Westinghouse Non-Proprietary Class 3

WCAP-16793-NP-A
Revision 2

July 2013

Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid

Westinghouse

WCAP-16793-NP-A
Revision 2

Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid

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July 2013

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This work performed under PWROG Project Number PA-SEE-0312.

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April 8, 2013

Mr. W. Anthony Nowinowski, Program Manager
PWR Owners Group, Program Management Office
Westinghouse Electric Company
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SUBJECT: FINAL SAFETY EVALUATION FOR PRESSURIZED WATER REACTOR OWNERS GROUP TOPICAL REPORT WCAP-16793-NP, REVISION 2, "EVALUATION OF LONG-TERM COOLING CONSIDERING PARTICULATE FIBROUS AND CHEMICAL DEBRIS IN THE RECIRCULATING FLUID" (TAC NO. ME1234)

Dear Mr. Nowinowski:

By letter dated October 12, 2011, the Pressurized Water Reactor Owners Group (PWROG) submitted Topical Report (TR) WCAP-16793-NP, Revision 2, "Evaluation of Long-Term Cooling Considering Particulate Fibrous and Chemical Debris in the Recirculating Fluid" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11292A020), to the U.S. Nuclear Regulatory Commission (NRC) staff for review. By letter dated January 29, 2013, an NRC draft safety evaluation (SE) regarding our approval of TR WCAP-16793-NP, Revision 2, was provided for your review and comment (ADAMS Accession No. ML12115A304). By letter dated March 6, 2013 (ADAMS Accession No. ML13093A082), the PWROG commented on the draft SE. The NRC staff's disposition of the PWROG comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The NRC staff has found that TR WCAP-16793-NP, Revision 2, is acceptable for referencing in licensing applications to the extent specified and under the limitations and conditions delineated in the TR and in the enclosed final SE. The final SE defines the basis for acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that the PWROG publish accepted proprietary and non-proprietary versions of this TR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include a "-A" (designating accepted) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, the PWROG and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

Sher Bahadur, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure:
Final SE

cc w/encl: See next page

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Project No. 694

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FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT WCAP-16793-NP, REVISION 2

"EVALUATION OF LONG-TERM COOLING CONSIDERING PARTICULATE, FIBROUS AND CHEMICAL DEBRIS IN THE RECIRCULATING FLUID"

PRESSURIZED WATER REACTOR OWNERS GROUP

PROJECT NO. 694

1.0 INTRODUCTION AND BACKGROUND

As a consequence of the U.S. Nuclear Regulatory Commission's (NRC's) evaluation of Generic Safety Issue 191 (GSI-191), "Assessment of Debris Accumulation on Pressurized Water Reactor (PWR) Sump Performance," in September 2004 the NRC issued Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors." GL 2004-02 requested that holders of operating licenses for PWRs perform evaluations of the emergency core cooling system (ECCS) and the containment spray system (CSS) to assess the potential for debris entrained in the circulated containment pool, following a loss-of-coolant accident (LOCA), to block restrictions within the ECCS recirculation flow path, including blockage within the reactor fuel assemblies.

In December 2004, the Nuclear Energy Institute (NEI) published NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML050550138), providing a method for licensees to resolve some aspects of GL 2004-02. The NRC staff safety evaluation (SE) of NEI 04-07 (ADAMS Accession No. ML050550156) found that additional guidance was needed in the area of blockage in the reactor vessel in order to adequately address the downstream effects of debris that passes through the ECCS sump strainer(s).

In response to the SE conclusions on NEI 04-07, the Pressurized Water Reactor (PWR) Owners Group (PWROG) sponsored development of Topical Report (TR) WCAP-16793-NP, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," Revision 0 (Reference 1). Reference 1 evaluated the effects of debris and chemical precipitates on core cooling for PWRs when the ECCS is aligned to the containment sump. The objective was to demonstrate that following a LOCA, long-term core cooling (LTCC) can be maintained to satisfy the requirements of Title 10 of the *Code of Federal Regulations* Section 50.46 (10 CFR 50.46). The TR intended to provide reasonable assurance that debris in the circulated containment pool will not prevent adequate cooling of the core.

The NRC staff reviewed TR WCAP-16793-NP, Revision 0, and issued requests for information (RAIs) in letters dated September 10 and September 20, 2007 (References 2 and 3, respectively). The responses to these RAIs are contained in a letter from the PWROG to NRC dated January 17, 2008 (Reference 4). On August 22, 2008, the NRC staff issued an additional set of RAIs (Reference 5).

ENCLOSURE

In response to the RAIs, contained in Reference 5, the PWROG undertook a test program to examine the synergistic effects of chemical precipitates, fiber, microporous insulation, and particulate on the flow of coolant through the core. In April 2009, the PWROG submitted Revision 1 to TR WCAP-16793-NP (Reference 6) incorporating limits on the amounts of debris that could be ingested in the reactor vessel such that adequate flow to the core can be maintained to accomplish LTCC. Reference 6 referenced two proprietary test reports that support the debris limits.

The debris limits in TR WCAP 16793-NP, Revision 1 (Reference 6), were developed through flow testing performed on mock AREVA NP, Inc. (AREVA) and Westinghouse Electric Company (Westinghouse) fuel assemblies at two separate test facilities. Westinghouse-designed fuel was tested at the Westinghouse Science and Technology Center and AREVA-designed fuel was tested at Continuum Dynamics Inc. (CDI). The mock fuel assemblies were approximately one-third the height of actual fuel assemblies and typically contained four to five intermediate grid straps. The fuel assembly test protocol and test results are described in proprietary reports WCAP-17057-P, Revision 0 (Reference 7), and EIR 51-9102685-000 (Reference 8) for Westinghouse and AREVA fuel designs, respectively. These proprietary reports were not part of TR WCAP-16793-NP, Revision 1. The NRC staff reviewed TR WCAP-16793-NP, Revision 1, and the proprietary test reports and issued RAIs in letters dated January 8 and 15, 2010 (References 9 and 10, respectively). The responses to these RAIs are contained in References 11, 12, and 13. Additional information on "mixed cores" (i.e., cores containing fuel from more than one vendor) was provided by the PWROG in Reference 14.

During the review processes, NRC staff noted that for some test conditions, there was an order of magnitude difference in the debris acceptance limit between the fuel supplied by Westinghouse and the fuel supplied by AREVA. In a May 25, 2010, Category 1 public meeting with the PWROG to discuss fuel assembly testing, NRC staff requested that the PWROG test each vendor's fuel assembly in the other vendor's test facility (cross-test) to confirm that the differences in behavior of the two fuel designs are due to differences in fuel design and not due to differences in the test facilities. The meeting summary can be found in Reference 15.

In response to the NRC request, the PWROG initiated several additional tests to address NRC staff's concerns. The additional testing, including a two-way cross-test, showed that the results obtained at one test facility, using one vendor's fuel assembly, could not be duplicated at the other test facility within the accepted tolerance limits.

As a result of the additional testing, the lower fiber acceptance limit of the two fuels was established as the acceptance limit for the fuel designed by both Westinghouse and AREVA. Westinghouse revised WCAP-17057-P, and issued it as Revision 1 (Reference 16) and AREVA issued proprietary test report EIR 51-9170258-000 (Reference 17). Also, the PWROG issued TR WCAP-16793-NP, Revision 2 (Reference 18), herein referred to as the "WCAP," to reflect the results of the additional testing and the basis for the (now lower) fiber acceptance limit.

The PWROG submitted two supplements to WCAP-16793-NP, Revision 2, as follows: "*Proposed Supplement to WCAP-16793-NP, Revision 2,*" PA-SEE-0312, Revision 4, dated May 25, 2012 (Reference 42) and "*Supplement to WCAP-16793-NP, Revision 2*", PA-SEE-0312, Revision 4, dated July 20, 2012 (Reference 43).

The following sections of this SE describe the NRC staff's review of TR WCAP-16793-NP, Revision 2 (Reference 18), the proprietary test reports WCAP-17057-P, Revision 1 (Reference 16), EIR 51-9102685-000 (Reference 8), and EIR 51-9170258-000 (Reference 17),

the RAI responses in References 11, 12, and 13, and the information in Reference 14 on mixed-fuel cores.

The general format followed in this SE, beginning with Section 3.1.3 (Executive Summary), is to first describe the TR WCAP-16793-NP evaluations and conclusions, by section or subsection, then to immediately follow with the NRC staff's evaluation of that section or subsection.

The NRC staff also reviewed the information provided in the proposed and final supplements to the WCAP (References 42 and 43) and found that there was no additional information provided that would alter the conclusions reached in this SE. Therefore, this SE does not contain detailed staff comments on the supplements contents.

Notes:

1. NRC staff has not issued a final SE for TR WCAP-16793-NP, Revision 0 or Revision 1 (References 1 and 6, respectively), because TR WCAP-16793-NP, Revision 2, supersedes the earlier versions.

2. Beginning with Section 3.2, the Subsection numbering system used in the Technical Evaluation section (Section 3) parallels the section numbering in TR WCAP-16793-NP, Revision 2 (Reference 18, hereafter referred to as the WCAP) (i.e., Section 3.2 of the SE corresponds to Section 2 of the WCAP, Section 3.3 of the SE corresponds to Section 3 of the WCAP, and so on).

2.0 REGULATORY EVALUATION

GL 2004-02 requested that holders of operating licenses for PWRs perform evaluations of the ECCS and the containment spray recirculation functions. These evaluations are to include the potential for debris blockage at flow restrictions within the ECCS recirculation flow path downstream of the sump strainer, including potential blockage at fuel assembly inlet debris strainers. Other potential flow restrictions are the spacer grids within the fuel assemblies. Debris blockage at such flow restrictions has the potential to impede or prevent the flow of coolant to the reactor core, potentially leading to inadequate LTCC.

The acceptance criteria for the performance of a nuclear reactor core following a LOCA are found in 10 CFR 50.46. The acceptance criterion dealing with the long-term cooling phase of the accident recovery is in 10 CFR 50.46(b)(5), which reads as follows:

> *Long-term cooling:* After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

At the request of the PWROG (Reference 19), the NRC staff provided guidance on (1) acceptance criteria for LTCC once the core has quenched and re-flooded, and (2) the mission time that should be used in evaluating debris ingestion effects on the reactor fuel. The NRC staff provided these clarifications in a letter dated August 16, 2006 (Reference 20). To summarize, long-term cooling capability must be provided despite potential challenges from

chemical effects (boron precipitation)[1] or physical effects (debris), as demonstrated by no significant increase in calculated peak cladding temperature (PCT). After quench and re-flood, moderate increases in cladding temperature (on the order of 100 to 200 degrees Celsius) could be acceptable, if appropriately justified. In addition, adequate core cooling performance during the ECCS mission time is demonstrated when bulk and local temperatures are shown to be stable or continuously decreasing with the additional assurance that any debris entrained in the cooling water supply would not be capable of affecting the stable heat removal mechanism due to sump strainer clogging or downstream effects.

