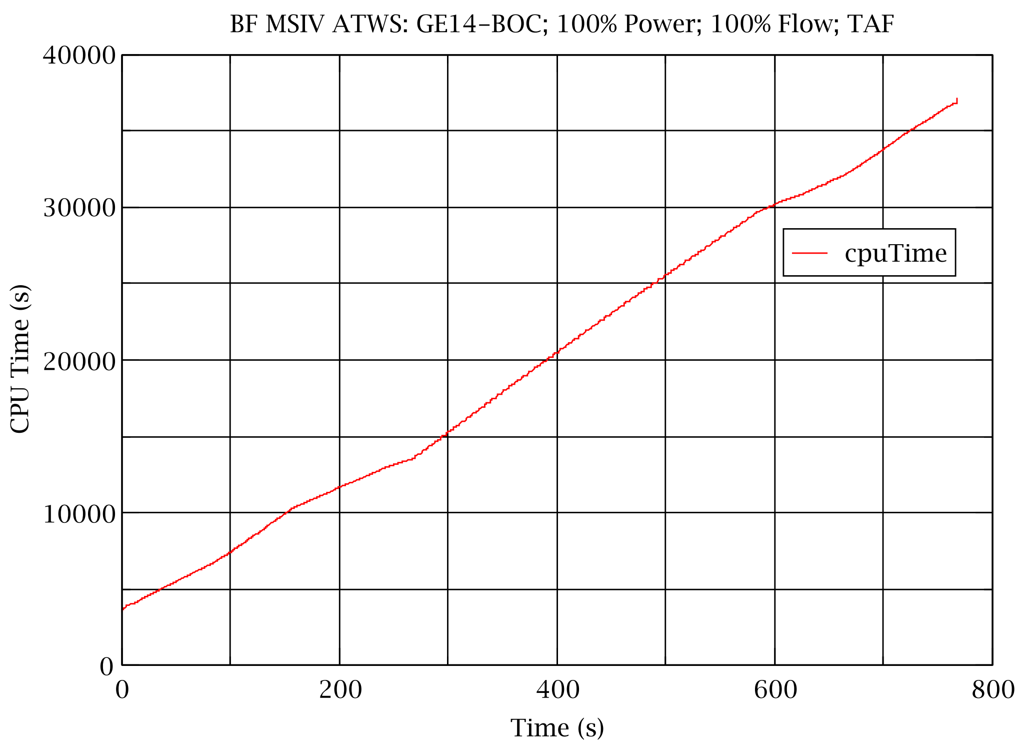

Figure 28. CPU Time

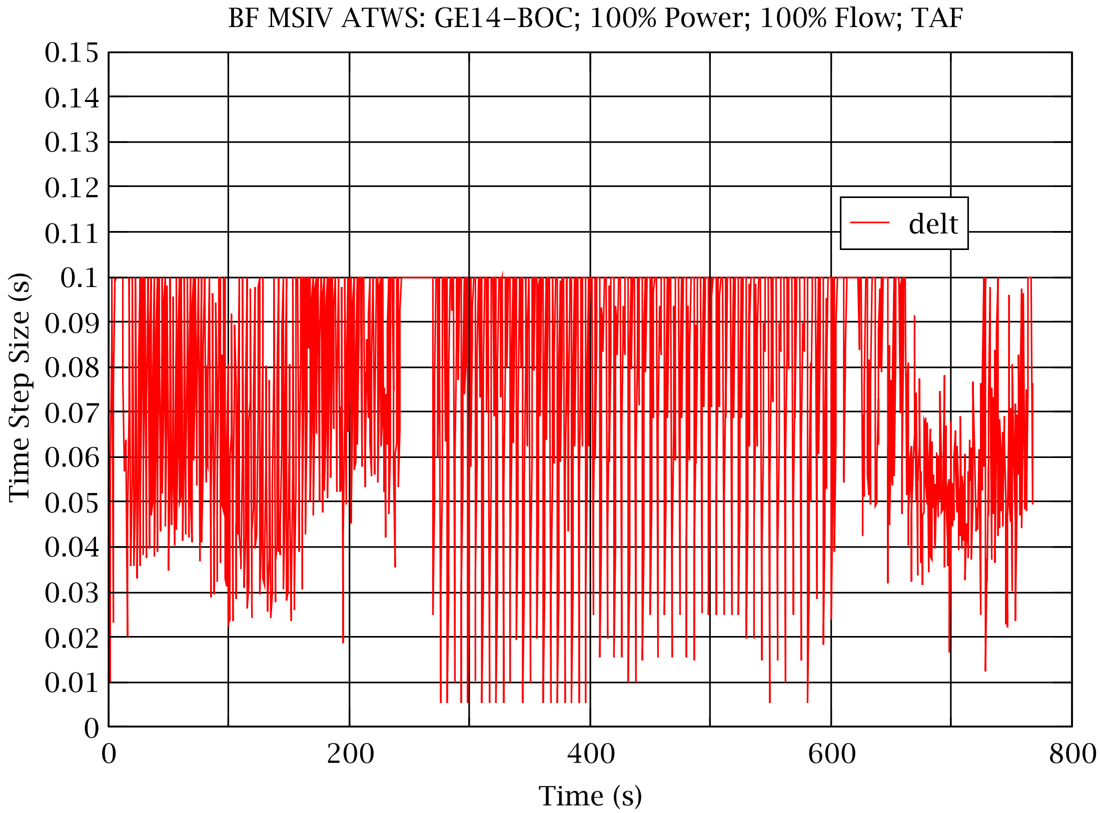

Figure 29. Time Step Size

4. RECOMMENDATIONS AND CONCLUSIONS

The recommendations below are given in order to a) improve the BWR ATWS model so that it can be used to assess the code's ability to calculate this type of event and b) to improve the ability of code users to run TRACE/PARCS more efficiently. The recommendations having to do with the model are divided into component models and initial conditions.

Modeling of Components

1. *The core model describing the compositions of fuel bundles should be replaced.*

The core model consists of a set of nuclear data (cross sections) and exposure (E) and moderator density (void) history (VH) distributions. Currently the cross sections are those received from UMich and the (E, VH) vectors are from GEH [8]. In order to model a more realistic BWR they should be consistent. The UMich cross sections, generated at ORNL using HELIOS, are for GE14 fuel but not the same GE14 fuel that GEH used for their core model. The former contain four axial composition zones with a different average enrichment (and loading) in the two interior zones and natural uranium end zones, whereas the fuel modeled by GEH uses six zones with different enrichments except in the end zones, which again contain natural uranium. Having cross sections for the core that GEH modeled for their ATWS analysis would allow an important comparison to be made as GEH has provided the initial power distribution throughout the core for their model.

2. *The branch calculations used to generate cross sections should take into account the range of boron concentrations expected in ATWS events.*

During an ATWS event the boron concentration can get to relatively high values and even though the power is reduced, calculating the sequence of events may still be of interest because of the potential for reduced cooling in the core. This requires a bounding boron concentration higher than in the present model.

3. *The number of thermal-hydraulic channels (CHANs) should be increased to 101 to allow for representation of all fuel bundles assuming one-eighth core symmetry.*

The current model has 24 channels and was developed for LOCA analysis. In an ATWS event the coupling between the neutronics and thermal-hydraulics is important and should be taken into account with a one-to-one correspondence between neutronic and thermal-hydraulic channels. Since the core being modeled does have one-eighth core symmetry, this can be carried out by using 101 CHANs. If lesser symmetry is needed for some reason (e.g., half core), more CHANs would have to be modeled.

