PWROG
PWR Owners Group

Program Management Office
4350 Northern Pike
Monroeville, Pennsylvania 15146

WCAP-16793-NP, Rev.0 (Non-Proprietary)
Project No. 694

March 2, 2009

OG-09-84

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-001

Subject: Pressurized Water Reactor Owners Group
Responses to the NRC Request for Additional Information (RAI) on WCAP-16793-NP, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid" (PA-SEE-0312)

References:

1. PWROG Letter, F. Schiffley to document Control Desk, "Submittal of WCAP-16793-NP, Revision 0, "Evaluation of Long-Term Core Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirulating Fluid," OG-07-264, June 4, 2007.

2. NRC Letter, Holly D. Cruz of NRR to Gordon Bischoff of PWROG, "Request for Additional Information Re: Pressurized Water Reactor Owners Group (PWROG) Topical Report (TR) WCAP-16793-NP, Revision 0, "Evaluation of Long-Term Core Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirulating Fluid", (TAC No. MD5891), August 22, 2008.

3. NRC e-mail, Paul Klein of NRR to Paul Pyle of Westinghouse, "Additional RAI on Retrograde Solubility", January 23, 2009.

In June 2007, the Pressurized Water Reactor Owners Group (PWROG) submitted WCAP-16793-NP (Non-Proprietary), Rev. 0, "Evaluation of Long-Term Core Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirulating Fluid," for review and approval (Reference 1). In August 2008, as a result of the Advisory Committee on Reactor Safeguards review of WCAP-16793-NP in March 2008, NRC staff provided a formal Request for Additional Information (RAI) (Reference 2) for WCAP-16793-NP.

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In January 2009, Paul Klein, NRC, transmitted an email containing an additional RAI associated with WCAP-16793-NP.

Enclosure 1 to this letter provides the RAI responses to the 14 questions received in Reference 2 and to the one question received in Reference 3.

These RAI responses are being provided to support issuance of the draft Safety Evaluation on WCAP-16793-NP.

Following receipt of the Safety Evaluation for WCAP-16793, this letter will be incorporated into the approved version and will be issued as WCAP-16793-NP-A, Revision 0.

If you have any questions concerning this matter, please feel free to call Ken Nemit at 412-374-6388.

Sincerely,

Kenneth J Nemit Approving for Dennis Buschbaum

Dennis Buschbaum, Chairman
Pressurized Water Reactor Owners Group

DEB:KJN:rfn

Enclosure 1

cc: PWROG Management Committee
PWROG Systems & Equipment Engineering Subcommittee
PWROG PMO
T.S. Andreychek
T.D. Croyle
D.J. Fink
R.J. Schomaker – AREVA NP
P.V. Pyle

1) WCAP-16793-NP, Appendix B, presents analyses of the effect of core inlet blockage using the WCOBRA/TRAC analysis code. Since the report was prepared additional analyses have been performed to determine the blockage level that would reduce core flow below that necessary to match coolant boil-off. Please provide documentation of the additional analyses that have been performed, including figures, for the integrated core inlet and exit flow, peak cladding temperature, core collapsed liquid level, core exit void fraction, and core pressure drop for the bounding conditions. The results should be presented for each case analyzed up to and including the blockage level for which boiloff is no longer satisfied.

RESPONSE:

Introduction:

Several additional WCOBRA/TRAC (WC/T) analyses were peformed in support of WCAP-16793-NP. The WC/T runs were performed at the request of the Advisory Committee for Reactor Safeguards (ACRS) with the purpose of determining the blockage level (either using a reduction in area or increase loss coefficient) that would reduce core flow below that necessary to match coolant boil-off. As requested by this RAI, the documentation includes figures of the integrated core inlet and exit flow, peak cladding temperature, core collapsed liquid level, core exit void fraction, and core pressure drop for the bounding conditions.

Method Discussion & Input:

The WC/T runs made in support of WCAP-16793-NP are described in Reference 1. As stated in the above Introduction Section, in order to assess the blockage level that would reduce core flow below that necessary to match coolant boil-off, modifcations were made to the flow area and loss coefficient input values used in the original runs and the calculations repeated.

The base case for the calculation results presented in this RAI response is Case 2, or the more restricted flow area case, from Section 6.0 of Reference 1. The Darcy equation defines pressure drop as being proportional to the form-loss coefficient and inversely proportional to the flow area squared. Using this principle, two separate approaches were taken to determine the blockage level needed to preclude sufficient flow into the core to provide for long-term core cooling. The first approach considered an area reduction while maintaining the form-loss coefficients. The second approach considered form-loss coefficient increases while maintaining the flow area constant.

- For the first approach, the flow area of the hot channel, Channel 13, was reduced. The input value of the hydraulic loss coefficient, C_D, for the other channels into the core, Channels 10, 11, 12 and 13 remained the same as the base case. As discussed on pages 26-27 of Reference 1, for this modeling approach, flow will only enter the core through the hot channel (Channel 13). To maintain the total core flow area, the adjacent channel (Channel 11, representing an "average channel") flow area was increased to offset the change in flow area to Channel 13. This change is needed to preserve the total core flow area; however, no flow will enter the core through Channel 11.
- For the second approach, the loss coefficients were increased in increments until boil-off could not be matched.

Areas Used in Reduced Flow Area Approach:

The flow area values used in the two flow area reduction cases are as listed below.

Channel 13 50% Flow Reduction Case:

Channel 13 Flow Area = 23.76 * (0.50) = 11.88 in^2

Channel 11 Flow Area = 1782 + 23.76 * (0.50) = 1794. in^2

Channel 13 80% Flow Reduction Case:

Channel 13 Flow Area = 23.76 * (0.20) = 4.752 in^2

Channel 11 Flow Area = 1782 + 23.76 * (0.80) = 1801. in^2

Due to time constraints, the transient run time was reduced from 2400 seconds to 1500 seconds for the calculations that were performed. The transient calculation time of 1500 seconds is sufficient to demonstrate whether the reduction in core flow would be sufficient to match boil-off.

C_D Values used in Increased Loss Coefficient Approach:

In order to determine the blockage level that would reduce core flow below that necessary to match coolant boil-off, the inlet core loss coefficients were increased in increments until boil-off could not be matched. The computer calculations made include uniform loss coefficients of 50,000, 100,000, and 1,000,000. The only changes required for these runs were updates to the variables used to activate the dimensionless loss coefficient ramp logic. For these cases, the C_D input value was changed from 10^9 to desired C_D value to reduce flow through peripheral channels, the average channels and the hot assembly channel instead of block flow. Also, the feature to allow the C_D value of all core inlet channels to vary as a function of time was enabled.

Three runs were made; C_D = 50,000, C_D = 100,000 and C_D = 1,000,000. The increase in C_D values to the desired values was accomplished over a 30 second time interval. The ramp up started at the time of switchover from injection from the BWST/RWST to recirculation from the sump, transient time t = 1200 seconds and was completed at transient time t = 1230 seconds.

Again, due to time constraints, the transient run time was reduced from 2400 seconds to 1500 seconds for the calculations that were performed. The transient calculation time of 1500 seconds is sufficient to demonstrate whether the reduction in core flow would be sufficient to match boil-off.

Results from Flow Area Reduction Runs:

The first flow reduction run performed reduced the hot channel (Channel 13) flow area by 50%, which yields a total core inlet flow reduction of 99.7% compared to an unblocked core. The requested plots for this case are shown in Figures 1 through 7. Figures 1 and 2 show comparisons of the integrated core inlet flow and the core boil-off rate. As shown, even with the increase in core blockage, the flow that enters the core is still in excess of the boil-off rate. Figure 3 displays the integrated liquid flow at the core exit. The figures illustrates that, although liquid in excess of that needed to keep the core quenched enters the core, every little liquid flow is present at the core exit after the blockage occurs. The Peak Cladding Temperature (PCT) is shown in Figure 4. There are no significant PCT excursions after the core is blocked. Figure 5 displays the collapsed liquid level of the average assembly core channel (Channel 11 of Figure 6.3.4-3 in Reference 1). The figure shows that the collapsed liquid level drops slightly at the time blockage occurs, however, the liquid level continues to increase even after the blockage to the hot channel (Channel 13) is fully implemented at 1230 seconds. The void fraction at the core exit shown in Figure 6 again illustrates that liquid is present at the top of the core which shows the flow that enters the core after blockage occurs is still in excess of the boil-off rate. The core pressure drop is displayed in Figure 7. The figure displays an increased pressure drop of roughly 2 psi as blockage at the core inlet is increased. As the conditions in the Reactor Coolant System (RCS) adjust to increase in core blockage, it is noticed that the core pressure drop fluctuates consistent with the core liquid level.

The next flow reduction run performed reduced the hot channel (Channel 13) flow area by 80%, which yields a total core inlet flow area reduction of 99.9%. The requested plots for this case are shown in Figures 8 through 14. Figures 8 and 9 show comparisons of the integrated core inlet flow and boil-off rate. As shown, with the increase in core blockage, the flow that enters the core can not match the boil-off rate. Since all the liquid entering the core at the inlet is boiled-off, there is no liquid flow at the core exit (as shown in Figure 10). In addition, Figure 11 shows that the PCT increases until the end of the transient once the core liquid level, shown in Figure 12, is reduced to a level that the core becomes unquenched. Continuing with the trend discussed above, the void fraction at the core exit (Figure 13) shows that only vapor is present. The core pressure drop is displayed in Figure 14. The figure displays an increased pressure drop of roughly 4 psi as blockage at the core inlet is increased and the core liquid level begins to stabilize.

These results indicate that a total core inlet area reduction of up to as much as 99.7% will still allow sufficient flow into the core to provide for removal of decay heat and assure long-term core cooling.

Results from Uniform Loss Coefficient Runs:

The first uniform loss coefficient run performed applied a uniform C_D of 50,000 at the core inlet. The requested plots for this case are shown in Figures 15 through 21. Figures 15 and 16 show comparisons of the integrated core inlet flow and boil-off rate. As shown, even with the increase of the loss coefficient at the inlet, the flow that enters the core is still in excess of the boil-off rate. (Note that the integrated mass flow behavior shown between time t = 1200 seconds and time t = 1250 seconds of Figure 16 is the result of the 30 second ramp-up of the hydraulic loss coefficient, C_D, to 50,000 that is initiated in the calculations at time t = 1200 seconds.) Figure 17 displays the integrated liquid flow at the core exit. The figures displays that liquid in excess of that needed to keep the core quenched enters the core and that liquid flow is present at the top of the core even after the increase of the loss coefficient at the inlet. The PCT is shown in Figure 18. There are no significant PCT excursions after the core inlet loss coefficient is increased. Figure 19 displays the collapsed liquid level of the average assembly core channel (Channel 11 of Figure 6.3.4-3 in Reference 1). The figure shows that the collapsed liquid level drops slightly at time blockage occurs, however, the liquid is maintained even after the increase in the loss coefficient at the inlet. The void fraction at the core exit shown in Figure 20 again illustrates that liquid is present at the top of the core which shows the flow that enters the core after the increase of the loss coefficient occurs is still in excess of the boil-off rate. The core pressure drop is displayed in Figure 21. The figure displays an increased pressure drop of roughly 2 psi as blockage at the core inlet is increased. As the conditions in the Reactor Coolant System (RCS) adjust to increase in core blockage, it is noticed that the core pressure drop fluctuates consistent with the core liquid level.

The second uniform loss coefficient run performed applied a uniform C_D of 100,000 at the core inlet. The requested plots for this case are shown in Figures 22 through 28. Figures 22 and 23 show comparisons of the integrated core inlet flow and boil-off rate. As shown, even with the further increase of the loss coefficient at the inlet, the flow that enters the core is still in excess of the boil-off rate. (Note that the integrated mass flow rate of Figure 23 shows a similar behavior as was shown in Figure 16. Again, this is due to the 30 second ramp-up of the hydraulic loss coefficient, C_D, to 100,000 that is initiated in the calculations at time t = 1200 seconds, but extends the behavior over a slightly longer period of time.) Figure 24 displays the integrated liquid flow at the core exit. The figures displays that liquid in excess of that needed to keep the core quenched enters the core and that some liquid flow is still present at the top of the core even after the increase of the loss coefficient at the inlet. The PCT is shown in Figure 25. There are no significant PCT excursions after the core inlet loss coefficient is increased. Figure 26 displays the collapsed liquid level of the average assembly core channel (Channel 11 of Figure 6.3.4-3 in Reference 1). The figure shows that the collapsed liquid level drops slightly at time blockage occurs, however, the liquid level recovers even after the increase in the loss coefficient at the inlet. The void fraction at the core exit shown in Figure 27 again illustrates that liquid is present at the top of the core which shows the flow that enters the core after the increase of the loss coefficient occurs is still in excess of the boil-off rate. The core pressure drop is displayed in Figure 28. The figure displays an increased pressure drop of roughly 2 psi as blockage at the core inlet is increased. As the conditions in the Reactor

Coolant System (RCS) adjust to increase in core blockage, it is noticed that the core pressure drop fluctuates consistent with the core liquid level.

The next uniform loss coefficient run performed applied a uniform C_D of 1,000,000 at the core inlet. The requested plots for this case are shown in Figures 29 through 35. Figures 29 and 30 show comparisons of the integrated core inlet flow and boil-off rate. As shown, with the increase in core blockage, the flow that enters the core can not match the boil-off rate. Since all the liquid entering the core at the inlet is boiled-off, there is no liquid flow at the core exit (as shown in Figure 31). In addition, it is displayed in Figure 32 that the PCT increases until the end of the transient once the core liquid level, shown in Figure 33, is reduced to a level that the core becomes unquenched. Continuing with the trend discussed above, the void fraction at the core exit (Figure 34) shows that only vapor is present. The core pressure drop is displayed in Figure 35. The figure displays an increased pressure drop of roughly 4 psi as blockage at the core inlet is increased and the core liquid level begins to stabilize.

The results indicate that an increase in the form loss coefficient at the core inlet of up to CD = 100,000 for the limiting plant and fuel load design will allow for sufficient flow into the core to remove decay heat and provide for long-term core cooling.

References:

1. WCAP-16793-NP, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," May 2007.

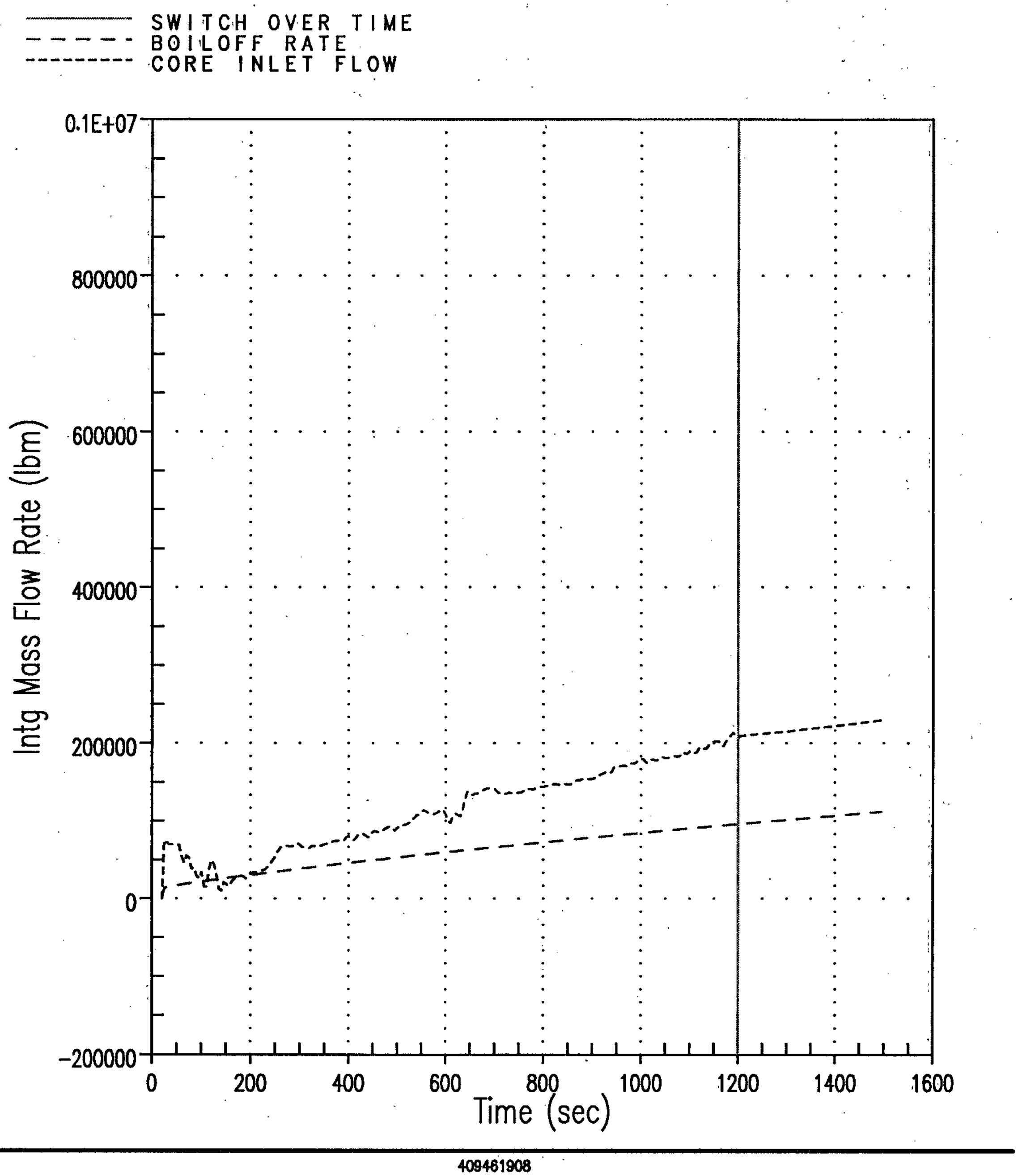

Figure 1: **Integrated Core Flow vs. Core Boil-off for Channel 13 Flow Reduction 50%**

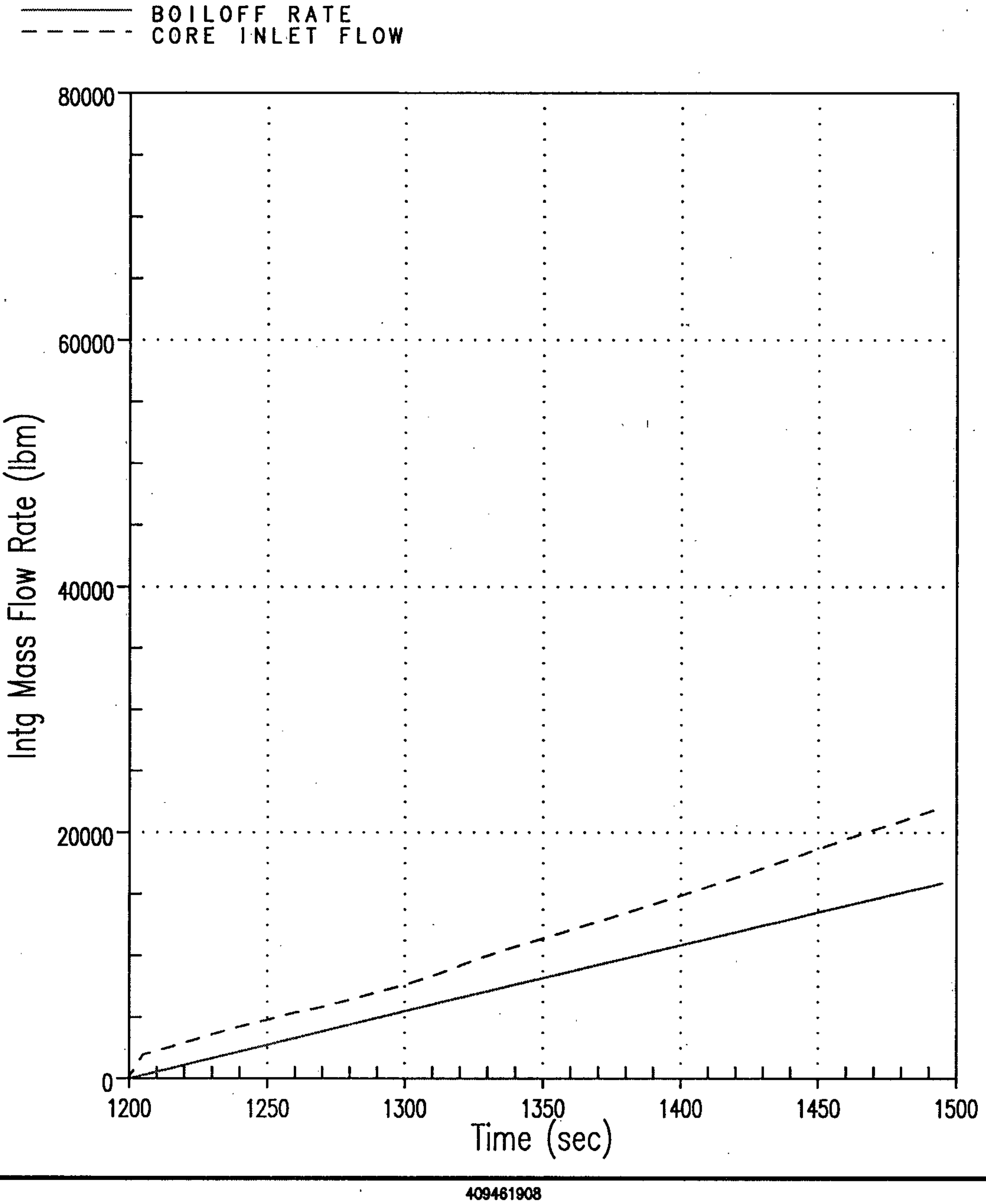

Figure 2: **Integrated Core Flow vs. Core Boil-off for Channel 13 Flow Reduction 50% Case (Shifted Scale)**

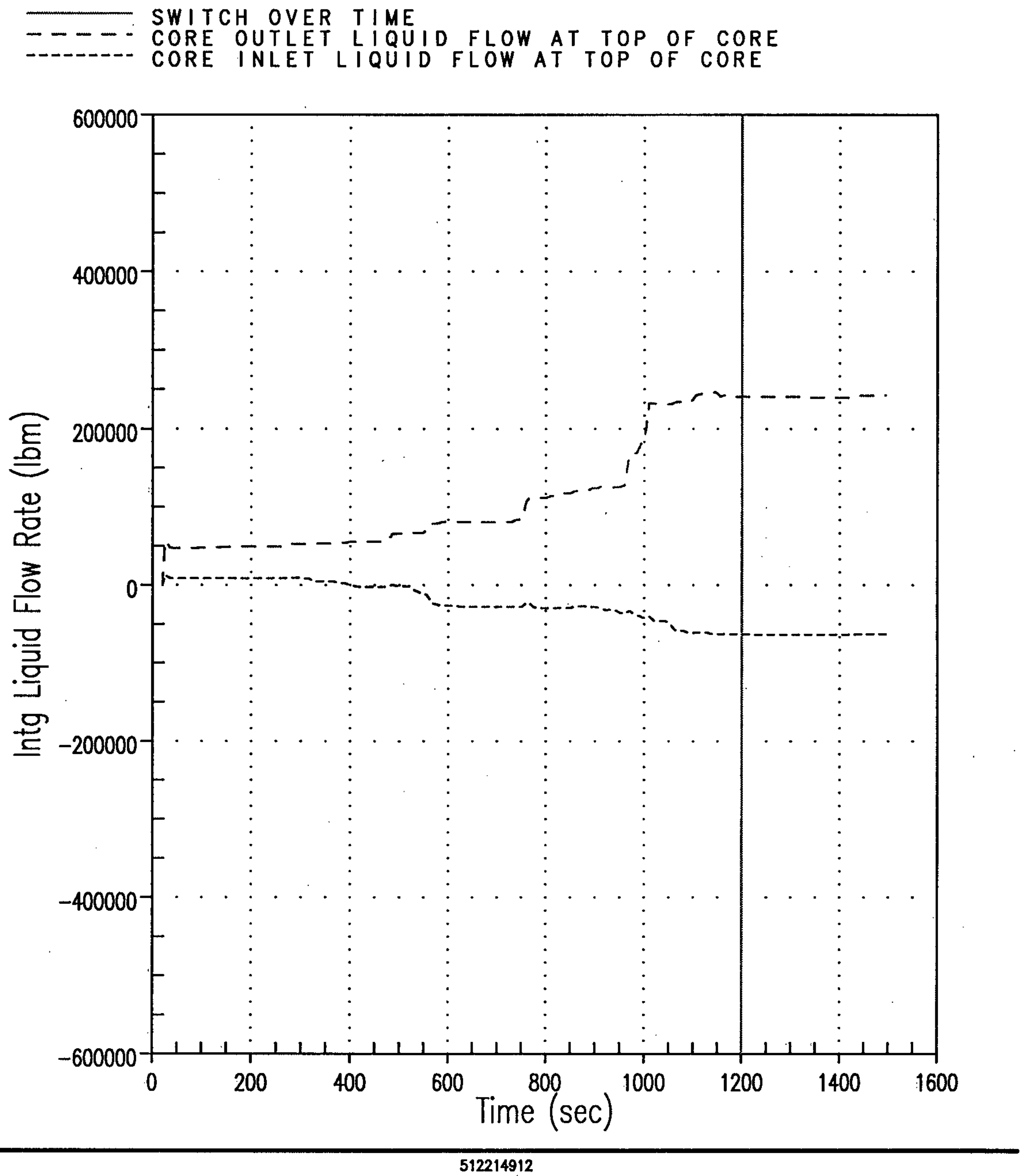

Figure 3: **Total Integrated Liquid Flow at the Top of the Core for Channel 13 Flow Reduction 50% Case (Positive/Outlet flow represents HA, GT, AVG channels; Negative/Inlet flow represent LP channel)**

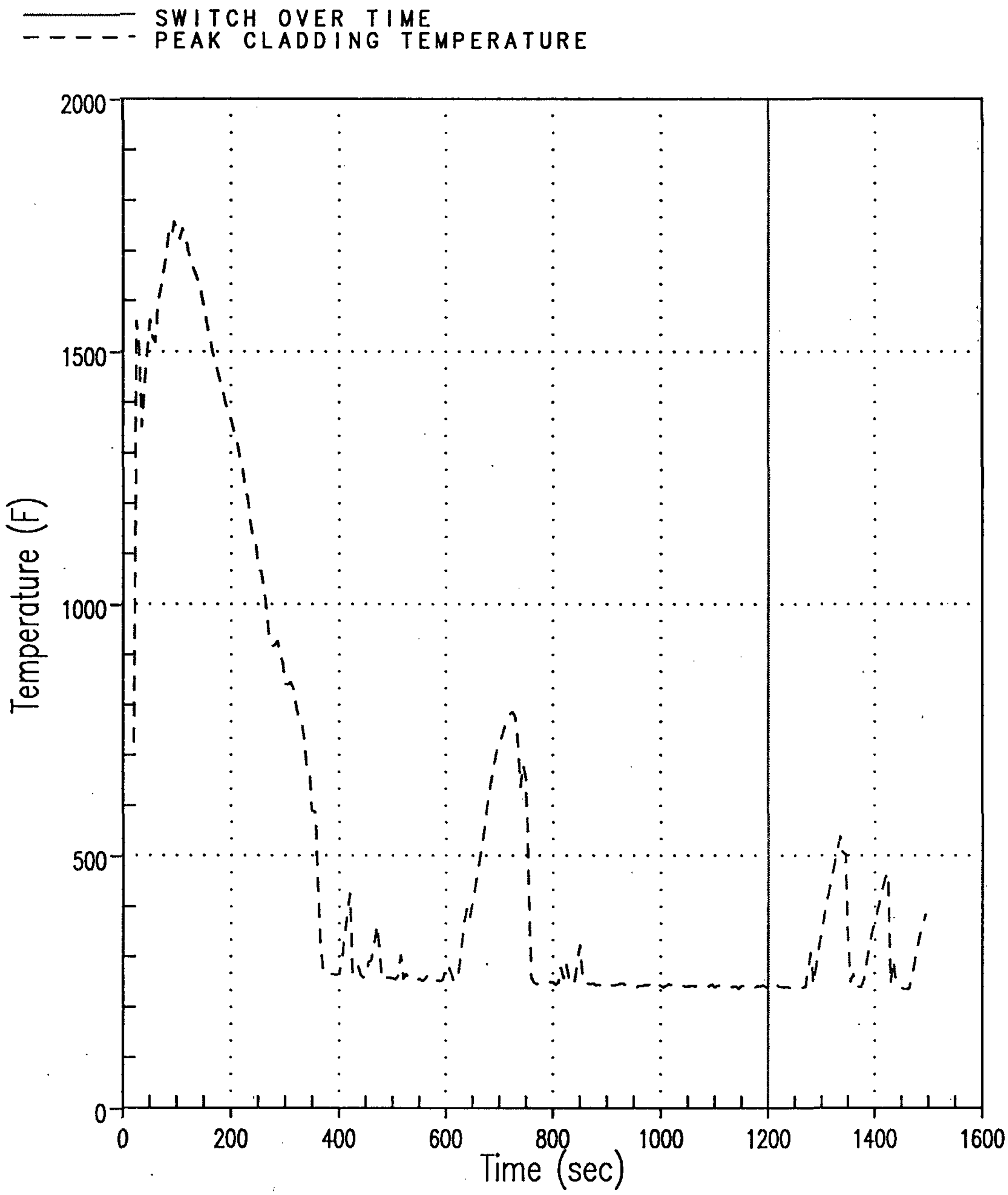

Figure 4: **Hot Rod PCT for Channel 13 Flow Reduction 50% Case**

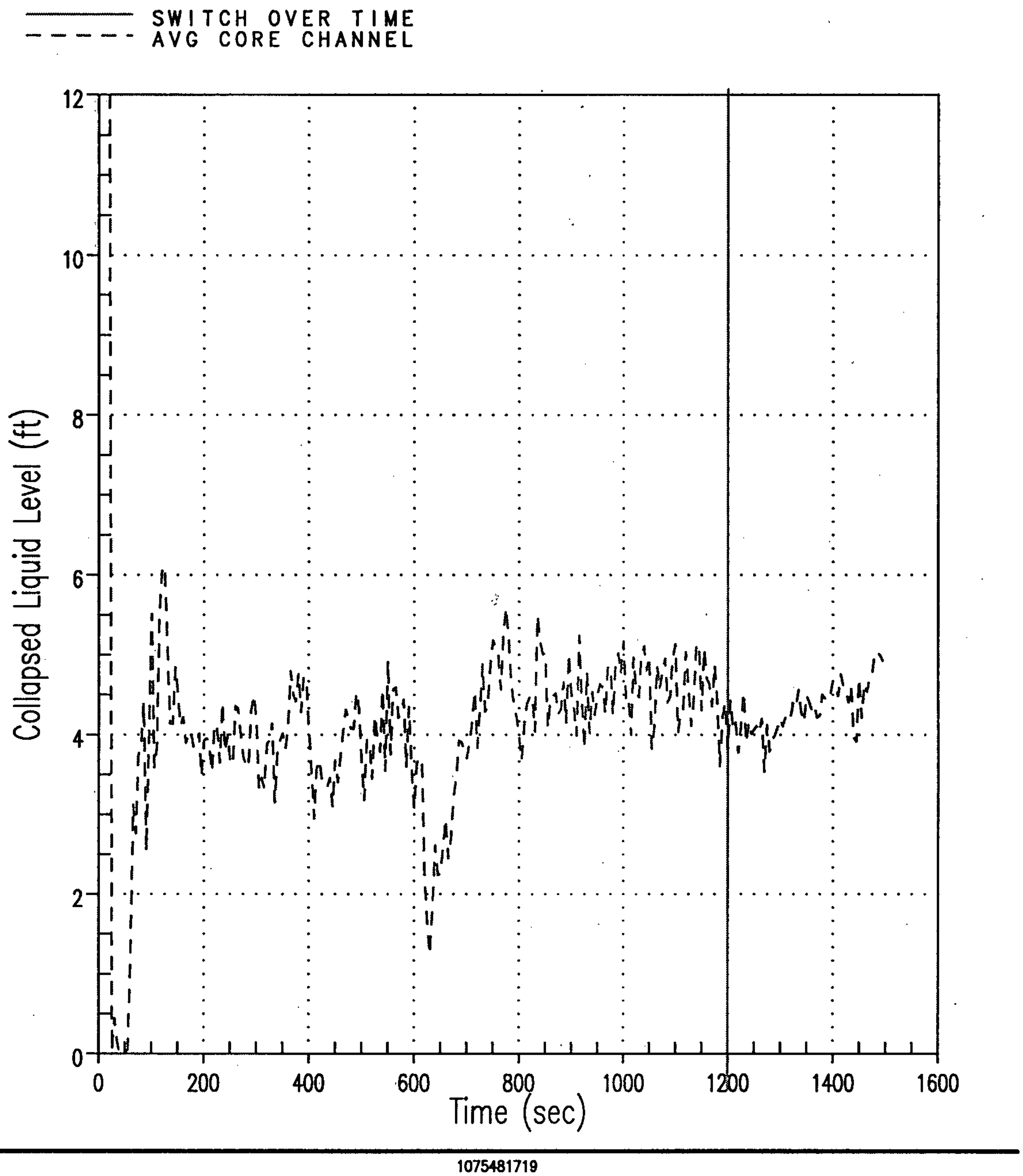

Figure 5: **Average Core Channel Collapsed Liquid Level for Channel 13 Flow Reduction 50% Case**

