Britt T. McKinney
Sr. Vice President & Chief Nuclear Officer

PPL Susquehanna, LLC
769 Salem Boulevard
Berwick, PA 18603
Tel. 570.542.3149 Fax 570.542.1504
btmckinney@pplweb.com

JUL 3 0 2007

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop OP1-17
Washington, DC 20555

SUSQUEHANNA STEAM ELECTRIC STATION PROPOSED LICENSE AMENDMENT NO. 285 FOR UNIT 1 OPERATING LICENSE NO. NPF-14 AND PROPOSED LICENSE AMENDMENT NO. 253 FOR UNIT 2 OPERATING LICENSE NO. NPF-22 EXTENDED POWER UPRATE APPLICATION RE: REACTOR SYSTEMS TECHNICAL REVIEW REQUEST FOR ADDITIONAL INFORMATION RESPONSES
PLA-6250

Docket Nos. 50-387 and 50-388

References: 1) PLA-6076, B. T. McKinney (PPL) to USNRC, "Proposed License Amendment Numbers 285 for Unit 1 Operating License No. NPF-14 and 253 for Unit 2 Operating License No. NPF-22 Constant Pressure Power Uprate," dated October 11, 2006.

2) PLA-6209 B. T. McKinney (PPL) to USNRC, "Proposed License Amendment Numbers 285 for Unit 1 Operating License No. NPF-14 and 253 for Unit 2 Operating License No. NPF-22 Reactor Systems Technical Review Request for Additional Information Responses," dated June 15, 2007.

3) PLA-6230 B. T. McKinney (PPL) to USNRC, "Proposed License Amendment Numbers 285 for Unit 1 Operating License No. NPF-14 and 253 for Unit 2 Operating License No. NPF-22 Extended Power Uprate Application Supplement to Request for Additional Information Responses," dated June 29, 2007.

Pursuant to 10 CFR 50.90, PPL Susquehanna LLC (PPL) requested in Reference 1 approval of amendments to the Susquehanna Steam Electric Station (SSES) Unit 1 and Unit 2 Operating Licenses (OLs) and Technical Specifications (TSs) to increase the maximum power level authorized from 3489 Megawatts Thermal (MWt) to 3952 MWt, an approximate 13% increase in thermal power. The proposed Constant Pressure Power Uprate (CPPU) represents an increase of approximately 20% above the Original Licensed Thermal Power (OLTP).

The purpose of this letter is to supplement the responses to NRC Questions transmitted to the Request for Additional Information transmitted to PPL in Reference 2.

A001
NRR

The Attachments contain the PPL responses.

Attachments 1 and 4 contain AREVA NP, Inc. proprietary information. As such, AREVA NP, Inc. requests that the proprietary information be withheld from public disclosure in accordance with 10 CFR 2.390 (a) 4 and 9.17 (a) 4. Attachments 2 and 5 contain the non-proprietary versions of these responses.

An affidavit supporting this request for withholding this information from public disclosure is provided as Attachment 3.

Also, Attachment 4 contains revised Figure 4.14 and 4.15 originally transmitted to NRC in Reference 3. Figure 4.14 is revised to correct a typographical error in the caption and Figure 4.15 is revised to accurately reflect the data points shown at less than 15 lbm/sec.

There are no new regulatory commitments associated with this submittal.

PPL has reviewed the "No Significant Hazards Consideration" and the "Environmental Consideration" submitted with Reference 1 relative to the responses. We have determined that there are no changes required to either of these documents.

If you have any questions or require additional information, please contact Mr. Michael H. Crowthers at (610) 774-7766.

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

Executed on: 07/30/2007

B. T. McKinney

Attachment 1: Proprietary Version of the Request for Additional Information Responses
Attachment 2: Non-Proprietary Version of the Request for Additional Information Responses
Attachment 3: AREVA NP, Inc. Affidavit
Attachment 4: Corrected Figures 4.14 and 4.15 of PLA-6230 (Proprietary Version)
Attachment 5: Corrected Figures 4.14 and 4.15 of PLA-6230 (Non-Proprietary Version)

Copy: NRC Region I,
Mr. R. V. Guzman, NRC Sr. Project Manager
Mr. R. R. Janati, DEP/BRP
Mr. F. W. Jaxheimer, NRC Sr. Resident Inspector

Attachment 2 to PLA-6250
Non-Proprietary Version of the Request for Additional Information Responses

Request for clarification/supplemental information in response to PLA-6209.

NRC Follow-up Question to PPL Response 7a:

Regarding the first rod listed in the table, discuss why the change in dryout location does not affect much change in the critical power ratio (CPR).

PPL Response:

In the calculation of critical power, the XCOBRA computer code performs an iterative calculation in which assembly power is pushed until minimum CHFR for a single axial node is determined to be 1.00, indicating boiling transition has occurred. In the SPCB, Revision 1 evaluation of this high power, high flow BOC test case, the CHF node was determined to be Node 25. However, Node 18 was the next most limiting node, and the corresponding CHFR for this node was 1.00246.

[

] As a result of this revision to the calculation process, Node 25 was no longer the limiting node, and Node 18 became the limiting node. Since only a small change in assembly power was needed to move the Node 18 CHFR from 1.00246 to 1.00, the critical power ratio calculated with SPCB Revision 2 is nearly identical to the value determined through use of Revision 1 of the SPCB critical power correlation.

NRC Follow-up Question to PPL Response 7b:

Explain why such a change in dryout location occurs for the first rod listed in the table.

PPL Response:

This case demonstrates the effect of the changes to the calculation of the Omega term, and subsequently the Tong factor calculation between Revision 1 and Revision 2 of the SPCB critical power correlation. [

]

[

]

The equation for [

] The development of the equation for CMIN is included in the transmittal letter (Reference A.3) for Reference A.1.

NRC Follow-up Question to PPL Response 13:

NRC Question 13a:

Specifically identify any operator actions that are credited in the analyses presented.

PPL Response:

The analysis credits the below listed operator actions which are also assumed for CLTP as described in FSAR Section 15.8.8.1:

1. SLC
2. Lower RPV level to target band specified in the EOPs
3. Maintain HPCI suction on the CST if sufficient time is available for operator actions
4. Inhibit ADS
5. Initiate Suppression Pool Cooling (SPC)
6. Raise RPV water level to normal range when the HSBW (Hot Shutdown Boron Weight) has been injected.

NRC Question 13b:

Qualitatively identify any differences in the scenarios analyzed that would arise from an assumption that the plant operators follow EOPs.

PPL Response:

There are no differences in operator actions in response to an ATWS for either the current or CPPU ATWS scenarios. The current and CPPU ATWS analyses are consistent with the EOP instructions for plant operators. Operator response times in the CPPU analysis for the time critical operator actions (initiate SLC and SPC) were determined by observing SSES operating shift crews on the SSES plant simulator. These operator response times were then used to calculate the expected containment pressure and temperature increases, which were determined to be acceptable. The actual observed operator response times (although increased from the original analysis) are conservative with respect to the response times assumed in the analysis as shown in the table below:

Operator Action	Analysis Assumed Time	Average time Observed on the Simulator	Maximum Time Observed on the Simulator
Initiate SLC Injection	120 seconds	95 seconds	102 seconds
Suppression Pool Cooling (SPC) - first heat exchanger	1100 seconds	791 seconds	945 seconds
Suppression Pool Cooling (SPC) - second heat exchanger	1600 seconds	1161 seconds	1354 seconds

The following identifies the operator actions that were used as inputs to the ATWS analysis. These actions were consistent with the BWR Owners' Group Symptom Based Emergency Operating Procedures and they utilize the Human Performance Error Prevention techniques implemented at SSES. The specific operator actions are:

- Ensure Reactor Mode Switch is in Shutdown;
- Initiate Alternate Rod Insertion;
- Inject Standby Liquid Control (within 2 minutes);
- Inhibit Automatic Depressurization System;
- Ensure Reactor Water Cleanup is isolated;
- Insert Source Range and Intermediate Range Neutron Monitors;

- Reduce Reactor Recirculation Pump Speeds to Minimum;
- Trip both Reactor Recirculation Pumps *(Note: During an ATWS with MSIV closure, the pumps automatically trip at -38 inches RPV level);*
- Maximize Control Rod Drive system flow *(this inserts control rods by drifting)*;
- Reduce Reactor Water Level to Target Band (-60 to -110 inches RPV level) *(Note: During an ATWS with MSIV closure, this will automatically occur as the turbine driven feedwater pumps lose motive force and HPCI/RCIC automatically inject at rated flows);*
- Manually insert control rods *(Note: this requires Rod Sequence Control and Rod Worth Minimizer systems to be bypassed);*
- Direct the venting of the scram air header *(this is an in-plant operator action);*
- Direct the scram solenoids to be de-energized by removing fuses in RPS logic *(this is an in-plant operator action);*
- Place Div 1 Suppression Pool Cooling in service *(1100 seconds);*
- Place Div 2 Suppression Pool Cooling in service *(1600 seconds).*

At SSES, the operator actions for a Pressure Regulator Failure Open ATWS and the MSIV Closure ATWS are the same. The Pressure Regulator Failure Open transient quickly results in an MSIV closure on low steam line pressure at 861 psig. Both events cause a loss of normal feedwater flow from a loss of steam supply to the reactor feedwater pump turbines.

NRC Question 13d:

For PCT plots, what is the significance of the measurement above BAF?

PPL Response:

The temperature profile for peak cladding temperature for limiting MSIVC ATWS represents the temperature that was calculated to occur at the 119 inch node {119 inches above the bottom of active fuel (BAF)}. The highest PCT was calculated to occur at this node.

The temperature profile for peak cladding temperature for limiting PRFO ATWS represents the temperature that was calculated to occur at the 99 inch node {99 inches above the bottom of active fuel (BAF)}. The highest PCT was calculated to occur at this node.

NRC Question 13e:

Identify the heat capacity temperature limit on the suppression pool temperature plots.

PPL Response:

The heat capacity temperature limit (HCTL) cannot be plotted on the suppression pool temperature plots that were provided in the PPL Response 13. The HCTL plots reactor pressure versus suppression pool level versus suppression pool temperature while the plots provided in Response 13 plotted suppression pool temperature versus time.