3.0 TECHNICAL EVALUATION

3.1 GENERAL

3.1.1 ECCS Description

Following a large LOCA, the CSS is actuated to suppress containment building pressure and the ECCS is actuated to cover the core and remove decay heat. (Note: Some plants may not initiate CSS). Initially, the source for this water is from stored locations, e.g., the refueling-water storage tank (RWST) at Westinghouse PWRs, the refueling water tank (RWT) at CE PWRs, or the borated water storage tank (BWST) at B&W PWRs. Water that is pumped into the reactor vessel is eventually discharged through the break into containment where it collects on the containment building basement floor and in the ECCS sump(s). As the stored water supply is exhausted, the CSS and ECCS are realigned to draw coolant from the containment sump. The coolant discharged from the reactor coolant system (RCS) and from the CSS is then circulated back into the RCS to provide for continued LTCC without the need for additional cooling water. There are two separate categories for LOCA break location depending on whether the break is upstream or downstream of the core (cold-leg side or hot-leg side), and two locations of initial core injection (cold-leg or downcomer, or upper plenum) depending on whether the plant design is a 2-loop Westinghouse upper plenum injection (UPI) plant, a B&W plant, a CE plant, or a 3-loop or 4-loop Westinghouse plant. The quantity of debris carried into the core, the quantity of debris deposited on fuel cladding surfaces and the head available to drive coolant into the core are greatly dependent upon the location of the pipe break. The effect of the different break locations on CE plants, Westinghouse 3-loop and 4-loop plants, Westinghouse 2-loop plants, and B&W plants is discussed below.

a. CE, Westinghouse 3-loop and 4-loop, and B&W plants

During a LOCA in a CE plant or a Westinghouse 3-loop or 4-loop plant, the ECCS pumps are aligned to inject stored borated water into the RCS cold-legs. In B&W plants, water is injected directly into the reactor vessel downcomer through nozzles located on the reactor vessel. These injection points are the same for all RCS break locations. When the stored water is nearly depleted, the ECCS pumps are realigned to take suction from the containment sump for circulation of coolant for an indefinite period of time, thus providing LTCC.

In the event of a hot-leg break, the coolant pumped into the cold-leg is forced into the reactor pressure vessel (RPV), down the downcomer and up through the reactor core toward the break. During the LTCC period, core flow, plus a small amount of core bypass flow, is equal to the total

[1] Section 8 of WCAP-16793-NP, Revision 2, states that the effects of boron precipitation on LTCC are being addressed by the PWROG in a separate program. Refer to section 3.8 of this SE for a description of the program and the NRC staff evaluation.

ECCS flow delivered to the cold-leg or downcomer. During LTCC, flow through the core may vary, depending on the system and number of pumps operating. The bulk of the testing for the hot-leg break scenario was based on the maximum expected ECCS flow rate (44 gallons per minute (gpm) per fuel assembly) because this flow rate was shown through testing at lower flow rates (e.g., single train operation or high-pressure safety injection) to result in the greatest pressure drop across the test assembly.

In the event of a cold-leg break at CE and Westinghouse reactors, and a cold-leg or injection line break in B&W reactors, ECCS coolant injected into the failed loop/pipe will exit the break and coolant injected into the intact loop/pipe will enter the downcomer annulus, ensuring that the downcomer is filled, at minimum, to the bottom of the cold-leg nozzle. During a cold-leg break, once the core has been re-covered, the flow of coolant entering the core is only that required to replenish boil-off (less than 3 gpm per fuel assembly). The excess coolant flows around the downcomer annulus and exits the RPV through the failed pipe. Therefore, the LTCC period following a cold-leg or injection line break represents a minimum core flow condition. Debris build-up at the core inlet and in the fuel assemblies under these conditions could affect heat transfer from the fuel cladding and could add to the resistance in the core inlet that must be overcome to drive adequate coolant into the core. This minimum flow condition is also used in the LTCC evaluations because it represents the lowest available head to drive coolant into the core.

b. Westinghouse 2-loop Upper-Plenum Injection (UPI) Plants

For Westinghouse 2-loop UPI plants, initial ECCS flow to the RPV from the stored borated water source is through the cold-legs and nozzles on the RPV located above the fuel (upper plenum). At the time of ECCS pump suction realignment to the ECCS sump (sump switch-over), flow to the cold-legs is secured and ECCS flow to the upper plenum is maintained. Therefore, during the LTCC period, the direction of ECCS coolant flow through the core is reverse of that for CE, B&W, and Westinghouse 3-loop and 4-loop plants.

For a hot-leg break or UPI line break scenario, the bulk of the flow into the reactor upper plenum during sump circulation flows out the failed hot-leg or UPI nozzle, carrying the bulk of the suspended debris out of the vessel. Water level is maintained at the break elevation, but not below the hot-leg or injection nozzle bottom elevation. The flow into the core is by gravitational force and is only that needed to replenish the coolant boiled away. Excess ECCS flow is discharged through the break. Therefore, the long-term core-cooling period following a hot-leg/UPI line break represents a minimum core flow condition for a UPI plant. Without a net flow through the core, boiling in the core will continue, causing debris and chemicals to concentrate. This minimum flow condition is used in the LTCC evaluations because it represents the lowest head available to drive coolant into the core.

For a cold-leg break scenario, ECCS flow delivered to the upper plenum flows through the core and out the break. Therefore, core flow, plus a small amount of core bypass flow, is equal to the total ECCS flow. During LTCC, flow through the core may vary, depending on the number of pumps operating. However, the maximum pressure drop is expected to occur at the maximum flow because the debris load would be the greatest. Therefore, maximum ECCS flow rates were used in the LTCC evaluations.

3.1.2 LTCC Considerations

At a meeting with the PWROG and Westinghouse on February 7, 2007, and at meetings of the Advisory Committee on Reactor Safeguards (ACRS) Thermal-Hydraulics subcommittee on May 15, 2007, and March 19, 2008, the NRC staff, the PWROG, and Westinghouse developed a list of issues that should be considered in the resolution of GSI-191 with regard to reactor core blockage. The WCAP addresses the identified issues by providing evaluations and conclusions as described below. This SE discusses the acceptability of each of these evaluations and conclusions in the SE sections noted.

a. Adequate flow to remove decay heat will continue to reach the core when the debris limits set in Section 10 of the WCAP are met. (WCAP Section 3, Appendix B, and Appendix C; SE Section 3.3.

b. Decay heat will continue to be removed even with debris collection at the fuel assembly spacer grids. (WCAP Section 4, Appendix C, and Appendix D; SE Section 3.4)

c. Fibrous debris entering the core region will not tightly adhere to the surface of fuel cladding. (WCAP Section 5; SE Section 3.5)

d. Using an extension of the chemical effects source-term method developed in TR WCAP-16530-NP-A (Reference 21), and fuel clad deposition model developed in the WCAP, a sample calculation using large debris loadings of fiberglass and calcium silicate was performed. The calculation showed a maximum deposition thickness and peak fuel cladding temperature well within the acceptance limits. (WCAP Section 7, Appendix E, and Appendix F; SE Section 3.7)

e. The three categories of protective coatings used inside reactor containment buildings have been evaluated to have negligible effect on the generation of precipitate. (WCAP Section 6; SE Section 3.6)

f. The PWROG has undertaken a program, outside of GSI-191[2], to address boric acid precipitation. The PWROG stated that, when complete, the program will examine the effects of concentration of boric acid and debris in the core. These evaluations will include an assessment of the effects of debris ingested in the reactor vessel on boron precipitation. Therefore, the evaluation of the effects of concentrating boric acid and debris contained in the circulated post-LOCA containment pool are outside the scope of the WCAP. (WCAP Section 8; SE Section 3.8) Although the WCAP does not address the effects of boric acid precipitation on LTCC, SE Section 3.8 includes a discussion of the staff's position on this issue.

g. The PWROG has addressed fuel cladding embrittlement by setting 800 degrees Fahrenheit as the maximum acceptable cladding temperature after the initial quench of the core. The 800 degrees Fahrenheit temperature was selected based on autoclave data that demonstrated oxidation and hydrogen pickup to be acceptable and not cause the fuel cladding to become brittle at temperatures up to the 800 degrees Fahrenheit limit. (WCAP Section 2 and Appendix A; SE Section 3.2)

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[2] The PWROG has undertaken a program to reexamine its methods for evaluating boron precipitation in the reactor vessel in response to NRC findings made during a technical audit of Westinghouse Topical Report CENPD-254-P (Reference 41).

h. The PWROG investigated the potential for chemical precipitates to increase the head loss through any fiber bed that may form at the core inlet and at the spacer grids by conducting flow testing on an approximately one-third height, 17 x 17 fuel assembly using fibrous, particulate, and chemical debris. The debris acceptance limits established in the WCAP are based on these tests. (WCAP Sections 3 and 4; SE Sections 3.3 and 3.4)

i. The PWROG investigated the potential for blockage in the reactor pressure vessel (RPV) when coolant is delivered to the top of the core. The WCAP concludes that the debris limit specified in Section 10 of the WCAP is also applicable to UPI plants. (WCAP Section 9 and Appendix G; SE Sections 3.9)

3.1.3 Executive Summary

The Executive Summary of the WCAP summarizes the significant in-vessel downstream effects phenomena, defines the acceptance criteria for successful core cooling associated with these phenomena, and specifies actions required by licensees to demonstrate adequate LTCC associated with these phenomena as follows:

a. The WCAP evaluations cover the following topical areas associated with the in-vessel downstream effects phenomena[3]:

   1. Blockage at the core inlet (WCAP Section 3),
   2. Collection of debris on fuel grids (WCAP Section 4),
   3. Collection of fibrous material on fuel cladding (WCAP Section 5),
   4. Protective coating debris deposited on fuel clad surfaces (WCAP Section 6),
   5. Chemical precipitates and debris deposited on fuel clad surfaces (WCAP Section 7), and
   6. Coolant delivered to the top of the core (WCAP Section 9).

b. The acceptance bases for the evaluation of the topical areas identified above are:

   1. The maximum clad temperature shall not exceed 800 degrees Fahrenheit.
   2. The thickness of the cladding oxide and the deposits of material on the fuel shall not exceed 0.050 inches in any fuel region.

   The cladding temperature and total deposit thickness acceptance bases are applied after the initial quench of the core and are consistent with the LTCC requirements stated in 10 CFR 50.46 (b)(4) and 10 CFR 50.46 (b)(5). The PWROG stated that the acceptance bases do not represent, nor are they intended to be, new or additional LTCC requirements. These acceptance bases allow demonstration that local temperatures in the core are stable or continuously decreasing and that debris entrained in the cooling water supply will not affect decay heat removal. The 800 degrees Fahrenheit temperature was selected based on autoclave data that demonstrated oxidation and hydrogen pickup to be acceptable at and below 800 degrees Fahrenheit. A discussion of the technical basis for the 800 degrees Fahrenheit temperature is given in Appendix A of the WCAP. The 0.050-inch limit for oxide plus deposits was selected to preclude the

[3] The effects of boron precipitation on long-term core cooling are being addressed by the PWROG in a separate program as outlined in Section 8 of the WCAP. See the NRC staff evaluation of Section 8 (SE Section 3.8) for the NRC staff position on this issue.

formation of deposits that would bridge the space between adjacent rods and block flow between fuel channels.

c. Utilities must evaluate site-specific fiber loading against the debris load acceptance criteria provided in the WCAP. The WCAP also states that plants with debris loads above the debris load acceptance criteria may demonstrate adequate LTCC capability through engineering evaluations of plant-specific conditions and/or plant-specific testing.

d. The WCAP states that in order to demonstrate reasonable assurance of LTCC, all plants must evaluate the areas identified [in paragraph (a)] above, demonstrate they are bounded by the maximum fuel cladding temperature and maximum deposit thickness requirements [paragraph (b) above] and evaluate the site-specific fiber loading against the developed debris load acceptance criteria. The WCAP states that:

1. Plants that follow the guidance provided in Section 10 of the WCAP can state that debris that bypasses the strainer will not build an impenetrable blockage at the core inlet. While any debris that collects at the core inlet will provide some resistance to flow, in the extreme case that a large blockage does occur, numerical analyses have demonstrated that core decay heat removal will continue.

2. Decay heat will continue to be removed even with debris collection at the fuel assembly (FA) spacer grids. Plants that follow the guidance provided in Section 10 can state that debris that bypasses the screen will not build an impenetrable blockage at the fuel spacer grids. This assertion is bolstered by numerical and first principle analyses.

3. Fibrous debris, should it enter the core region, will not tightly adhere to the surface of fuel cladding. Thus, fibrous debris will not form a "blanket" on clad surfaces to restrict heat transfer and cause an increase in clad temperature. Therefore, adherence of fibrous debris to the cladding is not plausible and will not adversely affect core cooling.