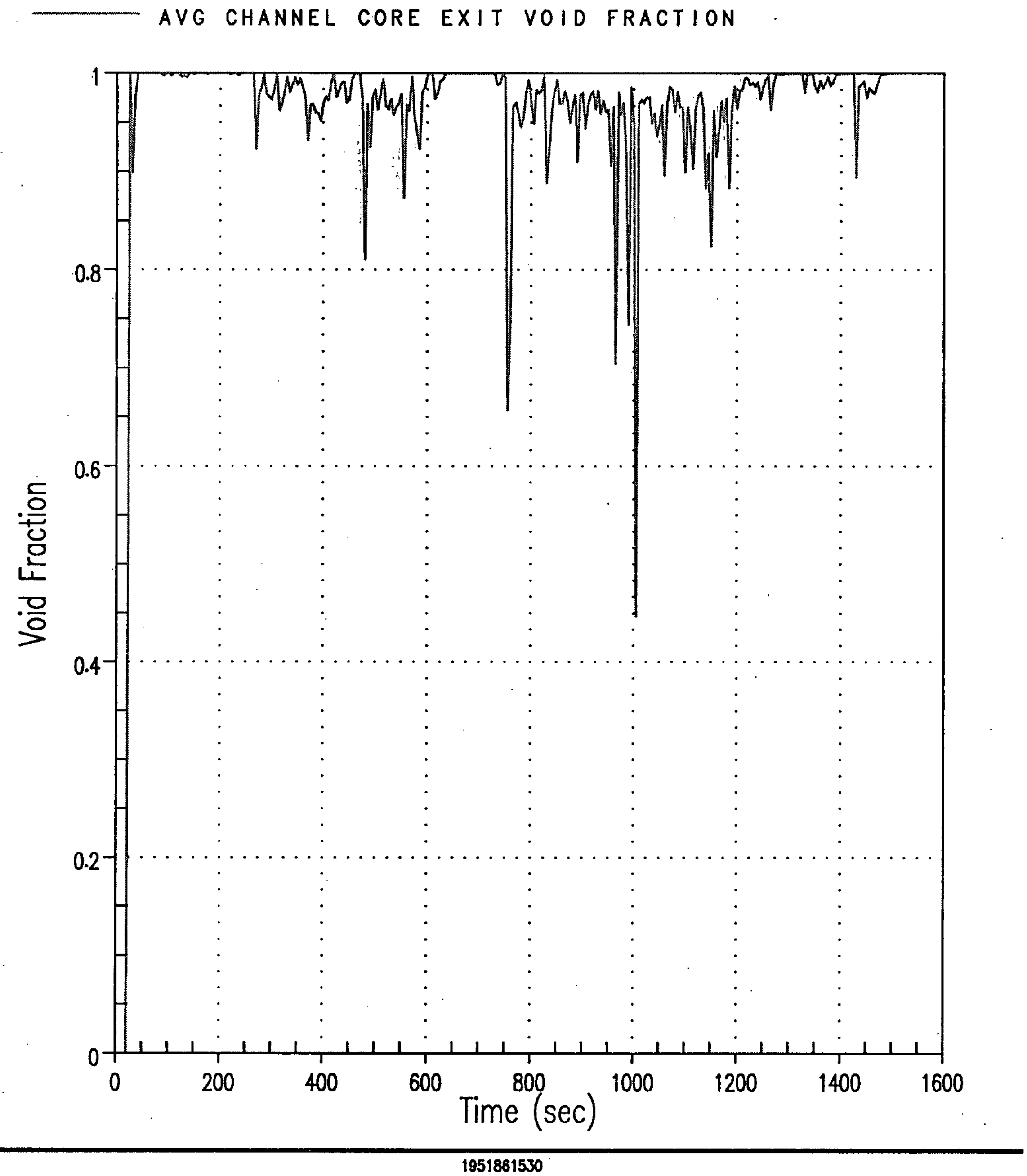

Figure 6: **Void Fraction at the Exit of the Average Core Channel for Channel 13 Flow Reduction 50% Case**

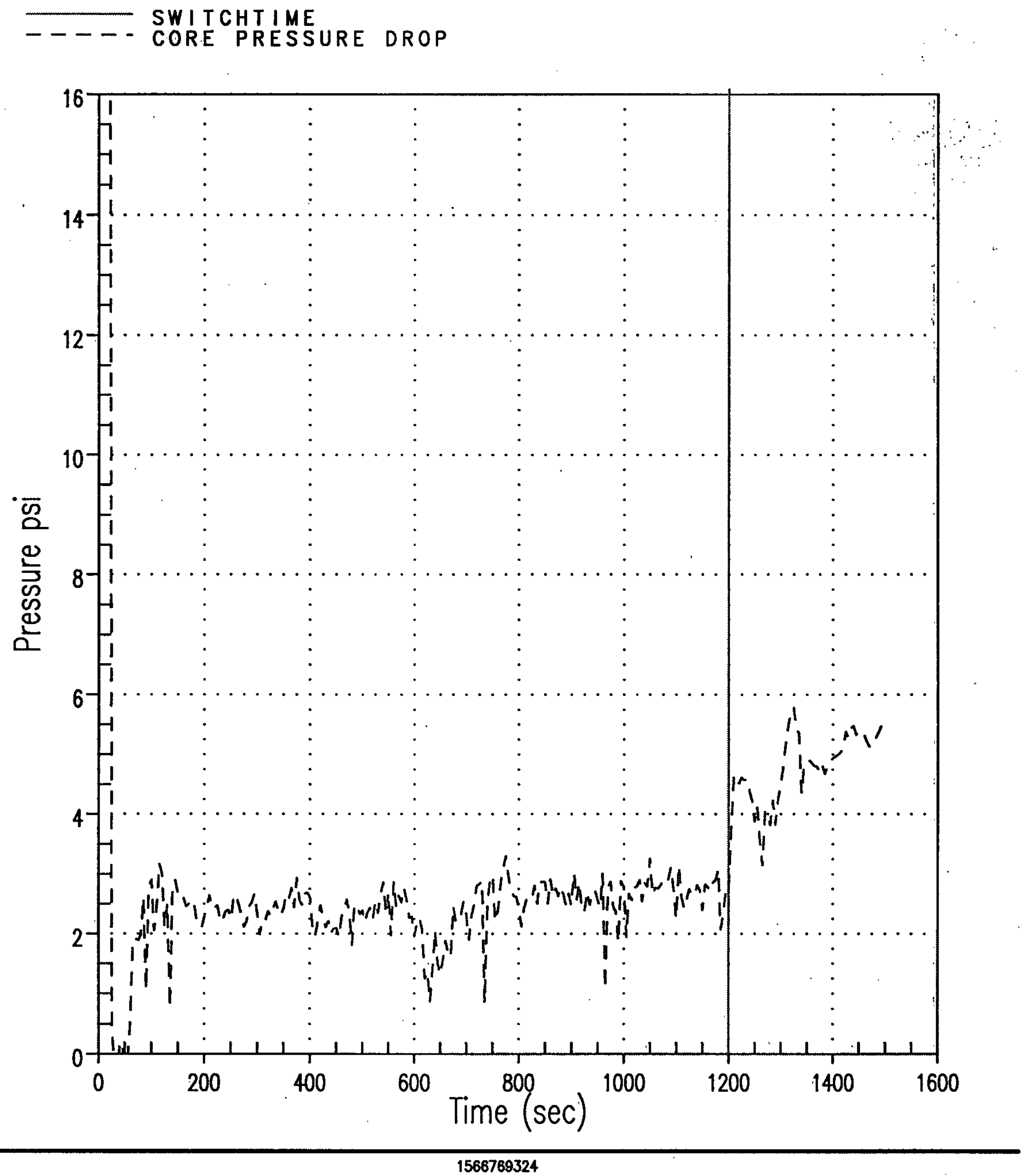

Figure 7: **Core Pressure Drop for Channel 13 Flow Reduction 50% Case**

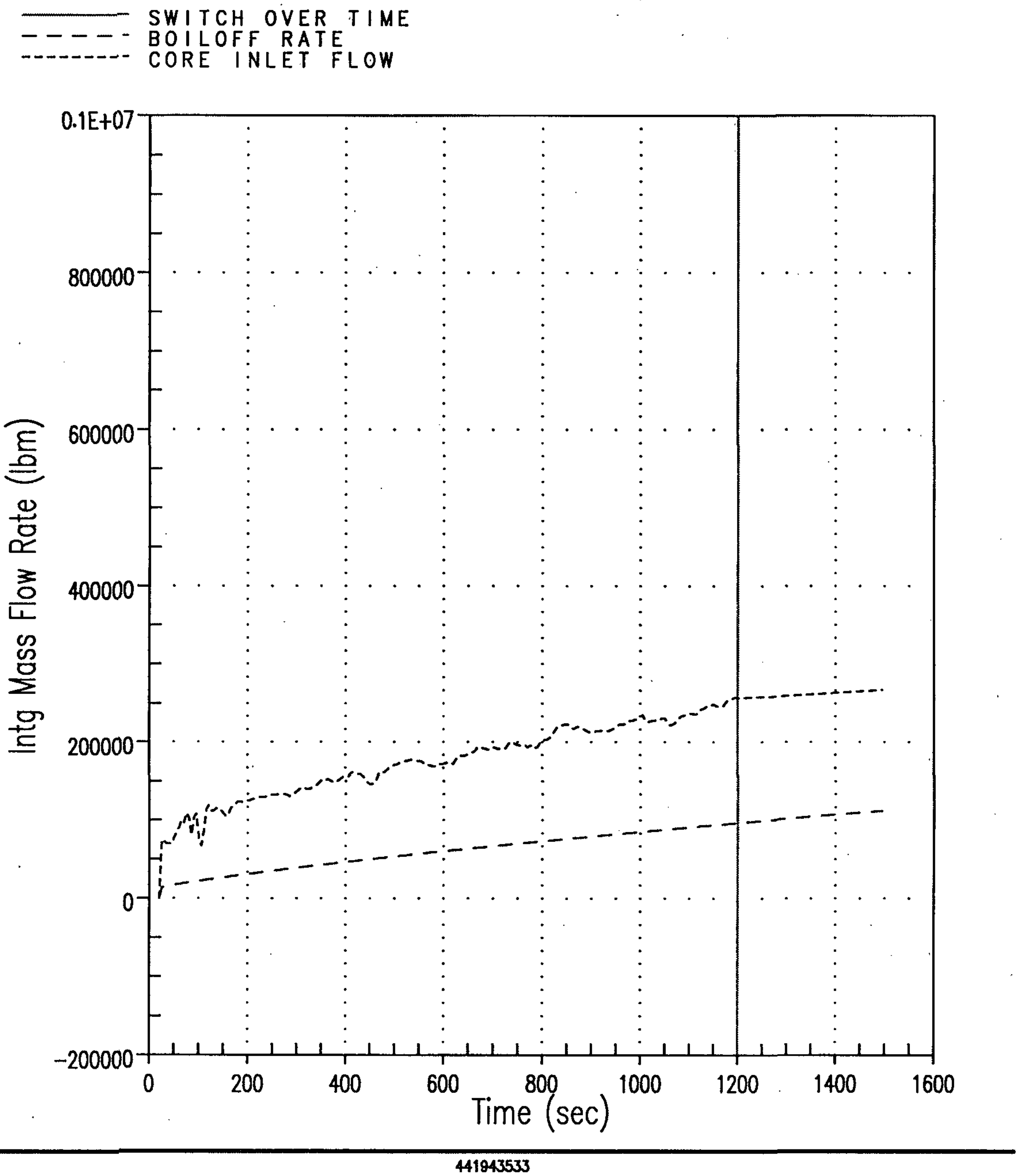

Figure 8: **Integrated Core Flow vs. Core Boil-off for Channel 13 Flow Reduction 80% Case**

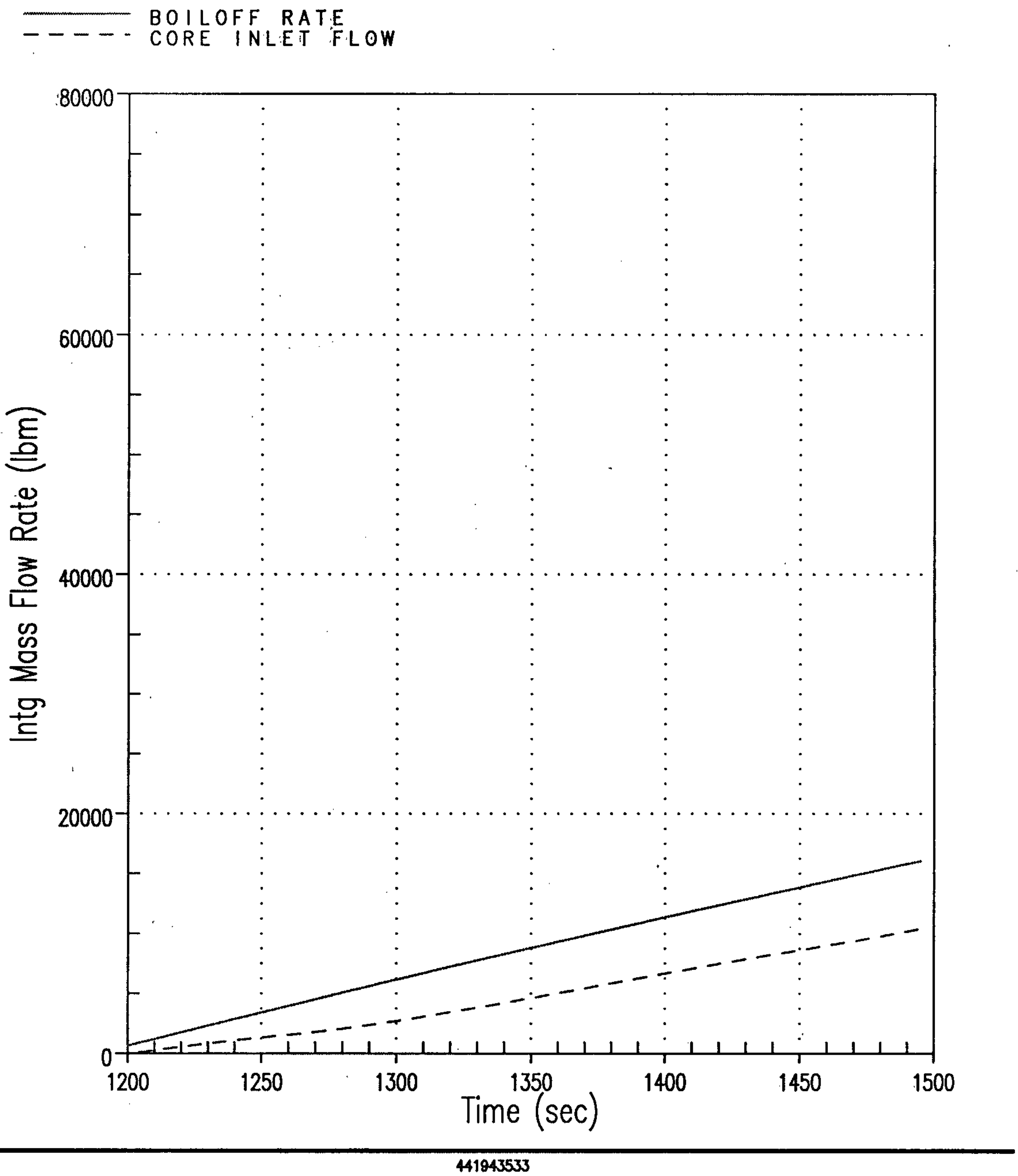

Figure 9: **Integrated Core Flow vs. Core Boil-off for Channel 13 Flow Reduction 80% Case (Shifted Scale)**

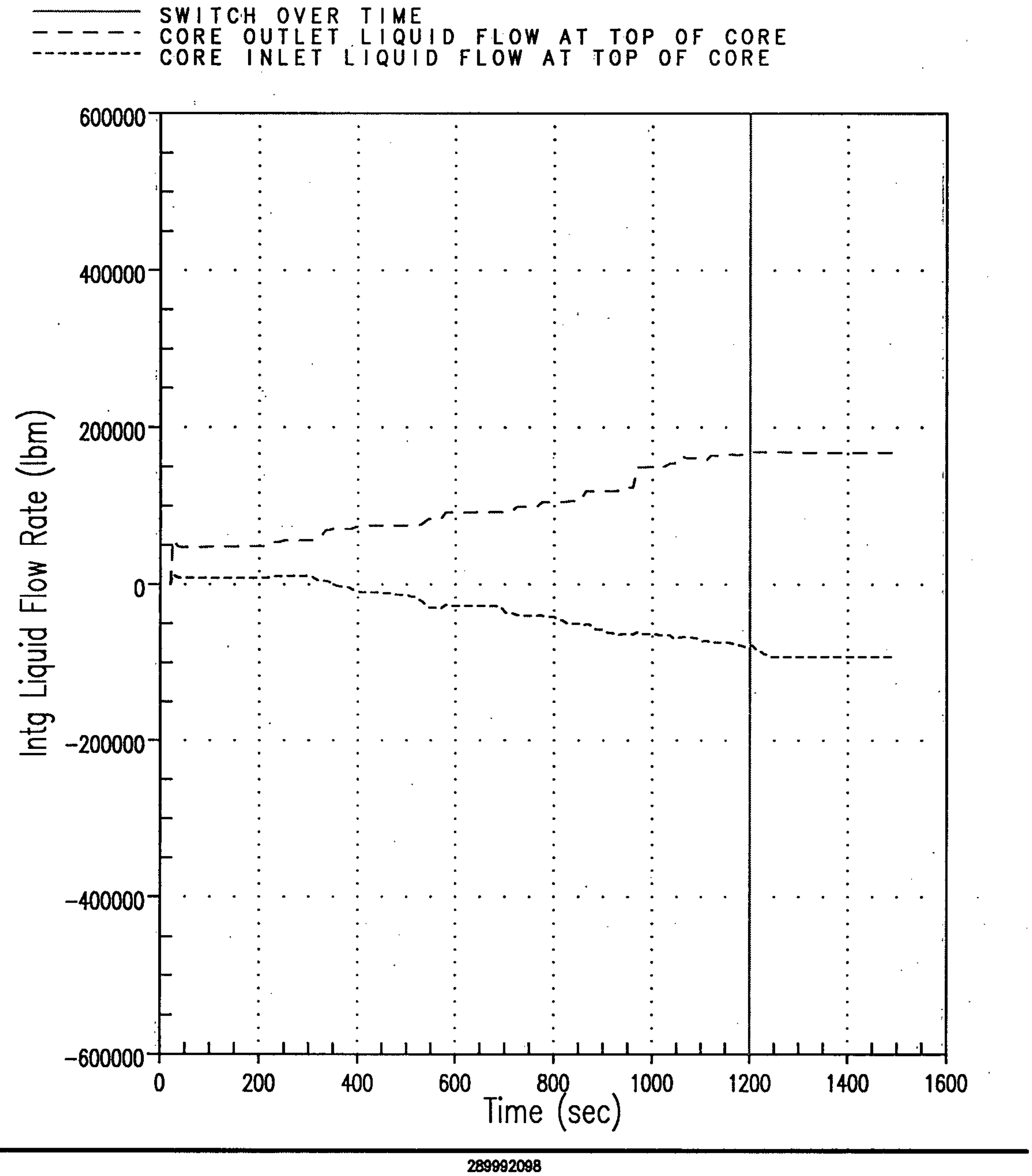

Figure 10: **Total Integrated Liquid Flow at the Top of the Core for Channel 13 Flow Reduction 80% Case (Positive/Outlet flow represents HA, GT, AVG channels; Negative/Inlet flow represent LP channel)**

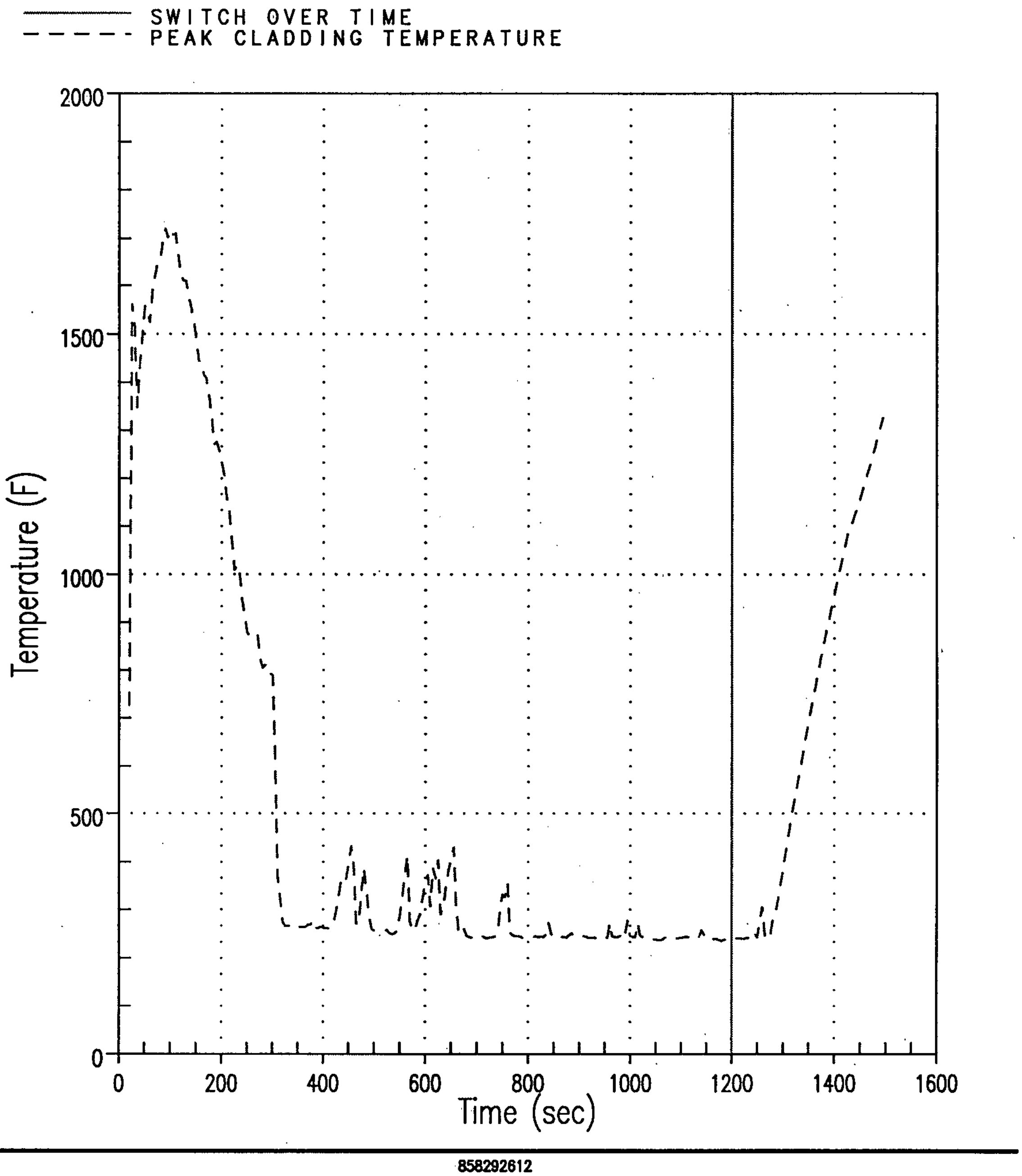

Figure 11: **Hot Rod PCT for Channel 13 Flow Reduction 80% Case**

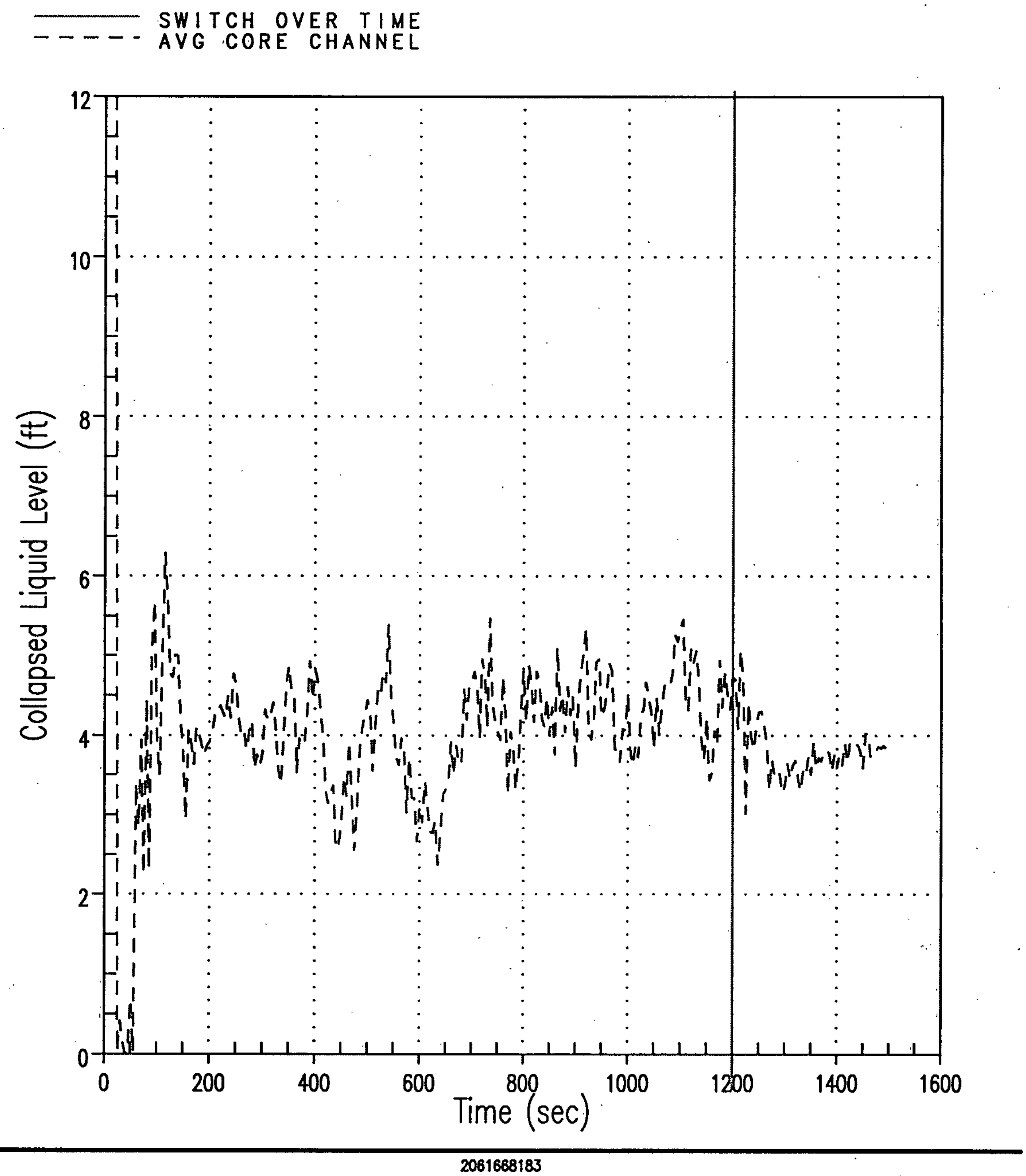

Figure 12: **Average Core Channel Collapsed Liquid Level for Channel 13 Flow Reduction 80% Case**

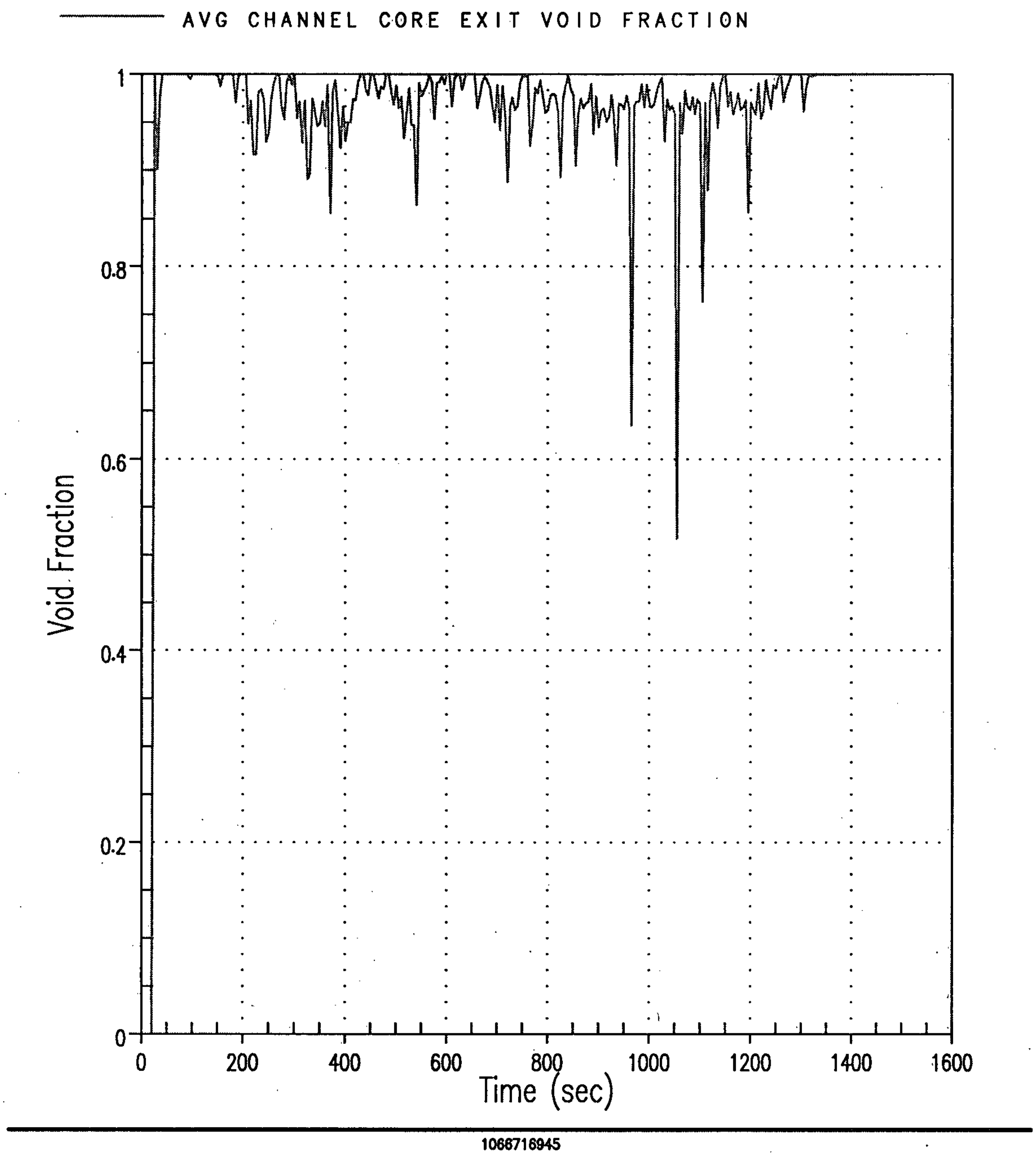

Figure 13: **Void Fraction at the Exit of the Average Core Channel for Channel 13 Flow Reduction 80% Case**

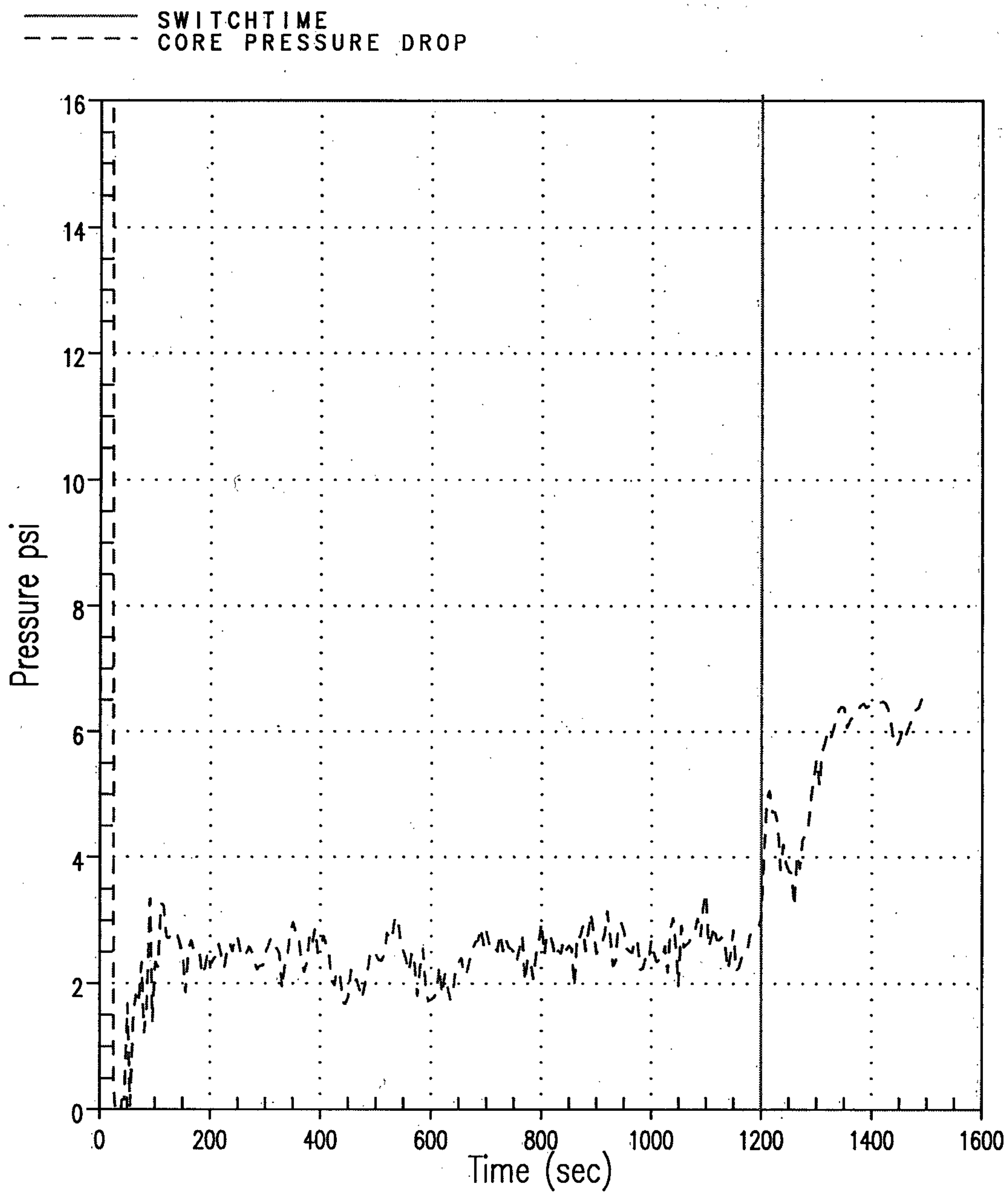

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Figure 14: Core Pressure Drop for Channel 13 Flow Reduction 80% Case