The HCTL curves are used in the SSES Emergency Operating Procedures to determine when a reactor blowdown is warranted. For ATWS, the EOPs direct the operators to not perform an emergency blowdown if the suppression pool temperature is above the HCTL during an ATWS event, unless all control rods have been inserted.

The SSES CPPU ATWS suppression pool temperature analysis conservatively assumed that the suppression pool is at the TS minimum level and maximum temperature at the start of the event. The cooling water to both RHR heat exchangers is assumed to be at 97°F versus the TS maximum of 88°F. Based on these conservative assumptions, the SP temperature would be above the HCTL curve with the RPV between 1000 and 1100 psig. Operation of both SPC loops will reduce the temperature below the HCTL curve.

Under nominal conditions, the peak suppression pool temperature may not exceed the HCTL curve during an ATWS event. The ATWS analysis was performed using conservative assumptions to produce bounding results.

NRC Follow-up Question to PPL Response 15-17:

NRC requests PPL review NUMARC 87-00 and provide applicable supporting analyses prescribed therein.

PPL Response:

Per Section 2.7.1(1) of NUMARC 87-00, Revision 1, the temperatures inside containment as a result of a SBO event are enveloped by the loss of coolant accident and high energy line break environmental profiles. This is true for SSES and thus no additional analyses are prescribed by the NUMARC document.

Regarding use of MAAP to predict suppression pool temperature at four hours following an SBO, a hand calculation was performed which assumes transfer of all the decay heat from the reactor directly to the suppression pool with no subsequent heat transfer. This calculation determines that at the end of four hours, the suppression pool temperature would be approximately 170°F.

Since the MAAP code credits passive heat sinks in containment and RCS leakage of 100 gpm to the drywell, the MAAP predicted result of 156.6 °F is considered reasonable.

NRC Follow-up Question to PPL Response 19:

NRC requests PPL review NUMARC 87-00 and provide applicable supporting analyses prescribed therein.

PPL Response:

Concerning the use of MAAP, NRC requests PPL provide a supplemental analysis done in accordance with NUMARC 87-00, Revision 1. NUMARC 87-00, Revision 1, Section 7.2.1 "Condensate Inventory for Decay Heat Removal" addresses the analysis method for determining the adequacy of condensate inventory.

According to Section 7.2.1 of NUMARC 87-00, Revision 1, the following formula is to be used to calculate the Condensate Inventory for Decay Heat Removal:

$B = A * (X\ GAL/MW_t) + C$

Where B = Required makeup

A = Reactor thermal power (MW_t)

X = 22.12 for 4 hour coping plant

C = Zero (SSES does not require a primary system cooldown to minimize reactor coolant pump seal leakage or to maintain decay heat removal capability.) [Basis- Section 2.5.2]

Therefore,

$B = 3952\ MW_t * 22.12\ GAL/MW_t + 0$

$B = 87{,}418.24\ GAL$

The SSES Unit 1 and Unit 2 Technical Specification 3.3.5.1, Function 3d (Condensate Storage Tank Level- low) setpoint of ≥36 inches above tank bottom (corresponds to 135,000 gallons (FSAR Table 6.3-8). Thus, the NUMARC method calculated required makeup (87 418) is less than the Technical Specification minimum available (135,000). Thus, based on the NUMARC analysis method, adequate condensate is available.

The NUMARC 87-00, Revision 1, method only calculates condensate makeup based on decay heat. It does not provide any guidance for calculating vessel makeup for steam supplied to the HPCI or RCIC turbines. Both the current and CPPU SSES SBO analyses meet and exceed the guidance from NUMARC 87-00, Revision 1. These analyses account for the reactor decay heat, RPV inventory leakage (35 gpm per Reactor Recirculation pump, 25 gpm for identified drywell leakage, and 5 gpm for unidentified drywell leakage), SRV discharge to the suppression pool (to maintain reactor pressure),

and steam supply to the HPCI/RCIC turbines. Therefore, the SSES SBO analysis provides a more conservative result than the NUMARC 87-0, Revision 1, analysis.

NRC Follow-up Question to PPL Response 21:

The licensee's response states, "The evaluation process confirms use of 23% of CPPU rated thermal power for the beginning of thermal limit monitoring is appropriate." The staff would like to understand how 23% was derived. The response appears to be too subjective; licensee is requested to clarify how 23% is actually calculated with quantitative details.

PPL Response:

The value of 23% was based on previous CPPU submittals for Browns Ferry Units 2 & 3 and Vermont Yankee. This value can be related to the power density of the Grand Gulf reactor which had an original power density of approximately 4.8 MW per bundle. At 25% power (grand Gulf thermal limits monitoring threshold) this yields an average bundle power of 1.2 MW. Since SSES has 764 bundles, a 1.2 MW average bundle power would yield a total core power of approximately 917 MWt. Dividing 917 MWt by the CPPU rated power of 3952 MWt yields a power level of 23.2% which is rounded down to yield 23%. Thus, the value of 23% is based on previous CPPU submittals and remains within the current BWR operating base.

NRC Follow-up Question to PPL Response 22:

Please clarify: "Therefore, the conclusion is applicable when the pressure is ≥785 psig and core flow is < 10 million lbm/hr." in the response. It seems to contradict other statements in the response.

PPL Response:

The following table shows the applicable Technical Specification sections for combinations of pressure and flow in the Safety Limits Technical Specification Section 2.1.1.

	Core Flow < 10 Mlb/hr	Core Flow ≥10 Mlb/hr
Pressure < 785 psig	TS 2.1.1.1 Power ≤ 23%	TS 2.1.1.1 Power ≤ 23%
Pressure ≥785 psig	TS 2.1.1.1 Power ≤ 23%	TS 2.1.1.2 MCPR ≥ MCPRSL

The PPL response states that: "the conclusion is applicable when the pressure is ≥785 psig and core flow is < 10 million lb/hr." The PPL response is referring to the lower left hand corner of the above table where the proposed power limit of 23% would apply. Based on the NRC question that asked: "Is the conclusion applicable for the scenario with reactor pressure greater than 785 psig?" The PPL response needed to clarify that the conclusion regarding the applicability of the thermal power limit was only for the case where core flow was < 10 million lb/hr since the MCPRSL would be applicable if core flow was ≥10 million lb/hr with pressure ≥785 psig.

NRC Follow-up Question to PPL Response 25:

Please justify how the "ANSI/ANS5.1-1979 + 6% uncertainty" decay heat model mentioned in the response satisfies ELTR1 requirement of 1979 ANS +10% decay heat for loss of feedwater transient.

PPL Response:

The decay heat was recalculated based on 1979 ANS + 10% and the loss of feedwater transient was reanalyzed. The change in the decay heat only caused minor changes in the results of the calculation. Therefore, the information presented in Section 9.1.3.1 in Attachment 4 of PLA-6076, "Constant Pressure Power Uprate" dated October 11, 2006, (Reference 1) remains applicable for the revised analysis.

NRC Follow-up Question to PPL Response 29:

Please provide the resultant peak fuel enthalpy (cal/g) in RWE at startup or low power for EPU and OLTP operation and justify the 170 cal/gm limit is met. Since RWE at low power was not re-analyzed in EPU application, please provide an estimate of the peak fuel enthalpy and justify the limit can be met. For CLTP and OLTP, please provide the FSAR analysis results.

PPL Response:

A Rod Withdrawal Error (RWE) at startup or low power is described in SSES FSAR Section 15.4.1.2. This event is defined in the FSAR as: while operating below the low power setpoint and coincident with a failure or bypass of the RWM, the operator makes a procedural error and withdraws an out of sequence control rod of maximum worth.

The OLTP calculation for an RWE at startup or low power determined that a fuel enthalpy of 60 cal/gm would result. Peak fuel enthalpy is not changed significantly by EPU for this low power cold operating condition and therefore this event did not need to be reanalyzed for extended power uprate due to the amount of margin to the fuel failure threshold of 170 cal/gm. Furthermore, since this is a cold event and the cold and low power reactivity characteristics of the fuel are unaffected by extending the allowable maximum operating power of the core, the consequences of this event are not affected by power uprate.

The NRC staff position for ELTR1 states in Appendix E, Section 2.4 that only the limiting transients need to be included in the uprate amendment request, but a list of all transients analyzed in support of power uprate should be included. As stated in ELTR1 Section E.2.2, the minimum list of events to be included in the power uprate evaluation (Table E-1) confirms that the existing set of reload analysis transients remain valid for power uprate. In regard to the limiting control rod withdrawal event, the RWE event identified in the ELTR table is the power operation RWE (as opposed to the low power RWE) since it is the only type of RWE that can potentially challenge a fuel design parameter. The power operation RWE has been analyzed and the results provided in Attachment 4, Table 9-2 of the PUSAR.

NRC Follow-up Question to PPL Response 46:

Please provide the CMWR value for SSES LOCA analysis and justify SSES 0.2% limit is met.

PPL Response:

AREVA performed SSES pre-CPPU CMWR calculations for ATRIUM™-10[2] fuel which determined CMWR = 0.19% when the maximum planar MWR was 1.62% (an 88% reduction between the maximum planar MWR and the core wide MWR). AREVA SSES CPPU calculations for ATRIUM-10 fuel determined the maximum planar MWR was 0.28% versus the pre-CPPU value of 1.62%. The decrease in the maximum planar MWR between the AREVA pre-CPPU and the CPPU SSES LOCA analyses was primarily because the pre-CPPU CMWR analysis was performed for LOCA results which used a very conservative assumption (no credit for spray heat transfer until ATRIUM-10 testing was completed).

The relationship between the maximum planar MWR and the core wide MWR from the SSES pre-CPPU CMWR calculation (1.62% → 0.19%) was used to conclude the SSES CPPU CMWR is less than 0.2% since the SSES CPPU maximum planar MWR is only 0.28%.