4. Protective coating debris, should it enter the core region, will not restrict heat transfer and cause an increase in clad temperature. Therefore, adherence of protective coating debris to the cladding is not plausible and will not adversely affect core cooling.

5. The chemical effects method developed in TR WCAP-16530-NP-A (Reference 21) was extended to develop a method to predict chemical deposition of fuel cladding. The calculation tool, LOCADM, can be used by each utility to perform a plant-specific evaluation. It is expected that each plant will be able to use this tool to show that decay heat would be removed and acceptable fuel clad temperatures would be maintained.

6. PWRs use boron as a core reactivity control method and are subject to concerns regarding potential post-LOCA boric acid precipitation in the core. In light of NRC staff and ACRS challenges to the simplified methods commonly used, it has recently become clear that additional insights and new methodologies are needed to answer fundamental questions about boric acid mixing and transport in the RCS and potential precipitation mechanisms that may occur both during the ECCS injection

phase and the sump recirculation phase after a LOCA. This will be addressed in a separate PWROG program.

7. The PWROG FA test results demonstrated that sufficient flow will reach the core to remove core decay heat. The debris load acceptance criteria developed is bounding and applicable to all PWR plants, including UPI plants.

e. Utilities are required to demonstrate acceptable LTCC with debris and chemical products present in the circulating fluid. Licensees will have to perform plant-specific LOCADM evaluations (Section 7 and Appendix E of Reference 18) and confirm that plant-specific conditions are bounded by the debris load acceptance criteria (Sections 3 and 10, and Appendix G of Reference 18). Plants with debris loads above the debris load acceptance criteria may demonstrate adequate LTCC capability through engineering evaluations of plant-specific conditions and/or plant-specific testing.

The WCAP states that these actions, along with reference to the WCAP, provide the basis for demonstrating that LTCC will not be compromised following a LOCA as a consequence of debris ingestion to the RCS and core.

NRC Staff Evaluation of Executive Summary Statements

a) The NRC staff reviewed the scope of the in-vessel downstream evaluation as described in the executive summary of the WCAP and found it adequate because it conforms to the list of agreed upon issues described in Section 3.1.2 of this SE. Further, NRC staff finds the position stated in the WCAP that post-LOCA boric acid precipitation analysis scenarios, assumptions and acceptance criteria and resultant methodologies that demonstrate adequate post-LOCA LTCC can be addressed in a separate PWROG program acceptable if debris limits approved by the staff in this SE are not exceeded. Larger debris loads require the potential for boric acid precipitation to be addressed in conjunction with the resolution of in-vessel downstream effects.

b) The NRC staff reviewed the WCAP statement that the cladding temperature and total deposit thickness acceptance bases are to be applied after the initial quench of the core because the statement is consistent with the requirements stated in 10 CFR 50.46(b)(4) and 10 CFR 50.46(b)(5). Also, the acceptance basis for the 800 degrees Fahrenheit temperature limit after re-flood and the 0.050 inch limit for oxide plus deposits are acceptable to NRC staff as discussed in detail in Section 3.7 of this SE.

c) The NRC staff reviewed the WCAP statement that utilities must evaluate site-specific fiber loading against the debris load acceptance criteria provided in the WCAP. The NRC staff finds this acceptable. The acceptance criteria stated in Section 10 of the WCAP have been demonstrated, through extensive testing, to allow adequate flow into the core. Also, the NRC staff finds that plants with debris loads above the debris load acceptance criteria may perform engineering evaluations and/or tests of plant-specific conditions to demonstrate adequate LTCC[4] capability. However, any tests or evaluations outside the parameters and

[4] In the context of GSI-191 in-vessel downstream effects evaluations, the WCAP references to LTCC generally refer to the capability to maintain adequate core flow in the presence of debris and the absence of deposits on fuel rods and in grid straps that would result in fuel clad temperatures exceeding 800 °F. The staff considers LTCC to include all phenomena needed to satisfy the requirements of 10 CFR

acceptance criteria stated in the WCAP and this SE shall be submitted to the NRC for review and approval. *This Condition is addressed further in Section 4.0, Item number 1, of this SE.*

d) The NRC staff reviewed the WCAP statement that in order to demonstrate reasonable assurance of LTCC, all plants must evaluate the areas identified in paragraph 3.1.3(a) above and demonstrate they are bounded by the maximum fuel cladding temperature requirement of 800 degrees Fahrenheit and maximum deposit thickness requirement of 0.050 inches in any fuel region. The 800 degrees Fahrenheit temperature is acceptable based on autoclave data that demonstrated oxidation and hydrogen pickup to be acceptable and not cause the cladding to become brittle at temperatures up to 800 degrees Fahrenheit. A discussion of the technical basis for the 800 degrees Fahrenheit temperature is given in Appendix A of the WCAP. The 0.050-inch limit for oxide plus deposits was selected to preclude the formation of deposits that would bridge the space between adjacent rods and block flow between fuel channels.

   Regarding item d.2 of paragraph 3.1.3, the NRC staff finds that, for plants that operate within the debris loads identified in Section 10 of the WCAP, debris that bypasses the screen will not build impenetrable blockage at the spacer grids. This had been demonstrated by the fuel testing. However, the NRC staff did not rely on the numerical and first principle analysis that supported this conclusion because the testing did not support the orifice behavior that was modeled in the analyses. This is discussed further in Section 3.3 of this SE.

e) The NRC staff finds the description of actions that are required of utilities to demonstrate acceptable LTCC with debris and chemical products present in the circulating fluid acceptable because it calls for licensees to perform plant-specific evaluations to demonstrate that they satisfy the debris limits, debris and oxide deposition limits and cladding temperature limits of the WCAP, as qualified by the Limits and Conditions stated in this SE. Further, licensees should confirm that their plant key evaluation inputs (ECCS flow rates, driving heads, etc.) are bounded by the values used in the WCAP, proprietary test reports (References 8, 16, and 17), and RAI responses (References 11 and 12). *This condition is addressed further in Section 4, Item number 1, of this SE.*

3.2 LONG-TERM CORE COOLING ACCEPTANCE BASIS *(WCAP-16793-NP, Revision 2, Section 2 and Appendix A)*

The WCAP defines the LTCC acceptance bases that are to be used to demonstrate that acceptable LTCC can be successfully maintained following the initial recovery (quench) of the core post-LOCA. The LTCC bases, applicable to GSI-191, are based on the requirements of 10 CFR 50.46, as clarified by the NRC in Reference 20. A detailed discussion of the criteria is contained in Appendix A of the WCAP. The acceptance bases, with justification, are summarized below.

1. Cladding temperatures during recirculation from the containment sump will not exceed a temperature of 800 degrees Fahrenheit.

50.46(b)(4) and (b)(5). One Phenomenon that must be considered for LTCC is boric acid precipitation. This TR does not evaluate the potential for boric acid precipitation.

The WCAP states that cladding temperatures at or below 800 degrees Fahrenheit maintain the clad within the temperature range where additional corrosion and hydrogen pickup over a 30-day period will not have a significant effect on cladding properties. At temperatures greater than 800 degrees Fahrenheit, there are occurrences of rapid nodular corrosion and higher hydrogen pickup rates that can reduce cladding mechanical performance. Long-term autoclave testing has been performed to demonstrate that no significant degradation in cladding mechanical properties would be expected due to a localized hot spot. This testing demonstrated that the increase in oxide thickness and hydrogen loading was limited at temperatures of less than 800 degrees Fahrenheit for periods of 30 days. With limited corrosion and hydrogen pickup, the impact on cladding mechanical performance is not significant. Therefore, no significant degradation in cladding properties would occur due to 30-day exposure at 800 degrees Fahrenheit and there would not be any adverse impact on the ability to cool the core. The autoclave results justify a maximum clad temperature 800 degrees Fahrenheit as an LTCC acceptance basis.

The PWROG has selected a cladding temperature of 800 degrees Fahrenheit as the acceptance basis for long-term cooling. The PWROG stated in Reference 4 that autoclave test data has demonstrated that oxidation and hydrogen pickup would be acceptable at and below the 800 degrees Fahrenheit temperature and that the reduction in cladding mechanical performance would be small. The cladding specimens in the autoclave tests were for fresh material. In RAIs contained in Reference 2, the NRC staff requested data for specimens which have undergone prior exposure to LOCA heat-up and quench conditions. The PWROG responded (Reference 4) by referring to autoclave test data and a literature review that indicates that susceptibility to localized accelerated corrosion occurs at temperatures in excess of 800 degrees Fahrenheit. The PWROG stated that it does not expect cladding properties to degrade due to a 30-day exposure to a temperature of 800 degrees Fahrenheit.

2. The deposition of debris and/or chemical precipitates will not exceed 50 mils on any fuel rod.

The WCAP states that for current fuel designs, regardless of vendor, the minimum clearance between two adjacent fuel rods, including an allowance for the spacer grid thickness, is greater than 100 mils. Therefore, a 0.050-inch (50-mil) debris thickness on a single fuel rod is the maximum deposition to preclude contact of the deposits on two adjacent fuel rods with the same deposit thickness. The 50 mil thickness is the maximum acceptable deposition thickness before bridging of adjacent fuel rods by debris is predicted to occur. The 50 mils of solid precipitation described here include the clad oxide, crud layer and debris deposition.

NRC Staff Evaluation

The NRC staff has reviewed the information provided to support the 800 degrees Fahrenheit maximum cladding temperature criterion and finds that the 800 degrees Fahrenheit temperature criterion provides reasonable assurance that LTCC will be successfully maintained. As discussed in Reference 4, this temperature represents a conservative boundary below which undesirable physical changes to the fuel cladding would not occur as a result of reheat following quench. Therefore, the NRC staff finds the limit to be conservative when used in conjunction with the engineering calculations discussed in Sections 3.2, 4.3, and 7 of the WCAP. The NRC staff accepts that the autoclave results are sufficient to justify the 800 degrees Fahrenheit as the long-term cooling temperature limit. If a licensee's final LOCADM calculation shows a cladding temperature that exceeds this value, cladding strength data must be provided for oxidized or pre-hydrided cladding material temperature in excess of that calculated. *This Condition is addressed further in Section 4.0, Item number 7, of this SE.*

The NRC staff has reviewed the information associated with the 0.050 inch debris thickness. The NRC staff finds that when used in conjunction with the engineering calculations discussed in Sections 3.2, 4.3, and 7 of the WCAP, a 0.050-inch limit on deposited debris thickness provides reasonable assurance that LTCC will be successfully maintained and cladding temperature will not exceed 800 degrees Fahrenheit. Although the bowing of fuel rods--if in opposite directions--may result in an increase in local cladding temperature, it is not expected to cause the cladding temperature to exceed the 800 degrees Fahrenheit temperature limit because: (1) the available margin that is described in Reference 4, Enclosure 1, RAI responses number 14 and 15, and (2) the margins available in the analyses discussed in Section 3.4.3 of this SE.

Therefore, the NRC staff finds that adherence to these acceptance bases and the methods and procedures discussed in this SE, including the conditions and limitations discussed in Section 4.0, will provide reasonable assurance of maintaining LTCC following a postulated LOCA.

3.3 BLOCKAGE AT THE CORE INLET *(WCAP-16793-NP, Revision 2, Section 3 and, Appendices B and G)*

3.3.1 Prototypical Fuel Assembly Testing

Section 3.1 of the WCAP discusses a prototypical test program designed to establish limits on the amount of debris that could bypass the ECCS sump strainer, enter the core, and still allow adequate flow to enter the core to ensure adequate LTCC. The WCAP states that a test protocol and test procedures were developed to include investigation of possible thin bed effects and that the debris used in testing represented debris that could be present in the RCS following a LOCA. Details of the testing are provided in Appendix G of the WCAP, proprietary test reports in References 8, 16, and 17, and RAI responses in References 11 and 12. The tests are also described in general terms in Section 4 of the WCAP.

3.3.1.1 Pressure Drop Considerations for Testing

Section 3.1.1 of the WCAP states that it must be demonstrated that the head available to drive flow into the core is greater than the head loss created by debris blockage. The WCAP also states that the available driving head is a plant specific value. The debris head loss is determined by testing.