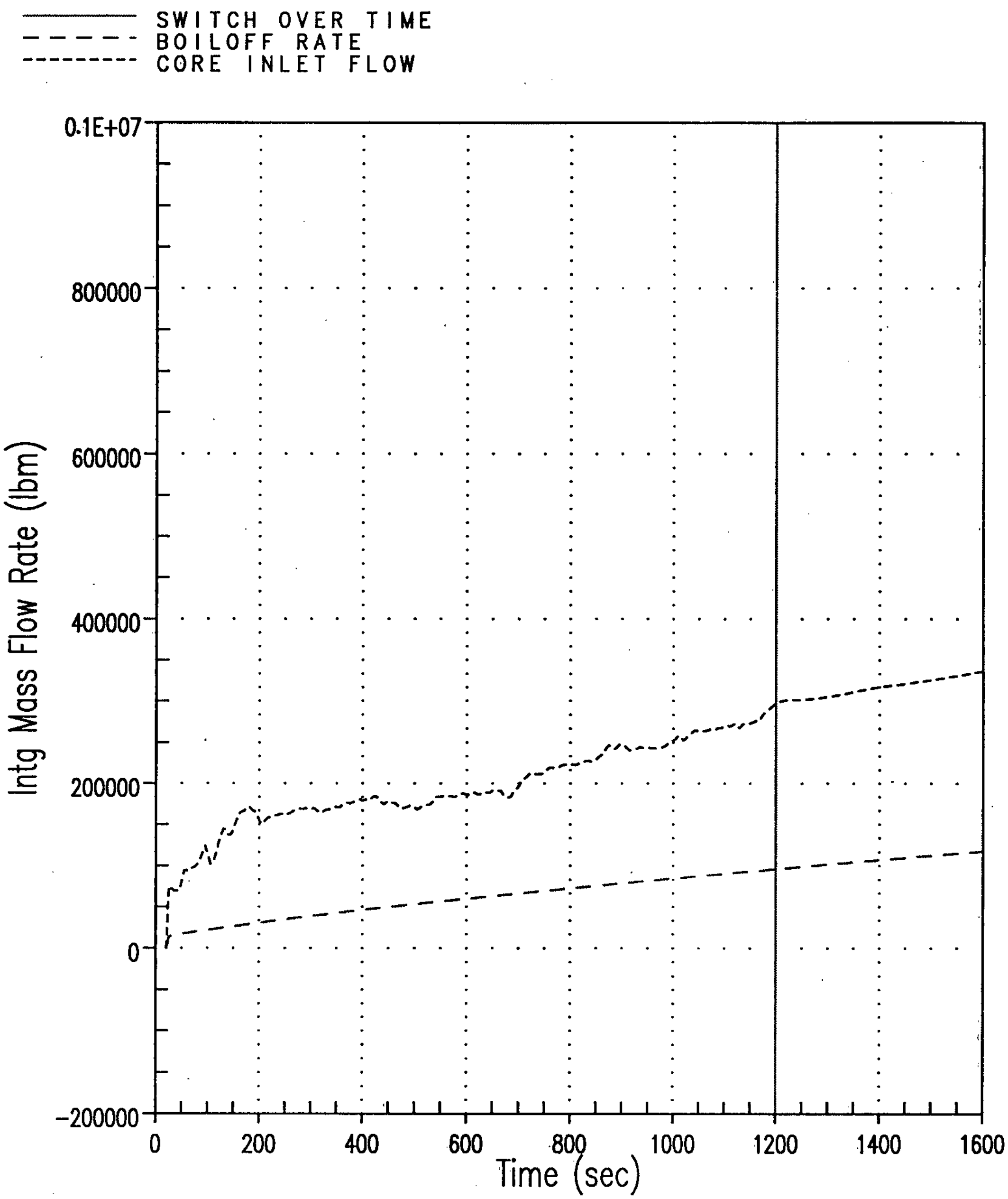

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Figure 15: Integrated Core Flow vs. Core Boil-off for Uniform C_D = 50,000 Case

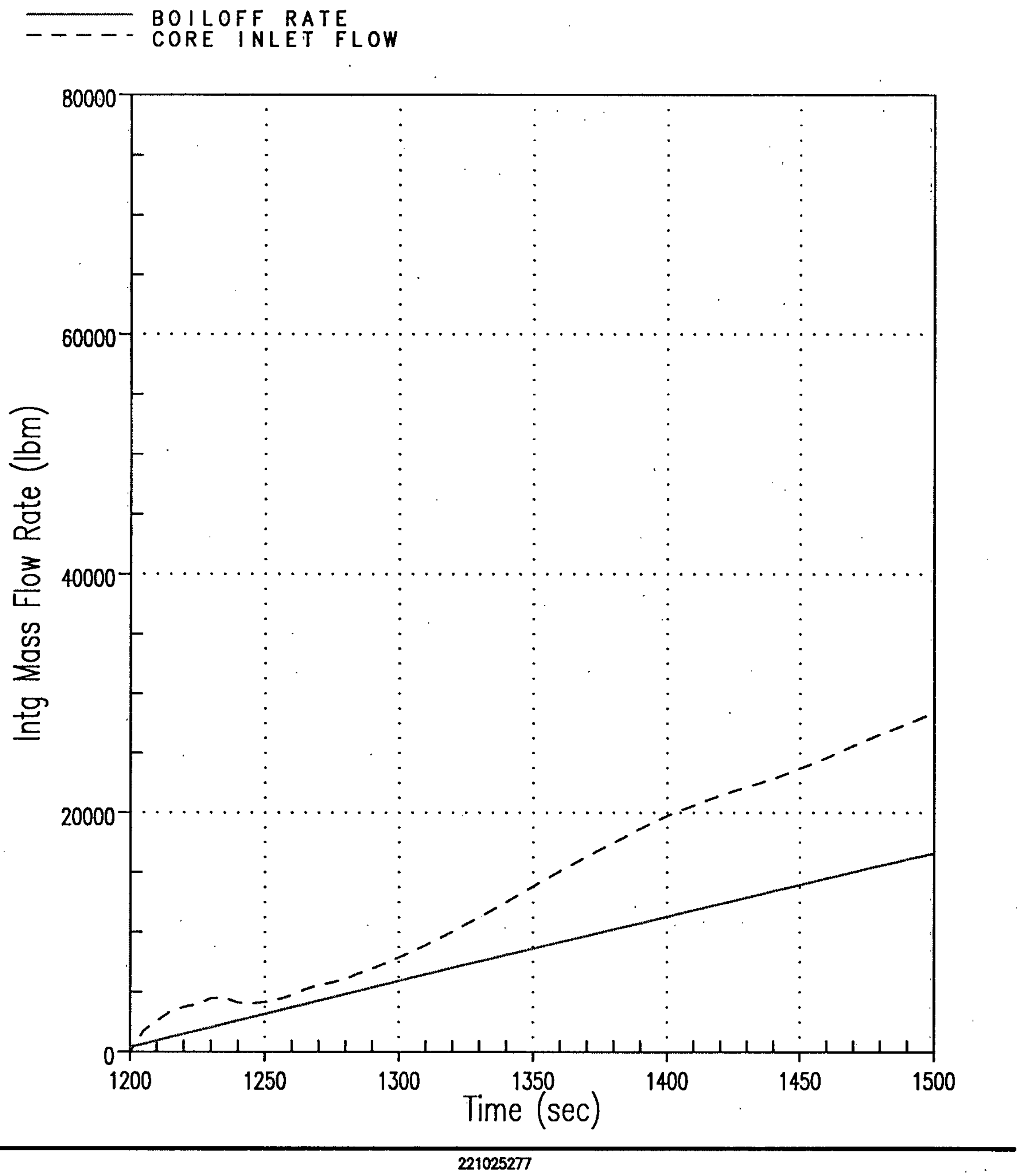

Figure 16: **Integrated Core Flow vs. Core Boil-off for Uniform C_D = 50,000 Case (Shifted Scale)**

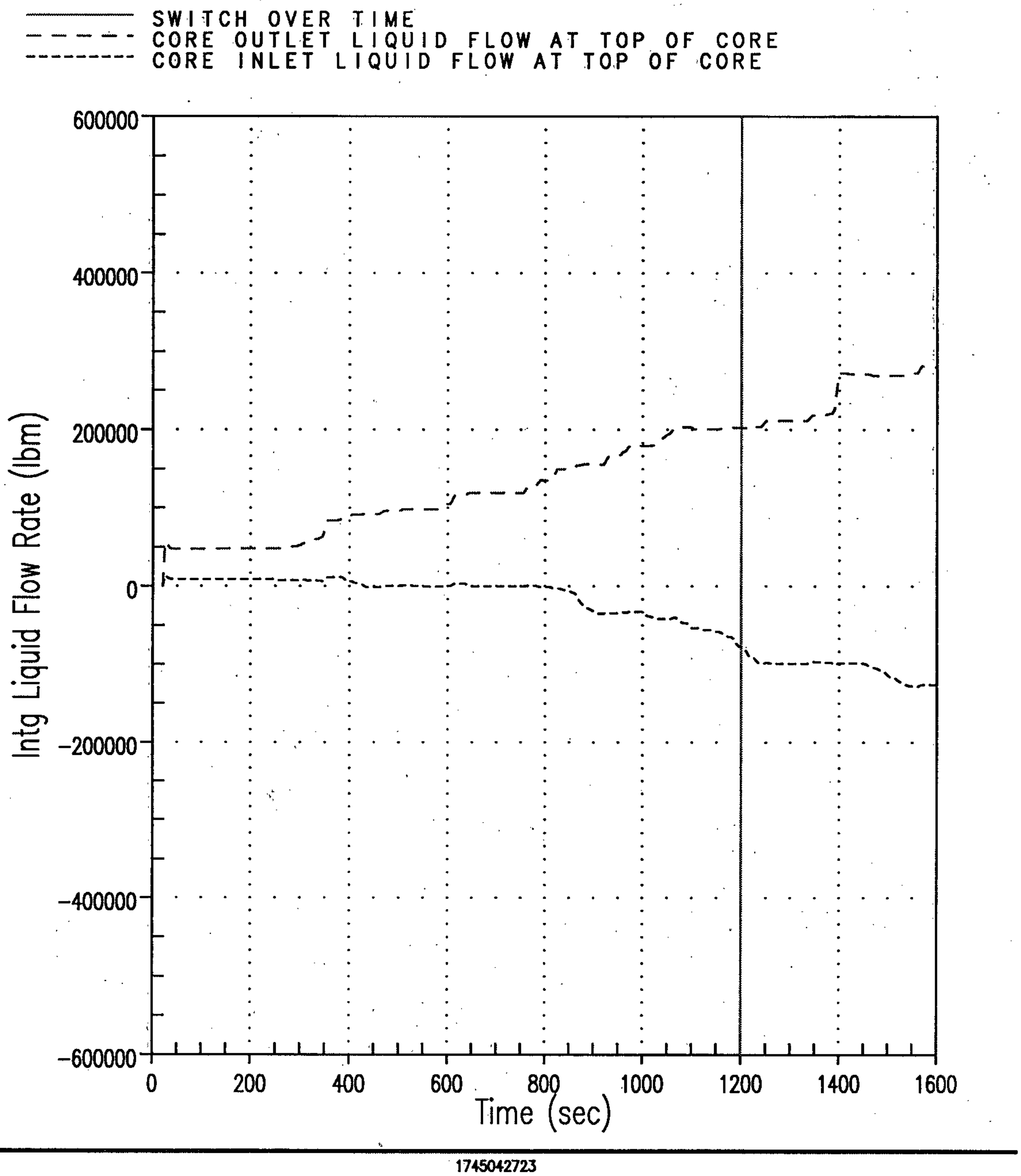

Figure 17: **Total Integrated Liquid Flow at the Top of the Core for Uniform C_D = 50,000 Case (Positive/Outlet flow represents HA, GT, AVG channels; Negative/Inlet flow represent LP channel)**

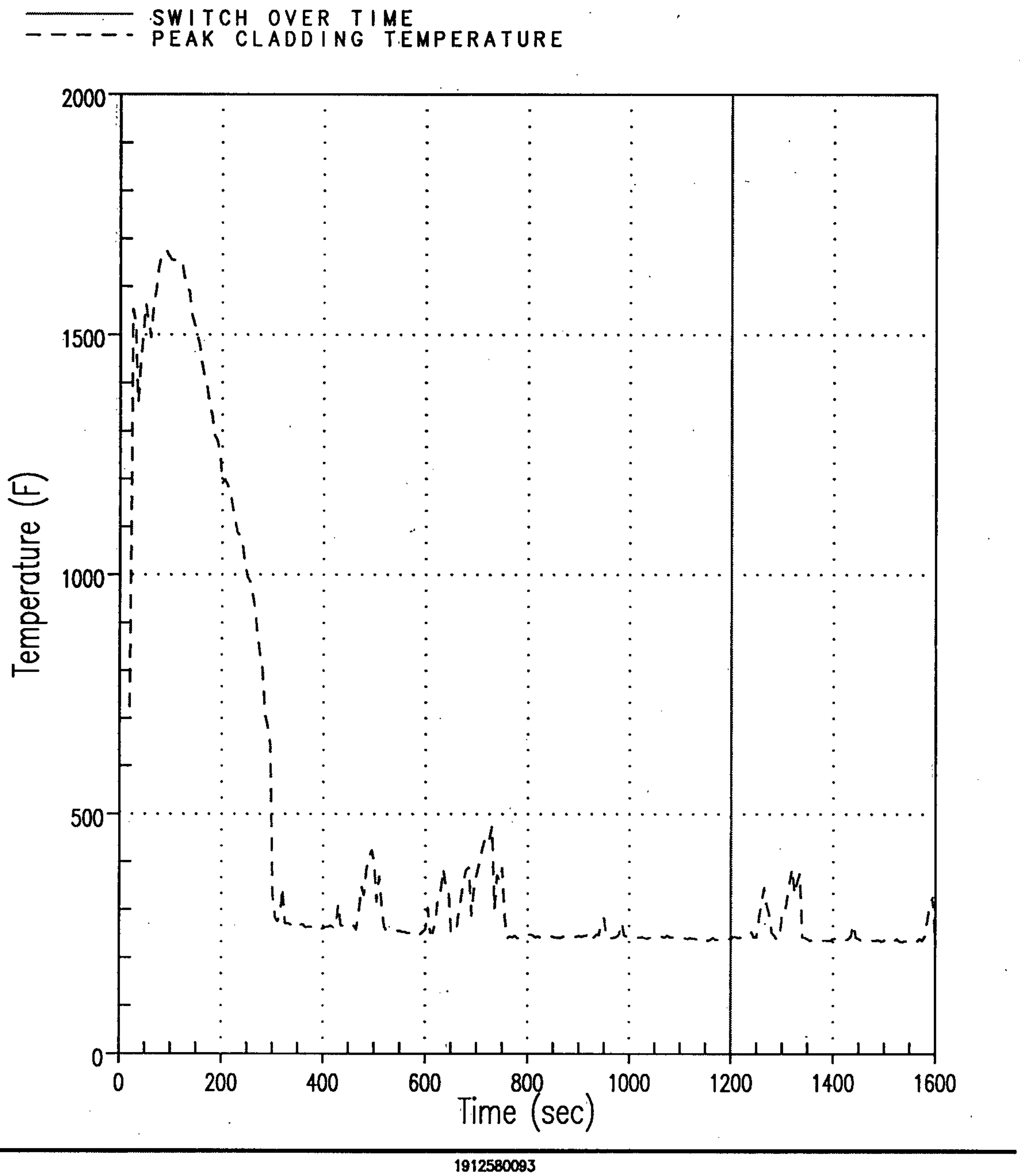

Figure 18: **Hot Rod PCT for Uniform C_D = 50,000 Case**

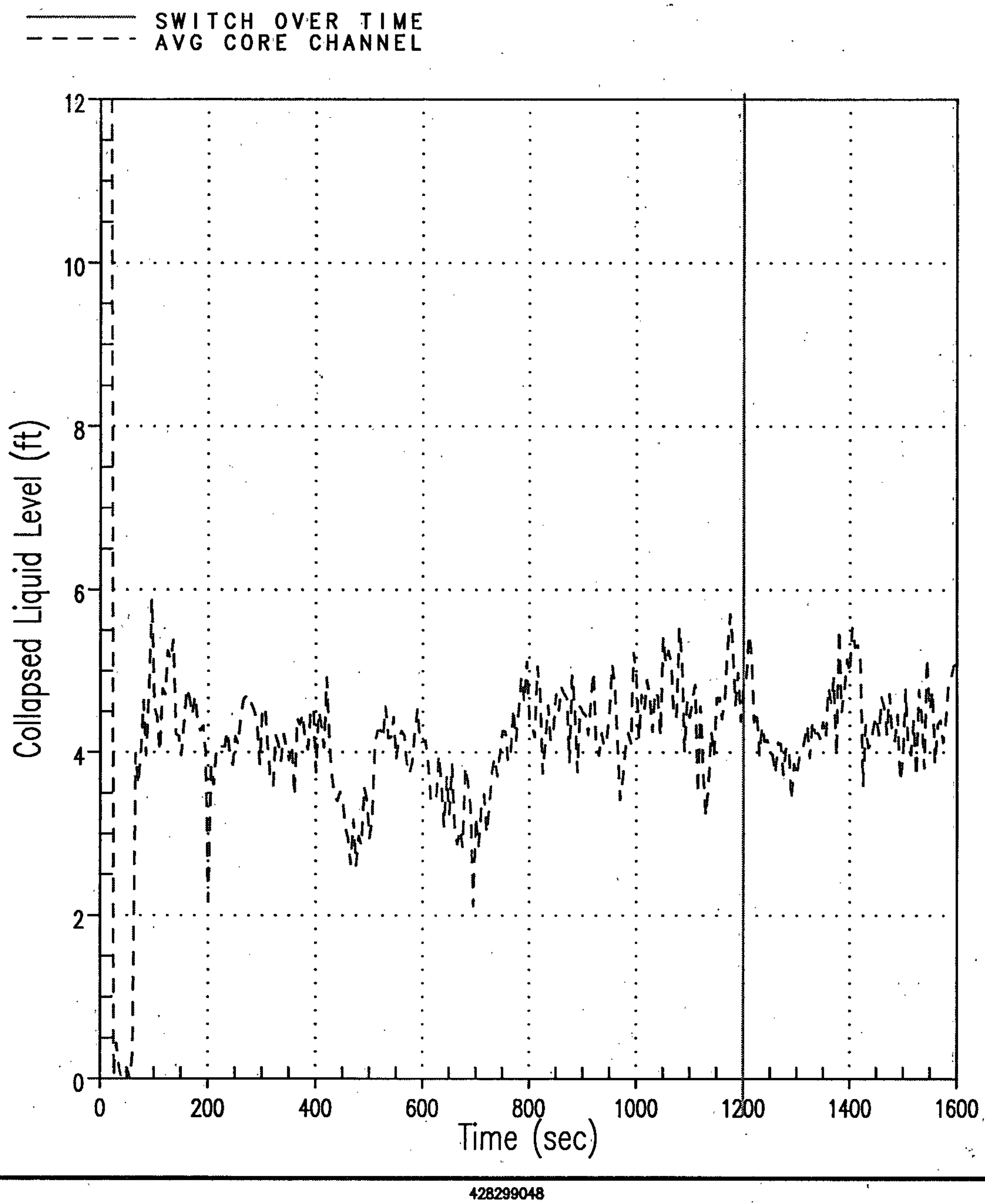

Figure 19: **Average Core Channel Collapsed Liquid Level for Uniform C_D = 50,000 Case**

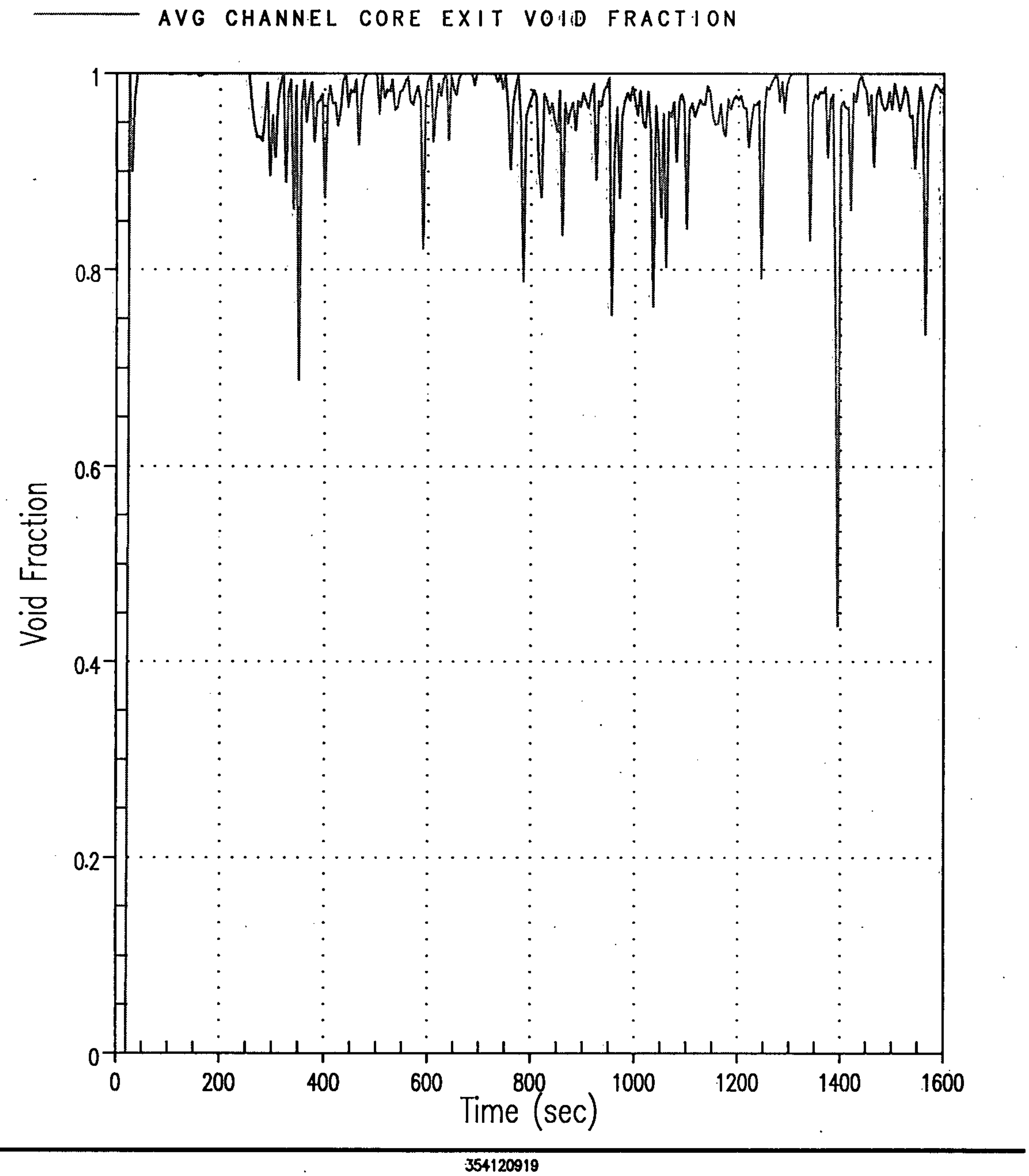

Figure 20: Void Fraction at the Exit of the Average Core Channel for Uniform C_D = 50,000 Case

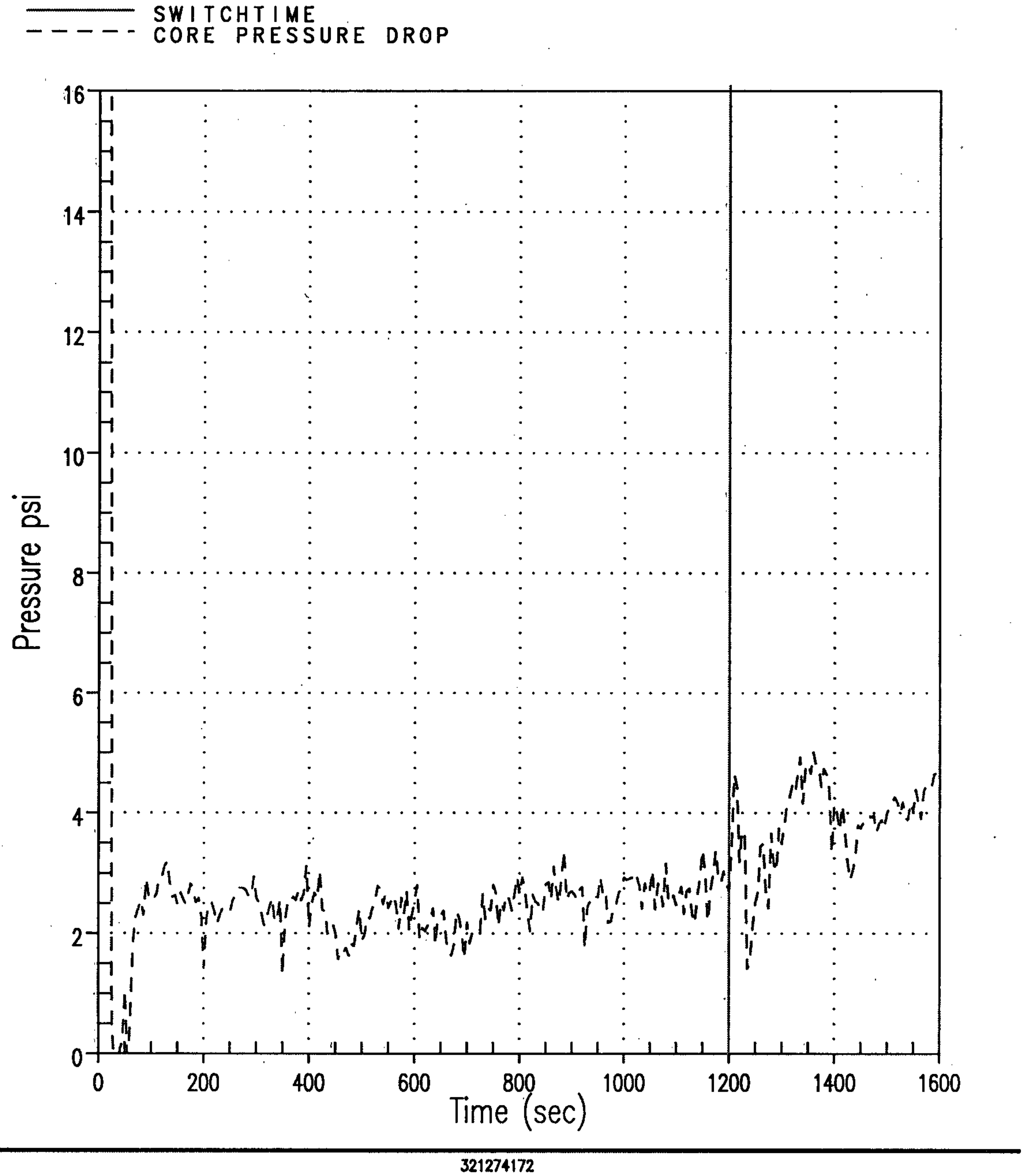

Figure 21: **Core Pressure Drop for Uniform C_D = 50,000 Case**

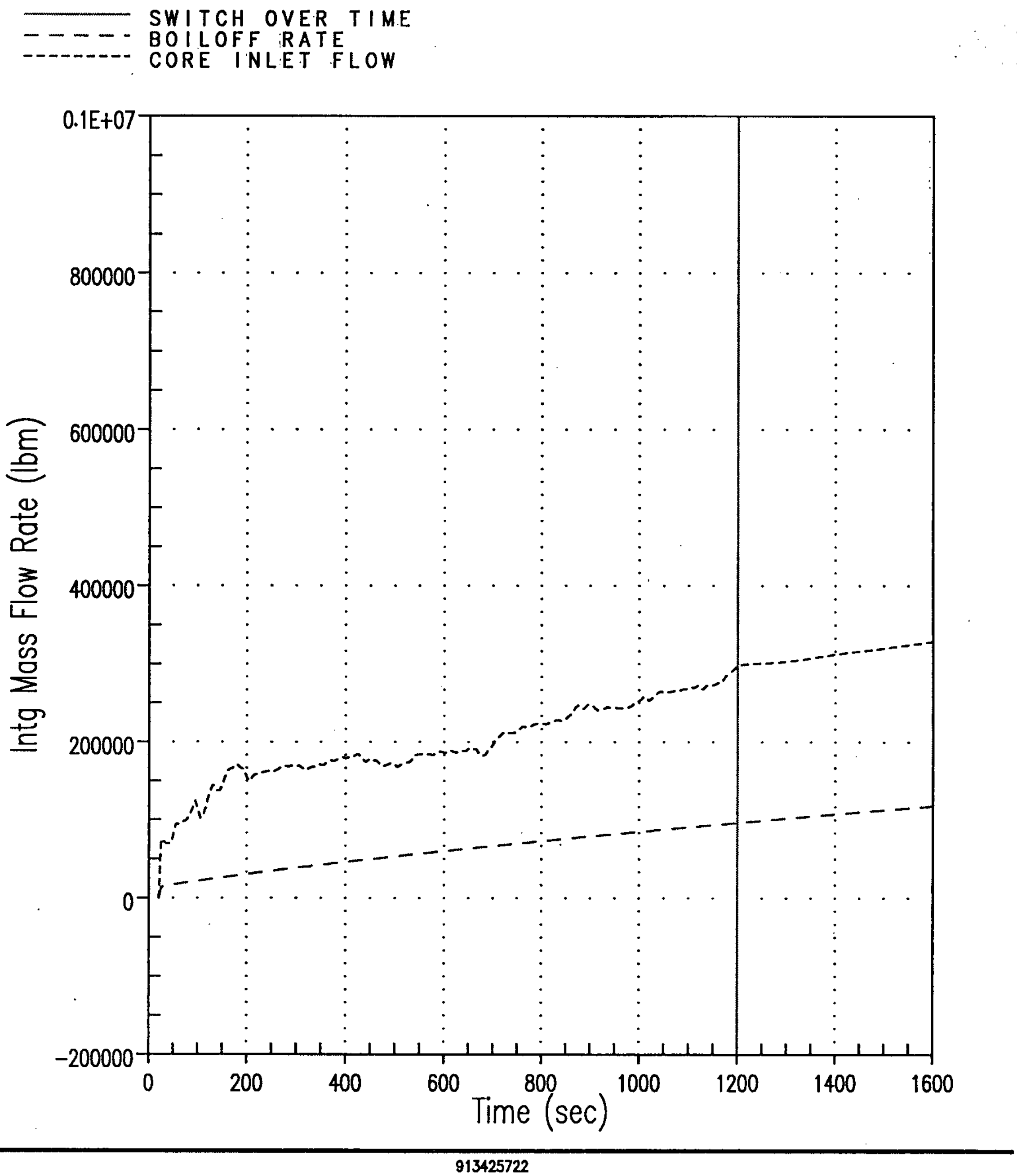

Figure 22: **Integrated Core Flow vs. Core Boil-off for Uniform C_D = 100,000 Case**

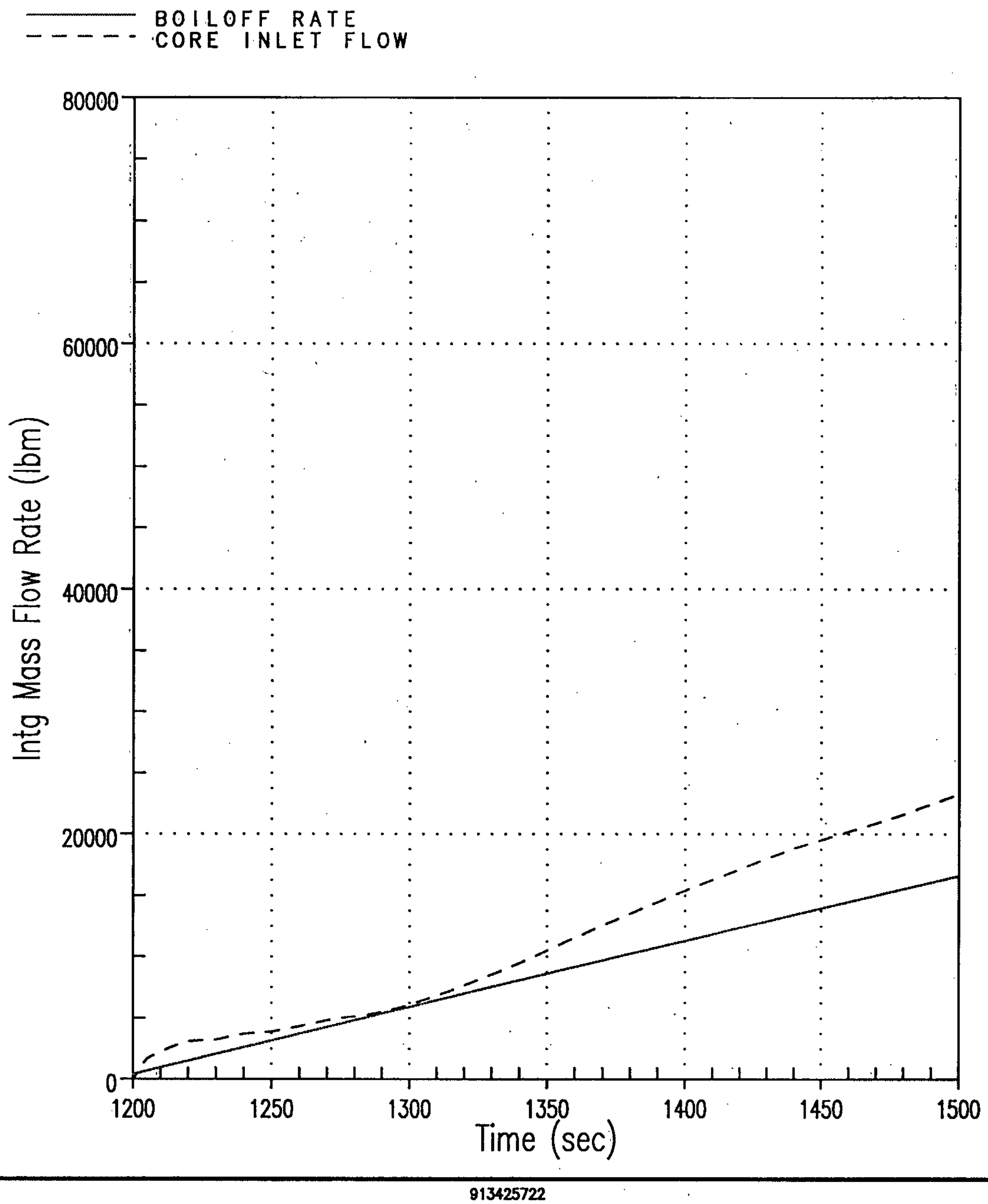

Figure 23: **Integrated Core Flow vs. Core Boil-off for Uniform C_D = 100,000 Case (Shifted Scale)**