New NRC Question:

The staff performed a small break LOCA with break size less than 0.1 ft2 at recirculation pump discharge side. The break characteristics include initial core flow of 108 Mlbm/hr, 102% power, top-peaked axial power profile and SF-BATT. The staff found the PCT in this break size range (less than 0.1 ft2) is much more limiting compared to break size at 0.7 ft2 case. Please provide small break LOCA analysis result at break size of 0.05 ft2 with same break characteristics. Please include major parameter plots as shown in the LOCA report (EMF-3242P, Revision 0) for TLO large break LOCA.

PPL Response:

AREVA performed the requested small break calculation for 0.05 ft^2, initial core flow of 108 Mlbm/hr, 102% power, top-peaked axial power profile and SF-BATT. The PCT for this small break was calculated to be 1296°F, which is lower than the limiting large break PCT and lower than the limiting small break PCT calculated with AREVA methods. The sequence of events for the requested small break calculation is provided in Table A.1 followed by a set of major parameter plots similar to the LOCA report.

---

[2] ATRIUM is a trademark of AREVA NP.

Table A.2 Event Times

Event	0.05 ft² Split PD SF-BATT 108F TOP Time (sec)
Initiate break, loss of offsite power	0.0
Initiate MSIV closure	2.0
Initiate scram (MSIV < 85% open)	2.5
MSIV fully closed	5.0
L2 low water level, HPCI signaled	125.9
L1 low water level, DG signaled	235.7
Jet pump suction uncovers	330.5
Recirc suction uncovers	522.5
Lower plenum flashes	360.2
DG power at ESS bus	260.8
HPCI flow starts	NA
LPCI pumps start	264.8
IL LPCI valve starts to open	522.3
IL LPCI flow starts	573.6
BL LPCI valve starts to open	NA
BL LPCI flow starts	NA
LPCS pump starts	272.3
LPCS valve starts to open	522.3
LPCS flow starts	555.6
ADS valve starts to open	355.7
RDIV closure starts	608.5
RDIV closure complete	641.5
Begin rated spray (TSPRAY)	679.6
End of blowdown	679.6
Bypass reflood	651.1
Core reflood	679.6
PCT	567.0

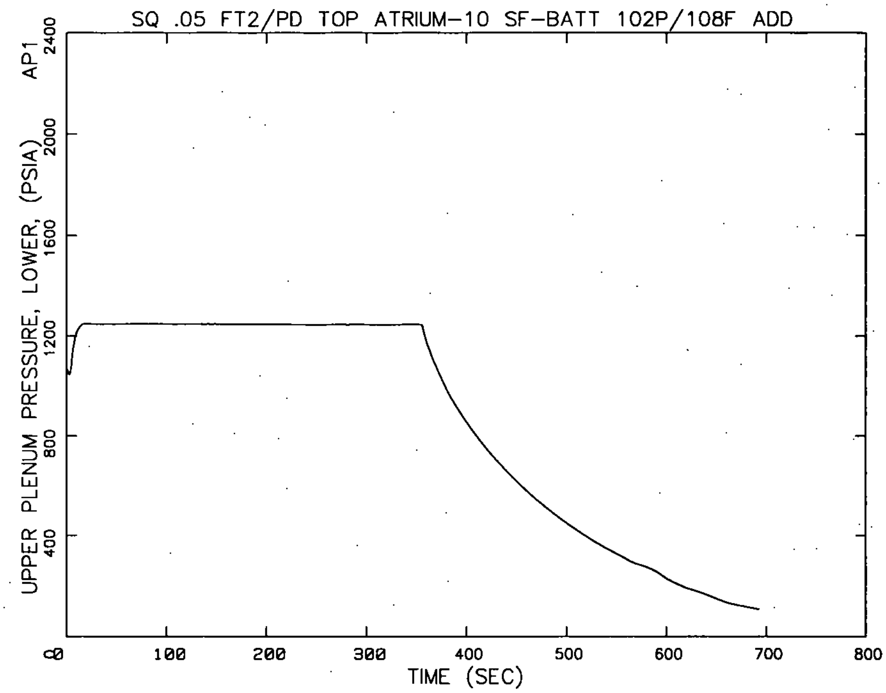

Figure A.1 0.05 ft^2 PD SF-BATT TOP 108F
Upper Plenum Pressure (Lower)