3.3.1.1.1 Available Driving Head

Section 3.1.1.1 of the WCAP describes the conditions under which coolant flows into the core. At the time of sump switchover, the core has been fully recovered, the fluid inventory in the RCS is above the top of the core and core decay heat is being removed by ECCS injection. At this point in the accident, flow into the core is only possible if the manometric balance between the downcomer and the core is sufficient to overcome the flow losses in the reactor vessel (RV) downcomer, the RV lower plenum, the core, and the RCS loops (or reactor vessel vent valves in the case of a B&W plant). The manometric differences are determined considering plant geometry, system water levels, core void fractions, and flow path resistances. The flow losses are calculated using the Darcy equation. Further, the driving head at the core inlet is dependent on whether the break is in the hot-leg or cold-leg.

For calculating the available driving head, Section 3.1.1.1 of the WCAP guides the user to Section 2.18 of WCAP Reference 19 (the RAI responses noted as Reference 13 of this SE).

3.3.1.2 Pressure Drop Due to Debris

Section 3.1.2 of the WCAP states testing was performed to determine the head loss due to various fibrous debris loads. The section provides a summary of the tests. Details of the testing are provided in Appendix G and associated test reports (References 8, 16, and 17). The test loop is described as a closed loop system that continuously circulated fluid and debris through the test assembly. The test chamber was sized to match the fuel assembly pitch based on fuel assembly spacing in the reactor core. The WCAP also states that the flow entering the bottom of the fuel assembly was uniform and constant and that all debris was available to form debris beds in the fuel assembly. The WCAP states that the features of the test apparatus promote a conservative debris capture in the test assembly.

3.3.1.3 Description of Tests

Section 3.1.3 of the WCAP states that a common test protocol was used to ensure that testing of the Westinghouse and AREVA fuel assemblies conducted at separate sites was consistent and that the test matrix, acceptance criteria, and procedures were developed based on the same protocol.

3.3.1.4 Discussion of Test Results

Section 3.1.4 of the WCAP states testing was performed at hot-leg and cold-leg break flow rates. In addition, the Section states that:

1. Test matrices used for the program are provided in Appendix G and the results are provided in References 8, 16, and 17.
2. The flow rate associated with a hot-leg break represented the limiting head loss condition.
3. The amount of particulate in the test affects the formation of the debris bed and the associated head loss. Also, testing was conducted at the particulate to fiber (p/f) ratio that resulted in the limiting head loss and that the head loss increased significantly due to chemical effects.
4. Fiber is the greatest variable for increasing head loss at the core inlet and is the only type of debris that requires a limit to prevent loss of LTCC. Additionally, several debris types including particulate, microporous insulation, cal-sil insulation, chemical precipitates, and fiber were included in the test program.

The section concludes that plants that have in core debris loadings that are within the limits of the debris masses successfully tested are bounded by the test program and that plants with debris amounts greater than those successfully tested can take other actions to ensure LTCC.

3.3.1.4.1 Impact of Thin Bed on Head Loss

Section 3.1.4.1 of the WCAP states that testing was performed by adding all particulate to the loop and batching fibrous debris into the test in small quantities, similarly to the NRC March 2008 letter on strainer head loss test guidance (Reference 29), to determine if a thin bed would form. It is stated that a thin bed was not observed in any of the tests, even when only small amounts of fiber were included.

3.3.1.4.2 Debris Settling in the Lower Plenum

Section 3.1.4.2 of the WCAP states that credit for settling in the lower plenum is not being considered as a means of demonstrating LTCC but that it may be applied with appropriate justification for other issues associated with the closure of GSI-191.

3.3.1.4.3 Alternate Flow Paths

Section 3.1.4.3 of the WCAP states that flow paths that bypass the core inlet were not considered when determining the limiting debris loads based on testing. The section further states that some plants may choose to credit alternate flow paths, but would have to provide justification that the flow paths are viable to ensure LTCC.

NRC Staff Evaluation

Section 3.1 of the WCAP and the fuel assembly test reports (References 8, 16, and 17), discuss fuel assembly testing that was conducted to determine the potential effects of post-LOCA debris on flow into the core following a postulated LOCA. These references describe prototypical fuel assembly testing performed at the Westinghouse Science and Technology Center in Churchill, Pennsylvania and Continuum Dynamics, Inc. (CDI) in Ewing, New Jersey, for Westinghouse and AREVA fuel designs, respectively. This testing was conducted to answer NRC staff questions regarding previous evaluations that postulated that blockage at the core inlet would not inhibit cooling of the fuel following a LOCA. The initial fuel assembly test program and the evaluation presented in WCAP-16793-NP, Revision 1 (Reference 6) resulted in several staff observations that led to additional fuel assembly testing. The results of the additional tests were presented to the NRC staff in response to RAIs and are documented in letters from the PWROG (References 11, 12, and 13). The NRC staff observed some of the tests to confirm that the testing was conducted as expected and to ensure that the NRC staff had a good understanding of the test facilities and procedures. These observations are documented in NRC staff-prepared trip reports (References 22, 23, 24, 25, and 26). Additional NRC staff questions resulted in the PWROG conducting additional tests to ensure that other test variables had been adequately considered during testing. The updated tests are documented in the WCAP, Westinghouse proprietary test report in Reference 16, and a new AREVA proprietary test report (Reference 17). The following provides a brief discussion of the testing that was conducted to address the LTCC issue. The majority of the NRC staff's detailed evaluation of the test program results is included in Section 3.4 of this SE.

Section 3.1.1 of the WCAP describes the basis for the driving head used in the fuel assembly testing. WCAP Section 3.1.1.1 uses a version of the Darcy flow equation to calculate the pressure drop in the reactor coolant system. The NRC staff noted the equation did not appear to include a term for two-phase flow losses through the core, contrary to NRC confirmatory calculations that showed that two-phase flow may exist in the core during recirculation. Therefore, in Reference 9, RAI number 5, the NRC staff requested that the PWROG confirm that calculations for system pressure drop also included two-phase flow in the core. The PWROG responded to RAI 5 in Reference 13 by stating that the pressure drop due to steam in the core is calculated to be 0.1 pounds per square inch (psi) and, therefore, is negligible when compared to the conservatively established 1.5 psi cold-leg break acceptance criterion used in the fuel assembly testing.

The NRC staff does not accept the PWROG response to RAI number 5 in Reference 13. In the response to RAI number 5, a hand calculation is presented to show that a one-phase core pressure drop is less than 0.1 psi. However, this equation does not include a 2-phase pressure drop multiplier to account for 2-phase flow that can increase the core pressure drop and reduce margin to the 1.5 psi total available pressure drop stated in the RAI number 5 response. Also, the method prescribed does not include the head required to clear the cross-over leg loop-seal (Westinghouse and CE plants) that could form later in the cold-leg break scenario. The NRC staff does not fully accept the method described in response to RAI 18 of Reference 13 (WCAP Reference 19, Section 2.18) for calculating the available driving head for cold-leg or hot-leg breaks because it attempts to envelope the entire fleet of PWRs. Core void fractions, venting through the loop seals or reactor vessel vent valves (RVVVs) may vary significantly by plant. The method, however, was adequate to establish driving head values to be used in testing. Based on the very low fiber limits stated in Section 10 of Reference 18, the quantity of debris reaching the core under cold-leg break conditions is very low, as explained in Section 3.8 of this SE, and, therefore, the verification of available driving head under cold-leg break conditions may be addressed in the boric acid precipitation evaluation program. *This Condition is addressed further in Section 4.0, Item number 1, of this SE.*

To enable plants to determine the fraction of the strainer bypassed debris that is transported to the core inlet, the PWROG provided additional guidance in proprietary test report RAI responses (References 11 and 12), and RAI response number 18 in Reference 13. The NRC staff has reviewed the guidance contained in RAI response number 2 in Reference 11, RAI response number 4 in Reference 12, and RAI response number 18 in Reference 13, and concludes that there is sufficient guidance provided to enable licensees to determine the available driving head, the quantity of fiber delivered to the core inlet, and the debris limits for their plant(s). The guidance consists of basic mathematical equations and uses ECCS design inputs that are available in the respective plant existing design-basis calculations. The debris limits are based on fuel assembly testing conducted to support the WCAP.

The most recent testing determined that the hot-leg break case is limiting from the perspective of the amount of debris reaching the core and LTCC. If debris limits for the hot-leg break are increased through plant-specific testing, the cold-leg break may become limiting. Therefore, if licensees increase the allowable hot-leg debris load, they should confirm that their plant is covered by valid fuel assembly tests for both the hot and cold-leg conditions. *This Condition is addressed further in Section 4.0, Item number 1, of this SE.*

Licensees should determine the available driving head for hot-leg and cold-leg breaks following the guidance in RAI response number 18 in Reference 13. Licensees' GL 2004-02 submittals should include the available driving head used for the hot-leg. If licensees maintain the 15 gram debris limit established for hot-leg breaks, the cold-leg break may be bounded by the hot-leg break. *This Condition is addressed further in Section 4.0, Item number 2, of this SE.*

The NRC staff finds that the test facilities, as described in WCAP Section 3.1.2, were designed with attributes that should result in a conservative head loss for a specified debris load and that a common test protocol was used for the tests as stated in Section 3.1.3. However, the NRC staff noted that there were some differences in test results between facilities. The reasons for the variations were not determined. (These issues are discussed in later sections of this SE). The NRC staff concluded that at a limit of 15 grams per fuel assembly, the results attained in the existing test facilities show that adequate margin to ensure flow to the core is maintained. The statement in Section 3.1.2 regarding neglecting liquid entrainment and bubbly flow is an over simplification.

The fuel assembly testing involved the sequential addition of particulate, fibrous, and chemical precipitate debris to the water being pumped, under simulated ECCS flow conditions, to the lower section of an unheated mock fuel assembly. The purpose of the testing was to determine the quantity of debris that could be transported to the core inlet without exceeding the driving head available to deliver the required coolant flow to the core. The NRC staff reported in References 22, 23, 24, 25, and 26 that the debris transported to and collected on the fuel assembly inlet nozzle or first spacer grid. Further, in some tests run at higher flow rates, the debris accumulated at grids located further up in the fuel assembly. Fibrous debris used in the demonstrations consisted of fibers sized to represent the sizes observed to bypass typical sump strainers during strainer head loss testing. Particulates and microporous debris were simulated using the same materials (silicon carbide and Microtherm®) used for the ECCS strainer testing. The following observations were noted by NRC staff representatives, who made multiple trips to each test facility to observe fuel assembly tests:

1. As fibers were introduced into the test rig, they would be captured at the fuel assembly protective grid (P-grid) or on spacer grids located further up in the assembly. Depending on the test conditions, some fibrous debris passed through the mock fuel assembly and circulated back to the assembly inlet. The suspended debris was captured on subsequent passes. All fibers, no matter how short, appeared to be susceptible to capture within the test assembly.

2. The location of debris capture within the fuel assembly depended on the flow rate through the assembly, the fuel design, and the debris mixture used in the test. For most fuel, the cold-leg flow rate resulted in the collection of almost all of the debris at the protective grid (first spacer grid for fuel designs with no protective grid). For the hot-leg flow rate, the debris collected at different locations in the fuel assembly depending on the fuel assembly type, debris mix, and flow rate. The fuel design with no protective grid tended to collect low particulate-to-fiber ratio debris mixes at the first spacer grid and high particulate-to-fiber ratio debris mixes throughout the assembly for hot-leg breaks.

3. As particulates were added, they immediately distributed with the flow of the fluid throughout the test fixture. Differences in water clarity as the tests progressed allowed the NRC staff to observe that, over time, the fiber bed efficiently captured the particulates.