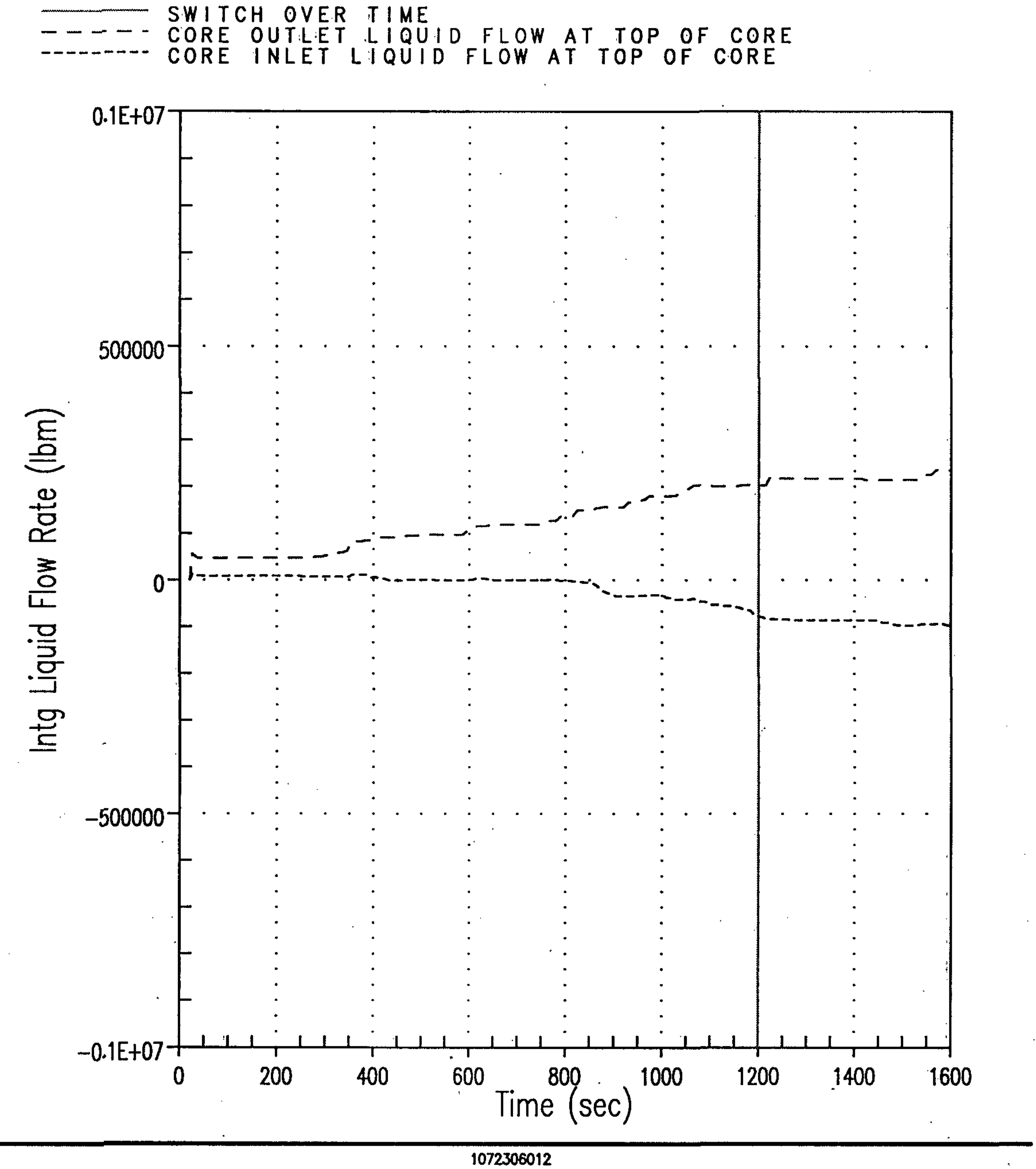

Figure 24: **Total Integrated Liquid Flow at the Top of the Core for Uniform C_D = 100,000 Case (Positive/Outlet flow represents HA, GT, AVG channels; Negative/Inlet flow represent LP channel)**

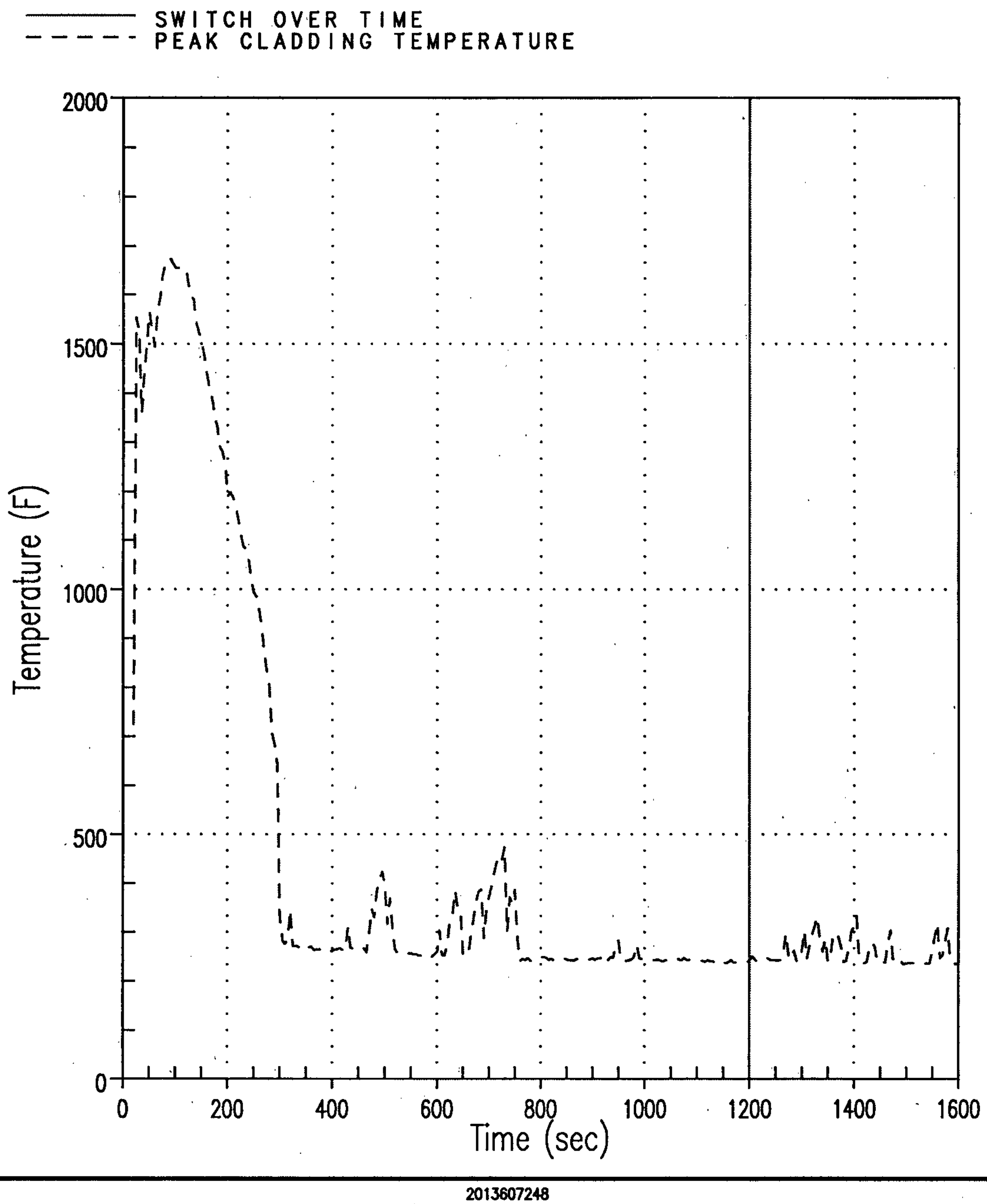

Figure 25: **Hot Rod PCT for Uniform C_D = 100,000 Case**

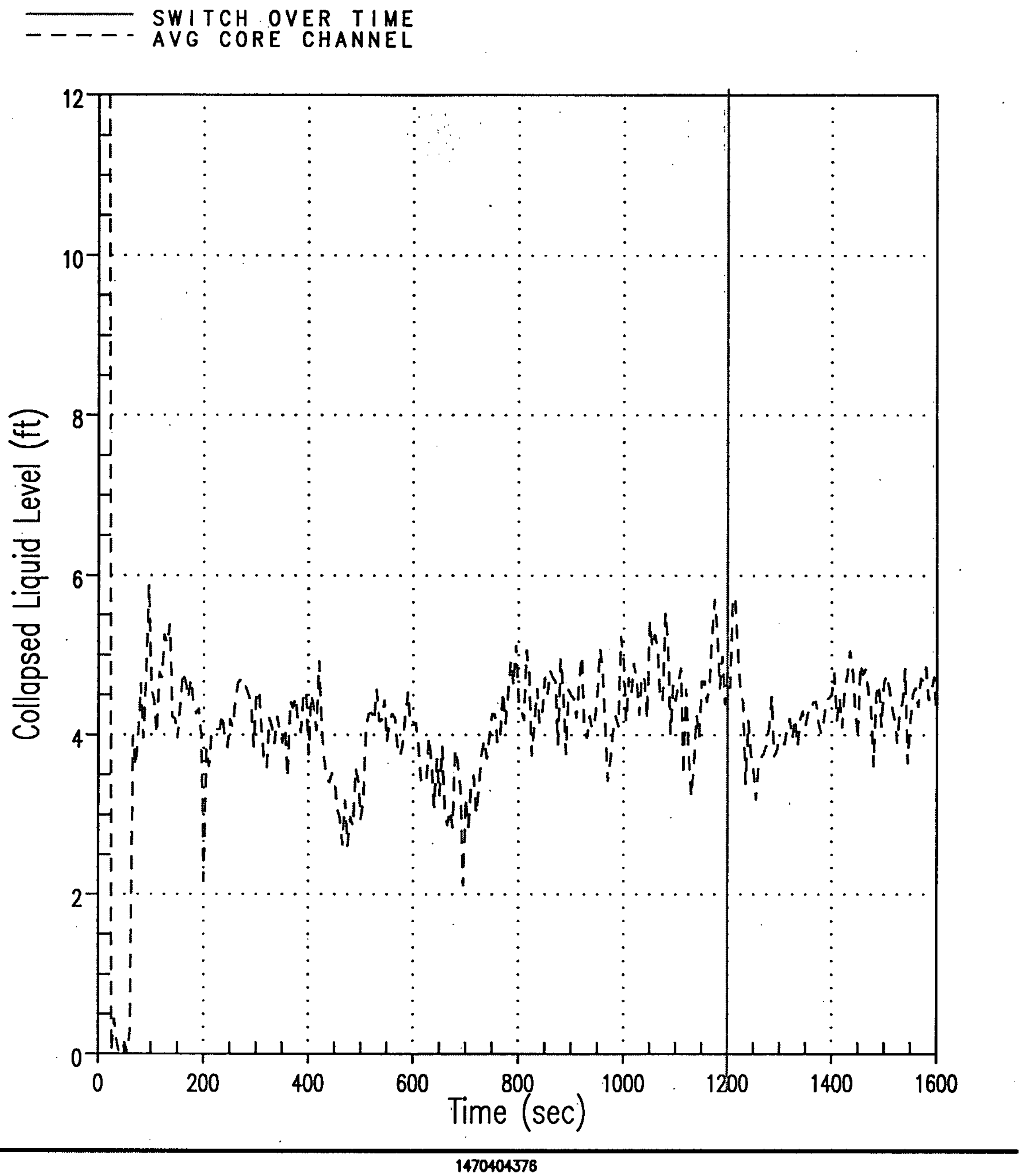

Figure 26: **Average Core Channel Collapsed Liquid Level for Uniform C_D = 100,000 Case**

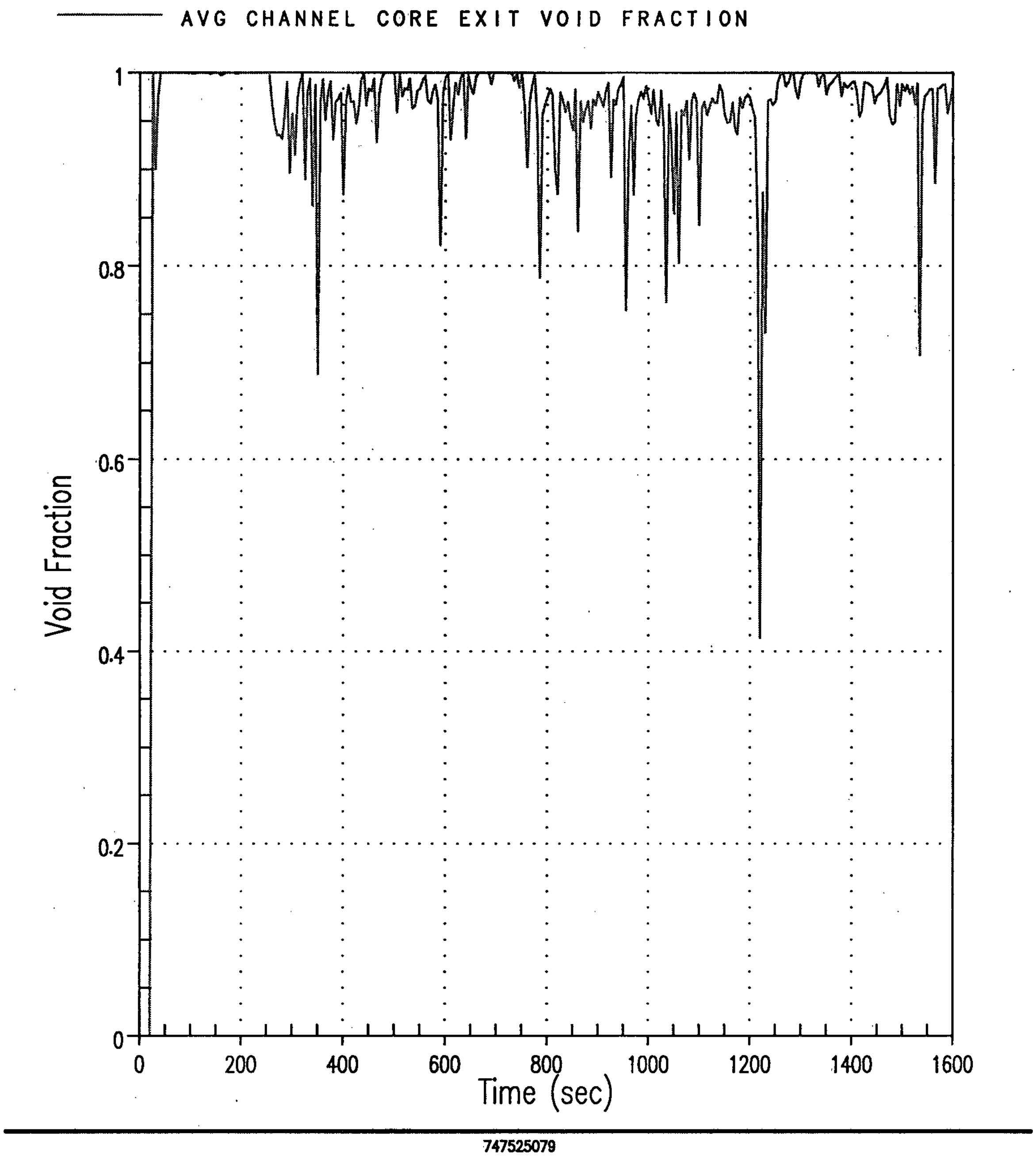

Figure 27: **Void Fraction at the Exit of the Average Core Channel for Uniform C_D = 100,000 Case**

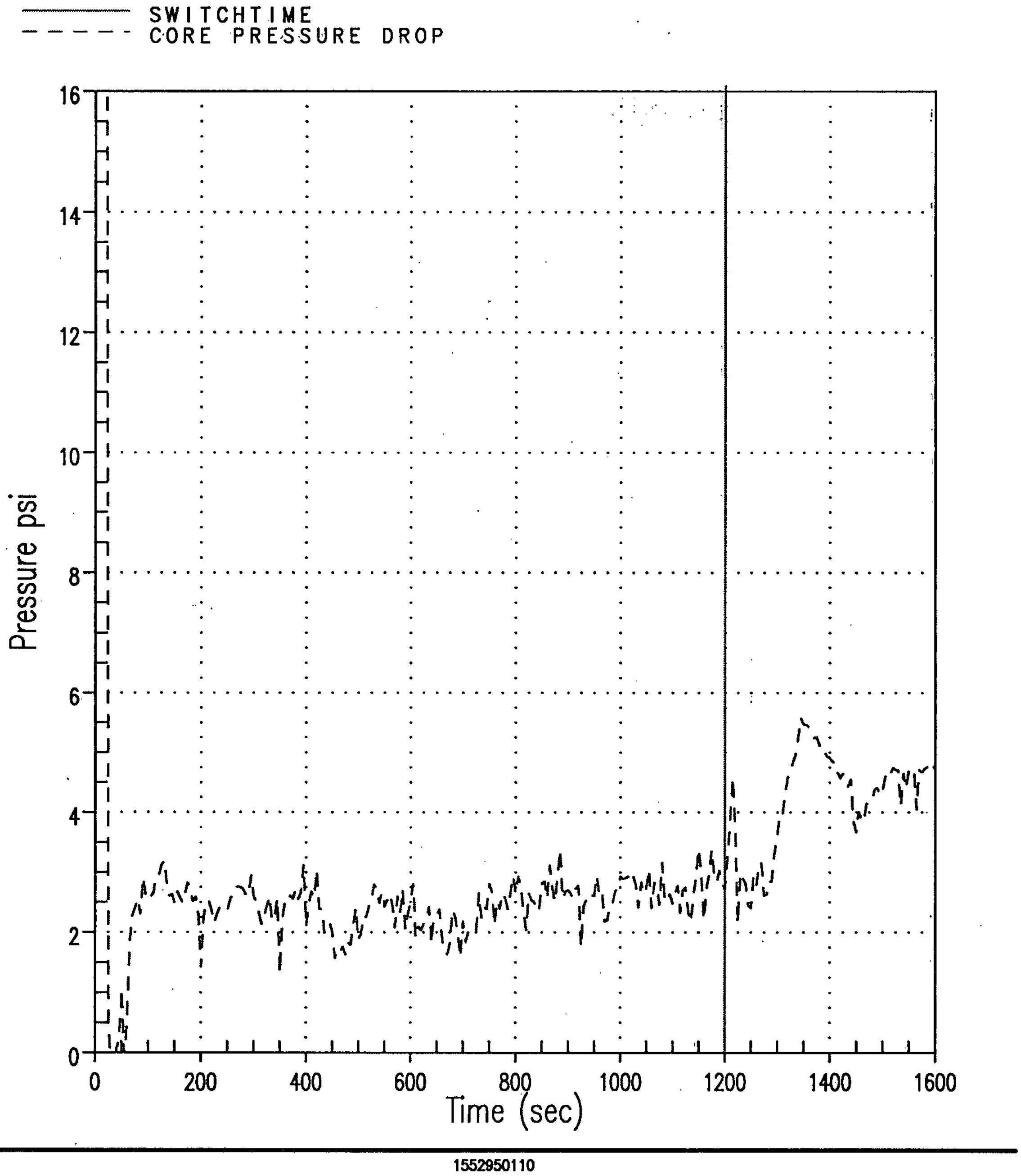

Figure 28: Core Pressure Drop for Uniform C_D = 100,000 Case

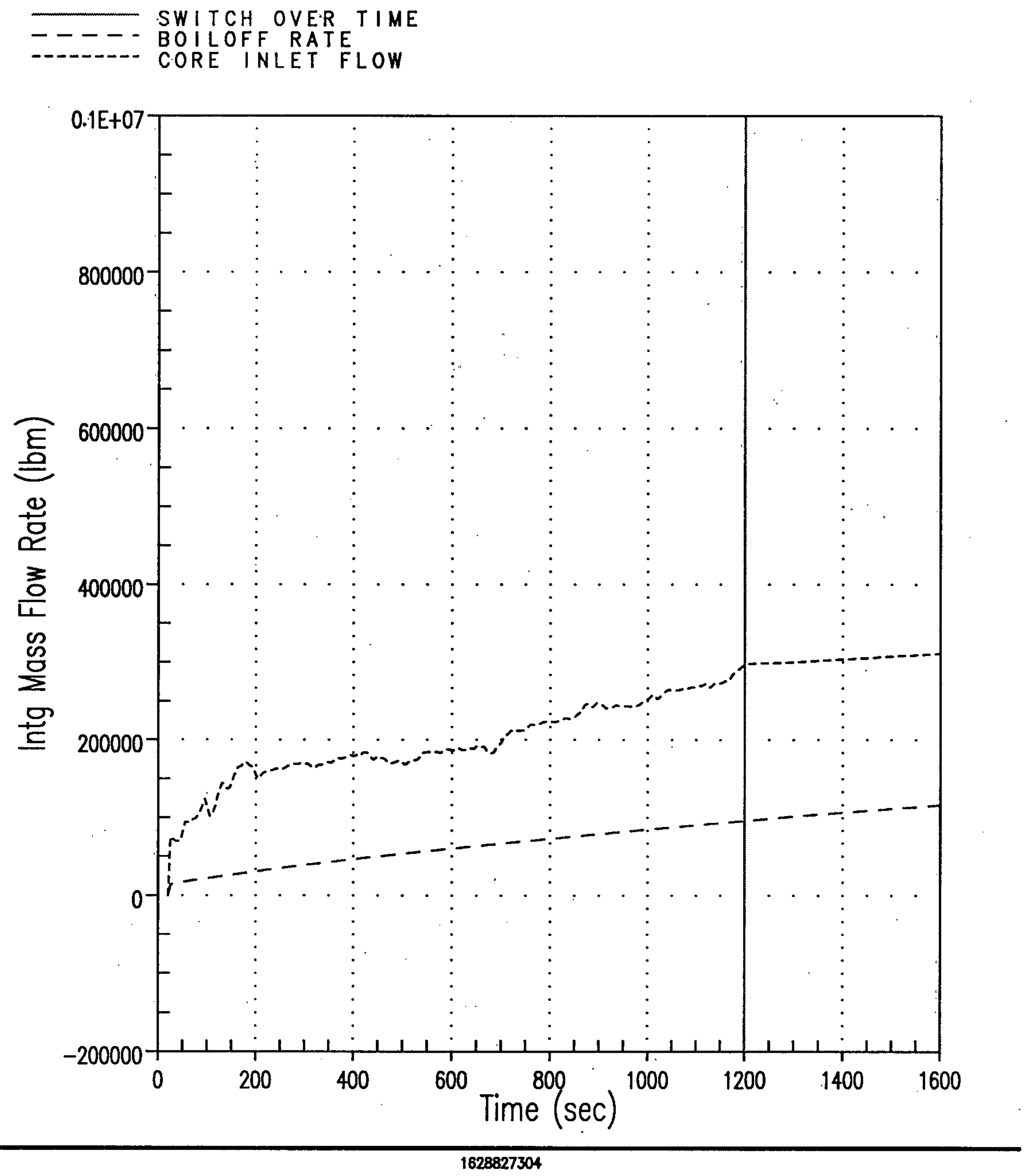

Figure 29: **Integrated Core Flow vs. Core Boil-off for Uniform C_D = 1,000,000 Case**

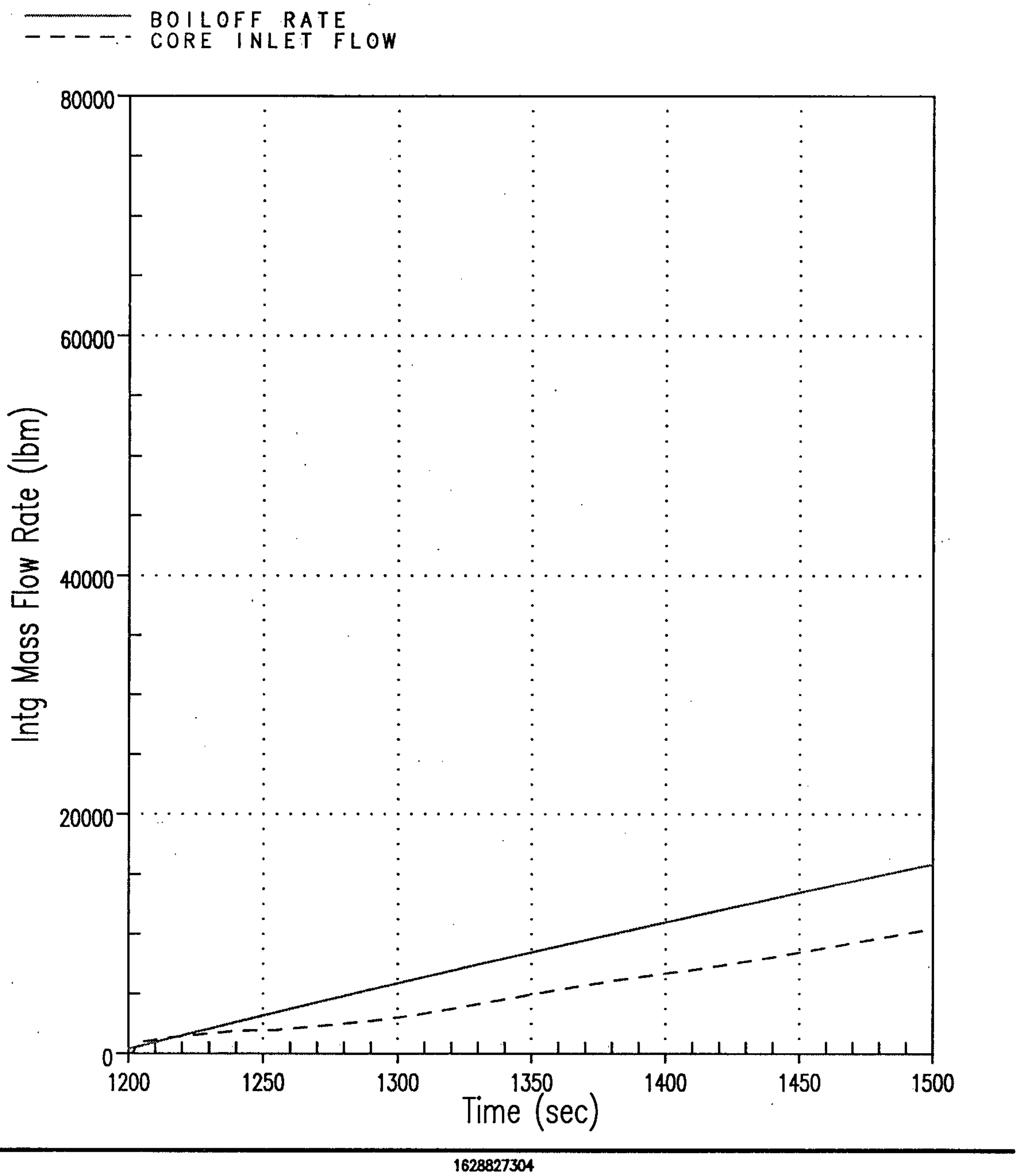

Figure 30: **Integrated Core Flow vs. Core Boil-off for Uniform C_D = 1,000,000 Case (Shifted Scale)**

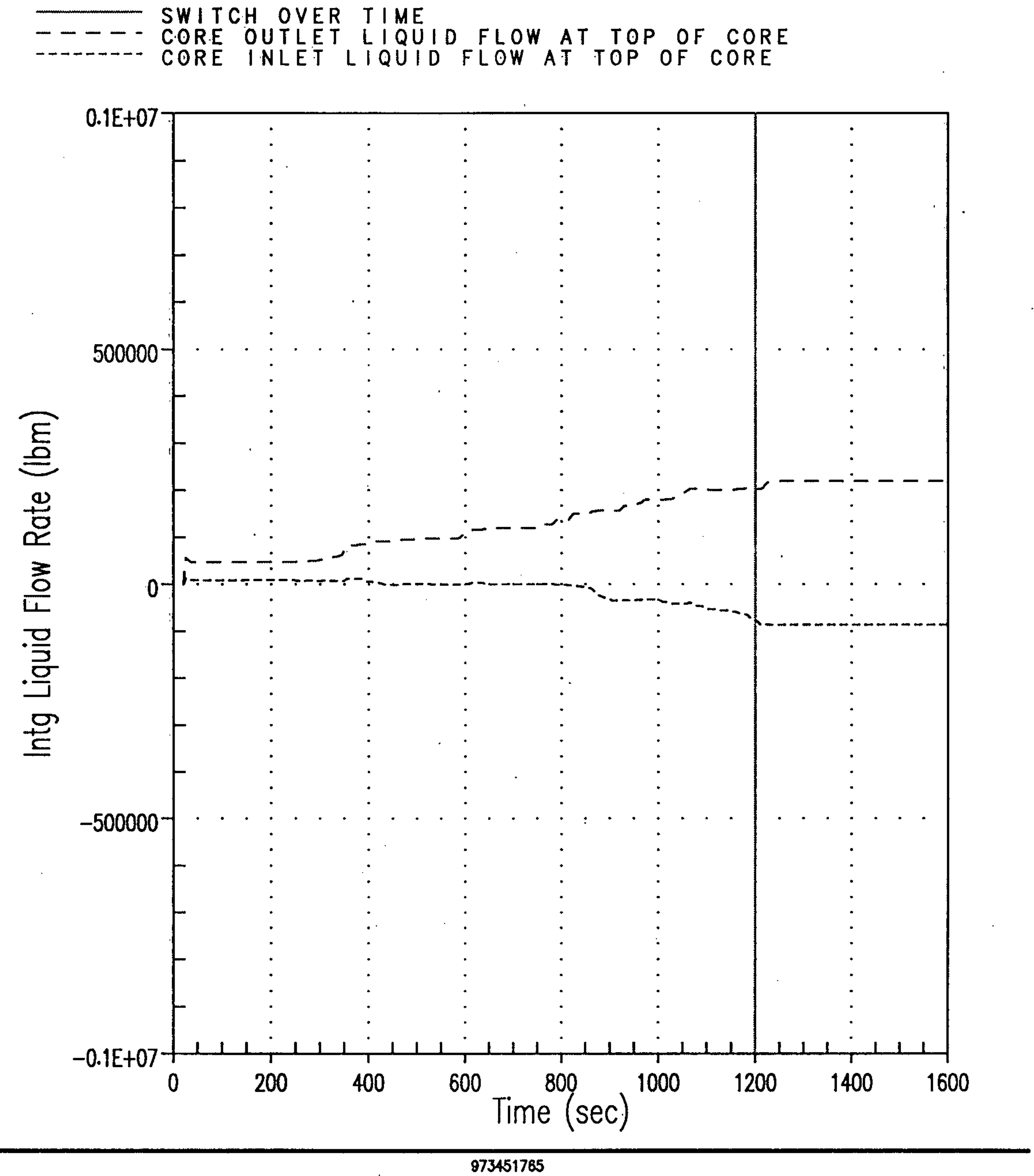

Figure 31: **Total Integrated Liquid Flow at the Top of the Core for Uniform C_D = 1,000,000 Case (Positive/Outlet flow represents HA, GT, AVG channels; Negative/Inlet flow represent LP channel)**