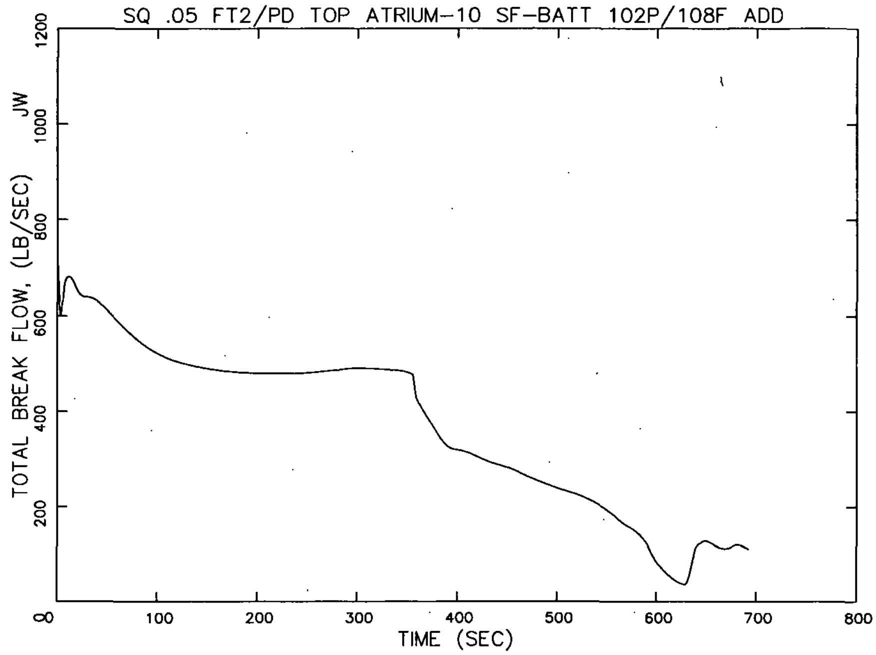

Figure A.2 0.05 ft^2 PD SF-BATT TOP 108F
Total Break Flow Rate

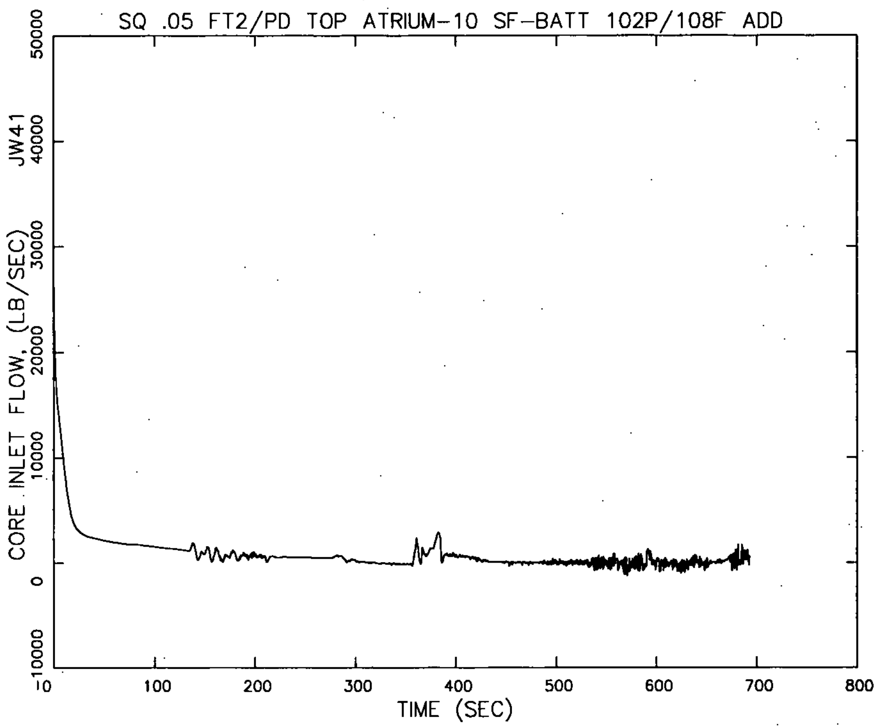

Figure A.3 0.05 ft^2 PD SF-BATT TOP 108F
Core Inlet Flow Rate

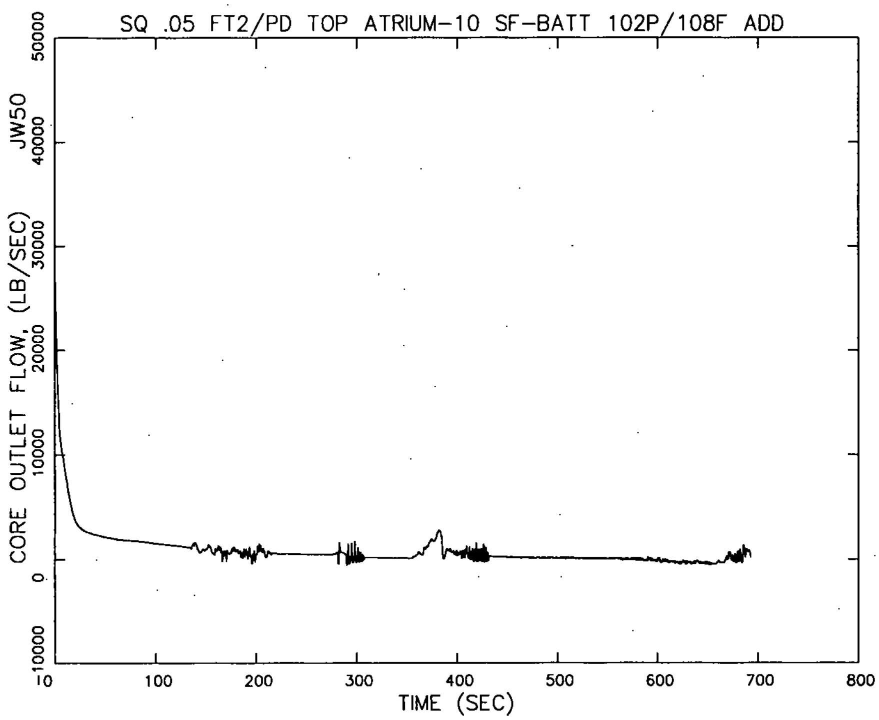

Figure A.4 0.05 ft^2 PD SF-BATT TOP 108F
Core Outlet Flow Rate

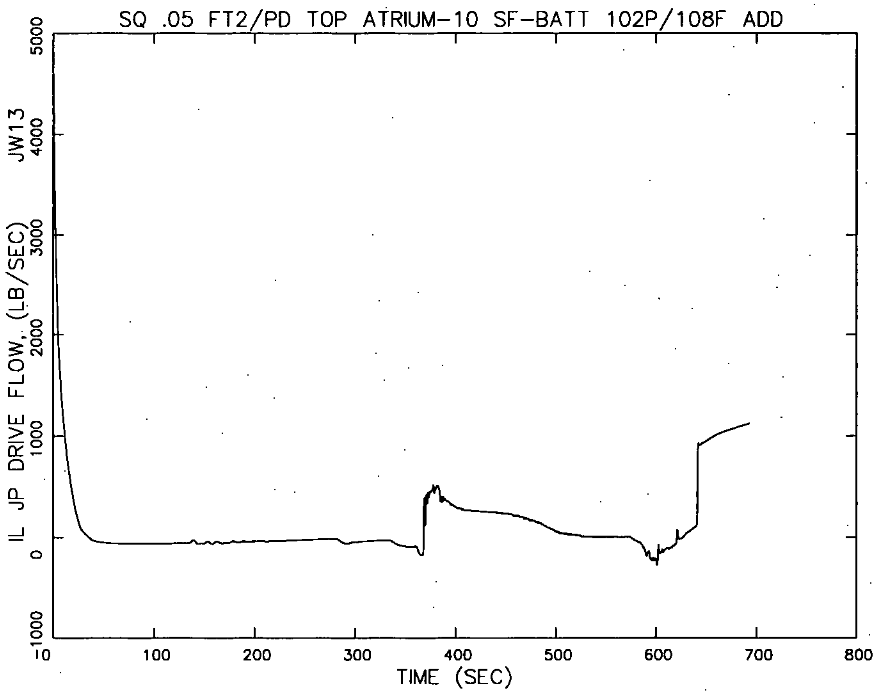

Figure A.5 0.05 ft² PD SF-BATT TOP 108F
Intact Loop Jet Pump Drive Flow Rate

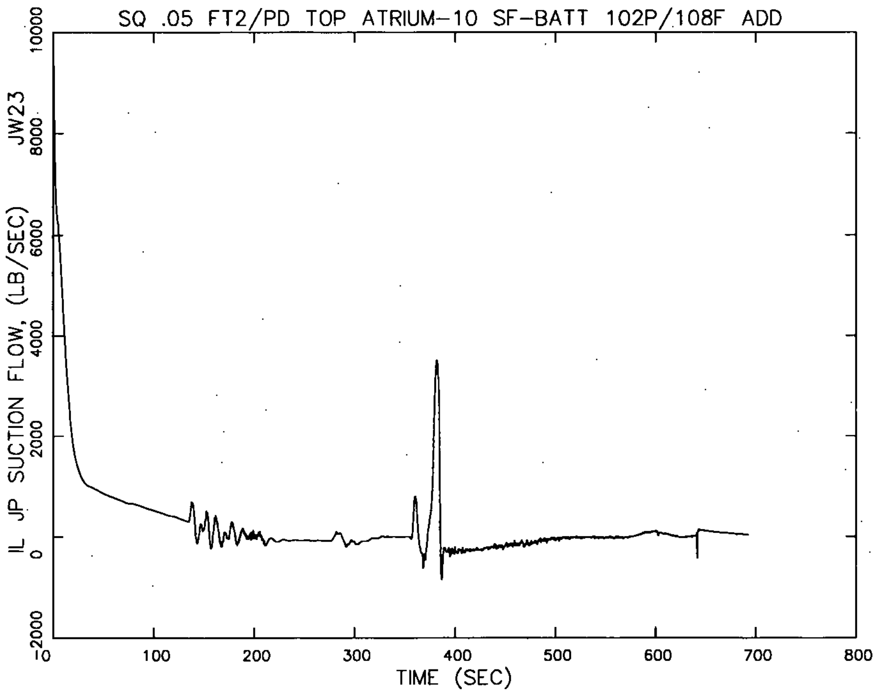

Figure A.6 0.05 ft^2 PD SF-BATT TOP 108F
Intact Loop Jet Pump Suction Flow Rate

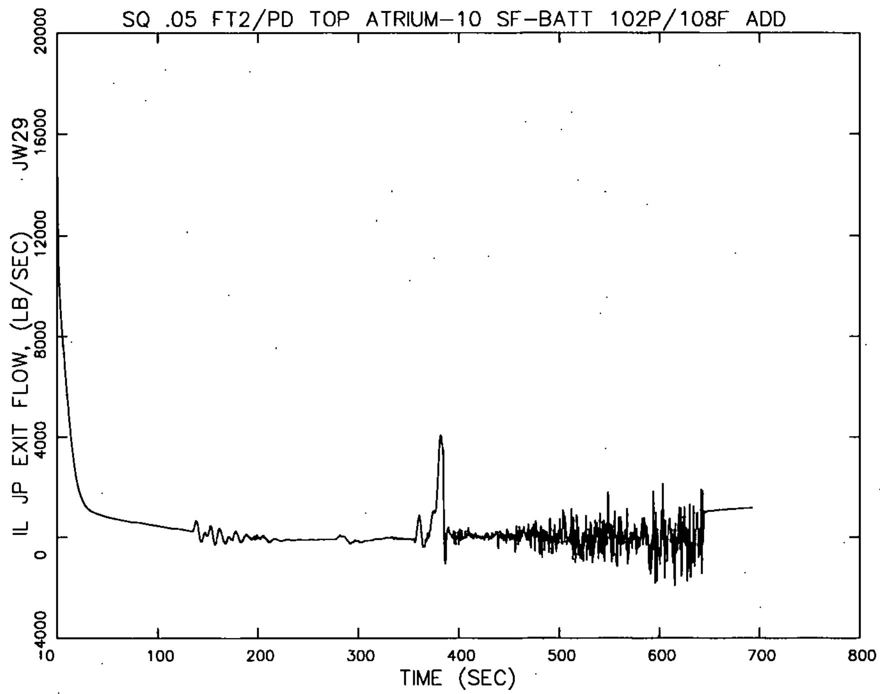

Figure A.7 0.05 ft^2 PD SF-BATT TOP 108F