The NRC staff generally agreed with the observations in Section 3.1.4 of the WCAP. The NRC staff noted that the head losses from the early cold-leg tests exceeded the cold-leg head loss acceptance criteria, and therefore, did not agree that the hot-leg case was limiting. RAI number 15 of Reference 9 documents this issue. Based on RAI number 15, additional cold-leg testing was conducted. Based on the test results, the acceptance criteria for cold-leg break debris loading were decreased significantly. Additional testing, conducted prior to the issuance of Revision 2 of the WCAP, resulted in a reduction in the debris limits established for the hot-leg break scenario. These limits were similar to the limits obtained for the cold-leg break scenario. With these lower limits specified in Section 10 of the WCAP, the NRC staff concluded that the hot-leg case is limiting because the volume of debris-entrained coolant (and thus the mass of debris) that could enter the reactor core during a postulated cold-leg break would be significantly less than 15 grams. However, if debris limits for a hot-leg break scenario are increased through additional plant-specific testing, the cold-leg break scenario debris loads should be re-evaluated.

The NRC staff finds that chemical and particulate have a significant effect on head loss when fiber is present in sufficient quantity. During the early tests which contained large amounts of particulate relative to the amount of fiber in the circulated water, the staff noted that head loss appeared to be inversely proportional to the amount of particulate included in the tests—higher ratios of particulate to fiber resulted in lower pressure drop across the test assembly. RAI number 4 of Reference 10 and RAI number 3 of Reference 9 expressed the NRC staff's concern that this phenomenon had not been explored by the fuel assembly testing. Additional testing was performed that determined that the head losses were sensitive to particulate loading and to chemical precipitates. At cold-leg flow rates, relatively high particulate-to-fiber ratios result in the limiting head losses. At hot-leg flow rates, low particulate-to-fiber ratios result in higher head losses. Chemical precipitates do not have a significant effect on pressure drop if the debris bed is fully saturated with particulates prior to the precipitate addition. These effects are discussed in more detail in Section 3.4 of this SE.

The NRC staff finds that, based on the limited quantity of strainer bypassed fibrous debris permitted in order to satisfy the hot-leg debris limit of 15 grams, the amount of debris reaching the core following a postulated cold-leg break would be significantly lower than 15 grams because a portion of the debris exits the cold-leg break with the ECCS flow. The NRC staff also found that the debris is well mixed with the coolant being pumped to the reactor vessel due to turbulence in the piping system.

The NRC staff finds that plants that have debris loadings within those defined by acceptable tests are bounded by the test program and that plants not bounded by the program can perform plant specific evaluations to demonstrate acceptable LTCC. Plants that establish higher debris limits than that stated in Section 10 of the WCAP shall include with their GL 2004-02 responses a description of the test protocol, the test parameters and test results used to establish the higher limits. *This Condition is addressed further in Section 4.0, Item number 1, of this SE.*

The NRC staff reviewed the statement in Section 3.1.4.1 of the WCAP that the fuel assembly testing was conducted using a procedure that attempted to form a thin bed on the fuel assembly and that a thin bed was not observed to form during the fuel assembly testing. Specifically, testing showed that higher fiber loading resulted in higher head losses and relatively low particulate-to-fiber ratios resulted in limiting head losses when chemical precipitates were added after the fiber/particulate debris bed had formed, especially for the hot-leg break cases. With respect to the formation of a thin bed, testing showed that higher fibrous loads are more challenging for head loss than lower fibrous loads. Therefore, the NRC staff's position is that limiting the fibrous debris to the maximum allowable amount determined by testing is necessary and that formation of debris beds with lesser amounts of fiber will not be more challenging to LTCC. However, plants that determine that chemical deposits will not affect in-vessel head losses should ensure that tests for their specific conditions search for limiting head losses using particulate loads that maximize head loss with no chemical precipitates included in the tests. *This Condition is addressed further in Section 4.0, Item number 12, of this SE.*

The NRC staff reviewed the statement in Section 3.1.4.2 of the WCAP and finds that not crediting debris settling in the lower plenum as described in Section 3.1.4.2 is conservative for FA testing because this maximizes debris reaching the fuel assembly. If credit for settling in the lower plenum is used in later plant-specific evaluations, it should be adequately justified. Additional testing details and the NRC staff evaluations regarding the various fuel assembly tests are provided in Section 3.4 of this SE.

The NRC staff reviewed the position stated in Section 3.1.4.3 of the WCAP and finds that alternate flow paths into the core have not been credited when determining debris loads limits and that some licensees could potentially credit these alternate flow paths via plant specific evaluations. The NRC staff has not received validated information on the availability or capability of alternate flow paths to demonstrate LTCC. If a licensee elects to take credit for alternate flow paths, such as core baffle plate holes, the licensee would need to demonstrate that the flow paths would be effective; i.e., that the flow holes would not become blocked with debris during a LOCA, and that debris that does pass through the alternate flow path does not adversely affect core cooling and that any changes to the flow patterns do not adversely impact boron precipitation. *This Condition is addressed further in Section 4.0, Item number 3, of this SE.*

3.3.2 WCOBRA/TRAC (WC/T) Evaluations of Blockage at the Core Inlet

Section 3.2 of the WCAP states that to further bolster the assertion that core cooling flow will be maintained, WCOBRA/TRAC (WC/T) analyses were performed to demonstrate that adequate flow is provided and redistributed within the core to maintain adequate LTCC. The WC/T code is used for evaluating best-estimate large break LOCA response.

The WCAP states that a bounding evaluation was performed using limiting assumptions to evaluate the consequences of core inlet blockage on LTCC. The calculation concluded that with 99.4 percent of the core inlet blocked, sufficient liquid could enter the core to remove core decay heat once the plant had switched to sump recirculation.

The WCAP states that the core inlet blockage simulations were designed to bound the U.S. PWR fleet. To ensure a bounding calculation, the limiting break type and the limiting vessel design were taken into consideration before selecting a plant model for the simulation.

The WC/T evaluations simulated the effects of debris buildup at the core inlet by ramping-up the dimensionless friction factor (CD) at the core inlet to a large number, resulting in a reduction of flow. After the core inlet resistance ramped up to its maximum value of about CD = 10^9 (which essentially eliminates all flow through the path), the simulations were run out to 40 minutes to show that the flow rate supplied to the core would be sufficient to remove decay heat and maintain a coolable core geometry with 99.4 percent of the core inlet area blocked. Appendix B of the WCAP contains a detailed description of the evaluation performed.

<u>NRC Staff Evaluation</u>

The NRC staff performed confirmatory analyses using the FLUENT and TRACE codes and obtained results consistent with the WC/T results. All of these analyses indicate that significant blockage would not preclude adequate coolant flow reaching the fuel to remove the decay heat. Figure A-1, in Appendix A of the WCAP, is a plot of the flow rate needed to match the boil-off rate in a Westinghouse 4-loop PWR. From this figure, it can be concluded that following the postulated LOCA, there is a decrease in required coolant flow with time, thus requiring greater inlet blockage to inhibit adequate core cooling.

However, in light of the results obtained in the fuel assembly tests documented in proprietary test reports (References 8, 16, and 17) and the associated RAI responses (References 11 and 12), the NRC staff does not accept these calculations as a basis for demonstrating that adequate flow will reach the core. The PWROG has not demonstrated that an open flow channel will remain that will allow the required flow into the core.

3.3.3 Additional WC/T Calculations

Section 3.3 of the WCAP described several additional WC/T analyses that were performed at the request of the ACRS with the purpose of determining the blockage level (either using a reduction in inlet area while maintaining a constant form-loss coefficient or increase in the loss coefficient while maintaining a constant flow area) that would reduce core flow below that necessary to match coolant boil-off.

The detailed documentation for these additional calculations is presented in Appendix B to the WCAP and includes time history plots 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. The pressure drop for flow through a porous medium is approximately proportional to the velocity. The WC/T uses a pipe flow relation where pressure drop is proportional to loss-coefficient and velocity-squared and inversely proportional to flow area squared. The WC/T analysis uses two approaches to approximate the debris bed behavior. The two approaches were taken to determine the blockage level needed to preclude sufficient flow into the core to provide for LTCC. The first approach considered an area reduction while maintaining the form-loss coefficients (simulating localized bore-holes in the debris bed). The second approach considered form-loss coefficient increases while maintaining the flow area constant (simulating uniform blockage across the core inlet).

In the second approach (Increased Loss Coefficient Approach) uniform core inlet loss coefficients of 50,000, 100,000, and 1,000,000 were used to determine when boil-off could no longer be matched.

The flow reduction simulation showed that a reduction of 50 percent of the hot channel flow area over the 99.4 percent area reduction case (Case 2 analysis described in Section 3.3.2 of the WCAP) yields a total core inlet flow reduction of 99.7 percent compared to an unblocked core. The evaluation shows that even with this increase in core blockage, the flow that enters the core is still in excess of the boil-off rate and the PCT remains well below 500 degrees Fahrenheit. However, a reduction in the hot channel flow area by 80 percent yields a total core inlet flow area reduction of 99.9 percent. At this degree of core blockage, the flow that enters the core does not match the boil-off rate. The WC/T analysis concluded that a total core inlet area reduction of up to as much as 99.7 percent will still allow sufficient flow into the core to provide for removal of decay heat and assure LTCC.

The uniform loss coefficient evaluations showed that an increase in the CD at the core inlet of up to 100,000 for the limiting plant and fuel load design would allow for sufficient flow into the core to remove decay heat and provide for LTCC.

Section 3.3 of the WCAP states that while the blockage simulated by these calculations will not occur with the fuel assembly debris loadings used in the fuel assembly tests, these WC/T calculations provide additional assurance that LTCC will not be compromised. The document concludes that sufficient liquid can enter the core to remove core decay heat once the plant has switched to sump recirculation with up to 99.4 percent blockage at the core inlet.

<u>NRC Staff Evaluation</u>

The NRC staff performed confirmatory analyses to determine the thickness of a uniform, particulate-entrained fiber bed distributed across the bottom of the fuel inlet that would allow adequate flow for LTCC. The analysis, however, did not consider the effects of chemical

precipitates. This analysis showed that in the absence of chemical precipitates, a relatively thick debris bed could form before flow to the core would be restricted to an unacceptable value.

However, as stated in the NRC staff evaluation Section 3.3.2 above, results obtained in the fuel assembly tests documented in the proprietary test reports (References 7, 8, and 17) and the associated RAI responses (References 11 and 12) showed that with fiber quantities exceeding the limits established in References 7, 8, and 17, a debris bed was formed with fiber, particulate and chemical precipitates that resulted in a pressure drop across the fuel assembly that exceeded the specified available driving head. Therefore, NRC staff does not accept these calculations as a basis for demonstrating that adequate flow will reach the core when large quantities of debris are transport to the core inlet. The basis for this conclusion is that testing with chemical precipitates included in the debris mix has shown that significant blockage can occur with debris quantities lower than those predicted by the calculations. *This Condition is addressed further in Section 4.0, Item number 4, of this SE.*

3.4 COLLECTION OF DEBRIS ON FUEL GRIDS *(TR WCAP-16793-NP, Revision 2, Section 4 and Appendices C and D)*

Section 4 of the WCAP states that debris not collected at the core inlet will pass through the fuel assembly bottom nozzle and, potentially, lodge within the fuel assembly, primarily at the fuel spacer grids. The WCAP used three supporting analyses (general analysis of debris build-up and its effects on LTCC, review of fuel assembly test data, and numerical calculations) to demonstrate that blockage will not impede LTCC.

3.4.1 General Discussion

Section 4.1 of the WCAP states that spacer grids provide the most likely locations for debris capture within the core. The most likely locations for debris capture within the spacer grid being at the "springs" and the leading edge of the spacer grid. Section 4.1 of the WCAP states that the sump strainer openings are 0.11-inch or less in diameter, thus, limiting the size of the debris that can enter the core.