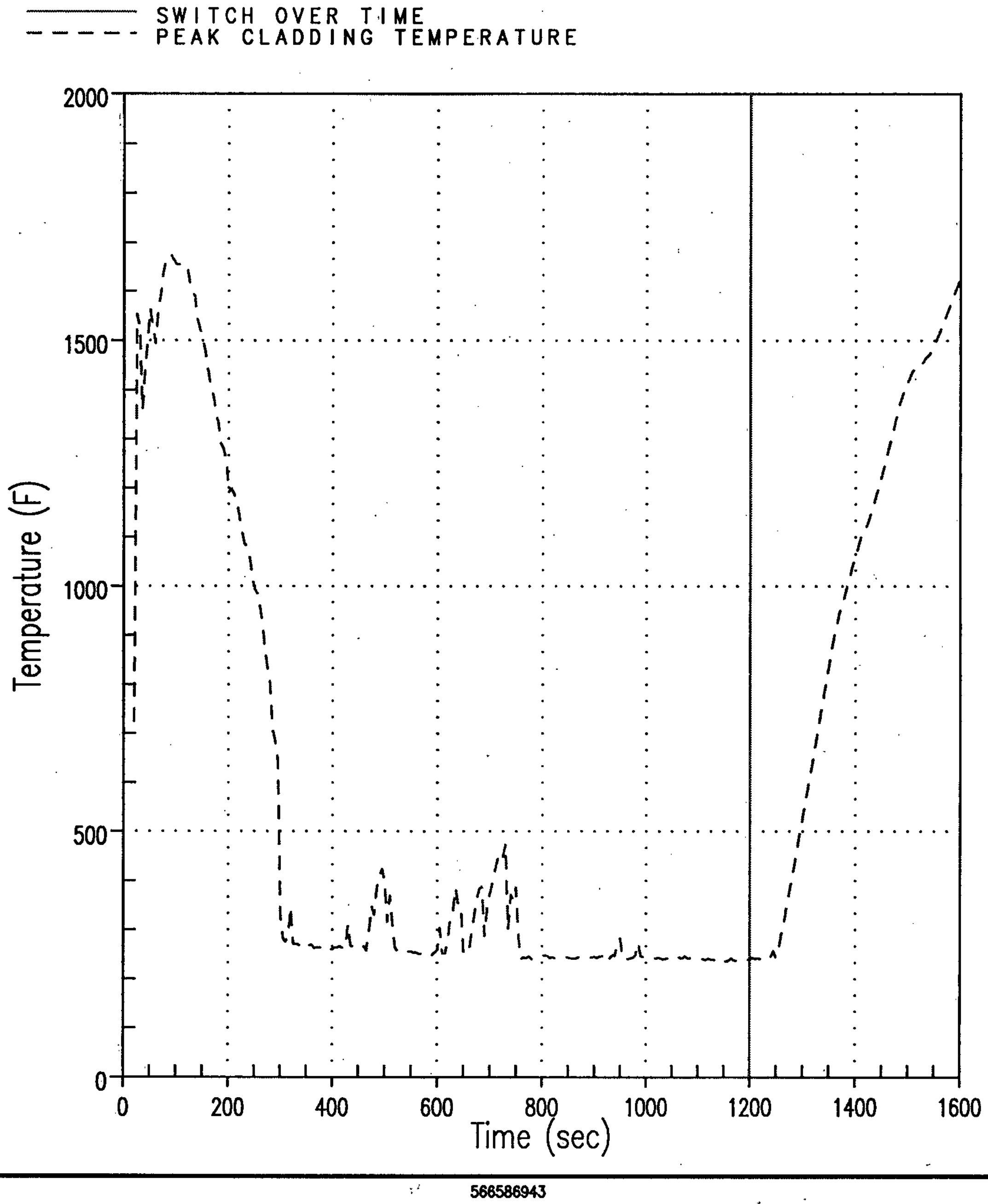

Figure 32: **Hot Rod PCT for Uniform C_D = 1,000,000 Case**

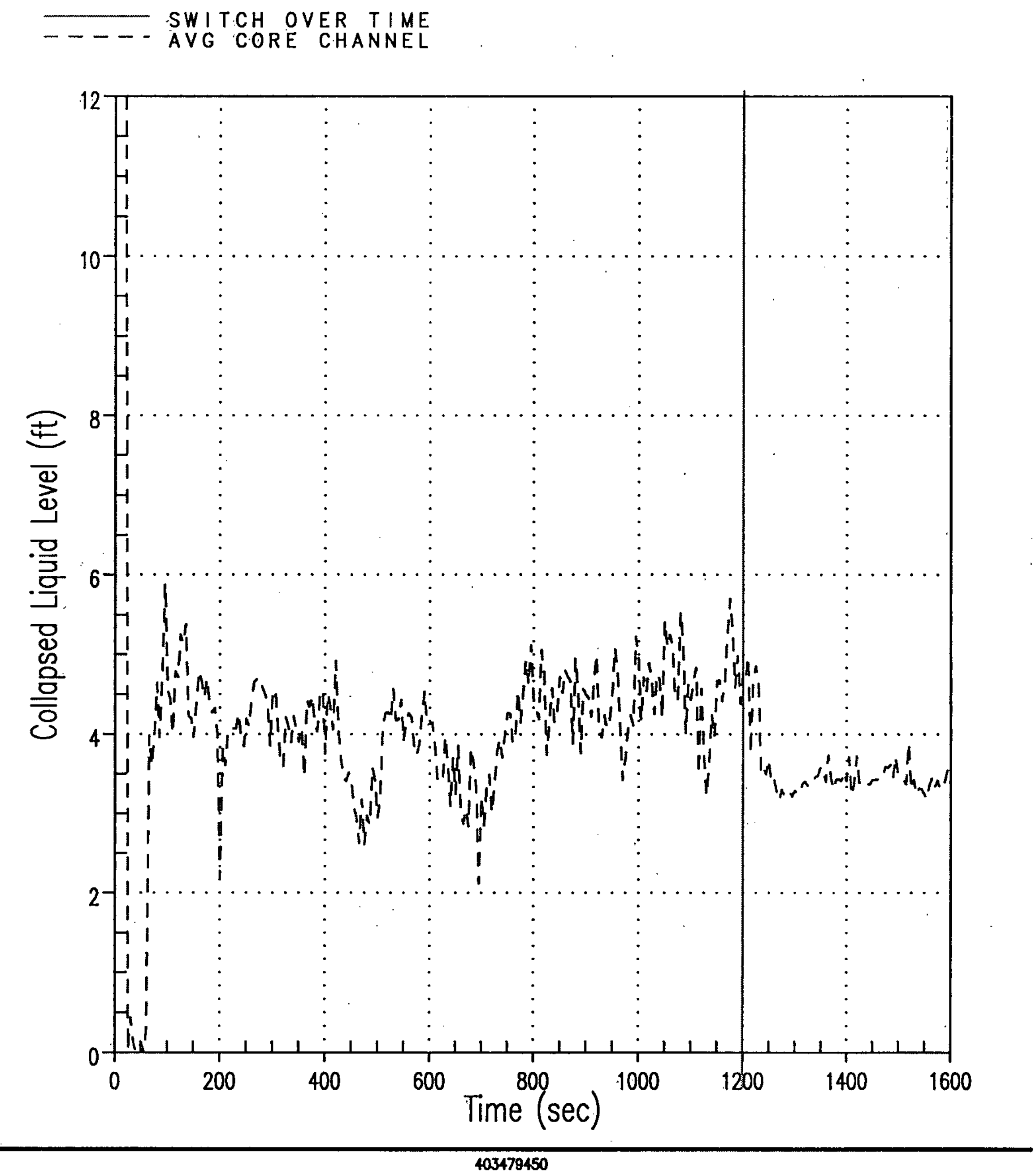

Figure 33: **Average Core Channel Collapsed Liquid Level for Uniform C_D = 1,000,000 Case**

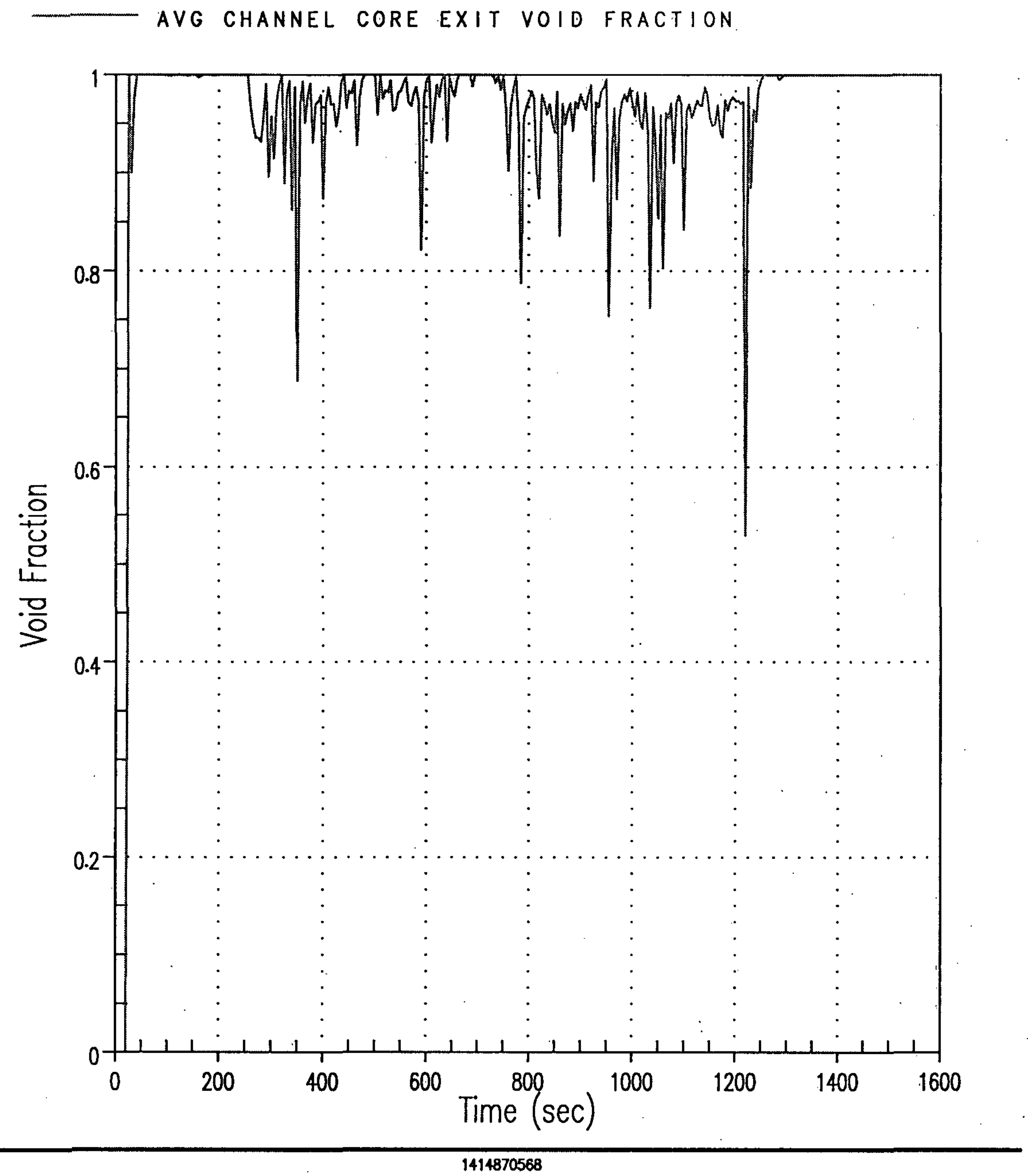

Figure 34: **Void Fraction at the Exit of the Average Core Channel for Uniform C_D = 1,000,000 Case**

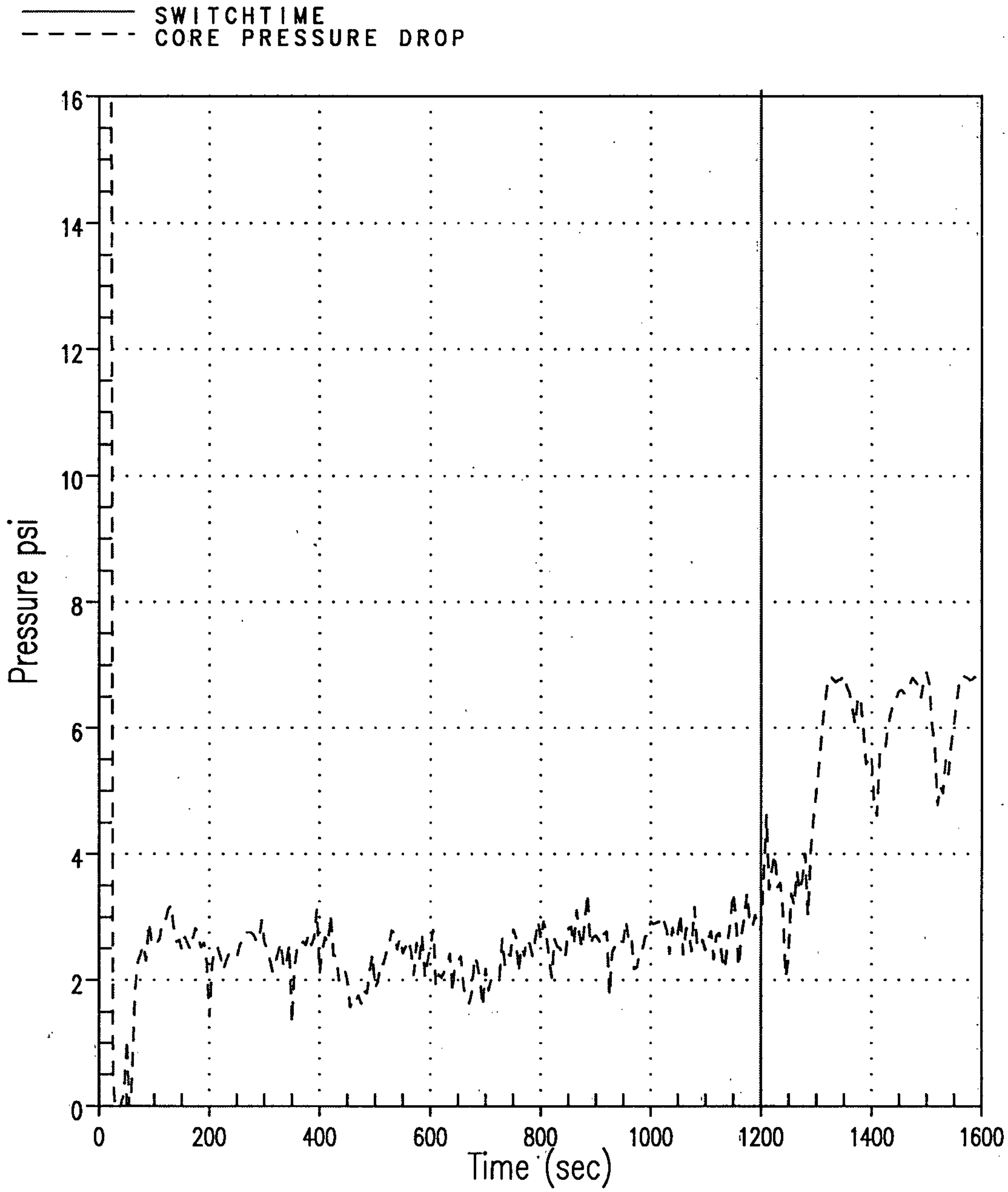

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Figure 35: Core Pressure Drop for Uniform C_D = 1,000,000 Case

2) Please clarify the WCAP assumptions regarding local blockage due to debris buildup. First, when a debris buildup of 110 mils or 50 mils is assumed at a spacer grid, please describe the assumed circumferential or azimuthal coverage of that debris layer. Given recent test observations, please justify any assumptions of less than full coverage. If the debris buildup bridges fuel rods, state whether the two rod heat sources are simultaneously applied in the analysis. Second, please discuss the same considerations regarding the layer of oxide, crud, and precipitate debris build-up on the fuel rods between spacer grids. Third, tests observed by staff indicate that fiber, particulate, and chemical precipitates can completely fill the grid space between adjacent rods in the first fuel assembly spacer grid. Please describe the effect of this debris build-up on local heating of pins and confirm that this affect is addressed in WCAP 16793-NP local effects analysis.

RESPONSE:

The text of Appendix C, "Fuel Clad Heat-Up Behind Grids," of WCAP-16793-NP describes the model, assumptions and inputs used to calculate the heat-up behind the grids. This calculation was a parametric study to assess the impact of thickness and thermal conductivity of debris and the thickness of the deposition on cladding in the boiling region.

The simulation of debris being applied to the model was accomplished as follows:

1. The space between the clad and the grid was modeled as being filled with debris. This is stated in the fifth bulleted item of Section C.5, "Assumptions." The assumption used in this model was that, "no convection occurs under the grids in the fuel rod assembly." Thus, although the model of Appendix C predates recent testing performed by the PWR Owners Group, modeling the space between the clad and the grid as having no convection is representative of current fuel assembly debris capturing test observations.

2. The deposition on the clad between grids was parametrically evaluated by varying the thickness between 0 mils, or no deposition, and 50 mils, or the maximum deposition considered. The deposition was modeled as occurring between the spacer grids in the fuel model. A deposition thickness of 50 mils on each of two adjacent rods will not fill the gap between rods as the fuel rod spacing is at least 110 mils or greater.

Thus, in all of the parametric calculations, the model did not allow for convective heat transfer from the fuel rod considering blockage within the grid straps, regardless of the thickness of the deposition modeled. Likewise, no convective heat transfer was modeled on the surface of the grid strap.

Conduction through space between the fuel clad outside diameter and the grid strap was, however, modeled. The same thermal conductivity was assigned to this space as was assigned to the deposition thickness between adjacent fuel assemblies.

Due to symmetry and the assumption of adiabatic surfaces along grid straps, modeling adjacent fuel rods in a grid strap was unnecessary. The model accounted for no convection behind adjacent grid straps.

The calculations of Appendix C did not model either an oxide or a crud layer on the fuel. However, the calculations described in Appendix D did consider both 17% oxidation of the fuel and a 100 micron layer of crud on fuel in the span between grids. As described on Page C-8, the temperatures of the model of Appendix D yields temperature predictions between 15°F to 86°F greater than those of the model in Appendix C. The discussion also notes that additional conservatism were used in the calculations presented in Appendix D; the bulk fluid temperature and heat flux used in the calculations of Appendix D are 25°F warmer and 25% higher, respectively, that those used in Appendix C. Thus, even with the additional conservatisms used in the calculations of Appendix D, the peak clad temperature behind a grid will remain well below the 800°F limit defined in Appendix A.

Subsequent to receipt of this RAI, and in response to comments from the Advisory Committee on Reactor Safeguards (ACRS), the PWR Owners Group initiated prototypical fuel assembly (FA) testing to establish limits on the debris mass (particulate, fibrous and chemical) that could bypass the reactor containment building sump screen and not result in unacceptable head loss that would impede core inlet flow and challenge long-term core cooling of the core. An overall test protocol and specific test procedures were developed to ensure that possible thin bed effects were investigated, and debris types and characteristics expected in the RCS were represented. Debris loads used in the test were based on sump screen bypass information provided by licensees. The fuel assemblies used in these tests included intermediate spacer grids. The results of these tests will be integrated into Revision 1 of WCAP-16793-NP as they pertain to debris buildup at the spacer grids.

3) For each of the following items, please discuss how the evaluation methods presented in WCAP-16793 will ensure that each plant that uses the methods will not incur unacceptable blockage at the core inlet or within the core (at grid spacers), considering the following:

   a. The potential for filtering debris beds on horizontal downward facing surfaces at typical core inlet flow rates that have been observed during strainer and fuel inlet blockage testing.

   b. Impacts of debris loading (fibrous, particulate, chemical).

   c. Impacts of fuel inlet nozzle, protective filter, and spacer grid designs.

   d. The potential impact of less than the maximum amount of postulated debris arriving at the core (thin bed). The staff believes that the potential for a fuel inlet thin bed is dependent on the protective filter above the inlet nozzle and the fuel inlet nozzle design, but has no test data to evaluate some of the designs. Filtering debris beds of less than 1/8 inch that have been observed during strainer testing.

   e. Impacts of plant-specific flow rates and available head for postulated cold and hot leg breaks.

   f. Justification for crediting settling in the lower plenum (if such credit is sought) based on lower plenum geometry, flow rates, and turbulence.

   Please include a discussion of how each plant is bounded by the WCAP analyses or how the WCAP prescribed methods will ensure that the plants have adequate guidance to perform a plant-specific evaluation of core inlet blockage. To the extent the WCAP attempts to extrapolate test results from one fuel assembly design to others, please provide the minimum and maximum fuel assembly inlet nozzle opening sizes including obstructions, such as due to spacer grids, for the fuel assembly designs involved. Please also include a description of the geometry of each fuel assembly inlet nozzle design in use, with dimensions, and identify the combinations of first fuel spacer-grid/inlet nozzle designs in use.

RESPONSE:

General Response:

Subsequent to receipt of this RAI, and in response to comments from the Advisory Committee on Reactor Safeguards (ACRS), the PWR Owners Group initiated prototypical fuel assembly (FA) testing to establish limits on the debris mass (particulate, fibrous and chemical) that could bypass the reactor containment building sump screen and not result in unacceptable head loss that would impede core inlet flow and challenge long-term core cooling of the core. An overall test protocol and specific test procedures were developed to ensure that possible thin bed effects were investigated, and debris types and characteristics expected in the RCS were represented. Debris loads used in the test were based on sump screen bypass information provided by licensees. The results of these tests will be integrated into Revision 1 of WCAP-16793-NP.

To use the results of this testing for closure of GSI-191, each plant will compare their plant-specific debris bypass load against the debris masses tested. Plants that have bypass debris loadings that are within the limits of the debris masses tested are bounded by the test. Several courses or actions have been identified for plants whose debris loads are outside of the limits tested. These actions include, but not limited to, reduction of problematic debris sources by removing or restraining the affected debris source or plant-specific fuel assembly testing.

The effects of differing fuel inlet nozzle designs were also considered in the test program. Both AREVA and Westinghouse have performed testing with their respective fuel inlet nozzles. Both vendors tested their various bottom nozzle designs and identified the limiting design (limiting was defined as the design that provided for the maximum pressure drop at the same flow and debris loading conditions). Each fuel bundle tested also had prototypical grids above the bottom nozzle debris capturing design features. Thus, the test data obtained from testing takes into account the fuel inlet nozzle, protective filter design features, and spacer grid designs. Descriptions of the fuel components tested, including bottom nozzles and grids, will be provided in proprietary submittals describing the testing performed and the results obtained.

Specific Responses to Specific Questions:

a. The ability of a model fuel assembly to capture fibrous, particulate and chemical surrogate debris has been tested with the objective of defining limits on the mass of debris that may bypass the reactor containment building sump screen and still provide for a sufficient low pressure drop across the model fuel assembly such that sufficient flow is provided to assure long-term core cooling requirements are satisfied. Plants that have bypass debris loadings that are within the limits of the debris masses tested are bounded by the test. Several courses or actions have been identified for plants whose debris loads are outside of the limits tested including, but not limited to, reduction of problematic debris sources by removing or restraining the affected debris source or plant-specific fuel assembly testing.

b. Testing has been performed to demonstrate and assess the ability of a model fuel assembly to capture fibrous, particulate and chemical surrogate debris. Based on that testing, limits have been defined on the mass of debris that may bypass the reactor containment building sump screen and still provide for a sufficient low pressure drop across the model fuel assembly such that sufficient flow is provided to assure long-term core cooling requirements are satisfied. Plants that have bypass debris loadings that are within the limits of the debris masses tested are bounded by the test. Several courses or actions have been identified for plants whose debris loads are outside of the limits tested including, but not limited to, reduction of problematic debris sources by removing or restraining the affected debris source or plant-specific fuel assembly testing.

c. The effects of differing fuel inlet nozzle designs have been assessed as both AREVA and Westinghouse have performed testing with their respective fuel inlet nozzles. Both vendors tested their various bottom nozzle designs and identified the limiting design (limiting was defined as the design that provided for the maximum pressure drop at the same flow and debris loading conditions). Both fuel bundles also had prototypical grids above the bottom nozzle debris capturing design features. Thus, the test data obtained from testing takes into account the fuel inlet nozzle, protective filter design features, and spacer grid designs. Plants that have bypass debris loadings that are within the limits of the debris masses tested are bounded by the test. Several courses or actions have been identified for plants whose debris loads are outside of the limits tested including, but not limited to, reduction of problematic debris sources by removing or restraining the affected debris source or plant-specific fuel assembly testing.

d. Testing was performed using the NRC March 2008 protocol of adding all particulate debris, then beginning to add the fibrous debris in small quantities so as to provide for the formation of a thin bed. Westinghouse performed several tests in this manner, with the NRC staff observing on such test. In all cases, NO thin bed was observed to form, even with very small quantities of fibrous debris. It was concluded by both the PWR Industry and the NRC that a thin bed was not likely to form.

e. Testing has been performed to define limits on the mass of debris that may bypass the reactor containment building sump screen and still provide for a sufficient low pressure drop across the model fuel assembly such that sufficient flow is provided to assure long-term core cooling requirements are satisfied. Testing used maximum hot-leg flow rates and maximum particulate debris loading, then varied fibrous debris loading to establish a limit on the mass of particulate, fibrous and chemical surrogate debris that could be bypassed by the sump screen and still provide sufficient flow to provide for long-term core cooling. Plants that have bypass debris loadings that are within the limits of the debris masses tested are bounded by the test. Several courses or actions have been identified for plants whose debris loads are outside of the limits tested including, but not limited to, reduction of problematic debris sources by removing or restraining the affected debris source or plant-specific fuel assembly testing.

Cold leg testing is being planned and the results will be reported in the next revision of WCAP-16793-NP.

f. Credit for settling in the lower plenum is not being considered as part of the demonstration of long-term core cooling for GSI-191 closure in WCAP-16793-NP (Reference 1). However, credit for settling in the lower plenum may be considered, with appropriate and applicable justification, for other issues associated with the closure of GSI-191.

4) Please provide information on potential flow paths that could bypass the fuel inlet to provide cooling in the event the core inlet becomes fully blocked with debris. Specifically, discuss the potential alternate flow paths (e.g., location, number, and sizes) for coolant to reach the core in the event that a complete blockage at the core inlet occurred. If these flow paths are credited for passing water to the core, please justify that they will not become blocked with debris and that they will pass adequate flow to the core to maintain cooling. Please also justify that these bypass flows will not result in problematic debris build up in the core.

RESPONSE:

Subsequent to receipt of this RAI, and in response to comments from the Advisory Committee on Reactor Safeguards (ACRS), the PWR Owners Group initiated prototypical fuel assembly (FA) testing to establish limits on the debris mass (particulate, fibrous and chemical) that could bypass the reactor containment building sump screen and not result in unacceptable head loss that would impede core inlet flow and challenge long-term core cooling of the core. The effects of differing fuel inlet nozzle designs were also considered in the test program. Both AREVA and Westinghouse have performed testing with their respective fuel inlet nozzles. Both vendors tested their various bottom nozzle designs and identified the limiting design (limiting was defined as the design that provided for the maximum pressure drop at the same flow and debris loading conditions). Each fuel bundle tested also had prototypical grids above the bottom nozzle debris capturing design features. Thus, the test data obtained from testing takes into account the fuel inlet nozzle, protective filter design features, and spacer grid designs.

This testing identifies debris loading limits that preclude the core inlet from becoming fully blocked with debris. Thus, if the core debris loading of plants fall within the limits of the debris loads tested, the core inlet will not become fully blocked with debris. Therefore, alternate flow paths are not considered in applying WCAP-16793-NP and are not credited or utilized in establishing acceptable debris loading conditions for long-term core cooling.

Several courses or actions have been identified for plants whose debris loads are outside of the limits tested including, but not limited to, reduction of problematic debris sources by removing or restraining the affected debris source or plant-specific fuel assembly testing.

In the event that a plant should choose to credit alternate flow paths for long-term core cooling, the plant would be expected to identify the number, size, flow capability and potential for blockage of the flow paths they are crediting.

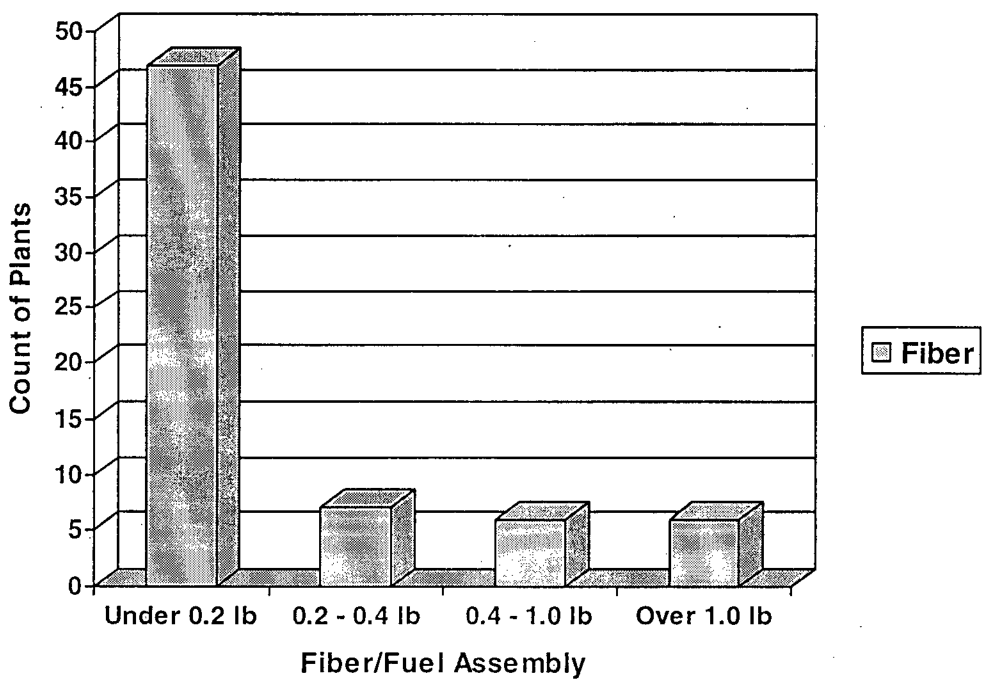
5) Please provide data that show the basis for the assumption that fibrous strainer bypass of 1 $ft^3/1000\ ft^2$ of strainer area provides a reasonable estimation of fiber bypass and that use of this number will not affect plant evaluations non-conservatively. Note that some protective fuel filters may be challenged by lower amounts of fiber (i.e. a thin bed) while some may be challenged by higher amounts. Please verify that these data were correlated with the area of the test strainer and that the results were not confounded by extrapolation to strainer areas less than the total of all strainers that may be in service during the event (i.e., strainer bypass estimates should assume all available strainer area is available for bypass).

RESPONSE:

Subsequent to receipt of this RAI, the PWROG initiated prototypical fuel assembly (FA) testing to establish limits on the debris mass (particulate, fibrous and chemical) to establish limits on the debris mass (particulate, fibrous and chemical) that could bypass the reactor containment building sump screen and not result in unacceptable head loss that would impede core inlet flow and challenge long-term core cooling of the core. The results of these tests will be integrated into Revision 1 of WCAP-16793-NP. As part of the effort to invoke this WCAP in the plant licensing basis, each plant will compare their plant-specific debris load against the masses tested. Therefore, the assumption of fibrous strainer bypass of 1 $ft^3/1000\ ft^2$ of strainer area is no longer relevant.

The assumption of fibrous strainer bypass of 1 $ft^3/1000\ ft^2$ of strainer area has been replaced by considering sump screen fiber bypass <u>mass</u> on a per fuel assembly basis. Total sump screen fiber bypass mass has been provided by licensees through a PWR Industry survey. This licensee-provided information was used to determine the amount of fibrous debris used in fuel assembly debris capture testing. The following chart shows the breakdown of bypass fibrous debris on a per fuel assembly basis for all plants that participated in the survey. The figure takes into account the number of fuel assemblies in the core of each plant reporting sump screen bypass values.

Figure 1: Survey Results of PWR Sump Screen Fibrous Debris Bypass



6) Please provide the following information for all fuel/core blockage tests that have been sponsored by the PWROG.

a. flow rates and bases including any variation in the flow rate during testing
b. debris types and size distribution for all debris added
c. amounts of each type of debris added to each test or subtest
d. bases for amounts and sizes of debris added to each test or subtest
e. scaling information for debris amounts and test flow rates
f. information regarding the prototypicality or conservatism of test facility flow pattern and settlement
g. head loss value experienced for each test or subtest including time dependent plots if available
h. observations of debris transport and accumulation including any settling with differences noted at different flow rates
i. behavior of debris during testing (agglomeration)
j. test methodology and setup
k. details of debris preparation and introduction
l. order and rate of debris addition
m. dimensions of fuel inlet test mock-up
n. design of fuel protective filter modeled in the test
o. photographs as available to assist in understanding the tests theoretical debris bed thickness based on as-manufactured fiber density

RESPONSE:

Introduction:

Subsequent to receipt of this RAI, and in response to comments from the Advisory Committee on Reactor Safeguards (ACRS), the PWR Owners Group initiated prototypical fuel assembly (FA) testing to establish limits on the debris mass (particulate, fibrous and chemical) that could bypass the reactor containment building sump screen and not result in unacceptable head loss that would impede core inlet flow and challenge long-term core cooling of the core. An overall test protocol and specific test procedures were developed to ensure that possible thin bed effects were investigated, and debris types and characteristics expected in the RCS were represented. Debris loads used in the test were based on sump screen bypass information provided by licensees. The results of these tests will be integrated into Revision 1 of WCAP-16793-NP.

The effects of differing fuel inlet nozzle designs were also considered in the test program. Both AREVA and Westinghouse have performed testing with their respective fuel inlet nozzles. Both vendors tested their various bottom nozzle designs and identified the limiting design (limiting was defined as the design that provided for the maximum pressure drop at the same flow and debris loading conditions). Each fuel bundle tested also had prototypical grids above the bottom nozzle debris capturing design features. Thus, the test data obtained from testing takes into account the fuel inlet nozzle, protective filter design features, and spacer grid designs. Descriptions of the fuel components tested, including bottom nozzles and grids, will be provided in proprietary submittals describing the testing performed and the results obtained.

Testing was performed using bounding debris loads and hot leg break flow rates. These tests demonstrated that for the bounding debris loads tested, the hot-leg flow rate through the fuel assembly mock-up was maintained with acceptable pressure drops. The results of these tests will be integrated into Revision 1 of WCAP-16793-NP. To be responsive to this RAI, the following summary is provided.