Intact Loop Jet Pump Exit Flow Rate

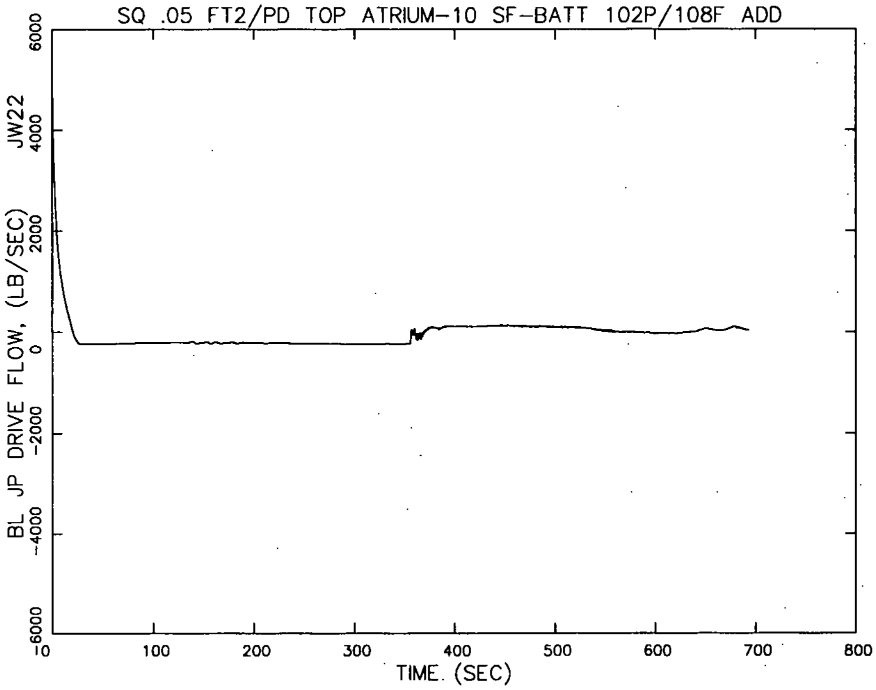

Figure A.8 0.05 ft^2 PD SF-BATT TOP 108F
Broken Loop Jet Pump Drive Flow Rate

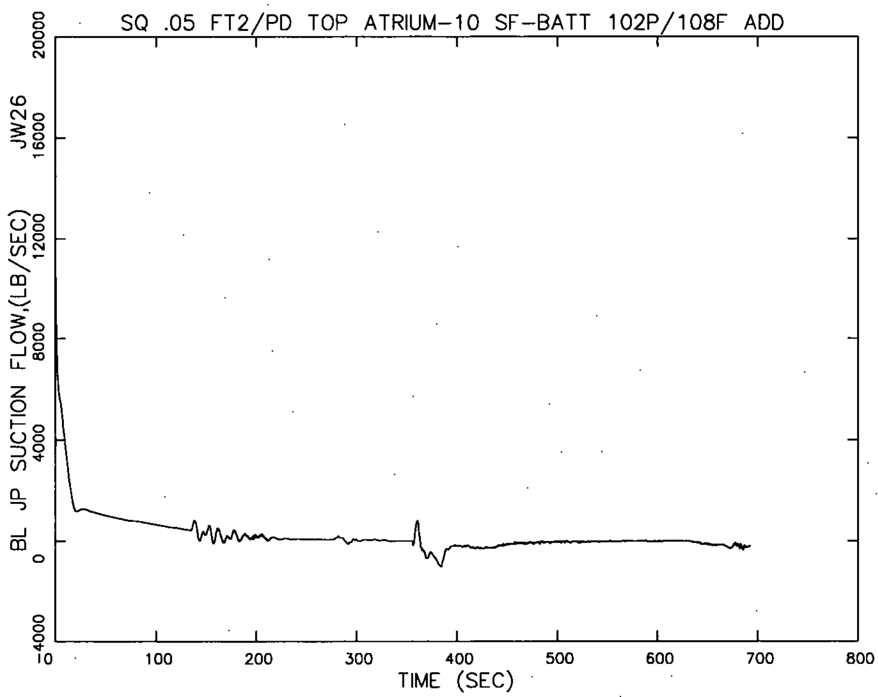

Figure A.9 0.05 ft^2 PD SF-BATT TOP 108F Broken Loop Jet Pump Suction Flow Rate

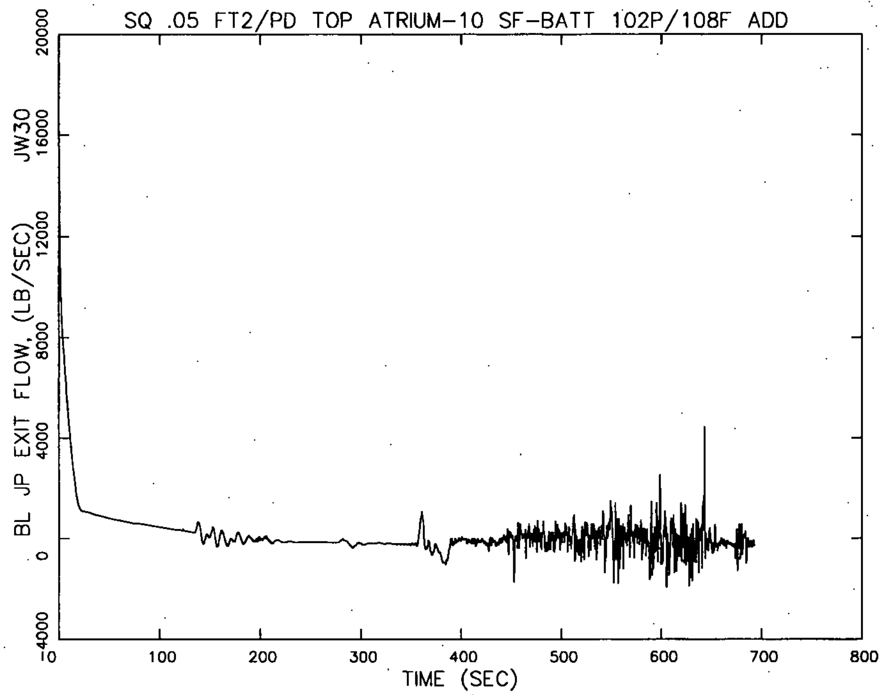

Figure A.10 0.05 ft^2 PD SF-BATT TOP 108F Broken Loop Jet Pump Exit Flow Rate

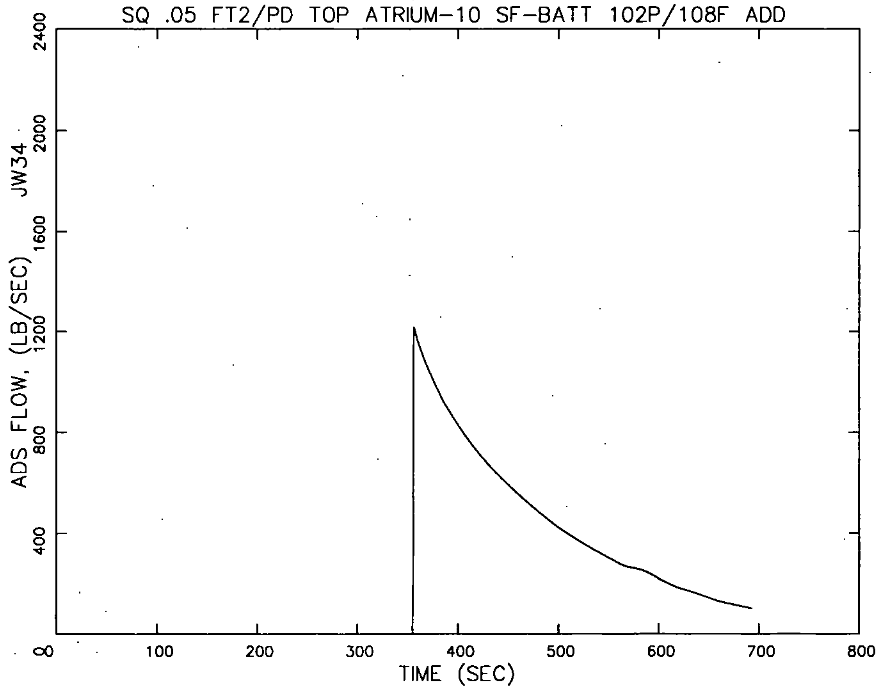

Figure A.11 0.05 ft[2] PD SF-BATT TOP 108F ADS Flow Rate

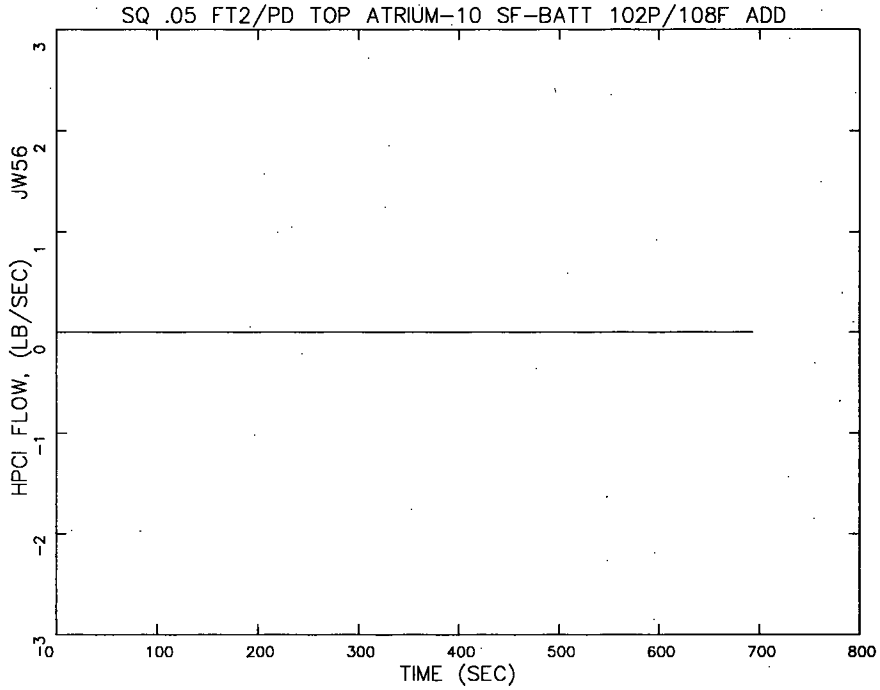

Figure A.12 0.05 ft^2 PD SF-BATT TOP 108F HPCI Flow Rate