Section 4.1 of the WCAP states that flow paths exist between fuel rods which will limit the extent and the consequences of debris build up as flow will divert away from the area where debris has deposited and is inhibiting flow. The WCAP also states that the debris that collects will allow weeping flow that will provide cooling for the cladding, that the packing factor will likely be less than about 60 percent, and that the debris bed will therefore not become impenetrable. The WCAP further states that boiling in the area of the blockage will occur with less than a 10 to 15 degrees Fahrenheit increase in the clad temperature over the adjacent cooling temperature, which will provide sufficient convective heat transfer to maintain the fuel rod a few degrees below the liquid saturation temperature. Based on the points above, the WCAP asserts that blockage at the spacer grids will not adversely affect LTCC.

3.4.2 Prototypical Fuel Assembly Testing

Section 4.2 of the WCAP states that a test program was conducted to determine acceptance criteria for the mass of debris that could be deposited at the core entrance and not impede LTCC. The WCAP states that, during testing, debris accumulated at the spacer grids, that fluid continued to pass through the debris bed, and blockage that occurred was conservative with respect to that which could occur following a LOCA.

Section 4.2 of the WCAP states that the testing contained the following conservatisms:

1. The ECCS volume must pass through or bypass the core, then exit the RCS via the break. Any debris that passes through the core or exits the break directly would pass through the sump strainer prior to re-entering the core. Some debris would likely be filtered out by the strainer and not be available to return to the core. The test allowed the debris to recycle multiple times if it passed through the test assembly.

2. If boiling occurs following a hot-leg break it will result in turbulence that will tend to remove debris from the spacer grids and confine blockages to isolated areas of the core.

3. Following a LOCA, the fuel assembly will bow due to the thermal transient on the fuel rods. This bowing will distort the flow channels, making some larger and some smaller, allowing flow around any blockages that form at the spacer grids.

4. Following a cold leg break the core flow will be turbulent enough that any debris that is not trapped will be continuously moved. The turbulence will ensure that coplanar blockage of the core does not occur.

Section 4.2 of the WCAP notes that at high p/f ratios that debris was observed to accumulate at the spacer grids and concludes that the testing defines an upper bound for the possible blockage that could occur at the fuel. The section also states that it was observed that debris bed formation did not occur at the spacer grids at limiting p/f ratios.

3.4.2.1 Topical Report WCAP-16793-NP, Appendix G and Proprietary Test Reports

Figure 3.4-1, taken from a nonproprietary AREVA submittal, is a schematic of the test facility used for the testing of the AREVA fuel assembly. The test facility used in the Westinghouse fuel testing is similar. Appendix G of the WCAP and fuel assembly test reports (References 8, 16, and 17) describe the test loop used for the fuel assembly testing. The WCAP states that the test loops (at Westinghouse and CDI) consisted of a mixing tank, a recirculation system, the test column, and monitoring instrumentation. The mixing tank was used as a water volume into which debris could be added and thoroughly mixed before being pumped to the test column. The recirculation system pumped the water through the fuel assembly at a controlled rate.

While the test loops used at Westinghouse and CDI were similar, they were developed separately so some differences exist. Under some conditions, the results of the tests conducted using the same debris loads were somewhat different. Notable differences between the facilities include the method used to maintain a homogenous mixture in the mixing tank (AREVA used a propeller blade-type mixer and Westinghouse used a circulating pump) and the flow diverter used at the inlet to the fuel assembly test tank (AREVA used a cone with the flat surface facing the stream inlet and Westinghouse used a cube with a corner facing the stream inlet). The chemical debris preparation methods varied slightly, and the water used in the test loops was local water so there may have been some effect from water chemistry. These differences could have resulted in differences in fiber characteristics, chemical debris characteristics, and flow patterns in the test tank. The NRC staff suggested that one or both fuel vendors' test-articles be tested in the other vendor's test facility (cross test) to validate that the differences in behavior during testing were due to fuel design and not due to the test facility. The PWROG developed a plan for conducting the cross tests, and provided the results in Appendix G of the WCAP. The tests are summarized below.

According to Appendix G of the WCAP and the proprietary test reports (fuel assembly test reports References 8, 16, and 17), the test columns possessed the following attributes: 1) the test column contained the fuel assembly and channeled the water through the assembly, 2) the column was constructed of clear material to allow observation of the debris deposition in the tested fuel assembly, 3) the test column also contained a chamber at the bottom to allow the water to be pumped into the assembly, 4) the lower chamber included a diffuser to prevent the incoming flow from impinging directly on the bottom of the fuel assembly, 5) above the diffuser was a simulated core support plate upon which the test fuel assembly rested, 6) the clearances between the fuel assembly and test column could be adjusted to the specified dimensions.

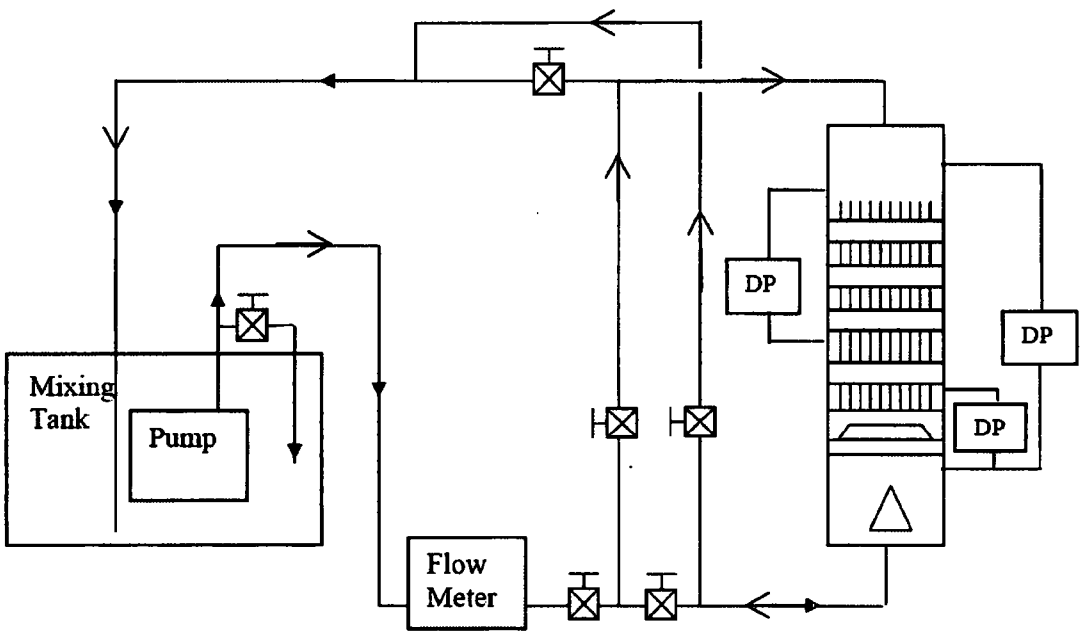

Figure 3.4-1 Schematic of Test Loop (Note that small black arrows indicate flow direction)

The WCAP states that the instrumentation provided measurement of fluid temperature, fluid flow rate, and differential pressure measurements across various locations in the fuel assembly. The instruments were connected to a data acquisition system for continuous data collection.

The types, amounts, and characteristics of the debris used in the fuel assembly testing are described in fuel assembly test reports (References 8, 16, 17, and 18). The debris characteristics were similar to those previously accepted by the NRC, in the SE of NEI 04-07 (ADAMS Accession No. ML050550156), for use in ECCS sump strainer head-loss testing. However, when appropriate, the debris was further refined to closely model that which had passed through an ECCS strainer. For example, particulate debris, normally prepared as 10 micron nominal diameter particles, was not further refined. Fibrous and microporous debris were further treated to ensure that they were prototypical of debris that had passed through a strainer. In general, the fibrous debris was fragmented into small individual fibers, and was measured to ensure that it met an expected size distribution that matched that of fibrous debris collected from strainer bypass tests. The fibrous debris target sizing is stated to be 77 percent shorter than 500 microns, 18 percent between 500 and 1000 microns, and 5 percent longer than 1000 microns. The actual size range of the debris used was 67 to 87 percent shorter than

500 microns, 8 to 28 percent between 500 and 1000 microns, and 0 to 15 percent longer than 1000 microns. Microporous debris was sieved through a screen having opening sizes characteristic of a strainer.

According to the WCAP, the order of debris addition followed NRC guidance (Reference 29) for strainer head loss testing. In general, particulate was added first, followed by batches of fiber (in an attempt to search for a thin bed effect), and finally chemicals were added. For tests that included microporous material or calcium silicate insulation (Cal-Sil), a portion of these materials were added prior to the fibrous debris and a portion added after the chemical precipitates. The WCAP states that this was intended to simulate erosion of these materials over the post-accident time period. The initial debris loads tested were scaled to bound the maximum amount of debris that could reach the core from most plants. However, the initial testing revealed that this amount of fibrous debris would not allow the head loss to remain within the allowable limit. Therefore, additional testing was conducted with reduced fibrous debris loading. In general, the initial Westinghouse testing was conducted with 200 grams of fiber per fuel assembly and AREVA testing was conducted with 150 grams of fiber per fuel assembly. Later testing used lower amounts of debris. Appendix G contains tables that summarize the test conditions and debris loads for the tests performed during the test program.

According to the WCAP, the specific flow velocities that existed in the mockup fuel bundle inlet were chosen to represent the maximum potential flow rates within a core following a large or small break-LOCA independent of plant design. These flow rates were chosen to ensure that the head losses attained during the test were maximized. (All flow rates discussed in this section are based on the flow through a single fuel assembly). For Westinghouse and B&W plants, the ECCS flow rates were set at 44.5 gpm and 3.0 gpm for flows associated with hot-leg and cold-leg breaks, respectively. For CE plants, the ECCS flow associated with a hot-leg break was specified as 11.0 gpm for plants with AREVA fuel and 6.25 gpm for plants with Westinghouse fuel. (Note: The hot-leg break flow rate was not driven by the fuel type but by the plant design parameters.) The flow associated with a cold-leg flow rate was specified as 3.0 gpm for all CE plants. All flow rates were stated to be plus or minus 10 percent. Based on the test reports and staff observations of testing, flow was controlled near the desired rate, and well within the proposed tolerances. Later testing was conducted to ensure that the maximum hot-leg flow rate was limiting when compared to lower flow rates that could occur if equipment failures occurred or for plants with lower ECCS flow rates. These tests were conducted at 15.5 gpm to model single train ECCS flow at a Westinghouse plant.

For Westinghouse-designed fuel, the cross section of the test assembly was prototypical of a 17x17 fuel assembly. A test was also run at Westinghouse for a 16x16 CE fuel design. The Westinghouse assembly was tested with a Standard P-grid, in lieu of the Westinghouse Alternate P-grid, because previous testing had shown that the Standard P-grid resulted in higher head losses than the Alternate P-grid. The CE fuel was tested with a Guardian Grid that is similar to the Standard P-grid. Test results showed that the head losses attained with the Standard P-grid were higher than those attained with the Guardian® Grid. Therefore, additional testing for Westinghouse fuel was conducted with the Standard P-grid.

For AREVA designed fuel, the cross section of the test assembly was prototypical of a 17x17 assembly. Testing of the AREVA fuel was conducted with three different inlet filters, Trapper Coarse Mesh®, Trapper Fine Mesh®, and Fuelguard®. Testing found that the Trapper Fine Mesh® resulted in excessive head loss, and continued testing of this filter was not performed since this fuel design is no longer being installed in U.S. plants. Future use of the Trapper Fine-Mesh® will need to be supported by additional testing. The Trapper Course Mesh® and

Fuelguard® filters exhibited acceptable head losses of similar magnitude. Because the Fuelguard® filter is the most commonly used AREVA inlet filter and because of the similar behavior of the two filters, AREVA testing was conducted primarily with the Fuelguard® filter.

According to Appendix G of the WCAP, the test program resulted in conservative head loss values for several reasons as follows:

1. Tests were conducted at limiting p/f ratios. As p/f ratios varied from the limiting value, head losses at a constant fiber load decreased. The WCAP states that plants may be able to determine plant specific p/f ratios that would allow larger fiber loads to be ingested while maintaining LTCC.