Test Overview:

A full area, partial height fuel assembly equipped with various fuel filters was used for the testing. Each assembly included a number of intermediate spacer grids including at least one intermediate flow mixing (IFM) grid or equivalent. Debris laden water was introduced to the bottom of the test region and flowed up through a simulated lower plenum region, through the simulated core support plate, and through the fuel assembly. As debris caught on the fuel assembly, the differential pressure was measured across various locations including the bottom nozzle and individual grids as well as across the entire fuel assembly. The differential pressure measurements were used to determine an acceptable debris load. The test loop was intended to test the debris capture characteristics of a full-area fuel assembly under the debris loading conditions of a hypothetical LOCA.

The output of this test program will be a set of acceptance criteria. The acceptance criteria will define maximum debris masses which, if passed through the reactor containment building sump screen, will result in an acceptable pressure drop at the core inlet. For a given plant to demonstrate acceptable long term core cooling, it will need to show that each of the plant specific sump screen bypass masses are bounded by the limits in the acceptance criteria.

Test Loop Description:

This section addresses the following parts of the RAI:

- Description of fuel inlet test mock-up
- Flow rates and bases including any variation in the flow rate during testing
- Information regarding the prototypicality or conservatism of test facility flow pattern and settlement
- Design of fuel protective filter modeled in the test

AREVA and Westinghouse performed the fuel assembly tests at different locations. However, both test facilities took great lengths to ensure conformity between both test loops. The answer below applies to both facilities.

The Westinghouse test loop for testing the debris capture characteristics of a full-width fuel assembly is shown in Figure 1. A schematic of this test loop is given in Figure 2. The AREVA test loop is shown in Figure 3. The schematic of this test loop is shown in Figure 4. The test loop is composed of four main parts:

- Mixing tank system
- Recirculation system
- Test column
- Computer monitoring system

Mixing Tank System:

The mixing tank system includes a plastic tank, a temperature control system, and a mixing system. The mixing tank is where debris can be added during the test. The tank design and mixing system helps preclude the settling and loss of debris on the bottom of the tank. The temperature of the water in the tank is controlled by either a heater element and/or by running water at a higher or lower temperature through a heater/chiller. The water temperature can be controlled from a low temperature of approximately 60°F to a high temperature of approximately 100°F, and the temperature of the water is measured continuously in the tank by a submerged thermocouple.

Recirculation System:

The recirculation system pumps the water from the tank, through the test column and back into the tank. A pump draws the water out of the bottom of the mixing tank. The recirculation system is continuous duty to accommodate longer tests.

Flow Rate:

Each test is performed at applicable hot-leg approach velocity. The bounding velocities below the core plate are:

- Westinghouse and B&W plants – 0.2 (+ 10%) feet per second.
- CE plants – 0.03 (+ 10%) feet per second (or 0.05 (+ 10%) feet per second for certain plants).

The flow rate is maintained during the test.

Test Column:

The test column contains the fuel assembly and simulates the geometry and many of the conditions that would be experienced inside of the reactor vessel. The test column includes a lower plenum region, a core support plate, the fuel assembly, and an upper plenum region. The debris laden water is introduced to the bottom of the lower plenum region. The design of this region is not prototypical of an RV lower plenum; it is designed instead to ensure that the debris remains well mixed in the fluid flow and precludes any debris settling, thereby ensuring that all debris introduced to the test column will reach the fuel assembly. The lower plenum region and the fuel assembly are divided by a simulated core support plate with 2.75" flow holes. The fuel assembly rests directly on this simulated core support plate. The region that contains the fuel assembly is made of Plexiglas for viewing during the test. This region is sized to represent the fuel assembly pitch for the test assembly that is being tested.

The debris and water enter through the bottom nozzle and flow up through the simulated core support plate. As debris catches on the fuel assembly, the differential pressure is measured constantly across the fuel filter as well as across the entire fuel assembly. There are extra ports available on the sides of the test column if a measure of the differential pressure across a specific portion of the fuel assembly as required.

Computer Monitoring System:

The computer monitoring system continuously records the following data:

- Temperature of the water in the mixing tank
- Flow rate
- Differential pressure measurements from ΔP gauges

This data can be recorded at a time interval chosen by the operator. The computer is also used to check the slope of the ΔP (pressure drop) or flow versus time graphs in order to determine if the curves have reached a point close enough to equilibrium.

Design of Fuel Protective Filter in the Test:

AREVA and Westinghouse performed tests with all relevant fuel filters.

Westinghouse*

- Fuel assembly with Westinghouse P-grid
- Fuel assembly with Guardian Grid

  *Note: The alternate p-grid design was not tested as previous test results had concluded that the standard p-grid was the limiting design.

AREVA

- 17x17 fuel assembly with AREVA FUELGUARD™ Grid
- 17x17 fuel assembly with AREVA TRAPPER™ coarse mesh screen
- 17x17 fuel assembly with AREVA TRAPPER™ fine mesh screen

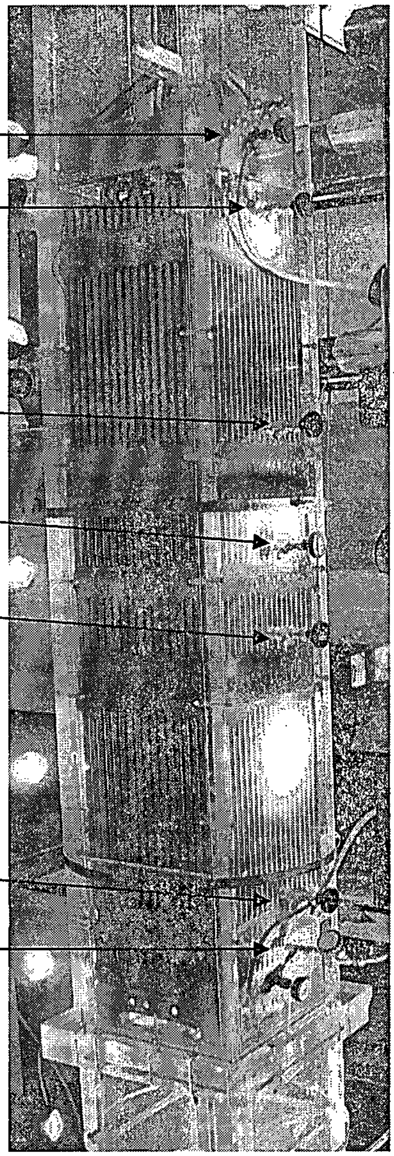
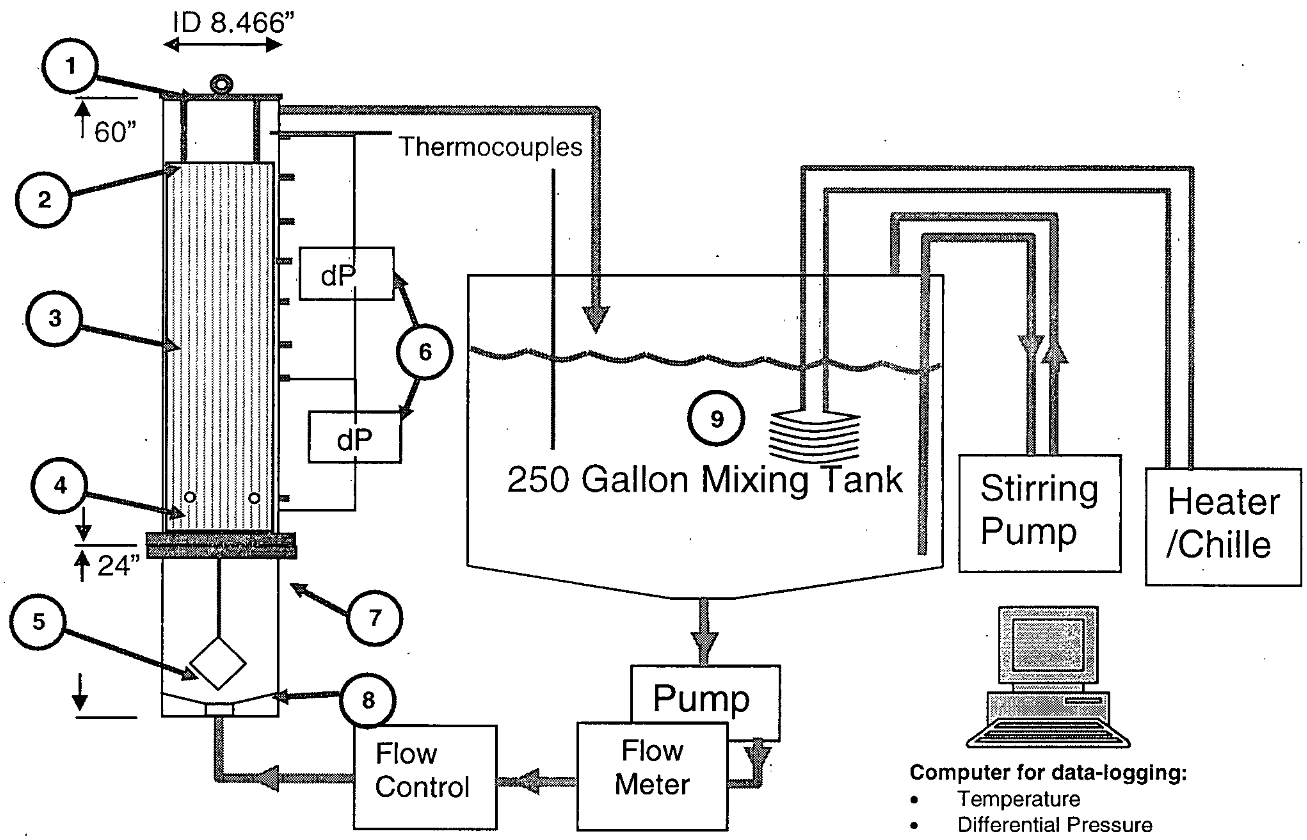


Figure 1: Photograph of the Westinghouse Fuel Test Vessel



Computer for data-logging:
- Temperature
- Differential Pressure
- Flow

Figure 2: Schematic of the Westinghouse Test Loop

1. Stainless steel top plate and lifting ring
2. Stainless steel hold-down bar
3. One-third height fuel assembly
4. Horizontal positioning set screws
5. Flow diverter (cube)
6. Differential pressure gauge
7. Port for measurement of differential pressure
8. Bottom flow cone
9. Temperature-regulation coil

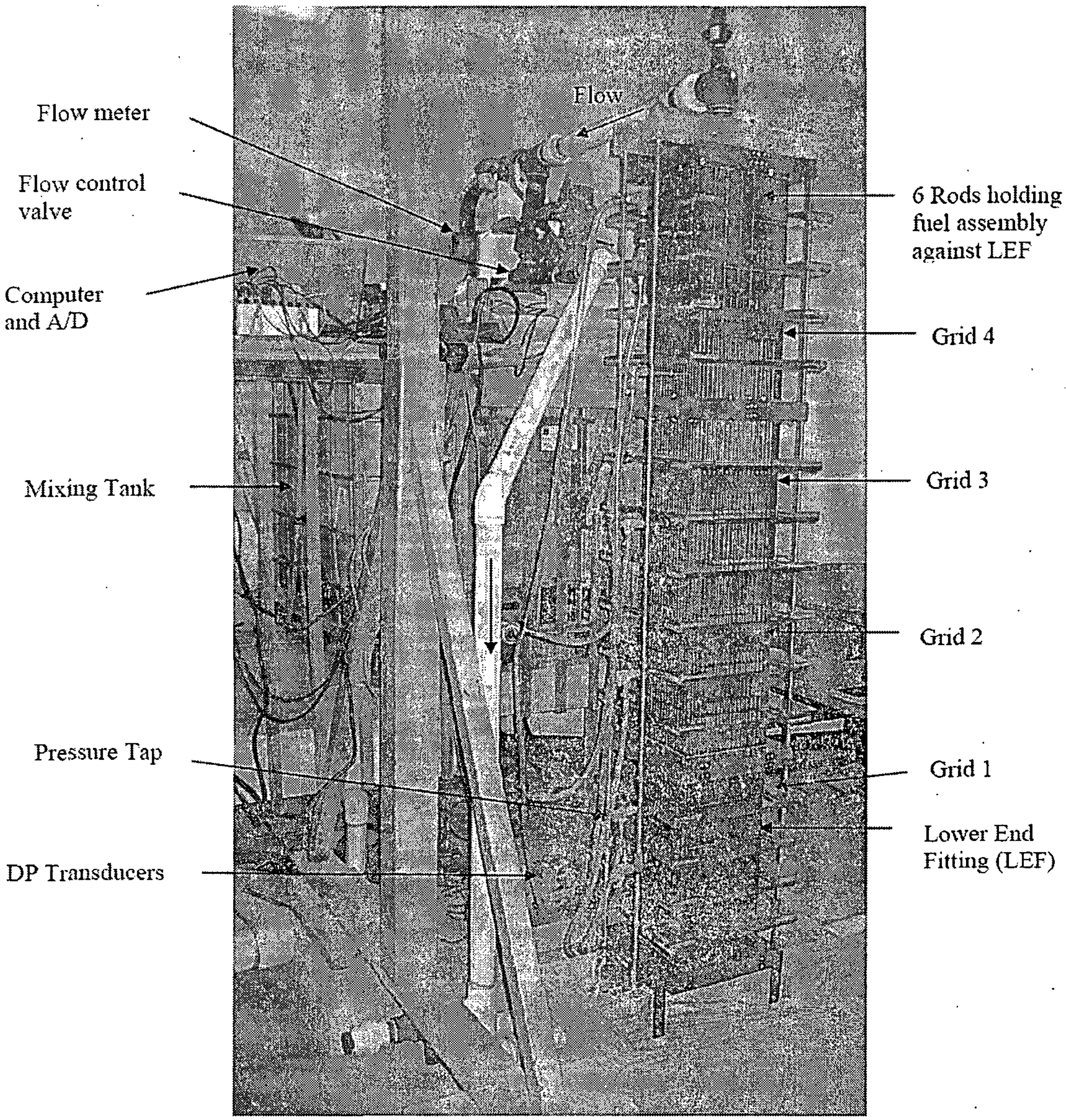


Figure 3: AREVA Test Loop

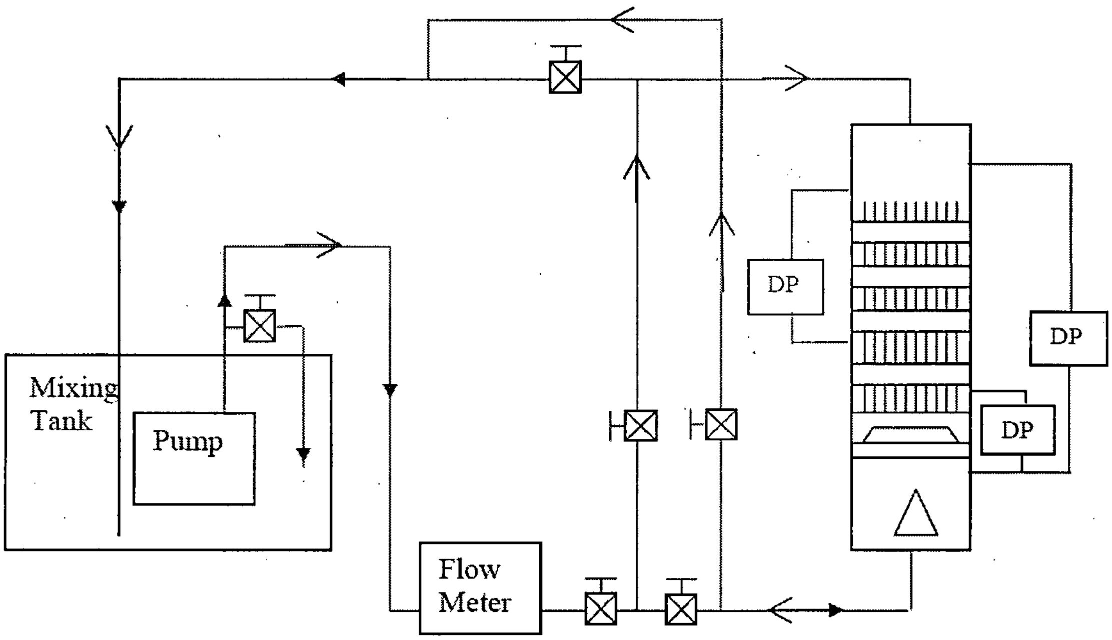


Figure 4: Schematic of AREVA Test Loop

Debris Discussion:

This section addresses the following parts of the RAI:

- Debris types and size distribution for all debris added
- Details of debris preparation and introduction
- Order and rate of debris addition
- Amounts of each type of debris added to each test or subtest
- Bases for amounts and sizes of debris added to each test or subtest
- Scaling information for debris amounts and test flow rates

Debris Type and Size Distribution:

The main debris materials added to the fuel nozzle testing were NUKON™ fiber, silicon carbide, Microtherm, calcium silicate, AlOOH chemical surrogate, and filtered tap water. The NUKON™ fiber was chopped and sized to match the industry reported average strainer bypass distribution per an acceptable procedure. Each batch was characterized by light microscopy to determine the distribution of the fiber lengths. The actual fiber distributions fall within the allowable limits of the target fiber distribution shown in Table 1.

Table 1: Values Specified for Fiber Length

Description	Specified Value	Range
Fiber length < 500μm:	77% + 10%	67%-87%
500μm < Fiber length < 1000μm:	18% + 10%	8%-28%
Fiber length > 1000μm:	5% + 10%	0%-15%

Silicon carbide powder with a nominal 9.5 micron particle size was used to simulate particulate debris. The actual particulate size was measured using scanning electron microscopy. This silicon carbide powder is used as a surrogate for the particulate debris in the reactor because of its chemical stability and the fact that the fine particulates collect within a fiber bed and result in conservative head losses. Silicon carbide has a relatively high specific gravity of about 3.2, which would normally cause it to settle out quickly. However, due to the small size of the particles and the test loop design and flow rates, this settling is minimized.

The microporous insulation, Microtherm, was obtained from Microtherm, Inc. The material was supplied in a pulverized form, and then was passed through a #7 sieve with a hole size of 0.11". The sieving is necessary to remove larger fibers and clumps of material that would not pass through the sump screen. Then the material was analyzed by scanning electron microscopy to characterize the material. The typical appearance of the Microtherm material used can be seen in Figure 5.

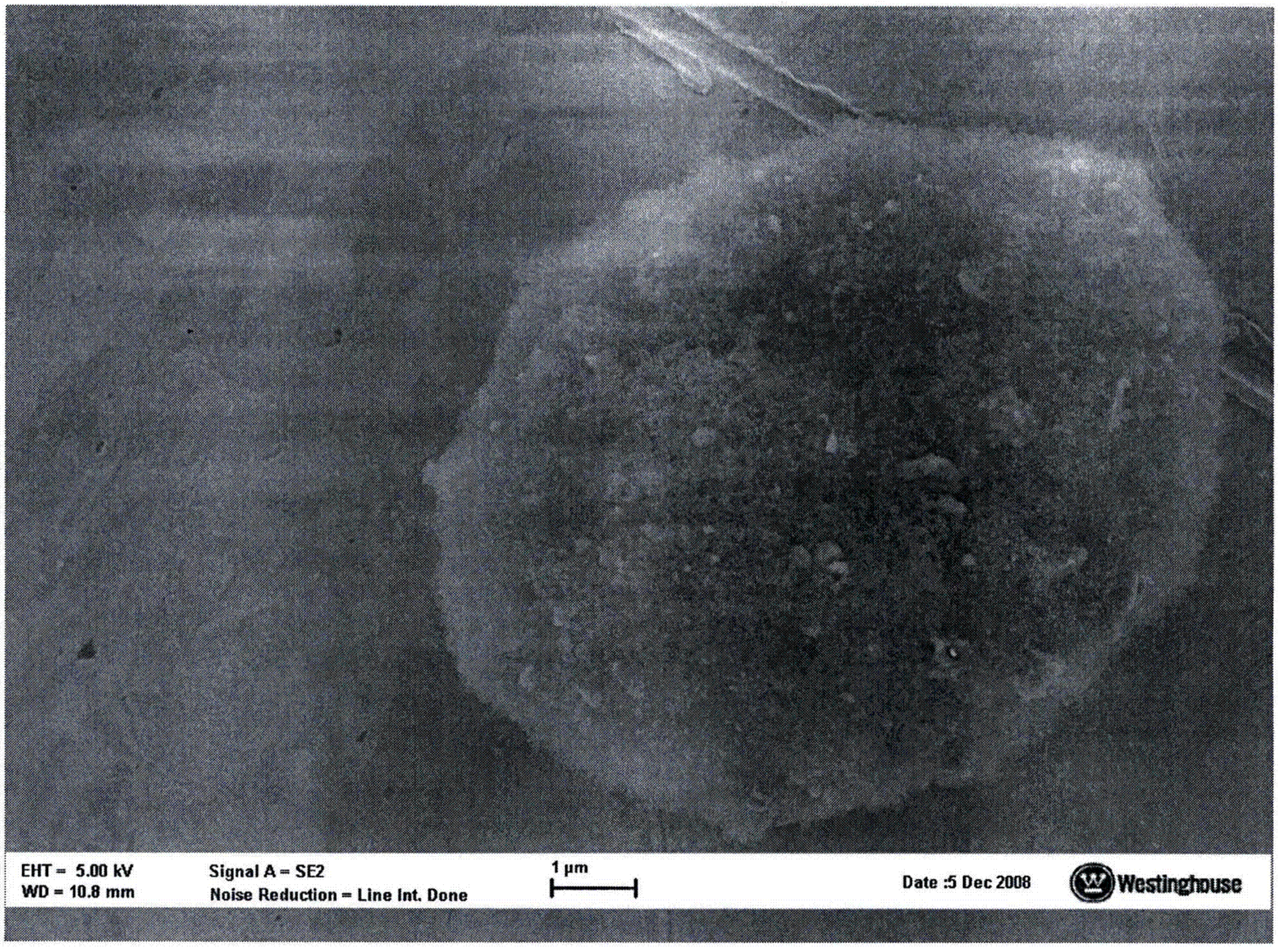


Figure 5: Microtherm Scan

Calcium silicate insulation material, Lot # S15-276, was obtained from Performance Contracting Incorporated (PCI). PCI obtained the material from Industrial Engineering Group and pulverized it into a fine powder by a hammer mill. Upon receipt at Westinghouse Science and Technology Department, the material was passed through a #7 sieve with a hole size of 0.11". Then the CalSil was analyzed by scanning electron microscopy to characterize the material. The typical appearance of the CalSil material used can be seen in Figure 6.

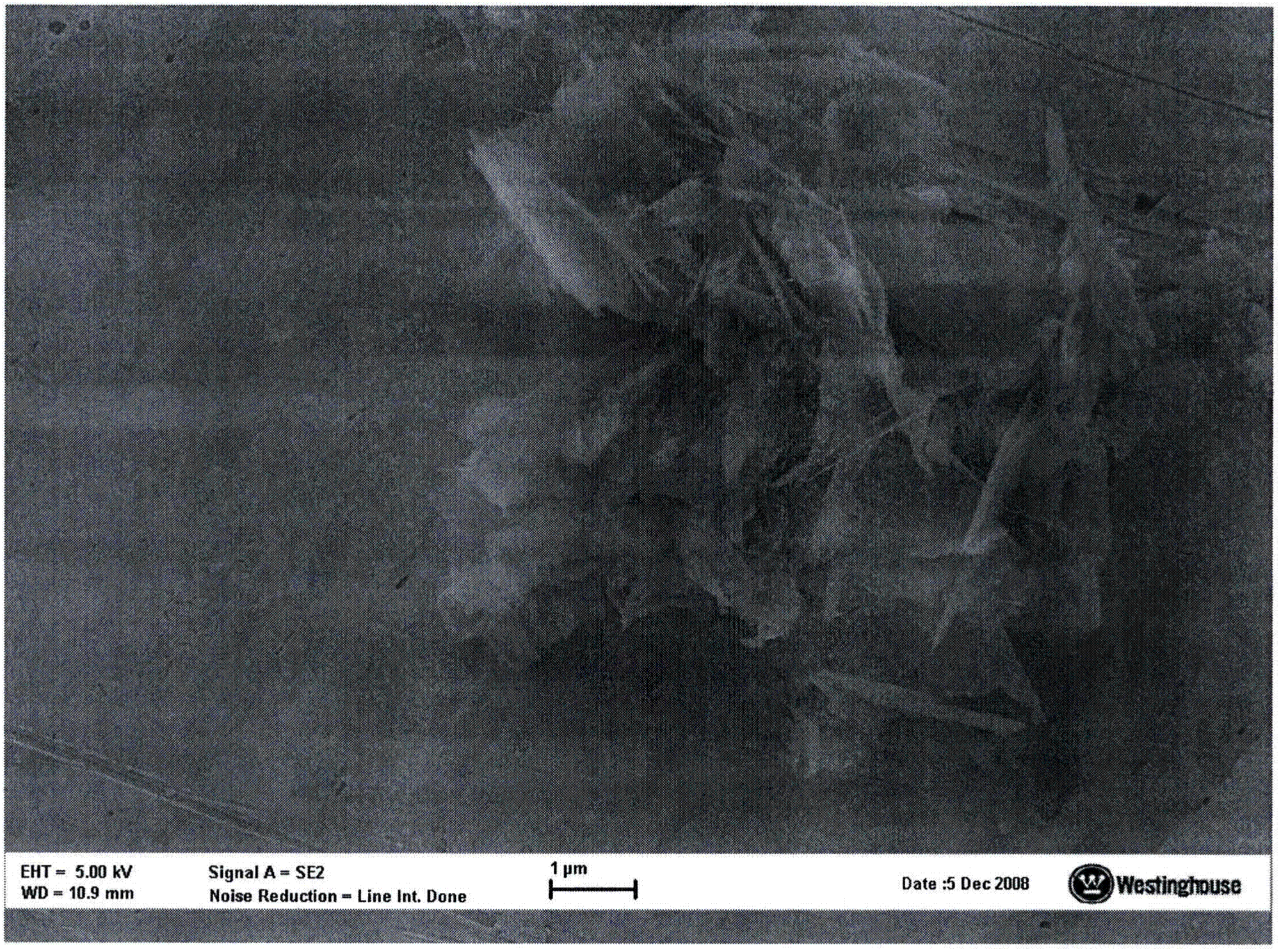


Figure 6: Calcium Silicate Scan

AlOOH was prepared according to the recipe in WCAP-16530-NP-A at a concentration of 11 g/L. The 1 hour settling volume of the precipitate met the criteria in WCAP-16530-NP-A.

Order of Debris Addition:

For tests that only included particulate, fiber and chemical, NRC guidance regarding the order of addition was followed. The entire particulate load was added first, followed by fiber in 10 gram increments and then by chemicals in specified increments. For tests that included particulate, fiber, chemical, calcium silicate and/or microporous material, the order of addition was varied slightly. Like the other tests, the entire particulate load was the first addition, then, to simulate the initial blast introduction of calcium silicate and/or microporous material, a specified amount of these materials were added, this was followed by fiber additions in 10 gram increments, then the chemical was added and the final additions were calcium silicate and/or microporous material to simulate the slow erosion of these materials during an accident.

Method of Debris Introduction:

This is a lengthy and site specific discussion. The method of introduction will be extensively covered in the test report associated with Revision 1 of WCAP-16793-NP.

Rate of Debris Addition:

The initial particulate addition of silicon carbide was introduced in its entirety. The remaining debris types required a wait of two loop turnovers between each addition.

Information Related to Debris Test Amounts:

It was communicated to the PWROG that an acceptance criteria for debris loads was being developed. In order to define debris test loads that were applicable to and bounds, to the extent possible, all PWRs, all plants were asked to provide their downstream debris values.

These values were then divided by the total fuel assemblies of each plant. This provided the test program with per fuel assembly debris values. These values were then used to determine the bounding conditions of the fuel assembly tests.

The amounts of each type of debris added to each test will be published in the next revision of WCAP-16793-NP.

Test Observations:

This section addresses the following parts of the RAI:

- Head loss value experienced for each test or subtest including time dependent plots if available
- Observations of debris transport and accumulation including any settling with differences noted at different flow rates
- Behavior of debris during testing (agglomeration)
- Photographs as available to assist in understanding the tests theoretical debris bed thickness based on as-manufactured fiber density

These will be discussed in-depth in proprietary test reports that will be submitted ton NRC and are associated with Revision 1 of WCAP-16793-NP.

7) For hot leg and cold leg breaks, some debris may bypass the fuel inlet because it flows to the containment spray system (CSS) instead of the emergency core cooling system (ECCS). Also, for cold-leg breaks, some flow bypasses the core by flowing out the break. If bypass is credited for a reduction of debris at the core inlet, please provide the basis for the magnitude of the reduction of debris entering the core.

RESPONSE:

Subsequent to receipt of this RAI, and in response to comments from the Advisory Committee on Reactor Safeguards (ACRS), the PWR Owners Group initiated prototypical fuel assembly (FA) testing to establish limits on the debris mass (particulate, fibrous and chemical) that could bypass the reactor containment building sump screen and not result in unacceptable head loss that would impede core inlet flow and challenge long-term core cooling of the core. An overall test protocol and specific test procedures were developed to ensure that possible thin bed effects were investigated, and debris types and characteristics expected in the Reactor Coolant System (RCS) were represented. Debris loads used in the test were based on sump screen bypass information provided by licensees.

The effects of differing fuel inlet nozzle designs were also considered in the test program. Both AREVA and Westinghouse have performed testing with their respective fuel inlet nozzles. Both vendors tested their various bottom nozzle designs and identified the limiting design (limiting was defined as the design that provided for the maximum pressure drop at the same flow and debris loading conditions). Each fuel bundle tested also had prototypical grids above the bottom nozzle debris capturing design features. Thus, the test data obtained from testing takes into account the fuel inlet nozzle, protective filter design features, and spacer grid designs. Descriptions of the fuel components tested, including bottom nozzles and grids, will be provided in proprietary submittals describing the testing performed and the results obtained.

Testing was performed using bounding debris loads and hot leg break flow rates. These tests demonstrated that for the bounding debris loads tested, the hot-leg flow rate through the fuel assembly mock-up was maintained with acceptable pressure drops. No credit was taken in these tests for flow bypassing the core due to operation of the containment spray system. The results of these tests will be integrated into Revision 1 of WCAP-16793-NP.

Following a LOCA, some debris may pass through the sump screens and enter the ECCS system. The ECCS system will deliver fluid and debris to the containment spray (CS) system and to the RCS. For the RCS and core evaluations, it is conservative to assume that all of the debris that passes through the sump screens reaches the RCS. Therefore, the WCAP-16793-NP methodology does not credit debris reduction by considering flow through the CS system.

However, licensees may choose to credit this flow path and reduce the debris that reaches the RCS by considering the following:

1) The flow split between what is delivered to the core and what is delivered to the postulated break location (i.e., cold-leg break versus hot-leg break),
2) The flow split between what is delivered to containment spray system and what is delivered to the reactor coolant system, and,
3) The time frame that the containment spray system is operational.

Of the debris that reaches the RCS, the amount that is transported to the core is dependent on the ECCS injection configuration and break location. ECCS is delivered to the RCS in two locations depending on the plant type. For most PWRs, ECCS is delivered to the cold legs or upper RV downcomer. For Westinghouse 2-loop designs, ECCS is also delivered to the RV upper plenum.

Licensees that may choose to credit partitioning of flow between the core and the break, or the reactor vessel and the containment spray system, would do so on a plant specific basis. These licensees would develop the technical basis for their crediting a reduction of debris loading ducted to the core based on the flow splits identified above, consistent with expectations for information NRC has identified that they need to evaluate these exceptions to the methods described in WCAP-16793-NP.

8) Following a LOCA, thermal energy stored in the thick reactor vessel shell and the reactor vessel baffle/barrel can influence the coolant temperature at the core inlet. For both a hot leg break and a cold leg break, please provide an estimate of the core inlet temperature as a function of time, starting at the onset of ECCS recirculation and ending when an equilibrium reactor vessel metal temperature has been reached. Please discuss how this temperature would affect:
   a. the solubility of aluminum-based precipitates,
   b. the solubility of calcium-based precipitates and,
   c. the potential for chemical precipitates to form in the vessel as a result of these phenomena.

<u>RESPONSE:</u>

The thermal energy stored in the thick reactor vessel (RV) shell and the RV baffle/barrel is small, as demonstrated below, and has no more than about a 5°F influence on the coolant temperature from the time it enters the RV until it enters the core inlet. This temperature rise in the RV is small and has results in no more than about a 5% change in solubility of aluminum-based and calcium-based precipitates. This change has no affect on the potential for chemical precipitates to form in the vessel as a result of these phenomena.

The postulated cold-leg break was chosen as this is the bounding case for heat-up of the coolant as it passes by the thick metal components of the RV. The low flow-rates associated with a cold-leg break (matching boil-off) provide the greatest residence time of the fluid next to the metal structures, allowing for the maximum heat-up of the coolant. A postulated hot-leg break, while having a larger velocity, also has a reduced residence time in the RV, minimizing the opportunity for coolant heat-up.

At the time that the Emergency Core Cooling System (ECCS) is realigned to draw suction from the reactor containment building sump from the Borated/Refueling Water Storage Tank (BWST/RWST), the heat transfer process between the thick metal components of the RV and the ECCS fluid in the RV is conduction limited. Under these conditions, there is little increase in temperature of the ECCS fluid as it passes by the thick-metal RV components and enters into the reactor core. The time history plots prepared from the WCOBRA/TRAC calculations reported in WCAP-16793-NP, confirm that this is conduction-limited heat transfer process, and that there is minimal temperature change of the coolant as it enters the RV and flows to the core.