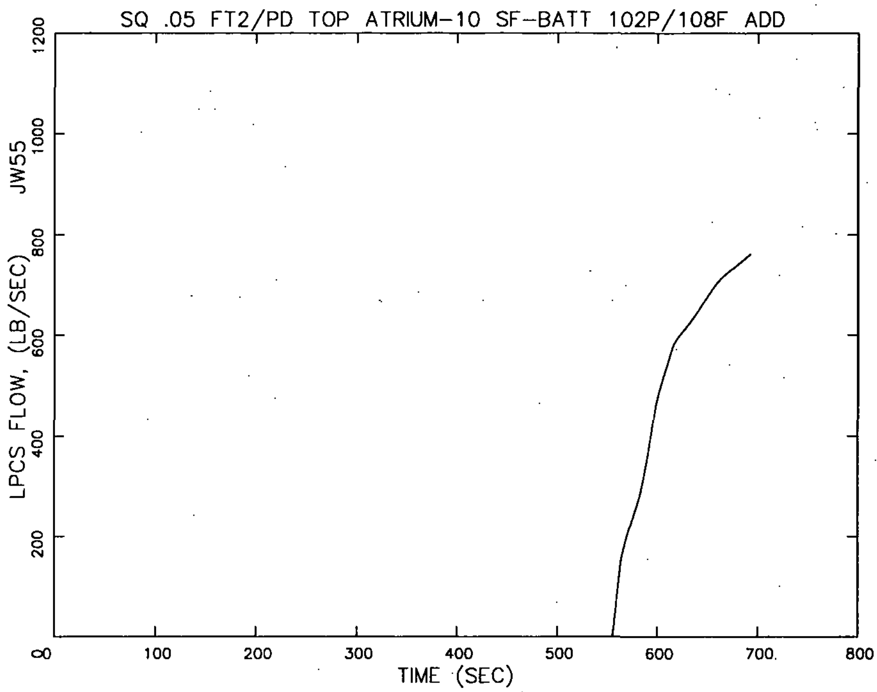

Figure A.13 0.05 ft^2 PD SF-BATT TOP 108F LPCS Flow Rate

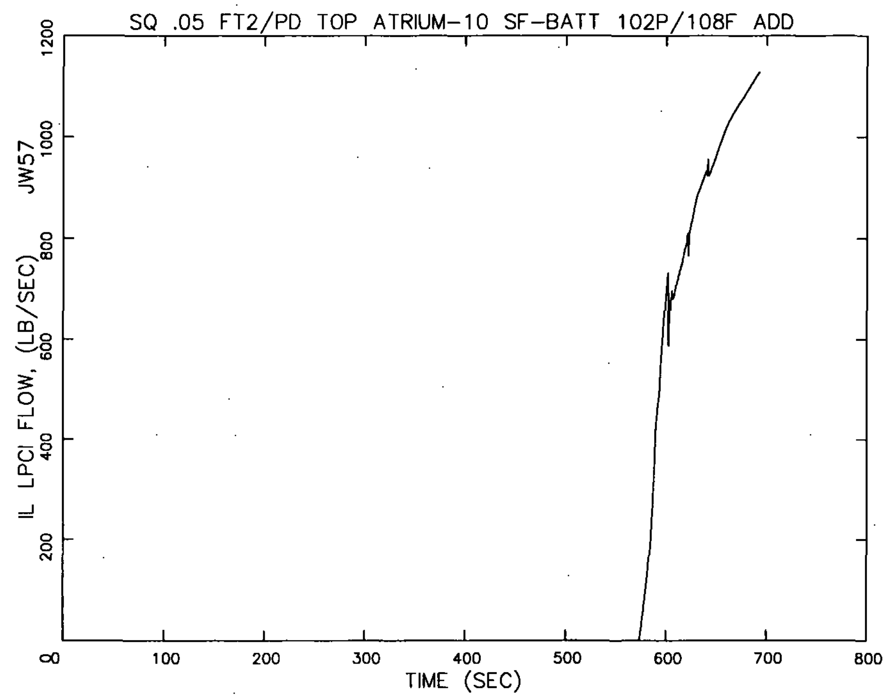

Figure A.14 0.05 ft^2 PD SF-BATT TOP 108F
Intact Loop LPCI Flow Rate

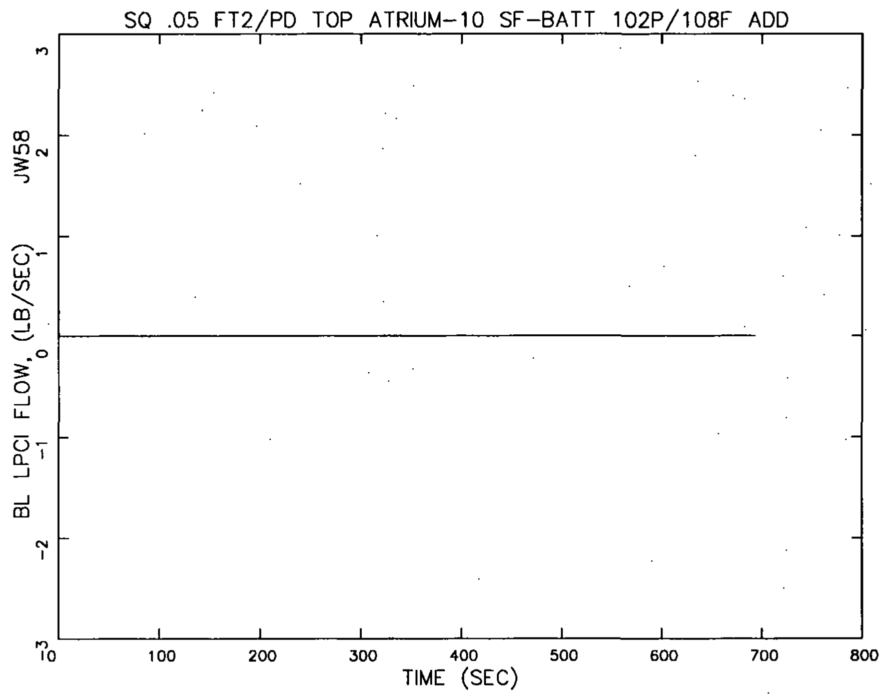

Figure A.15 0.05 ft^2 PD SF-BATT TOP 108F
Broken Loop LPCI Flow Rate

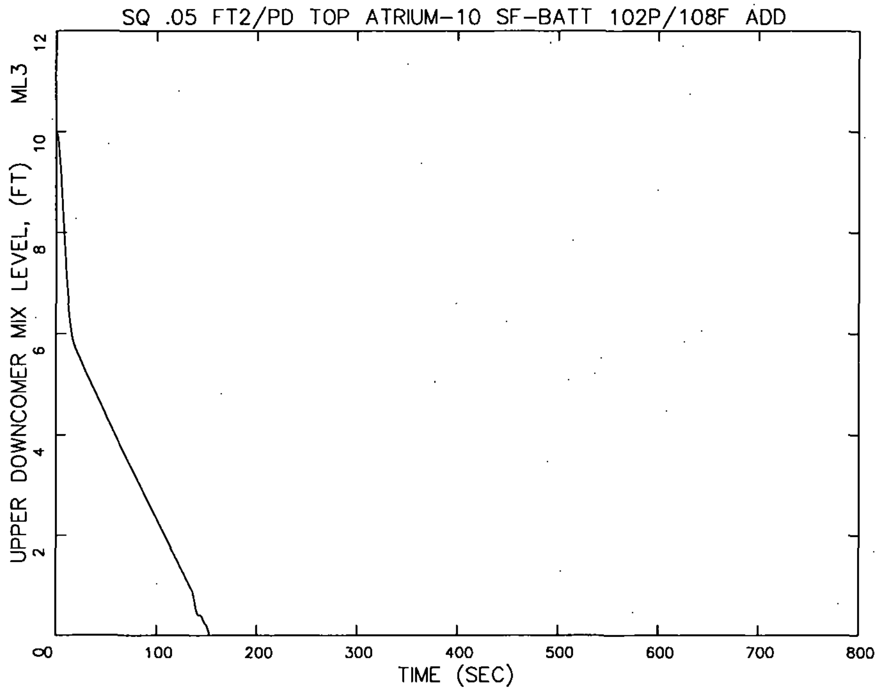

Figure A.16 0.05 ft² PD SF-BATT TOP 108F
Upper Downcomer Mixture Level

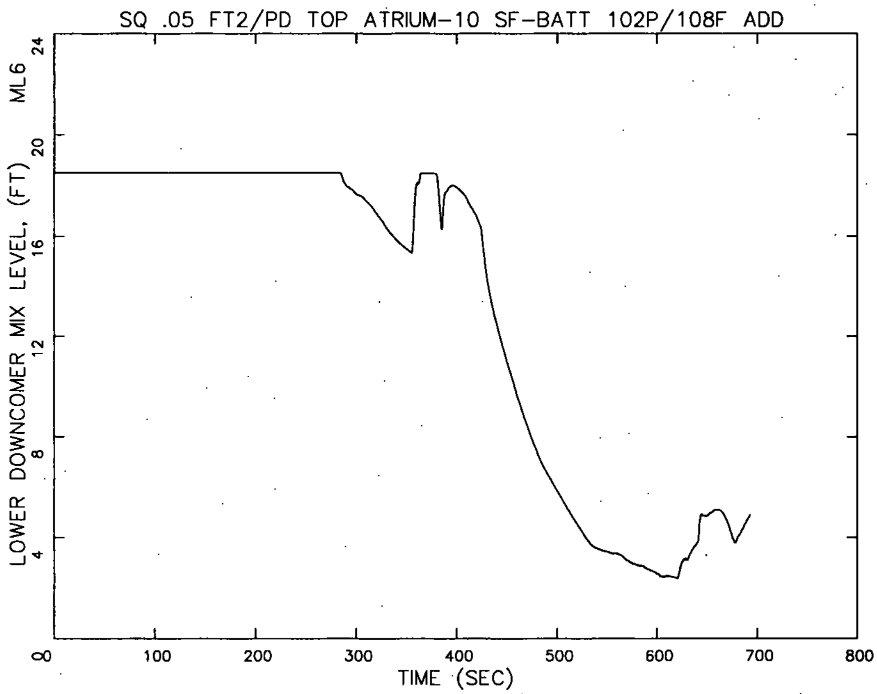

Figure A.17 0.05 ft^2 PD SF-BATT TOP 108F
Lower Downcomer Mixture Level

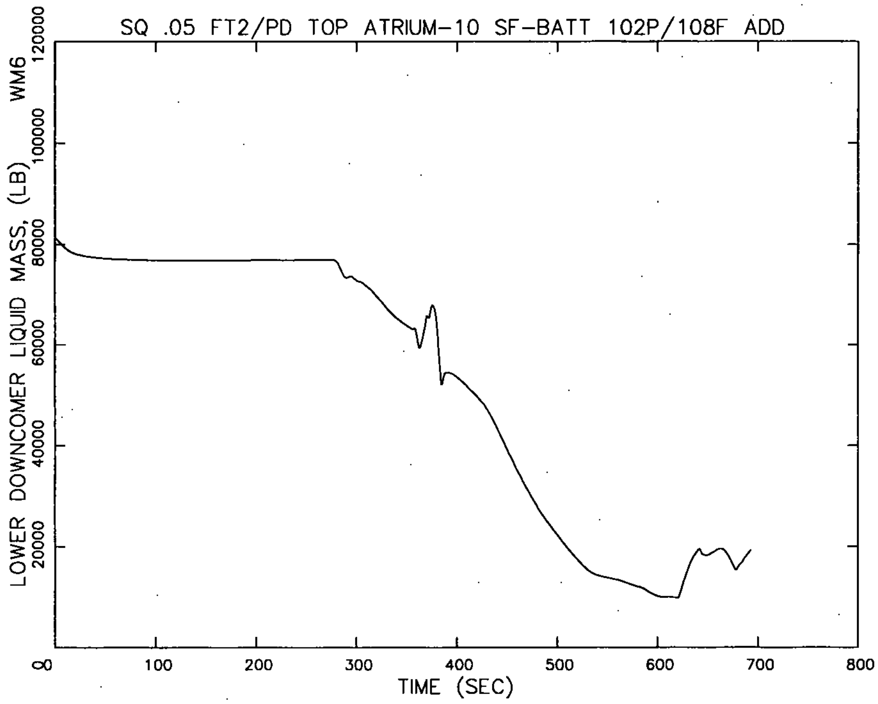

Figure A.18 0.05 ft² PD SF-BATT TOP 108F