4. *The PARCS full-core model should be reduced to a quarter-core model.*

The BF core has one-eighth core symmetry and can be modeled in PARCS using a quarter-core representation. This would save considerable run time.

5. *The fuel thermal properties should be a function of exposure.*

Currently fuel properties (e.g., gap conductance) are input the same for all fuel. If these properties were a function of exposure the feedback between fuel and coolant would be modeled more accurately. The ability to do this exists in TRACE.

6. *A model for the suppression pool is needed to be able to calculate pool temperature.*

The ability to calculate pool temperature would allow for determining whether it was within acceptable limits and also would provide the signal for ADS actuation (currently done on time as an operator action). The suppression pool model was in the ISL-2009 LOCA deck and should be placed back in the model.

7. *There appears to be a disconnect between the geometry specified for the core barrel relative to the geometry for the fuel in the core. This needs to be investigated.*

The problem has to do with the alignment of the boundaries of the axial nodes in the CHAN component and the heat structures representing the core barrel.

8. *The model runs with an ideal steam separator but not with the two- or three-stage models. This needs to be investigated.*

This may only be a problem for the version of the code being used, and not a problem in more recent version.

9. *Computational fluid dynamics (CFD) should be used to determine the effect of mixing boron in the lower plenum.*

Currently SLCS actuation on time makes the assumption that the delay in getting boron from the injection point in the lower plenum to the core inlet is known. The mixing of boron is modeled by introducing boron at a VESSEL level close to the core inlet. CFD analysis would confirm what the extent of mixing and the arrival time might be.

10. *Parts of the emergency core cooling system (ECCS) should be added to the model.*

The following parts of the ECCS were removed from the ISL-2009 input deck in order to simplify the calculation: high pressure coolant injection (HPCI), low pressure coolant injection (LPCI), core spray (CS), and suppression pool cooling. These systems would be actuated under certain ATWS scenarios and should be placed back into the model.