Figure 1, Comparison of Reactor Vessel Metal Temperature at Bottom of Fuel; Outside Diameter versus Inside Diameter, and Figure 2, Comparison of Reactor Vessel Metal Temperature at Top of Fuel; Outside Diameter versus Inside Diameter, are time history plots of the temperature of the inner and outer RV metal nodes of the <u>W</u>COBRA/TRAC calculations for a postulated cold-leg break. From Figures 1 and 2, it is noted that the temperature of the inner RV metal node at the top and bottom of the core is relatively unchanged over the 300 seconds following switchover from BWST/RWS injection to recirculation from the reactor containment building sump. Over this same time period, the outer RV node is predicted to drop by about 30°F. These figures demonstrate that the heat transfer process is conduction limited.

Figure 3, Comparison of Fluid Temperature at Top and Bottom of Downcomer, shows that there no more than about 5°F temperature gain in the coolant as it passes from the top to the bottom of the downcomer. Likewise, Figure 4, Comparison of Fluid Temperature at Top and Bottom of Baffle, shows a similar behavior. It is noted that the initial 10°F temperature difference diminishes to about a 5°F temperature difference within about 150 seconds of switchover from BWST/RWST injection to recirculation from the reactor containment building sump. Figure 5, Comparison of Fluid Temperature in Lower Plenum to Core Inlet, shows that the coolant at the core entrance is calculated to be generally slightly warmer but within about 5°F of the coolant in the RV lower plenum. Figures 6 and 7, Comparison of Fluid Temperature Between Core Inlet and Inside Baffle, and Comparison of Fluid Temperature Between Core Outlet and Inside Baffle, respectively, shows the calculated fluid temperatures at the core inlet and core outlet to be within less than about 5°F of each other throughout the calculation time period. More importantly, over the last 100 seconds of the calculation period, comparisons show almost no temperature difference between the fluid in the core and in the baffle.

Based on these comparisons for a postulated cold-leg break, it is concluded that the thermal energy stored in the thick RV shell and the RV baffle/barrel has no more than about a 5°F influence on the coolant temperature from the time it enters the RV until it enters the core inlet for either the cold-leg or hot-leg break scenarios. This conclusion is applicable to all plants, as is demonstrated by considering the Biot number, N_{Bi}, for this scenario. The Biot number is the ratio of surface conductance to internal conduction of a solid;

$$N_{Bi} = \frac{H \times L}{k}$$

where:

H	=	Surface heat transfer coefficient
L	=	Thickness of the solid
k	=	Thermal conductivity of the solid

At the time of initiation of recirculation from the reactor containment building sump, there is no boiling in the downcomer and the convective heat transfer coefficient between the thick metal and the coolant is dependent upon local flow rate and is evaluated to between less than $3 \frac{Btu}{hr - ft^2 - {}^oF}$ for a postulated hot-leg break. The thickness of a reactor vessel is about 8 inches. For evaluating a Biot Number, one-half of the thickness or 4 inches (0.33 ft.) will be used. The thermal conductivity of mild (carbon) steel is about $28 \frac{Btu}{hr - ft \ - {}^oF}$. Thus, the Biot Number for this scenario would be;

$$N_{Bi} \leq 0.036$$

The above calculation demonstrates that the dominate resistance to heat transfer from the reactor vessel thick metal during recirculation is due to the convective resistance between the reactor vessel surface and the fluid.

The stainless steel cladding on the inside of the reactor vessel was ignored for this evaluation. Stainless steel is about 1/3 as conductive as mild (carbon) steel. Although the cladding is thin, inclusion of this material in the evaluation of a Biot Number would further favor the convection limited process.

The fluid temperature rise of ≤ 5°F predicted by WCOBRA/TRAC calculations for a postulated cold-leg break is small in comparison to that needed to change solubility limits and is evaluated to have no affect on the solubility of aluminum-based precipitates, the solubility of calcium-based precipitates and the potential for chemical precipitates to form in the vessel as a result of the release of stored thermal energy from thick-metal components of the RV.

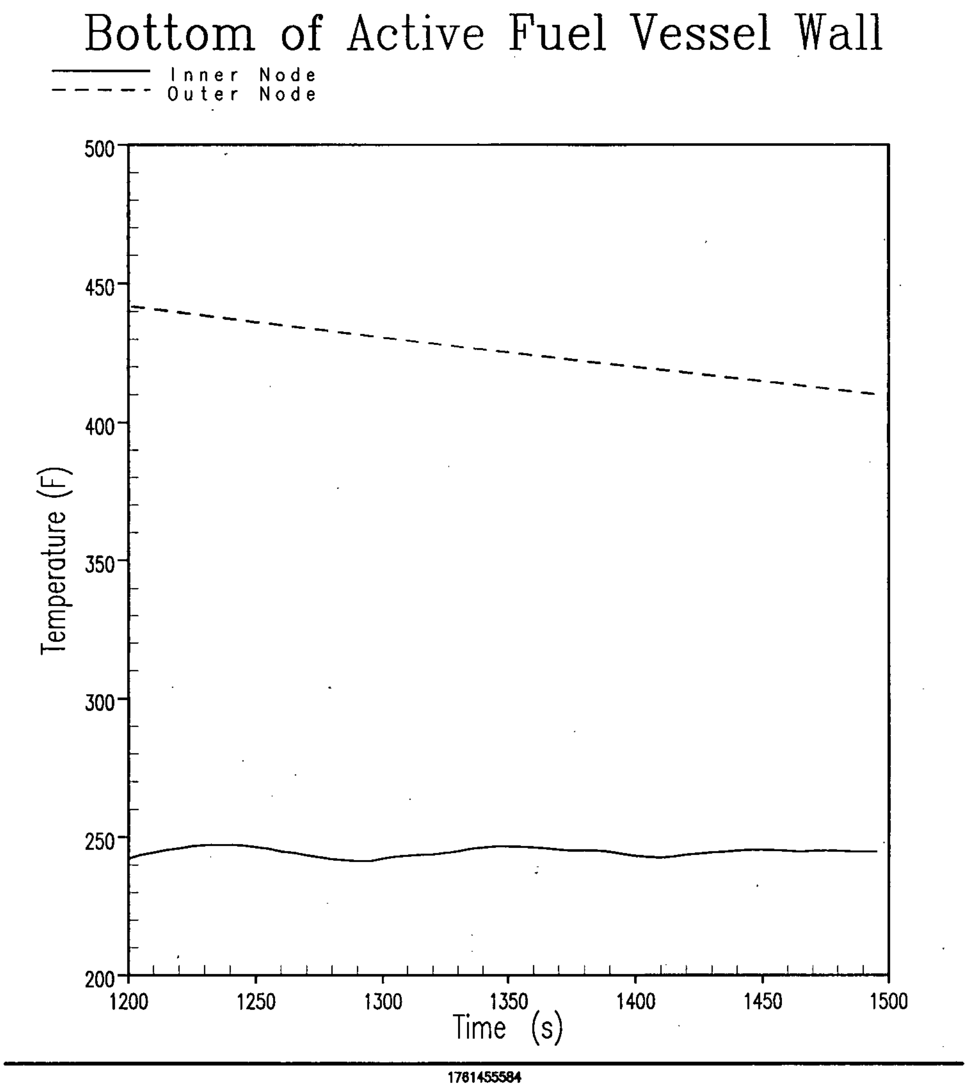


Figure 1 Comparison of Reactor Vessel Metal Temperature at Bottom of Fuel; Outside Diameter versus Inside Diameter

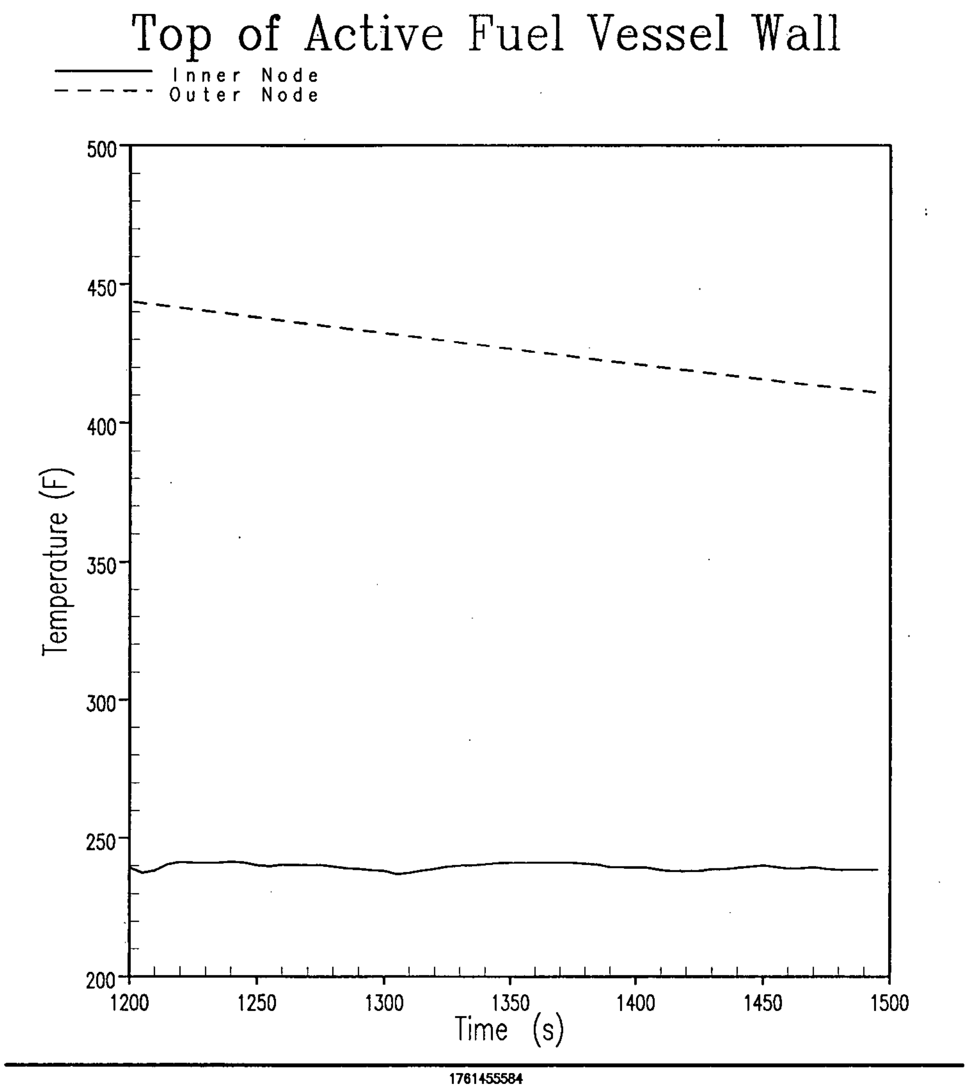


Figure 2 Comparison of Reactor Vessel Metal Temperature at Top of Fuel; Outside Diameter versus Inside Diameter

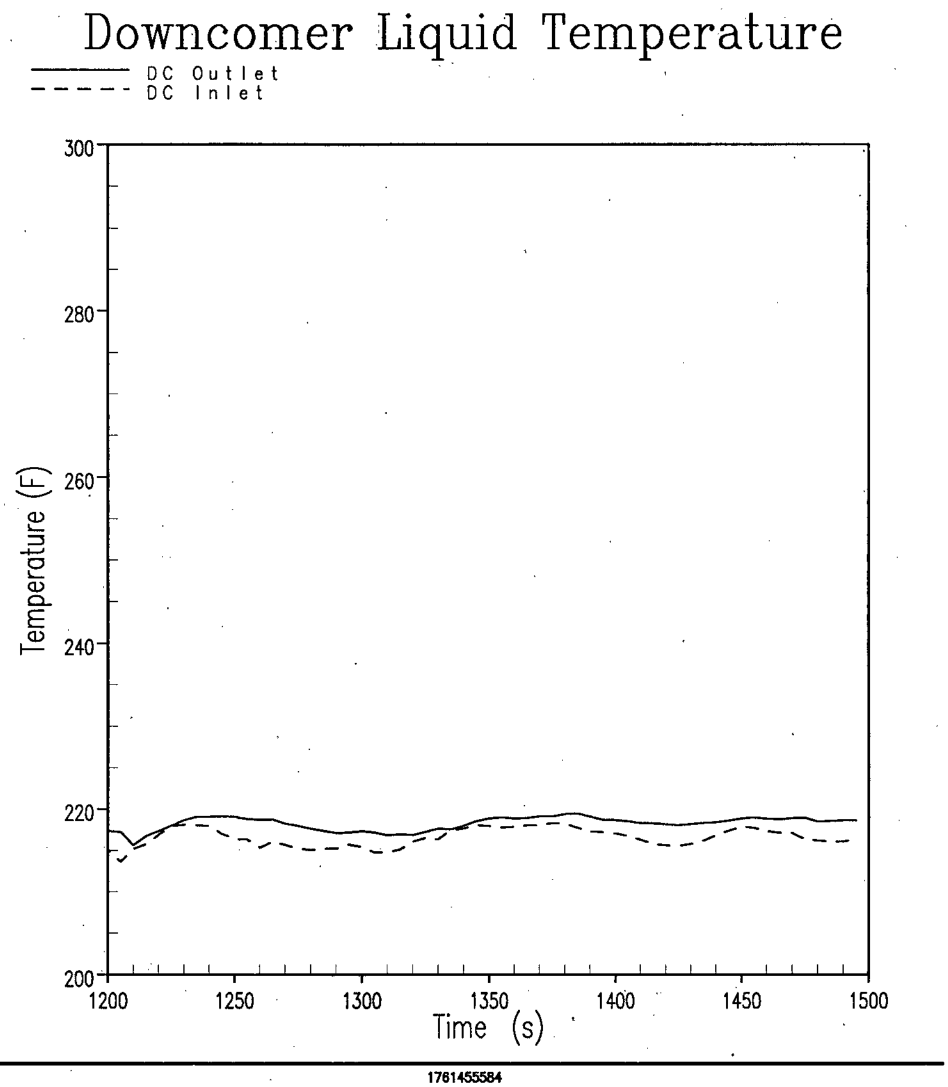


Figure 3 Comparison of Fluid Temperature at Top and Bottom of Downcomer

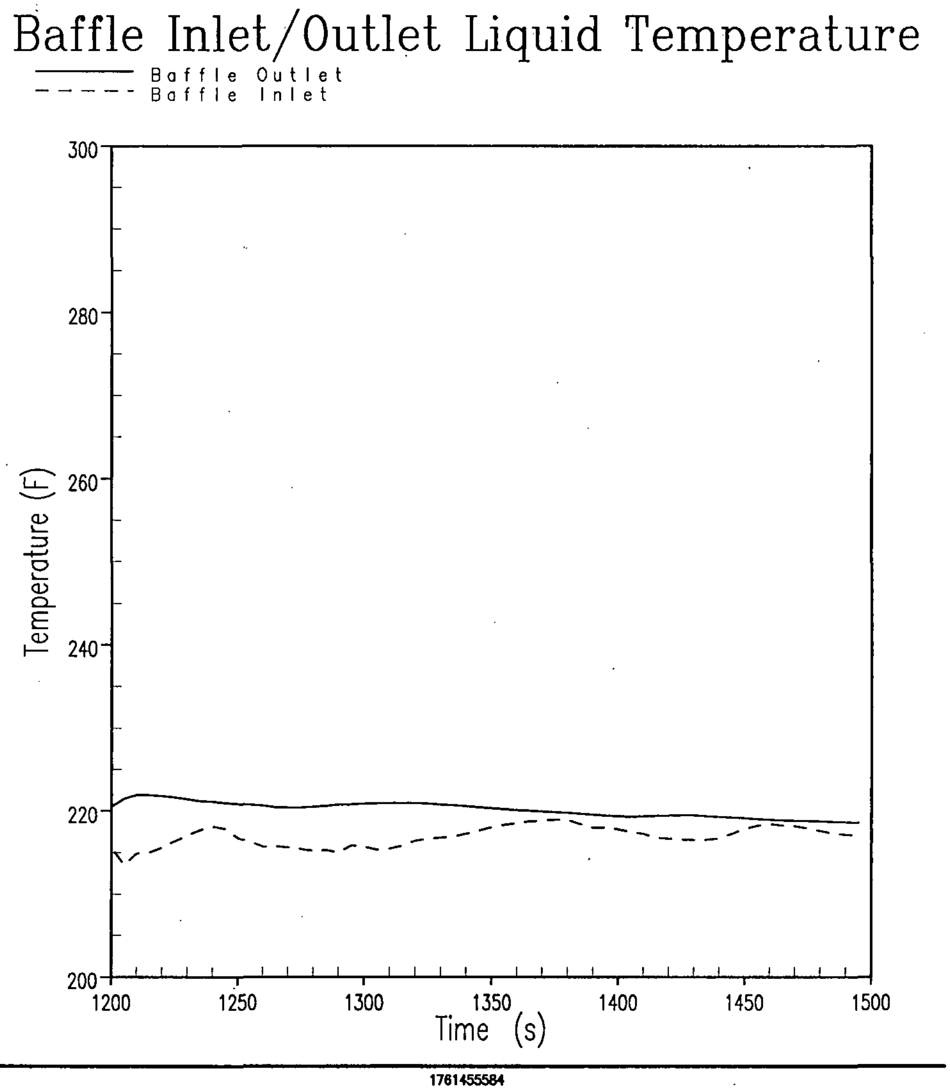


Figure 4 Comparison of Fluid Temperature at Top and Bottom of Baffle

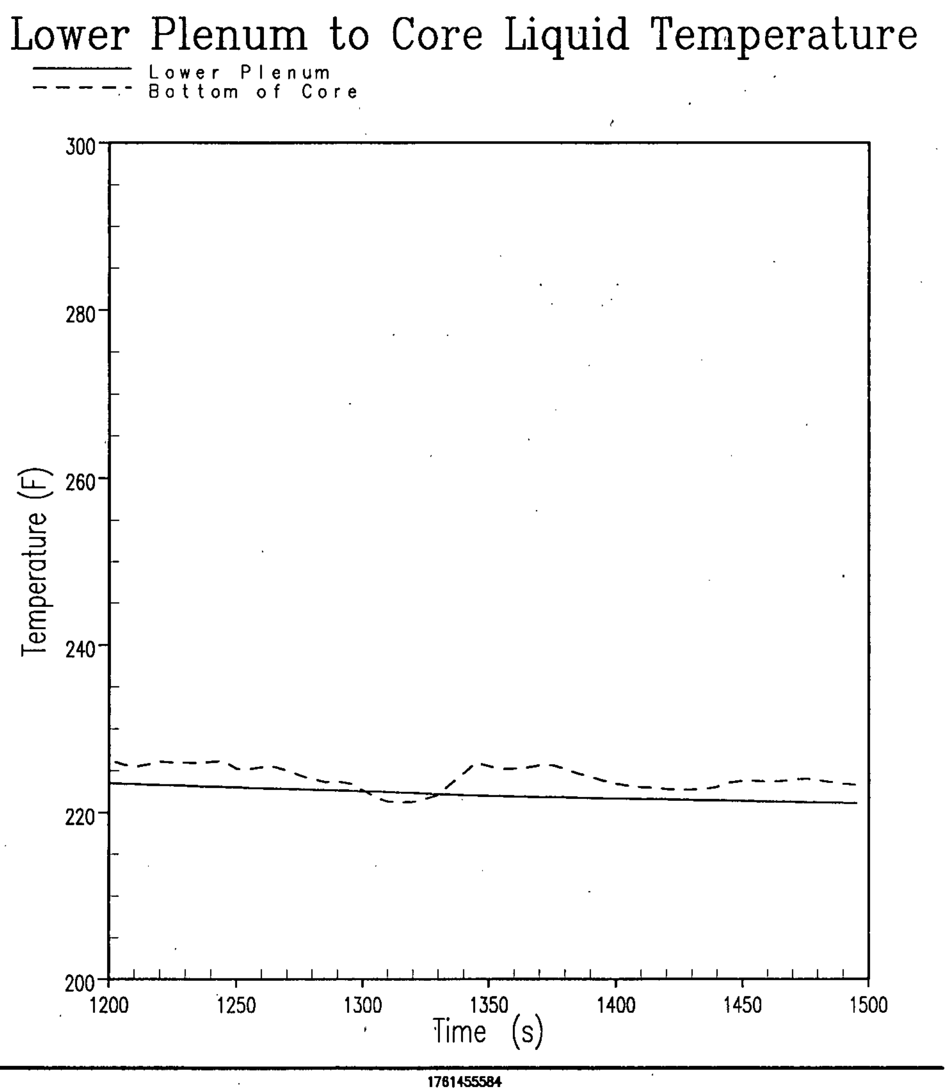


Figure 5 Comparison of Fluid Temperature in Lower Plenum to Core Inlet

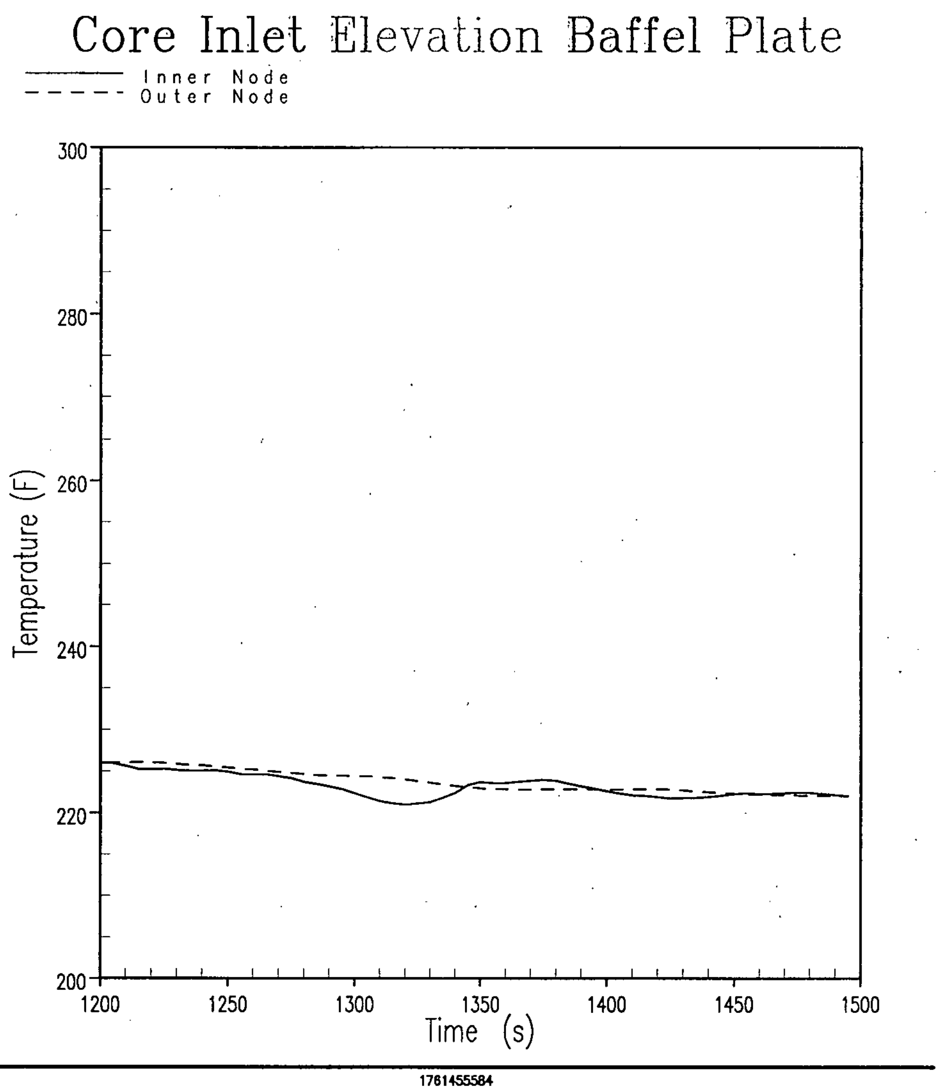


Figure 6 Comparison of Fluid Temperature Between Core Inlet and Inside Baffle

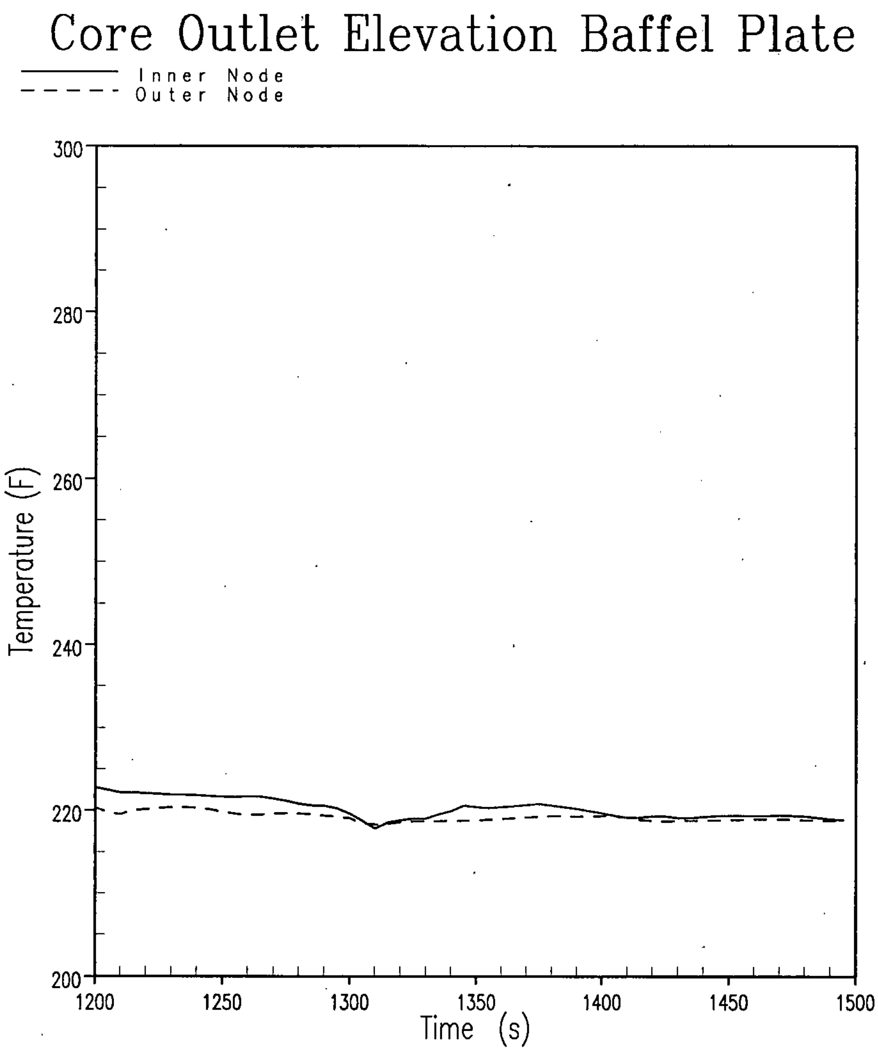


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Figure 7 Comparison of Fluid Temperature at Core Outlet and Top Baffle

9) The topical report does not provide specific guidance to licensees concerning the evaluation of potential chemical effects on a debris bed formed at the core inlet. Various factors can affect potential chemical precipitate interaction with a debris bed on the core inlet. For example, plant-specific amounts of LOCA debris and sump strainer surface area will determine the amount and type of debris materials that bypass the sump strainer. Bypass particulate such as microporous insulation and calcium silicate will influence the filtering properties of a debris bed differently than latent dirt particulate. Elevated temperature can either increase the solubility of precipitates or decrease the solubility of certain precipitates. Please discuss how pressure drop at the core inlet could result from chemical precipitate interaction with a debris bed. Also, please discuss your plans for providing guidance in the WCAP for licensees to evaluate this potential phenomenon.

RESPONSE:

Subsequent to receipt of this RAI, the PWROG began prototypical fuel assembly (FA) testing to establish limits on the debris mass (particulate, fibrous, microporous and chemical) that could be tolerated in the reactor core and the long-term core cooling function continue to be successfully achieved. Debris loads used in the test were based on sump screen bypass information provided by licensees and post-accident chemical precipitate loads based on evaluations using the methods of wCAP-16530-NP-A (Reference 1). The FA testing will be reported in proprietary submittals that will be made in support of Revision 1 of WCAP-16793-NP. The results from these FA tests will be integrated into Revision 1 of WCAP-16793-NP. As part of the effort to invoke this WCAP in the plant licensing basis, each plant will compare their plant-specific debris load against the FA debris masses tested. Revision 1 of WCAP-16793-NP will also include guidance on how that comparison is to be accomplished.

Reference:

1. WCAP-16530-NP-A, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," March, 2008.

10) In addressing the effects of core inlet blockage on the availability of the lower-plenum mixing volume to delay the onset of boron precipitation, the WCAP states, "In the extreme, core inlet blockage could inhibit mixing between the core region and the lower plenum and would effectively reduce the credited mixing volume contribution of the lower plenum and core baffle region in the analysis of record." The report also states, "Only total or severe core inlet blockage would effectively isolate the lower plenum from the core region." Please provide an analysis of the degree of core isolation and reduction in mixing capability expected for the degree of core blockage created by the quantity of bypassed debris evaluated in the WCAP to be acceptable. Also, please address the effect the density gradients between the liquid in the core and the liquid in the lower plenum may have on localized fluid velocities and the transport of debris to and/or into the core.

RESPONSE:

Subsequent to receipt of this RAI, and in response to comments from the Advisory Committee on Reactor Safeguards (ACRS), the PWR Owners Group initiated prototypical fuel assembly (FA) testing to establish limits on the debris mass (particulate, fibrous and chemical) that could bypass the reactor containment building sump screen and not result in unacceptable head loss that would impede core inlet flow and challenge long-term core cooling of the core. An overall test protocol and specific test procedures were developed to ensure that possible thin bed effects were investigated, and debris types and characteristics expected in the RCS were represented. Debris loads used in the test were based on sump screen bypass information provided by licensees. The results of these tests will be integrated into Revision 1 of WCAP-16793-NP.

The effects of differing fuel inlet nozzle designs were also considered in the test program. Both AREVA and Westinghouse have performed testing with their respective fuel inlet nozzles. Both vendors tested their various bottom nozzle designs and identified the limiting design (limiting was defined as the design that provided for the maximum pressure drop at the same flow and debris loading conditions). Each fuel bundle tested also had prototypical grids above the bottom nozzle debris capturing design features. Thus, the test data obtained from testing takes into account the fuel inlet nozzle, protective filter design features, and spacer grid designs. Descriptions of the fuel components tested, including bottom nozzles and grids, will be provided in proprietary submittals describing the testing performed and the results obtained.

Testing was performed using bounding debris loads and hot leg break flow rates. These tests demonstrated that for the bounding debris loads tested, the hot-leg flow rate through the fuel assembly mock-up was maintained with acceptable pressure drops. The maintenance of core flushing flow with full 30-day debris loads (particulate, fibrous and chemical) precludes boric acid precipitation for hot-leg breaks. Additional details on the possibility of localized blockage following a hot leg are provided in the response to RAI #11, which follows.

For a postulated cold-leg break, the core flow rate is determined by core boil-off, which is on the order of 500 gpm or less at the time of initiation of recirculation from the reactor containment building sump and continually decrease as the time passes. Flow in excess of core boil-off spills from the reactor vessel downcomer and out the cold-leg break and into the reactor containment building sump where it is again recirculated through the sump screen. In this scenario, both the debris loads provided to the core and the core flow rates are considerably lower than for a postulated hot-leg break.

From the fuel assembly testing performed with postulated hot-leg break flows and debris loading conditions, flow through the gaps at the fuel assembly bottom nozzles and gaps at grid straps was observed. These gaps would exist at cold-leg flow rates and continue to provide coolant to remove decay heat. A test using cold-leg break flows and debris loads has been conducted to prove this principle (Note: test will be performed by mid-February 2009.)

In addition, the timing of events should be considered in evaluating boric acid precipitation concerns as described below.

- Following a large LOCA, realignment of the Emergency Core Cooling System (ECCS) and Containment Spray System (CSS) to draw suction from the reactor containment building sump can occur from between about 20 minutes to about 60 minutes after break initiation, depending upon the size of the Borated/Refueling Water Storage Tank (BWST/RWST) and number of ECCS and CSS trains in operation.

- Over a period of time that is determined by plant design the next 1 to 2 hours, again depending upon the number of trains in operation and their pumping capability, the ECCS will recirculate roughly the entire sump volume through the sump screens.

  - For B&W and Westinghouse plants, this duration may range from about 30 minutes to 2 hours, depending upon the number of trains operating and their pumping capacity.

  - For CE plants, this duration may take several hours more.

- For a postulated hot leg LOCA, most of the ECCS flow is through the core. Testing has shown that the core inlet did not totally block at the debris loads representing 30 days of sump recirculation.

- For the postulated cold-leg break:

  - For B&W and Westinghouse plants, approximately 1/5th of the ECCS flow may reach the core inlet.

  - For CE plants, as much as approximately ½ of the ECCS flow may reach the core inlet.

- Thus, the debris loading associated with a cold-leg break is a fraction of that of the hot-leg break. Therefore there is less debris at the core inlet for the cold-leg break than for a hot-leg break.

- Hot leg recirculation (from high head/low flow pumps) causes core flow reversal and debris loading at the core inlet is terminated.

- Also, since the hot leg recirculation dilutes the boric acid in the core, the boric acid precipitation concerns are alleviated and the lower plenum mixing volume is no longer needed.

These events, and their timing, further mitigate the concern regarding the impact of debris collection at the core inlet and on grids on potential boric acid precipitation.

Taking the discussion of sequencing and timing of events given above in conjunction with the limits on sump screen bypass set by testing performed to date, the reactor vessel lower plenum would continue to be available as a mixing volume to mitigate for boron precipitation.

With respect to boron precipitation, it is also noted that the Pressurized Water Reactor Owners Group (PWROG) has a program in place to develop a new Post LOCA-Boric Acid Precipitation Analysis Methodology. That program will utilize the same bounding debris loading determine by the fuel-debris testing in considering the effects of debris in the recirculating coolant on boric acid precipitation analysis methodology.