Lower Downcomer Liquid Mass

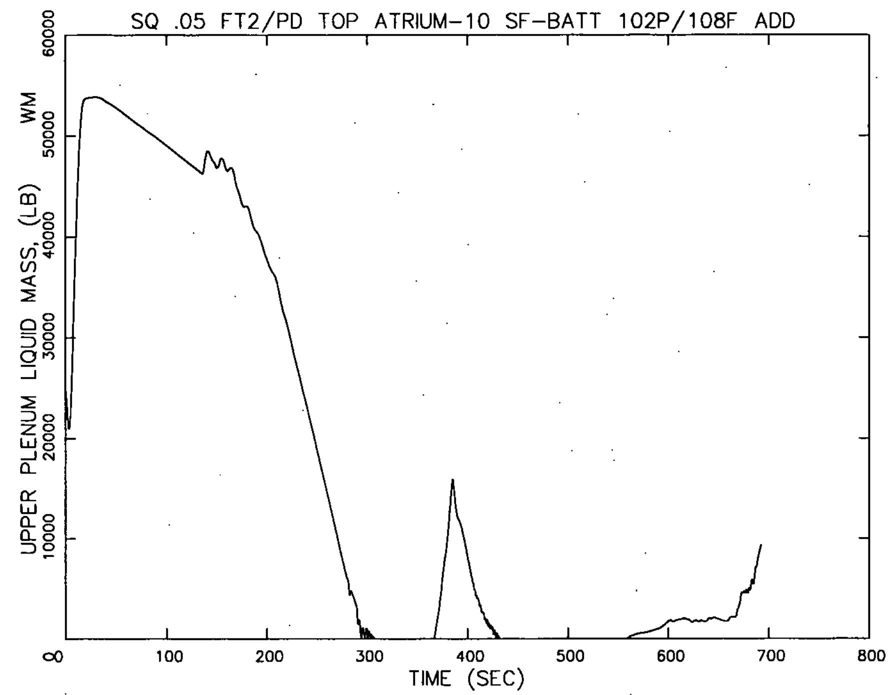

Figure A.19 0.05 ft^2 PD SF-BATT TOP 108F
Upper Plenum Liquid Mass

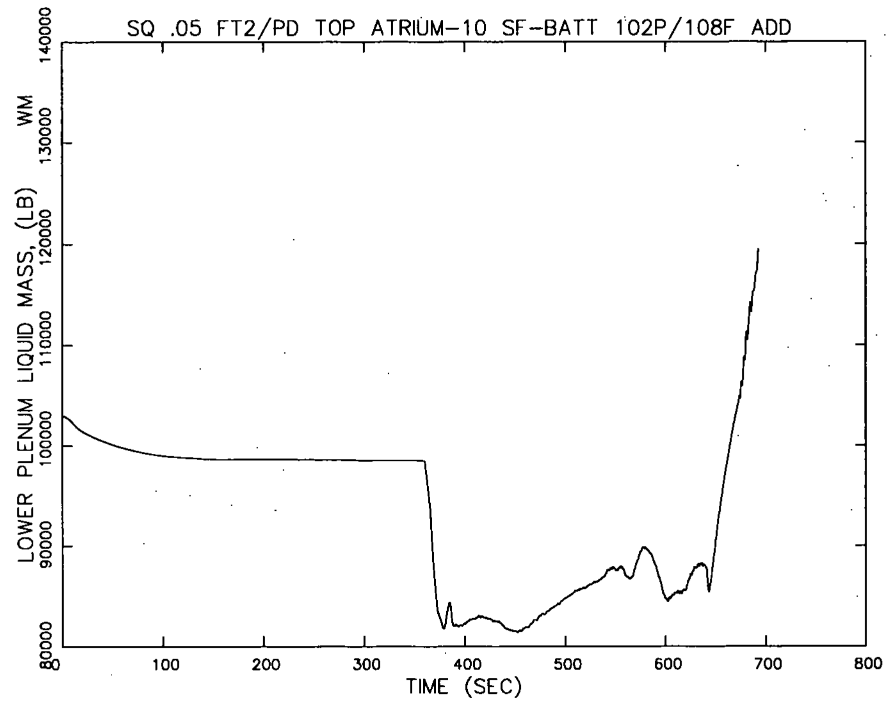

Figure A.20 0.05 ft^2 PD SF-BATT TOP 108F
Lower Plenum Liquid Mass

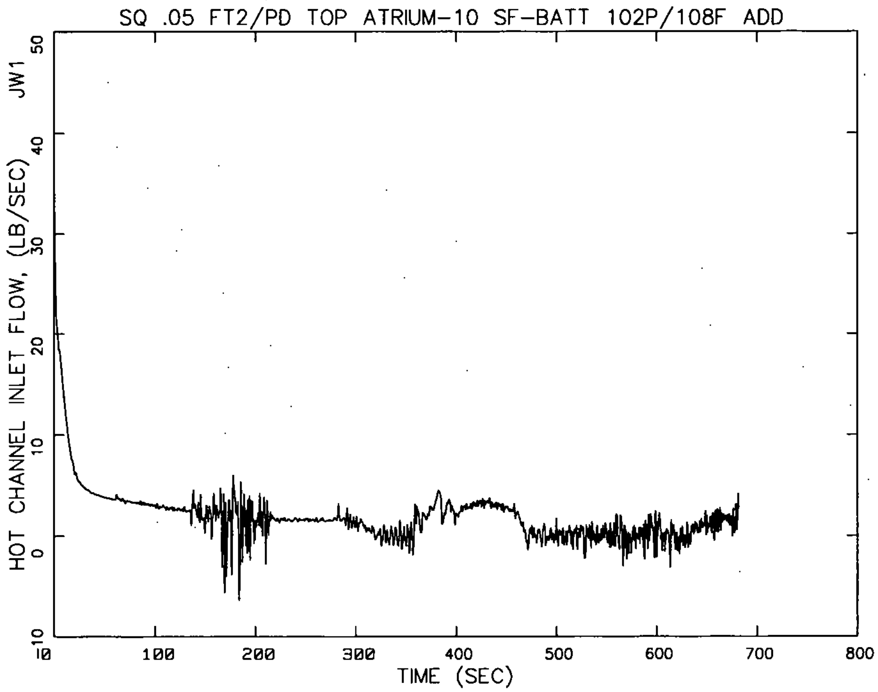

Figure A.21 0.05 ft^2 PD SF-BATT TOP 108F
Hot Channel Inlet Flow Rate

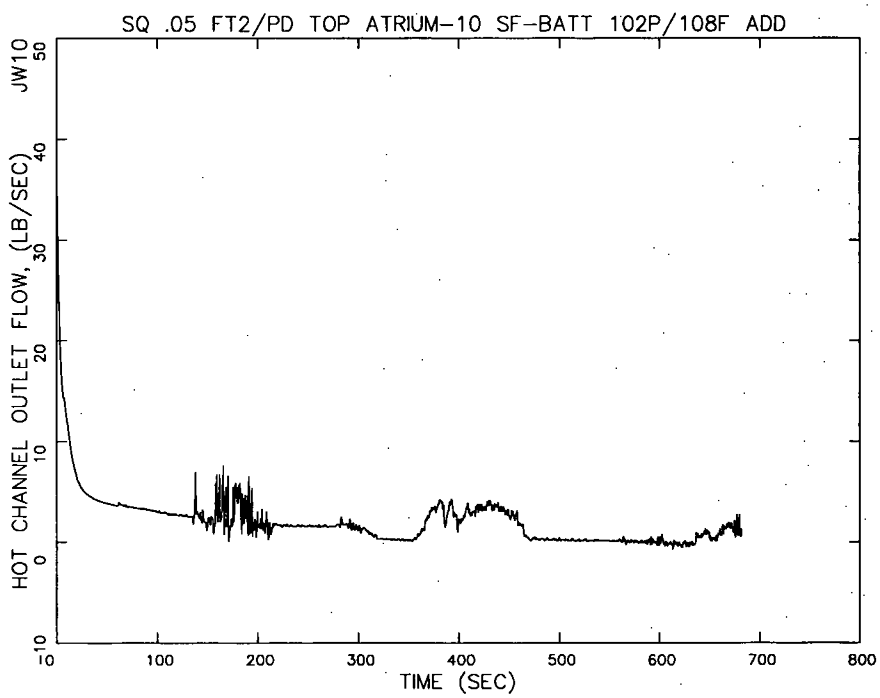

Figure A.22 0.05 ft² PD SF-BATT TOP 108F
Hot Channel Outlet Flow Rate

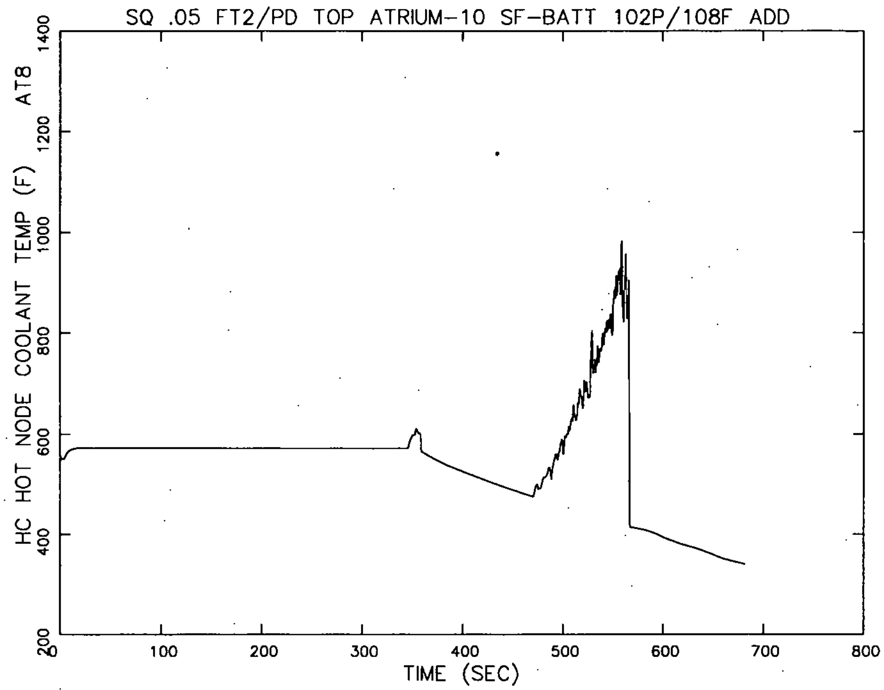

Figure A.23 0.05 ft^2 PD SF-BATT TOP 108F
Hot Channel Coolant Temperature at the Limiting Node