11. *Feedwater control system should be refined.*

Results from the initial simulation of an ATWS initiated by MSIV closure showed more than expected oscillations in feedwater flow. A closer examination of the feedwater control system revealed discrepancies in some of the input parameters between the ISL-2005 TRAC-B input deck and an updated version generated by NRC. After making changes to the feedwater control system using input parameters from the ISL-2005 TRAC-B input deck, the oscillations in feedwater were significantly reduced and appeared more reasonable.

12. *The model for SRV Bank 1 needs to be modified to represent the correct number of valves.*

The ISL-2009 LOCA deck [5] has the SRV fractional flow area representing only one valve and not four aas actually assigned to SRV Bank 1.

Initial Conditions

13. *Initial conditions for different statepoints should be obtained from GEH in order to investigate different ATWS events.*

Having the initial conditions (e.g., control rod patterns) for various statepoints will allow for calculating different ATWS events. For example, having the conditions at 120% power and 80% flowrate will allow for a test of MELLA+ conditions. If statepoints are available at times in the cycle different from that already modeled in TRACE/PARCS, then the corresponding (E, VH) distributions would also have to be supplied.

14. *The emergency operating procedures for Browns Ferry should be consulted to determine if the scenario being modeled conforms to that which would be expected at the plant.*

TRACE/PARCS Usage

15. *Improvements should be made in the error messages which frequently are too general to be of use in debugging problems.*

16. *The input variables in PARCS should be updated.*

The manuals describing the PARCS input (up through V3.0) have input variables that are no longer used. This should be corrected.

17. *SNAP should be made more useful for developing PARCS input.*

Although SNAP is very useful for developing TRACE input, this is not the case for developing PARCS input.

18. *AptPlot should be made more integrated with PARCS output.*

There is no documentation on generating plots from PARCS output using AptPlot.

19. *Sample problems for TRACE/PARCS should be updated so that they work with the latest version of the code.*

Several of the sample problems that utilize PARCS have not been updated and cannot run with the current version of TRACE/PARCS without modification.

20. *Users should have access to the source code and to the code-developers website.*

The ability to learn as much as possible about the code is a help in learning to work with the code.

21. *TRACE/PARCS execution could be made more efficient by running in parallel (multi-task) mode.*

TRACE offers a coarse-grained parallel (multi-task) mode of operation. An example is the spawning of PARCS as a satellite process. The execution of TRACE could be made more efficient by taking advantage of multi-core, multi-thread CPUs and splitting the system model into multiple TRACE processes.

In conclusion, the objectives of the project have been met. A model for carrying out ATWS analysis has been developed, albeit with known deficiencies. This model allows for the calculation of key parameters after closure of the MSIVs. It allows for RPT, water level control, injection of boron from the SLCS and actuation of the ADS.

5. REFERENCES

1. TRACE V5.0 User's Manual, U.S. Nuclear Regulatory Commission, October 8, 2008.

2. T. Downar, Y. Xu, and V. Seker, "PARCS V2.7 User Manual," Purdue University, October 2008.

3. "Development of the TRACE Thermal-Hydraulic System Analysis Code," Letter to the Honorable Dale E. Klein, Advisory Committee on Reactor Safeguards, September 24, 2008.

4. "Applicability of TRACE Thermal-Hydraulic System Analysis Code TO Evaluate the ESBWR Design and Related Matters," Letter to Mr. R.W. Borchardt, Advisory Committee on Reactor Safeguards, July 29, 2009.

5. Browns Ferry Calculation Notebook for TRACE LBLOCA with CONTAIN, Information Systems Laboratories, Inc., 2009.

6. D.A. Barber et al., "Technical Support for Review of GE BWR MELLA+ Topical Report, NEDC-33006P," ISL-NSAD-TR-04-24, Rev. 1, Information Systems Laboratories, Inc., February 2005.

7. A. Ulses, personal communication, U.S. Nuclear Regulatory Commission, January 2010.

8. GEH responses to requests for additional information February 6, 2009 (MFN 08-166), U.S. Nuclear Regulatory Commission.

9. J.S. Baek and D. Wang, "Browns Ferry-3 TRACE Calculation Notebook," Information Systems Laboratories, Inc., December 21, 2007.

10. L-Y. Cheng et al., "Calculation Notebook: Browns Ferry Model for TRACE," Brookhaven National Laboratory, March 24, 2010.

11. G. Raitses et al., "Calculation Notebook: Browns Ferry Reactor Core Model for TRACE/PARCS," Brookhaven National Laboratory, March 24, 2010.

12. Browns Ferry Nuclear Plant, Final Safety Analysis Report, Amendment 20, Figure 1.6-28, September 2003.