2. The tests were conducted at constant flow rates. For both the hot and cold-leg cases flow rates may decrease resulting in lower velocities through the beds and lower head losses.

3. The testing did not credit alternate flow paths that bypass the core inlet.

4. The tests are applied assuming that debris will collect uniformly in all fuel assemblies. This is unlikely to occur in a full core due to variations in flow, bundle power, and fuel assembly orientation.

5. Boiling would disrupt any bed formation allowing flow to cool the core. The boiling will occur differently for hot and cold-leg breaks.

6. Some debris in the plant would likely settle. The test facilities used agitation to encourage all debris to reach the fuel assembly.

Appendix G of the WCAP states that the key findings from the testing were as follows:

1. Based on testing with several types of debris it was determined that the amount of particulate included in the test affects the formation of the debris bed and resulting head loss. Also, that fiber is the limiting variable and is the only debris type that requires a limit.

2. The hot-leg break flow rate (highest flow rate) is the limiting condition.

3. The test facilities and procedures are repeatable, but small changes in the test loops can result in significant changes in test results.

Appendix G of the WCAP concludes that the test program was conservative and that fiber is the limiting debris type for LTCC. The appendix provides conclusions regarding debris loads that are identical to those presented in Section 10 of the main body of the WCAP. The WCAP states that the allowable fiber limits defined for a plant, in conjunction with analyses performed by licensees, can demonstrate that LTCC requirements are met. Further, the WCAP states that plants whose debris loads do not fall within the debris load parameters defined by the tests have alternate actions available to ensure that LTCC requirements will be met. The acceptance criteria are presented in Section 10 of the WCAP.

Additional details of the testing are contained in fuel assembly test reports (References 8, 16, and 17). These are proprietary test reports that describe the testing for the two vendors in more detail. Information from the test reports is discussed below.

a) Westinghouse Fuel Testing Report

The executive summary of Westinghouse Fuel Assembly Test Report, WCAP 17057-P, Revision 1 (Reference 16), states that the following observations can be made regarding the testing.

1. The tests and evaluations from the test facility are reliable and can be used to draw meaningful conclusions regarding the impact of debris on in-vessel effects.

2. Testing shows that the amount of particulate influences the formation of the debris bed and resulting head loss across the fuel assembly.

3. The hot-leg (maximum) flow rate represents the limiting test condition and should be used for testing to determine debris limits.

4. The Westinghouse test facility provides repeatable results and repeat-tests are not necessary to define debris limits.

5. The test program evaluated the impact of various debris types, determined that fiber is the limiting factor, and is the only debris type that requires a limit.

6. Although the majority of the testing was conducted with a single type of fuel assembly it is applicable to all fuel assembly designs included in the test report.

7. The test program is conservative and bounds all plant types.

8. 46 tests were conducted to define fiber limits for Westinghouse fuel assemblies.

The test report concludes that plants that are within the limits of the test parameters will meet LTCC requirements. Plants that are not bounded by the tests may take other actions to demonstrate LTCC.

The test report, WCAP-17057-P, Revision 1 (Reference 16), was updated from Revision 0 to incorporate additional testing to support the WCAP and to address NRC staff concerns with the conclusions from the earlier test program as documented in the NRC RAIs contained in Reference 9.

The following lists the basic steps for the testing of the Westinghouse fuel.

1. The fuel filter design to be tested is selected and mounted on the partial-height FA. The assembly is inserted into the test loop, and the loop is filled with water.

2. Debris quantities are measured and verified.

3. The pump is started, and the flow is set to the desired flow rate. The clean head loss is recorded.

4. Particulate debris is added to the mixing tank, and the head loss across the mock fuel assembly is recorded.

5. Fiber is added to the mixing tank in batches with at least two turnover times allowed to pass between batches. Fiber is added batches until the desired debris amount is added or the head loss limit of the test apparatus is reached. Head loss is recorded.

6. Chemical precipitates are added and head loss across the fuel assembly is recorded.

7. Head loss is allowed to reach a predefined steady state for test termination.

8. The final head loss readings are recorded, and the test is terminated.

WCAP-17057-P, Revision 1 (Reference 16), provides additional information regarding the characteristics of particulate, fibrous, chemical, Cal-Sil, and microporous debris used.

WCAP-17057-P, Revision 1 (Reference 16), discusses acceptance criteria for the testing. The report states that the acceptance criteria are plant specific and are determined based on the driving head available to push water through the core following an RCS break. The report discusses allowable head losses for hot and cold-leg conditions and states that the available driving head must be greater than the head loss due to debris and other flow losses to ensure LTCC.

WCAP-17057-P, Revision 1 (Reference 16) provides information on the fuel designs and plant designs for the plants that utilize Westinghouse fuel. The report also provides information regarding the test loop which has been previously discussed in this SE. This information also provides inputs to the determination of the test matrix for the testing that Westinghouse planned to run to quantify acceptable debris loads for the various plants.

The test fuel assemblies used in the fuel blockage tests were partial length assemblies, approximately 4.5 feet tall for Westinghouse plants and 3 feet tall for CE plants. The assemblies included bottom nozzles, protective filters, spacer grids, fuel rods (without fuel pellets), guide tubes, and instrument tubes. The fuel rods and other tubes in the assembly were closed to ensure that water could not flow through them.

The WCAP states that the gap was modeled into the test assembly by including a gap around the perimeter of the test assembly. The width of the gap was equal to one half of the gap between actual fuel assemblies in the reactor core. Flow through the gap was not measured.

WCAP-17057-P, Revision 1 (Reference 16), provides a summary of the testing that was conducted at Westinghouse, including the test of the AREVA fuel (cross-test), and also includes a significant amount of data, including head loss plots, from the tests. The technical report provides a discussion of the test results for each of the various fuel inlet filters. The test report concludes that the Standard P-grid filter resulted in the highest head loss when compared to the Guardian Grid and the Alternate P-grid filters. The comparison with the Alternate P-grid was based on a previously conducted fuel testing program, and the comparison with the Guardian Grid was based on the tests conducted within this test program.

Section 6.1 of WCAP-17057-P, Revision 1 (Reference 16), evaluates the facility for reliability and repeatability. The section describes the tests performed (cross tests) to evaluate potential causes for unexpected differences in results between tests conducted under similar conditions.

Westinghouse performed a root cause investigation to determine the reason for differences in test results. No root cause was identified, but five potential causes were postulated. They are: air entrainment, changes to inlet pipe location, inconsistent debris addition and mixing, variation in fiber characteristics, and physical phenomenon. Three of the potential causes were eliminated leaving only air entrainment and inlet pipe location changes as likely causes. Air entrainment was not confirmed as a cause. Testing found that inlet pipe location had an effect on test results with the most conservative results occurring with the inlet pipe submerged.

The test report states that the head loss across a fuel assembly is a function of the debris bed structure and the velocity of the fluid through the debris bed. As such, the Westinghouse test report (Reference 16) concludes, that for the same debris loading, the head loss at a lower flow rate would be reduced from that at a higher flow rate. The technical report notes that CE plants have significantly lower ECCS flow rates than Westinghouse PWR plants, due to the system design. It was concluded, based on a comparison of head loss test results, the Standard P-grid bounded the head loss that could occur with the Guardian Grid at the CE plant flow rates.

The test report states that some fuel designs have no inlet filters (Westinghouse determined that this is applicable only to the Westinghouse 2-loop UPI plants) and that these designs are bounded by the testing conducted for fuel assemblies with inlet filters, based on the limiting flow clearances of the inlet filters. Revision 0 of the Westinghouse test report (Reference 7) describes a test conducted to determine the head loss across the core for a plant that is designed for UPI, where the coolant is injected into the top of the core and flows down through the fuel assemblies. The fuel assembly used for this test was the same as the Westinghouse assembly described above even though UPI plants do not employ fuel filters. The maximum hot-leg break flow for the UPI plants is significantly lower than that for the majority of PWRs because for the hot-leg break, the majority of the flow exits the RCS through the break instead of flowing through the core.

The test report discusses the effect of the p/f ratio on fuel assembly head loss. Test results indicate that for cold-leg tests the limiting ratio (on a mass basis) is about 45:1. For hot-leg flow rates the limiting p/f ratio is 1:1. The effects of other debris types are also discussed in the report. The report concludes that microporous insulations and Cal-Sil behave similarly to particulate during in-vessel tests. The effects of chemical precipitate were also evaluated. The report concludes that chemical effects are minimized when the p/f ratio was high, but have a significant effect at lower p/f ratios. The report also concludes that only a small amount of chemical precipitate was required to reach the maximum head loss regardless of p/f ratio.

WCAP-17057-P, Revision 1 (Reference 16), discusses the effects of break location and flow rate on in-vessel head loss. The test program evaluated flow rates associated with cold-leg and hot-leg breaks and included varying hot-leg flows that could occur due to lower flow from smaller pumps and single failures of redundant pumps. The report concludes that a lower flow rate would form a less resistant debris bed. The report concludes that the highest flow rate results in the limiting head losses.

The test report discusses the application of the test results to cores that employ multiple fuel types or mixed cores. In a letter dated June 24, 2010 (Reference 14), and the Westinghouse test report (Reference 16) the PWROG recognizes that plants may have different fuel designs within a single core at various times, and state that the most conservative approach for applying the acceptance criteria developed by the testing is to consider the limiting fuel design. The report (Reference 16) notes that all Westinghouse fuel designs have the same debris limits.

b) AREVA Fuel Testing Reports

The AREVA tests are described in AREVA proprietary test reports EIR 51-9102685-000 (Reference 8) and EIR 51-9170258-000 (Reference 17). The first AREVA report describes the original round of testing and the second report describes additional testing tests conducted in response to NRC staff questions regarding the earlier tests documented in EIR 51-9102685-000 (Reference 8).

AREVA test report EIR 51-9102685-000 (Reference 8) notes that the debris load used for the TRAPPER Fine Mesh® fuel filter was lower than the other types of inlet filters. With the debris loads used during the testing, the fine mesh filter is more efficient at trapping debris at the fuel assembly inlet such that the debris bed contains more debris and head loss is higher. However, head loss measurements indicate that differential pressure occurred across the spacer grids as well as the filter. The tests run with the Trapper Coarse Mesh® and Fuelguard® filters incurred very little, if any, head loss at the bottom filter (bottom nozzle) with most of the losses occurring at the spacer grids.

AREVA test report EIR 51-9102685-000 (Reference 8) asserts that as the fuel inlet becomes blocked, the velocity of the flow through the gap between fuel-assemblies will increase until it is high enough to prevent debris from collecting in the gap. Therefore, the report asserts that the gap is self cleaning such that excessive head loss will not occur.

AREVA test report EIR 51-9102685-000 (Reference 8) discusses the use of AREVA fuel on a limited basis in cases when customers install small clusters of approximately 4 or 8 assemblies for test purposes. The report states that the limited number of assemblies installed under these conditions will not adversely affect LTCC considering the debris that may be present following a LOCA. Therefore, the report concludes that the debris limit associated with the resident fuel may be applied to the entire core if a limited number of alternate fuel assemblies are installed.

AREVA test reports EIR 51-9102685-000 (Reference 8) and EIR 51-9170258-000 (Reference 17) conclude that the test results indicate that, if the amounts of debris defined in the acceptance criteria were present at the core inlet, LTCC would still be assured.

AREVA test reports EIR 51-9102685-000 (Reference 8) and EIR 51-9170258-000 (Reference 17) state the following observations based on earlier and more recent testing:

1. The debris bed morphology and the flow rate are key parameters affecting head loss.

2. High flow rates result in more limiting head losses.

3. Low p/f ratio tests resulted in higher head losses for hot-leg flow rates.

4. The debris deposition within the fuel assembly is dependent on the p/f ratio. Higher p/f ratios result in the debris collecting at multiple spacer grids and lower p/f ratios result in debris collecting at the lower end grid.