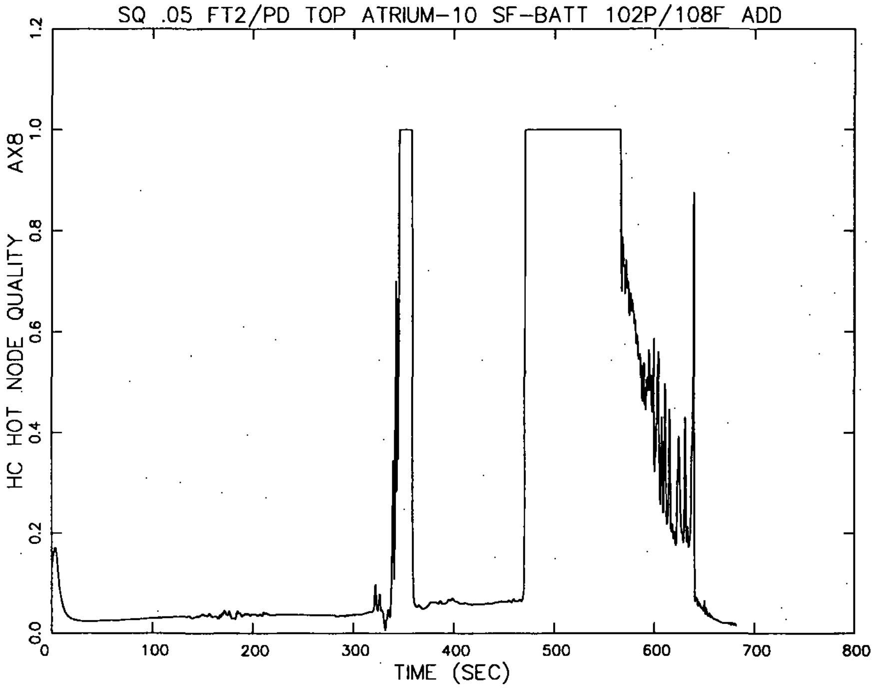

Figure A.24 0.05 ft^2 PD SF-BATT TOP 108F
Hot Channel Quality at the Limiting Node

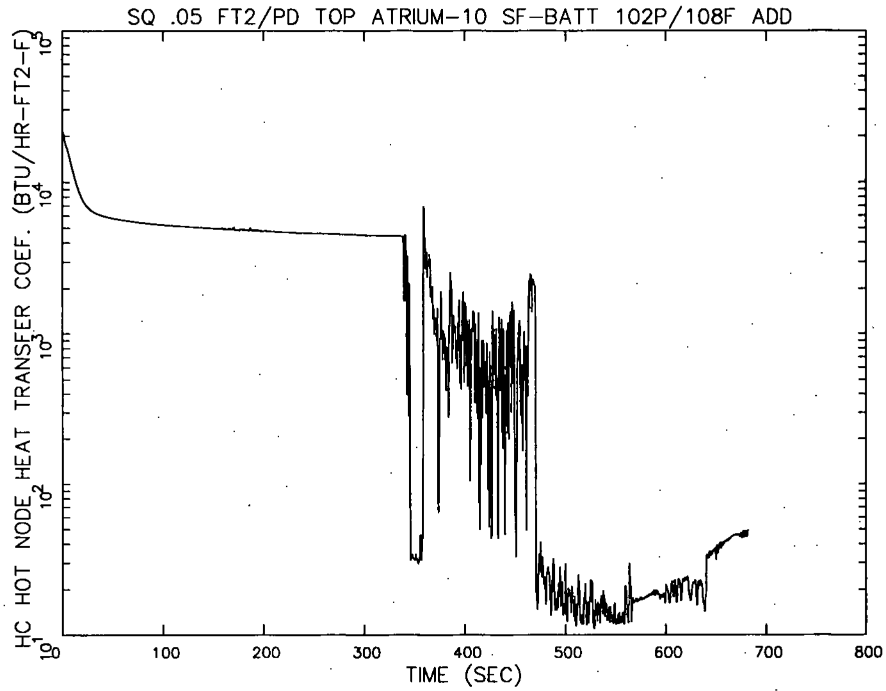

Figure A.25 0.05 ft^2 PD SF-BATT TOP 108F
Hot Channel Heat Transfer Coeff. at the Limiting Node

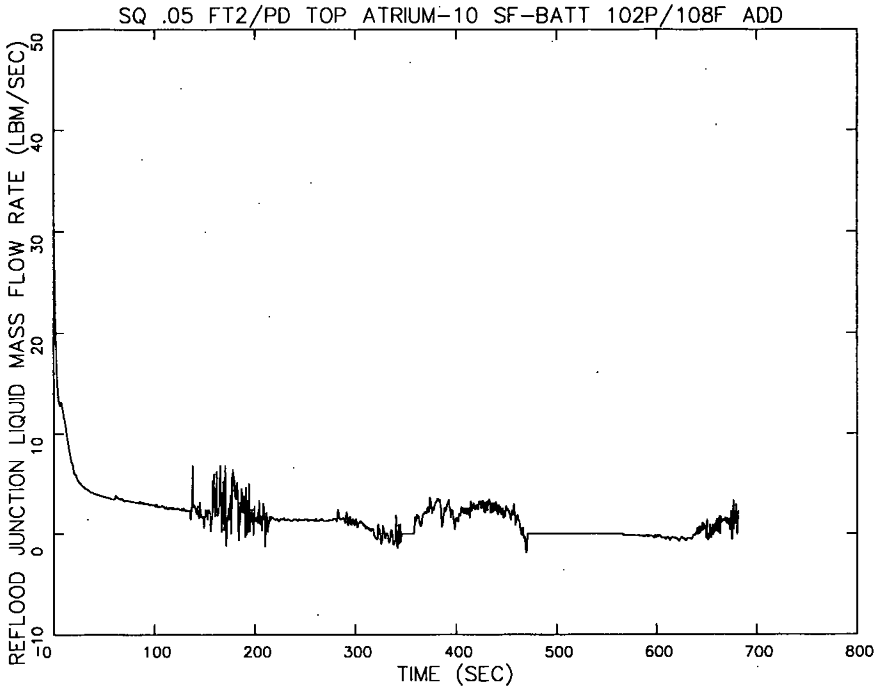

Figure A.26 0.05 ft^2 PD SF-BATT TOP 108F
Hot Channel Reflood Junction Liquid Mass Flow Rate

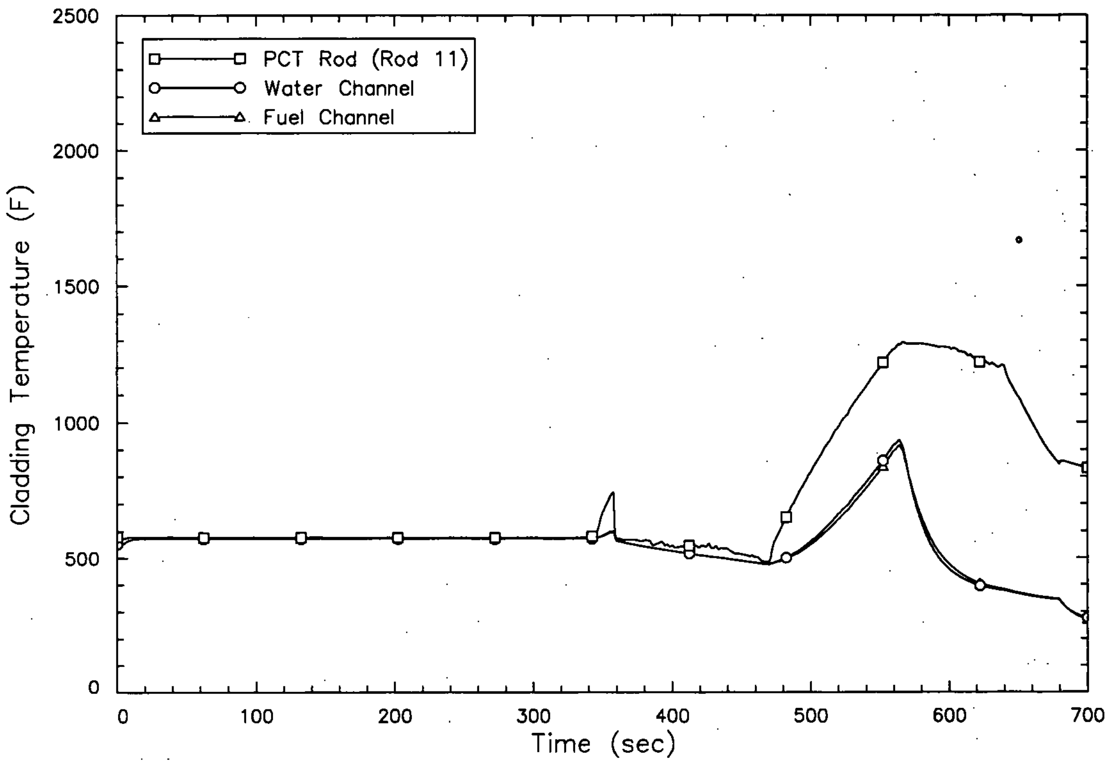

Figure A.27 0.05 ft^2 PD SF-BATT TOP 108F
Cladding Temperatures

Attachment 5 to PLA-6250
Corrected Figures 4.14 and 4.15 of PLA-6230
(Non-Proprietary Version)

[

]

Figure 4.14 Comparison of Interpolation Process Using Void Fractions of 0.0, 0.4, and 0.8 and Void Fractions of 0.0, 0.45, and 0.9

[

]

Figure 4.15 Comparison of the Measured Local Quality for ATRIUM-10 Void Data and Exit Quality for Typical Reactor Conditions

Attachment 3 to PLA-6250
AREVA NP, Inc.
Affidavit

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Jerald S. Holm. I am Manager, Product Licensing, for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the AREVA NP information contained in the PPL letter PLA-6250, *Susquehanna Steam Electric Station Proposed License Amendment No. 285 for Unit 1 Operating License No. NPF-14 and Proposed License Amendment No. 253 for Unit 2 Operating License No. NPF-22 Extended Power Uprate Application Re: Reactor Systems Technical Review Request for Additional Information Responses*, and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

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

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be

withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information".

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

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

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

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

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

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

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

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available,

on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

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

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

SUBSCRIBED before me this 26th

day of July, 2007.

Susan K. McCoy
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 1/10/2008