11) The WCAP does not include a discussion on boron precipitation associated with a hot-leg break. To address the effect of localized blockage on localized boron precipitation for hot-leg break scenarios, please address the following:

a. In the event of localized debris accumulation at the core inlet, has it been demonstrated that adequate flow would travel through or around the debris such that excessive boron build up is prevented? What quantity of boron would be expected to precipitate downstream of the local blockage?

b. Discuss, in terms of boron precipitation, the effects of local blockage on first grid structure above the fuel inlet nozzle.

c. Describe how boric acid control measures would be effective at controlling the potential localized precipitate buildups, as well as controlling general boron precipitation.

If localized boron precipitation were to occur due to local debris accumulation or lack of mixing between the core and the lower plenum, please state and justify the conclusion regarding whether the 800 F peak cladding temperature acceptance criterion would be met.

RESPONSE:

For most Pressurized Water Reactors (PWRs) currently in operation in the US, following a postulated hot leg break, coolant provided by the Emergency Core Cooling System (ECCS) is ducted into the cold legs or upper downcomer and must pass through the core region to reach the break. Some flow may pass through the baffle region, but the majority of the flow will pass through the core. For these plants, the core flow is approximately equivalent to the ECCS flow rate. With the continuous flow through the core, the boric acid concentration does not increase substantially; any concentrated by boiling is continuously flushed upward through the core and out of the break. Consequently, bulk boric acid precipitation following a hot leg break for most PWRs currently in operation in the US is not a concern unless the core inlet becomes nearly completely blocked. As described later in this response, fuel assembly debris capture testing has been recently performed to enable plants compare their plant-specific conditions to those tested with the objective of demonstrating that sufficient flow is maintained to provide for core cooling following a postulated hot-leg break. With continual flow through the core, bulk precipitation of boric acid following a postulated hot leg break will not occur.

Some PWRs introduce coolant directly into the reactor vessel upper plenum using an upper plenum injection (UPI) design. The effect of debris on UPI plants is discussed in Section 2.7.2, "Upper Plenum Injection Plants," of WCAP-16793-NP, Revision 0. In addition, responses to RAI 9, RAI 10, RAI 33 and RAI 44 in the first set of RAIs received from NRC on WCAP-16793-NP address debris collection by and within the core following a postulated hot-leg break. Finally, to address a request from NRC reviewers, the response to RAI 7 from that collection of RAIs also identified a licensing basis boric acid precipitation analysis for a UPI plant. Also, subsequent to receipt of the RAI, the PWR Owners Group initiated prototypical fuel assembly (FA) testing to establish limits on the debris mass (particulate, fibrous and chemical) that could enter the core and not result in unacceptable head loss that would impede core inlet flow and challenge long-term core cooling. This testing addresses the UPI plant configuration.

This RAI implies that localized blockage could induce some localized boric acid precipitation. By way of a response, it is insightful to review the conditions required for boric acid precipitation. In order to concentrate boric acid to the solubility limit, it must concentrate by a factor greater than 20:1 over the typical initial RCS boric acid concentration. This means that in order to concentrate to the precipitation point, the water in an isolated volume in the core would need to be converted to steam and be replaced 20 times. Since the concentrating mechanism is boil-off, the conditions would require that the steam escape the isolated volume without disrupting the isolation barrier and would also require that the makeup flow into the control volume be precisely the same amount as boil-off. If steam flow out of the isolated volume promotes liquid flow out of the volume, or if liquid flow into the volume exceeds boil-off by some degree, the boric acid concentration buildup will cease. Sample calculations (provided in Attachment B) show that the liquid mass flow into the volume of less than 110% boil-off mass flow will be sufficient to dilute the volume. A completely isolated volume is ruled out as incredible; this would result in overheating of the fuel. Therefore, the sample calculations indicate that the semi-isolated volumes would need only flow communication of 10% above boil-off.

As identified in the first paragraph of this response, subsequent to receipt of this RAI, and in response to comments from the Advisory Committee on Reactor Safeguards (ACRS), the PWR Owners Group initiated prototypical fuel assembly (FA) testing to establish limits on the debris mass (particulate, fibrous and chemical) that could bypass the reactor containment building sump screen and not result in unacceptable head loss that would impede core inlet flow and challenge long-term core cooling of the core. An overall test protocol and specific test procedures were developed to ensure that possible thin bed effects were investigated, and debris types and characteristics expected in the RCS were represented. Debris loads used in the test were based on sump screen bypass information provided by licensees. The results of these tests will be integrated into Revision 1 of WCAP-16793-NP.

The effects of differing fuel inlet nozzle designs were also considered in the test program. Both AREVA and Westinghouse have performed testing with their respective fuel inlet nozzles. Both vendors tested their various bottom nozzle designs and identified the limiting design (limiting was defined as the design that provided for the maximum pressure drop at the same flow and debris loading conditions). Each fuel bundle tested also had prototypical grids above the bottom nozzle debris capturing design features. Thus, the test data obtained from testing takes into account the fuel inlet nozzle, protective filter design features, and spacer grid designs. Descriptions of the fuel components tested, including bottom nozzles and grids, will be provided in proprietary submittals describing the testing performed and the results obtained.

Testing was performed using bounding debris loads and hot leg break flow rates. These tests demonstrated that for the bounding debris loads tested, the hot-leg flow rate through the fuel assembly mock-up was maintained with acceptable pressure drops. The maintenance of core flushing flow with full 30-day debris loads precludes boric acid precipitation for hot-leg breaks. This conclusion is applicable to all Pressurized Water Reactors (PWRs) that introduce their ECCS flow into the reactor coolant system cold legs.

The intent of the test program was, in part, to determine the amount of debris loading passing through the reactor containment building sump screen ("debris bypass") that would impede core flow. To use the results of this test program for closure of GSI-191, each plant will compare their plant-specific debris bypass load against the debris masses that were tested and determined to be acceptable. Plants that have bypass debris loadings that are within the limits of the debris masses tested are bounded by the test; the debris build-up for that plant will not impede core flow below that required to remove core decay heat.

The test FA included a bottom nozzle and a number of intermediate spacer grids. The relationship between blockages at each of these locations and the possibility of localized boric acid precipitation is discussed separately.

1. At the core inlet, the initial results from tests that simulated flow rates following a postulated hot leg break indicate that some debris buildup might occur on the fuel filters. As stated above, the test results showed that the head loss due to the debris buildup is not sufficient to block flow from entering the core. If the fuel filter doest not block completely, then continuous flow is assured and no localized precipitation will occur. If the filter blocks completely, gaps between fuel assembly bottom nozzles and the fuel assemblies themselves will allow flow to enter the core region.

2. If a small localized region of the bottom nozzle fuel filter blocks completely, then flow will continue around the blockage through the fuel filter. Due to the core quenching process and the core power shape, boiling near the bottom of the core is lower than at higher elevations. If the region downstream of the blockage becomes starved of flow and begins to boil, then the boil-off will be replaced by the liquid that flowed around the blockage or through the gaps between the fuel assemblies. This liquid inflow will assure that these regions remain well mixed and preclude localized precipitation at the core inlet.

3. The tests also showed that, at the core spacer grids, some debris build up might occur. However, initial test results indicate that the buildup does not preclude flow through the debris bed; the blockage is not complete and the fluid remains well mixed such that localized boric acid precipitation will not occur.

4. However, even if a solid localized blockage occurs, once the flow passes through the gaps between the FA spacer grids or around a small localized blockage, the low pressure region just downstream of the blockage caused by boiling will assure that the flow will mix into these regions, just as at the core inlet. Therefore, continuous flow is assured and no localized precipitation will occur at the spacer grids.

Therefore, for plants that have bypass debris loadings that are within the limits of the debris masses tested are bounded by the test, maintaining coolant flow through the fuel precludes local blockages. This flow also precludes local boric acid precipitation.

The LOCADM methodology considers the deposition of all chemical constituents in the sump fluid onto the fuel rod except the highly soluble chemicals of boron and sodium. The LOCADM methodology includes acceptance criteria specifying that the fuel clad temperatures must stay below 800°F. The ultimate ability of boric acid or sodium borate to insulate the fuel rods is limited since orthoboric acid and sodium borate precipitates have melting points well below 800°F.

ATTACHMENT B

HAND CALCULATION - CALCULATION OF MINIMUM RECIRCULATION DILUTION FLOW

Question: For a hot leg break could localized blockage create a localized isolated region that would be sufficiently isolated so as to create the conditions for localized boric acid precipitation?

For a hot leg break the bulk core conditions would be diluted since SI flow in excess of core boil-off would go through the core and out the break.

Conditions for Boric Acid Concentration to Increase to the Precipitation Point

1. There would need to be boiling in the isolated region since boiling is the process by which water is removed, replaced by a boric acid solution, thus increase the boric acid concentration in the isolated volume.
2. Steam would need to escape the control volume with minimal disruption of control volume isolation.
3. The boil-off makeup flow must flow into the control volume with minimal disruption of control volume isolation.
4. Liquid in isolated volume must be evaporated and replaced 20 times before boric acid solubility limit is reached (i.e. from 2500 ppm to 50,000 ppm)

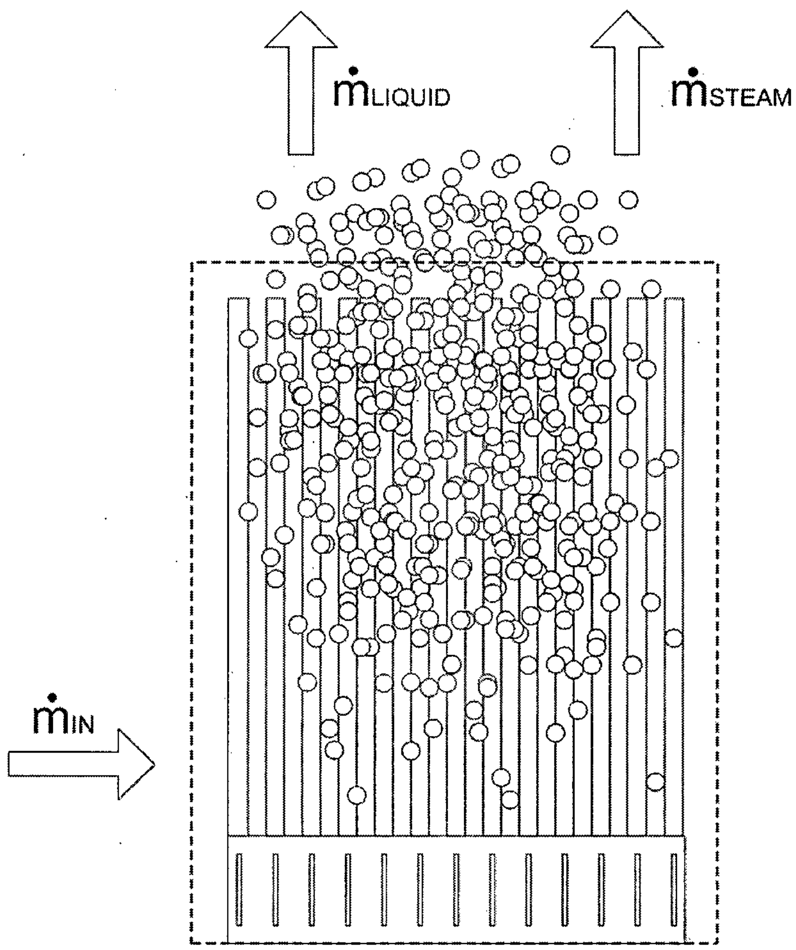
Calculation of the Isolation Efficiency to Permit Boric Acid Concentration to Exceed the Solubility Limit

1. Assume isolation is not perfect (i.e. semi-isolated).
2. Boric Acid Solubility Limit = 29.27 wt. % in control volume.
3. Some liquid in the isolated volume escapes with the steam exiting the isolated region (interfacial drag).
4. Dilution flow into the control volume replaces liquid that goes out of the control volume.
5. Liquid surrounding control volume is at bulk core conditions.

Conclusion: The conditions for isolation sufficient to cause a buildup in boric acid are unlikely. However if a buildup did occur, only a small amount of dilution flow must penetrate the volume to promote or maintain dilution.

Question: If a buildup of boric acid did occur in an isolated volume, how much flow out of (or into) the control volume (as a percentage of the boil-off rate in the semi-isolated volume) is necessary to maintain dilution below the solubility limit?

1. Assume isolation is not perfect (i.e. semi-isolated).
2. Boric Acid Solubility Limit = 29.27 wt. % in control volume.
3. Some liquid in the isolated volume escapes with the steam exiting the isolated region (interfacial drag).
4. Dilution flow into the control volume replaces liquid that goes out of the control volume.
5. Liquid surrounding control volume is at bulk core conditions.



Assumptions;

$\dot{m}_{in}$ = liquid mass entering control volume

$\dot{m}_{steam}$ = boil-off inside the control volume

$\dot{m}_{liquid}$ = liquid leaving the control volume with the steam

$\dot{m}_{liquidW}$ = water in the form of liquid leaving the control volume with the steam

$\dot{m}_{liquidBA}$ = boric acid in the form of liquid leaving the control volume with the steam

Basic boric acid relationships are as follows:

$$\text{Weight Fraction of Boric Acid} = \text{Boron [ppm]} / 174{,}840 \quad \text{(a)}$$

Using conservation of mass:

$$\dot{m}_{in} = \dot{m}_{steam} + \dot{m}_{liquid} \quad (1)$$

$$\dot{m}_{liquid} = \dot{m}_{liquidW} + \dot{m}_{liquidBA} \quad (2)$$

$$\dot{m}_{in} = \dot{m}_{inW} + \dot{m}_{inBA} \quad (3)$$

Dilution will occur at point where the liquid in ($\dot{m}_{in}$) is just greater than that needed to keep the core region at the boric acid solubility limit. The equilibrium point would occur when:

$$\dot{m}_{inBA} = \dot{m}_{liquidBA} \quad (4)$$

Using (1) and (2):

$$\dot{m}_{in} = \dot{m}_{steam} + \dot{m}_{liquidW} + \dot{m}_{liquidBA}$$

Then using Equation (4):

$$\dot{m}_{in} = \dot{m}_{steam} + \dot{m}_{liquidW} + \dot{m}_{inBA} \quad (5)$$

Using the basic relationship (a):

$$\dot{m}_{inBA} = \dot{m}_{in} \times PPM_{CORE} / 174{,}840$$

and using Equation (5):

$$\dot{m}_{in} = \dot{m}_{steam} + \dot{m}_{liquidW} + \dot{m}_{in} \times PPM_{CORE} / 174{,}840 \quad (6)$$

The concentration in the isolated volume will be the boric acid solubility limit, BA_{LIMIT}. Working on $\dot{m}_{liquidW}$:

$$\dot{m}_{liquidW} = \dot{m}_{liquid} \times (1 - BA_{LIMIT})$$

Using (1) to eliminate $\dot{m}_{liquid}$

$$\dot{m}_{liquidW} = (\dot{m}_{in} - \dot{m}_{steam}) \times (1 - BA_{LIMIT}) \quad (7)$$

Putting (7) into (6)

$$\dot{m}_{in} = \dot{m}_{steam} + (\dot{m}_{in} - \dot{m}_{steam}) \times (1 - BA_{LIMIT}) + \dot{m}_{in} \times PPM_{CORE} / 174{,}840$$

Rearranging:

$$\dot{m}_{in} = \dot{m}_{steam} + (\dot{m}_{in} \times (1 - BA_{LIMIT}) - \dot{m}_{steam} + \dot{m}_{steam} \times BA_{LIMIT} + \dot{m}_{in} \times PPM_{CORE} / 174{,}840)$$

Reducing:

$$\dot{m}_{in} \times [(BA_{LIMIT}) - PPM_{CORE} / 174{,}840] = BA_{LIMIT} \times \dot{m}_{steam}$$

And finally:

$$\dot{m}_{in} = \dot{m}_{steam} \times [BA_{LIMIT} / (BA_{LIMIT} - PPM_{CORE} / 174{,}840)]$$

Or:

$$\dot{m}_{in} = R_{dilution\text{-}mass} \times \dot{m}_{steam} \text{ where:} \tag{8}$$

$$R_{dilution\text{-}mass} = \text{Dilution Ratio} = BA_{LIMIT} / (BA_{LIMIT} - PPM_{CORE} / 174{,}840) \tag{9}$$

If one considers liquid volumetric flow out of the semi-isolated volume, the ratio of the volume of liquid that must exit with the steam would be as follows.

$$R_{dilution\text{-}volume} = (R_{dilution\text{-}mass} - 1) \times (\rho_{steam} / \rho_{liquid}) + 1$$

Where:

$\rho_{steam} = 0.037$ lbms/ft^3

$\rho_{liquid} = 59$ lbms/ft^3

At the solubility limit of 29.27 wt.% (i.e. $BA_{LIMIT} = 0.2927$), the values for $R_{dilution\text{-}mass}$ for $R_{dilution\text{-}volume}$ for different core boron concentrations are given in Table 1. Note that for a hot leg break, the core will be continuously diluted and will approximately be at the sump boron concentration.

Table 1

PPM_{CORE}	**Bulk Core (wt.%)**	**$R_{dilution\text{-}mass}$**	**$R_{dilution\text{-}volume}$**
2500	1.43	1.051	1.000042
3000	172	1.062	1.000039
4000	2.29	1.084	1.000053

Conclusion: If even a small amount of liquid mass escapes the semi-isolated volume with the steam (< 10% above the steaming rate inside the semi-isolated volume), the semi-isolated volume will remain below the boric acid solubility limit. On a volume basis, the volume ratio of the liquid to steam exiting the semi-isolated volume to provide dilution is < .005 %.

12) Emergency Operating Procedures typically specify use of hot leg injection at some point in the LOCA recovery period to reverse the core flow and control boron concentration and precipitation. Please identify the time into the accident at which each PWR class of design will employ hot leg injection. Also, discuss the effect of the change in flow distribution on the debris bed that has formed at the core inlet and fuel spacer grids.

RESPONSE:

Action times from analyses of record and identified in plant Emergency Operating Procedures (EOPs) for hot leg injection and/or other actions to prevent boric acid precipitation are discussed in the PWR Owners Group Letter OG-06-200 (Reference1). The action times from Table A-3 of that letter are listed in Table 1 of this RAI response for hot leg injection and/or other actions to prevent boric acid precipitation for all classes of currently operating Pressurized Water Reactors (PWRs).

Subsequent to receipt of this RAI, and in response to comments from the Advisory Committee on Reactor Safeguards (ACRS), the PWR Owners Group initiated prototypical fuel assembly (FA) testing to establish limits on the debris mass (particulate, fibrous and chemical) that could bypass the reactor containment building sump screen and not result in unacceptable head loss that would impede core inlet flow and challenge long-term core cooling of the core. The effects of differing fuel inlet nozzle designs were also considered in the test program. Both AREVA and Westinghouse have performed testing with their respective fuel inlet nozzles. Both vendors tested their various bottom nozzle designs and identified the limiting design (limiting was defined as the design that provided for the maximum pressure drop at the same flow and debris loading conditions). Each fuel bundle tested also had prototypical grids above the bottom nozzle debris capturing design features. Thus, the test data obtained from testing takes into account the fuel inlet nozzle, protective filter design features, and spacer grid designs. Descriptions of the fuel components tested, including bottom nozzles and grids, will be provided in proprietary submittals describing the testing performed and the results obtained.

Testing was performed using bounding debris loads and hot leg break flow rates. These tests demonstrated that for the bounding debris loads tested, the hot-leg flow rate through the fuel assembly mock-up was maintained with acceptable pressure drops. The maintenance of core flushing flow with full 30-day debris loads (particulate, fibrous and chemical) for cold-leg recirculation demonstrates that flow paths through the core remain available. These flow paths would be available from the top of the core to the lower plenum for the lower hot-leg recirculation flow (< 150 gpm).

For a postulated cold-leg break, the core flow rate is determined by core boil-off, which is on the order of 500 gpm or less at the time of initiation of recirculation from the reactor containment building sump and continually decrease as the time passes. Flow in excess of core boil-off spills from the reactor vessel downcomer and out the cold-leg break and into the reactor containment building sump where it is again recirculated through the sump screen. In this scenario, as described in the response to RAI #10, both the debris loads provided to the core and the core flow rates are considerably lower than for a postulated hot-leg break. Thus, similar to the hot-leg break scenario, and considering the lower debris loading for the cold-leg break scenario, flow paths would again be available from the top of the core to the lower plenum for the lower hot-leg recirculation flow (< 150 gpm).

Given the above discussion, the debris collection that has formed at the core inlet and fuel spacer grids is expected to have no effect on the flow distribution in the core during hot-leg recirculation.

It is also noted that the Pressurized Water Reactor Owners Group (PWROG) has a program in place to define, develop and obtain NRC approval of a new Post LOCA-Boric Acid Precipitation Analysis Methodology. That program will utilize the same bounding debris loading determine by the fuel-debris testing, discussed above, that demonstrated that mixing will occur between the core region and the lower plenum.

Reference:

1. PWROG Letter OG-06-200, "Suspension of NRC Approval for Use of Westinghouse Topical Report CENPD-254-P, Post LOCA Long Term Cooling Model, Due to Discovery of Non-Conservative Modeling Assumptions During Calculation Audit, PA-ASC-0290", June 19, 2006. (ML061720175)

Table 1: Time of Initiation of Core Flushing Flow

From Letter OG-06-200 (ADAMS ML061720175)

Group	No. of Plants	Plant Design	Action Time (hrs)[a]
B-1	3	B&W	4.94 – 7.56
B-2	1	B&W	24
C-1	4	CE	3 – 10
C-2	4	CE	3 – 6.5
C-3	2	CE	20
C-4	1	CE	13
C-5	1	CE	6
C-6	1	CE	3
C-7	1	CE	8.5
W-1	24	W	5.5 – 14
W-2	8	W	3 – 6.5
W-3	2	W	3
W-4	2	W	5
W-5	2	W	8
W-6[e]	1	W-UPI	20
W-7	3	W-UPI	NA
W-8	2	W-UPI	14

NOTES TO TABLE 1:

Only those notes applicable to Table 1, above, were taken from Table A-3 of PWROG Letter OG-06-200. There designation associated with the note remains the same as in PWROG Letter OG-06-200.

NA = Not Applicable

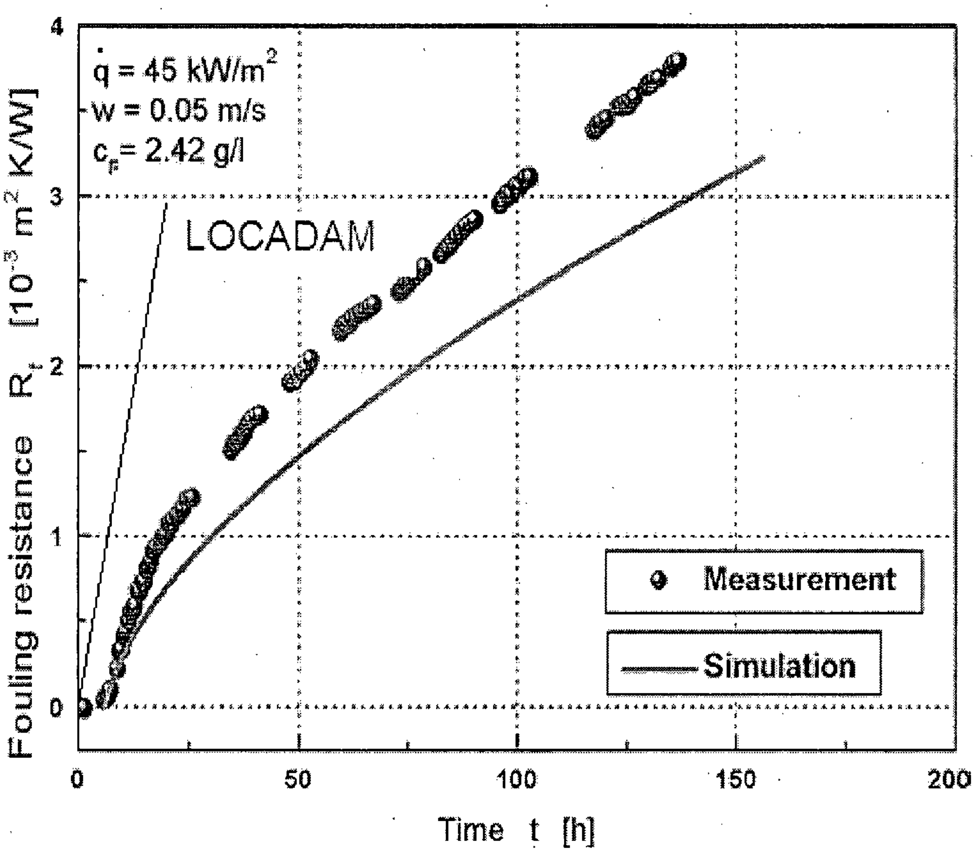
[a] = EOP Action time is latest hot leg switchover time, time to switch to simultaneous injection, or other actions to initiate core dilution.

[e] = For W 2-Loop UPI plants, UPI flow provides flushing flow for cold leg breaks and UPI/core region mixing prevents boric acid precipitation for hot leg breaks

13) Part of the LOCADM Model validation involved benchmarking against an experiment in which calcium sulfate solution entered an electrically heated tube and formed deposits on the heat transfer surface, Brahim et al. The heat fluxes were high enough to cause boiling within the deposits, according to the author's calculations. The LOCADM-predicted deposition rate was calculated to be higher than the deposition rate determined experimentally. Please discuss any additional LOCADM validation that has been performed, or will be performed, involving experimentally determined deposition rates to confirm that the amount of deposit predicted by LOCADM is conservative.

RESPONSE:

As described in WCAP-16793-NP, the LOCADM model conservatively assumes that all fiber and chemical products passed by the sump screen and transported to fuel surfaces by boiling will deposit on the fuel cladding. The assumption that all fiber and chemical products will deposit is a conservative assumption. The amount of conservatism is demonstrated in the plot comparing the deposition predicted by LOCADM over time to the experimental data of Brahim et.al. and is shown in the figure below. Considering the amount of conservatism in the LOCADM predictions to the observed test data, no additional LOCADM validation has been or is planned to be performed.



Fahmi Brahim, Wolfgang Augustin, Matthias Bohnet, "Numerical simulation of the fouling process" International Journal of Thermal Sciences 42 (2003) 323-334

14) For each plant type and configuration, please provide the driving head available for hot-leg and cold-leg break scenarios to push flow into the core across the core inlet.

RESPONSE:

At the time of sump switchover, the core has been fully recovered and the fluid inventory in the Reactor Coolant System (RCS) is above the top of the core. The core decay heat is being removed by Emergency Core Cooling System (ECCS) injection. The driving head at the core inlet is dependent on the break location.

For postulated cold leg pump discharge (CLPD) breaks, the ECCS from each cold leg runs to the break, ensuring that the downcomer is full to at least the bottom of the cold leg nozzles. The core level is established by the manometric balance between the downcomer liquid level, the core level, and RCS pressure drop through the loops. The core flow is only what is required to make up for core boiling to remove the decay heat. For postulated cold leg breaks, most of the ECCS spills directly out of the break. The situation is similar for cold leg pump suction (CLPS) breaks.

For a break in the hot leg, the ECCS must pass through the core to exit the break. The driving force is the manometric balance between the liquid in the downcomer and core. Should a debris bed begin to build up in the core, the liquid level will begin to build in the cold legs and into the Steam Generators (SGs). As the level begins to rise in the SG tubes, the elevation head to drive the flow through the core increases as well. The driving head reaches its peak when the shortest SG tube has been filled (in the W and CE plant designs) or the SG and hot legs to the spillover elevation have filled (in the B&W plant design) and the flow begins to spill over. Once the ECCS flow reaches the elevation of the shortest tubes, the spillover flow is sufficiently large that no increase in water level to the higher tubes is achieved. This is conservative as it provides for the minimum static head available in the steam generator tubes. The core mixture level will be at least to the hot leg nozzle elevation, and the core flow rate will equal the ECCS flow rate.

Core flow is only possible if the manometric balance between the downcomer side and the core is sufficient to overcome the flow losses in the Reactor Vessel (RV) downcomer, RV lower plenum, core, and loops at the appropriate flow rate.

$$\Delta P_{avail} = \Delta P_{dz} - \Delta P_{flow}$$

where:

ΔP_{avail} = Available head to drive flow into the core

ΔP_{dz} = Elevation head between downcomer side and core

ΔP_{flow} = Flow losses in the RV downcomer, RV lower plenum, core and loops (W & CE designs) or reactor vessel vent valves (B&W designs)

The manometric differences are determined considering plant geometry and core void fractions. The flow losses are calculated using the Darcy equation.

For a postulated hot leg break, the value of ΔP_{dz} is evaluated by taking elevation difference between the elevation of the shortest tube in the steam generator (W and CE plant designs) or the hot leg spillover elevation (B&W plant designs) and the elevation the bottom of the inside diameter of the hot leg. For a postulated hot leg break, no voiding in the coolant passing through the core and out the hot leg is assumed at time of initiation of recirculation from the reactor containment building sump. This provides for the evaluation of a minimum driving head for the fleet of PWRs of approximately 13 psid.

The temperature of the SG secondary side inventory may be above the saturation temperature associated with the containment pressure, which could cause boiling the SG and a reduction in the driving head of the water column. The 13 psid value does not account for either density variations in the column that may result due to heating of the water in the steam generator tubes by the warmer SG secondary side inventory, or the pressure increase on the cold leg water column that would result from the generation and venting of steam due to heat transfer from the warmer SG inventory to the coolant in the SG tubes.

The effect of effect of density due to heating of the coolant in the SG tubes on available head is bounded by comparing the density of water at saturated condition to the density of subcooled water. This approach conservatively uses extreme values and ignores density gradients that would be present in the actual system which, if considered, would provide less extreme values. At 1200 seconds, most reactor containment buildings are at a pressure of about 40 psia or less. The sump fluid temperature is at or near saturated conditions; this is about 267°F for 40 psia. Allowing for a conservatively large amount of cooling of the recirculating fluid by heat exchangers in the Emergency Core Cooling (ECC) or Containment Spray (CS) lines, an ECC fluid temperature of 200°F is used for this evaluation. Comparing the density differences at 40 psia for saturated conditions and 200°F water,

$$\Delta H = \frac{58.31 \, {}^{lb_m}\!/_{ft^3}}{60.11 \, {}^{lb_m}\!/_{ft^3}} = 0.97$$

From the equation above, there would be no more than a 3% reduction in available head due to the heating of the water column in the SGs. Using this conservative approach would reduce the approximately 13 psid acceptance criteria to approximately 12.6 psid. It is also observed that, as the containment tends to cool, the difference between the saturation density and the subcooled density (assuming a 60°F temperature difference) decreases. Thus, this density effect diminishes as the containment pressure continues to decrease following the postulated accident. Further, the conservatisms included in this calculation will overcome this small potential variation in results such that it can be ignored. One such conservatism is that the whole water column is uniformly heated to a temperature of 267°F.

Per the Darcy equation, as flow rate increases, the pressure drop will increase. The highest flow rate through the core is achieved with a hot leg break and the minimum driving head for operating plants is approximately 13 psid. As the flow decreases, the pressure drop decreases. The available cold leg driving head for operating plants is approximately 3 psid. Testing for the PWROG used these values as guidelines for the test design.

Specific plant values for the acceptable minimum driving head for operating plants will be included in proprietary data reports that are being prepared for the AREVA and Westinghouse testing.

15) The LOCA-DM program used to calculate deposition in the reactor core following a LOCA assumes that species present in the coolant deposit as the water boils away at the hot surface of the fuel cladding. Please discuss how deposition of material with retrograde solubility (e.g., calcium scales) is adequately accounted for by the LOCA-DM model. If applicable, provide results from any experiments (not previously discussed in WCAP-16793-NP) that show deposits predicted by LOCA-DM account for deposition from any mechanism, including retrograde solubility, in a conservative manner.

RESPONSE:

LOCADM looks at two categories of deposit formation on fuel cladding surfaces.

Category 1 deposition is driven by the boiling process. Evaporation at the fuel rod surface draws coolant through a preexisting crud deposit to the surface of the fuel rod. The amount of coolant drawn to the surface is determined by the boiling rate which is determined by a number of factors such as the decay heat flux, the deposit thickness and thermal conductivity, and bulk temperature of the coolant at the location being modeled. LOCADM assumes that any dissolved debris material that is drawn to the cladding surface will precipitate as a solid. The exact mechanism is not addressed since it is conservatively assumed that all of the dissolved debris material deposits. By considering all dissolved material to deposit due to the boiling process conservatively accounts for retrograde solubility.

Category 2 deposition includes all other mechanism by which dissolved material or chemical precipitates could be deposited on the fuel after transport by convection or diffusion. Due to the high level of uncertainties in flow, chemistry, and temperature gradients within the core, an empirical correlation was used to predict Category 2 deposition. This correlation was developed from operational core deposition data during normal operation and included factors for heat flux and impurity concentrations. Like the boiling case, retrograde solubility calculations are not done explicitly, but are included in the empirical correlation.

A sensitivity study was done examining the effect of increasing the Category 2 deposition above the level specified by the empirical correlation. It was discovered that increasing the Category 2 deposition to simulate an increased level of deposition due to retrograde solubility actually decreases the peak deposit thickness and cladding temperature. This occurs because Category 2 deposition mirrors the power profile during normal operation, and deposits are spread over a broad area of the core. This reduces solution concentrations and deposit buildup in the more localized boiling deposits is reduced.

In summary, LOCADM deals with Category 1 deposition in a conservative manner by assuming that all dissolved debris and chemical product are deposited. It deals with Category 2 deposition in a representative manner, and increasing the amount of deposition from retrograde solubility would actually be non-conservative since it would spread deposits out making them thinner and reducing fuel cladding surface temperatures.

No additional experimental data not previously discussed in WCAP-16793-NP are available.