5. The AREVA bottom nozzle is not the limiting debris accumulation location.

AREVA test report EIR 51-9170258-000 (Reference 17) discusses the results of the cross tests (testing of Westinghouse and AREVA assemblies in the other vendor's test facility) and makes the following conclusions:

1. The CDI and Westinghouse test loops behave differently. Both head losses and debris deposition patterns are different. In the CDI loop, the lower debris bed attained a higher head loss value prior to break through than in the Westinghouse loop.

2. The AREVA and Westinghouse assemblies behaved very similarly in the Westinghouse and CDI test loops.

3. There is no significant difference between the behavior of AREVA and Westinghouse fuel when tested under similar conditions.

NRC Staff Evaluation

a) Overview and Evaluation of Assumptions Regarding Fuel/Debris Behavior

The NRC staff reviewed the prototype fuel assembly test procedure and test results in detail. In addition, the staff witnessed several tests at Westinghouse and CDI. The trips to observe the testing are documented in trip reports References 22 through 28. The NRC staff evaluated the test facilities, the debris used during testing, the test methodology, the test fuel assemblies, and the test results. The NRC staff found that the test program resulted in measured head losses that reasonably bound those that could occur in a reactor following a postulated LOCA because the testing was conducted under conditions that reasonably represent those in the plant, as documented in trip reports References 22 through 28. The testing did not model all aspects of the post-LOCA environment. However, the testing did contain some conservatism as discussed further in this evaluation. The NRC staff did not agree with all of the assertions made in the topical report and the related test reports. Following the NRC staff's review of the test program and the conclusions that were drawn from the program, the NRC staff issued RAIs by letters dated January 8 and January 15, 2010 (References 9 and 10, respectively), regarding information presented in WCAP-16793-NP, Revision 1 (Reference 6), and the associated proprietary test reports (References 7 and 8). The majority of the RAIs issued regarding information presented in fuel assembly test reports (References 8, 16, and 17) are discussed in this section. Following the RAI responses, the PWROG conducted additional fuel assembly testing as described in References 16 and 17. Based on this testing, the NRC staff concluded that a limit of 15 grams of fibrous debris per fuel assembly was an acceptable upper limit. The NRC staff also concluded that debris loads above 15 grams may be acceptable, but should be evaluated on a plant specific basis.

The NRC staff concluded that the test facilities and partial height fuel assemblies used in the test program provided a model that could provide a realistic or conservative head loss value at the debris loads tested. The test fuel assembly, although partial height, provided adequate locations for debris to collect so that the effects of the debris could be determined.

Following the initial NRC staff review of the head loss test program, several issues were identified that caused Westinghouse and AREVA to perform additional testing to better define debris loads for various plant configurations and break scenarios. The NRC staff formally issued the questions as RAIs to WCAP-16793-NP, Revision 1 (Reference 6),] and to the proprietary test reports (References 7 and 8). The RAIs are contained in letters dated January 8, 2010 (Reference 9) and January 15, 2010 (Reference 10). As a result of the RAIs, Westinghouse issued Revision 1 to WCAP-17057-P (Reference 16), AREVA issued EIR 51-9170258-000 (Reference 17) to supplement the original test report (Reference 8), and the PWROG issued the WCAP. This NRC staff evaluation, while following the format of the WCAP, also addresses the RAIs in References 9 and 10. Because the fuel assembly tests were

performed by two different entities and reported in separate documents, the RAIs, although often covering similar issues, were directed toward the individual reports. Therefore, the RAI responses addressed in this SE are grouped together, as much as possible, by topic.

The NRC staff finds that debris that passes through the sump strainer that is not collected at the core inlet/bottom nozzle may be trapped within the fuel assembly. The supporting analyses used to evaluate the potential for core blockage are discussed below. The NRC staff reviewed the debris surrogates used in the head loss testing and determined that they were adequate to represent debris that could pass through the sump strainer and enter the reactor vessel. This conclusion is based on observations of strainer bypass testing and adherence in the fuel testing to NRC staff guidance for other aspects of LOCA debris generation, transport, and head loss testing. Based on the above, the NRC staff finds the methodology used to determine the fibrous debris sizing acceptable. However, licensees should verify that the size distribution of fibrous debris used in the fuel assembly testing represents the size distribution of debris expected downstream of the plant's ECCS strainer(s) *This Condition is addressed further in Section 4.0, Item number 13, of this SE.*

As discussed in Section 4.1 of the WCAP, the NRC staff finds that the spacer grids provide the most likely locations for debris capture within the core. However, the location of the debris capture within the spacer grid is not easily predicted. Debris may collect at any location within the spacer grid and, as was observed during testing, the entire grid may become filled with a debris bed. The NRC staff notes that a few strainers installed in the PWR fleet have openings larger than the 0.11 inches stated in Section 4.1 of the WCAP. Because this dimension was not explicitly used in the evaluation, the staff did not consider it a significant issue. However, the NRC staff finds that the opening size limits the size of the debris that can enter the core. The NRC staff does not agree with the statements in the WCAP that particulate and fibrous debris less than or equal to 0.04 inches will pass through the grid structures based on observations that showed that once a debris bed begins to form, debris of any size may be captured in the grids. The capture of fine debris, smaller than the openings within the fuel assembly, was observed during every fuel assembly head loss test conducted by Westinghouse and AREVA. The NRC staff issued RAI number 7 in Reference 9 to gain additional understanding of the assertion that small debris would pass through the spacer grids. In RAI response number 7 in Reference 13, the PWROG stated that the statement that small debris could not be captured will be deleted from the topical report. The WCAP does not contain the statement regarding the small debris passing through the core. This is appropriate because the fuel assembly testing resulted in prototypical capture of debris such that the effects of the smaller fibers and particulate debris were included in the measured head losses.

The NRC staff does not agree that flow paths exist between fuel rods that will limit the extent and the consequences of debris build up as flow will divert away from the area that has debris inhibiting the flow. The NRC staff's understanding is that, as debris builds in some areas of the spacer grid, flow will divert from these areas. However, the debris laden fluid will then have the potential to deposit the debris in the open areas of the grid. The NRC staff noted that testing showed that debris had the ability to fill the open volumes within the spacer grids. This was evident in one test that was designed to simulate the intersection between the corners of four fuel assemblies (creating a cross shaped gap). This test experienced similar blockage with debris despite having more open area than the single fuel assembly test design. The NRC staff finds that the debris that collects will allow flow through the bed that will provide cooling for the cladding as long as the amount of debris that reaches the fuel assembly is limited to that defined as acceptable by the fuel assembly test program and accepted by the NRC staff.

The NRC staff does not agree that the packing factor will likely be less than about 60 percent, and that the debris bed will therefore not become impenetrable under all debris loads, but finds that adequate flow to maintain LTCC will be maintained as long as the debris reaching the core inlet is maintained within the limits determined by testing and approved by the NRC staff. The NRC staff noted that the packing factor reference in the WCAP may not be applicable to debris beds that include fiber and fine particulate.

The NRC staff has not validated the assertion that boiling in the area of the blockage will occur with less than a 10 to 15 degrees Fahrenheit increase in the clad temperature over the cooling water, which will provide sufficient convective heat transfer to maintain the fuel rod a few degrees below the liquid saturation temperature based on the testing. However, calculations discussed in Section 3.4.3 of this SE have found that cooling will be maintained with debris accumulated between the fuel rods and the grid straps. Therefore, the NRC staff finds that blockage at the spacer grids will not adversely affect clad-to-coolant heat transfer if debris loading is maintained within the guidelines determined by the fuel assembly testing as accepted by the NRC staff.

The NRC staff finds that some aspects of the fuel assembly test program were conservative and that the quantities of various types of debris at limits defined by the test program are acceptable. The claimed conservatisms are discussed further below.

The NRC staff notes that the evaluations conducted in the topical report and the testing did not account for the potential for boric acid precipitation and that this issue could affect LTCC in some cases. The NRC staff concluded that for a hot-leg-break scenario at a fibrous debris limit of 15 grams per fuel assembly, LTCC would not be challenged because adequate coolant can flow through the core to maintain boric acid concentrations below the saturation limit. For the cold-leg-break or hot-leg break scenarios where the licensee wishes to justify a higher fibrous debris limit such that flow through the core is decreased, the NRC staff concluded that boric acid concentration may affect LTCC. These effects should be addressed by industry as described in Section 8 of the WCAP.

b) Evaluation of Hot-Leg-Break Case

Following a hot-leg break, the ECCS flow passes through the core, and then exits the RCS via the break. The NRC staff finds that some debris would pass through the core and exit the break. Some of this debris would likely be captured on the sump strainer and would not be available to deposit at the core inlet or in the core. As debris builds up on the strainer, this capture will become increasingly effective. Thus a test method which allows the debris to cycle through the fuel assembly multiple times without an intervening strainer, (the testing performed by the PWROG) results in a conservative quantity of debris depositing within the fuel assembly for all fuel designs that do not trap all debris on the first pass through the fuel assembly.

The NRC staff finds that, if boiling occurs following a hot-leg break, it would result in turbulence that may tend to remove debris from some areas of the spacer grids. However, boiling effects may not be significant following a hot-leg break because the high flow rates through the core may suppress boiling. Boiling would not likely occur in the lower portions of the core because the ECCS fluid is sub-cooled when it enters the core. Therefore, the NRC staff does not find that conservatism should be recognized for boiling. The NRC staff issued RAI number 19 in Reference 9 to request additional information regarding this claimed conservatism. The response to the RAI in Reference 13 states that boiling that occurs would increase the void fraction and therefore increase the driving head by decreasing the density of the fluid in the core

which would allow a higher debris load. The RAI response also references calculations that evaluated the potential for precipitation within the core and fuel heat up. The NRC staff finds with the assertion that boiling in the core would increase the available driving head for the ECCS flow into the core from an elevation head perspective, but notes that the 2-phase flow increases head-loss. The NRC staff does not view it as conservative. Also, boiling may not occur for a hot-leg break. The NRC staff reached this conclusion because any voiding would physically reduce the manometric balance between the core and the downcomer, but boiling would likely be suppressed due to higher flow rates associated with hot-leg breaks.

The NRC staff finds that following a LOCA, the fuel assemblies may bow due to the thermal transients in the fuel rods and that this bowing may distort the flow channels, making some channels larger and some channels smaller, thus allowing flow around any blockages that form at the spacer grids. However, the NRC staff does not recognize this phenomenon as being conservative with regard to the testing because the degree of bowing is not defined and it has not been demonstrated to result in openings large enough to prevent debris bed formation. Further, the fuel assembly tests have shown that debris accumulation occurs, primarily, at the fuel inlet fitting/nozzle and grid straps where bowing is not expected to be significant. This applies to both the hot-leg and cold-leg conditions.

The NRC staff finds that the testing defines an upper bound for the possible blockage that could occur at the fuel for the hot-leg break. However, the conclusions in Appendix G of the WCAP implied that testing showed that the Westinghouse fuel limit for fibrous debris is 25 grams. The NRC staff review of the test results concluded that 15 grams of fiber per fuel assembly is an appropriate debris limit for all fuel types. The single test of Westinghouse fuel with 25 grams of fiber required the flow through the fuel assembly to be reduced significantly to remain within the allowable head loss of the test facility. The NRC staff did not consider a single test with marginal results to be adequate justification for a debris limit of 25 grams of fiber per fuel assembly. The tests run with 15 grams of fibrous debris had significant margin with respect to head loss across the fuel assembly. Therefore, in the absence of additional testing and/or evaluation, the NRC staff considers 15 grams per fuel assembly to be an appropriate fiber limit for all fuel types.

c) Evaluation of Cold-Leg-Break Case

For a cold-leg break, the ECCS flow that enters the core is only that which replaces fluid lost to boiling. This flow can be less than half of the total ECCS flow to the reactor, and the remainder of the flow spills out of the RCS break. Therefore, the entire amount of the debris entrained in the coolant would not be expected to reach the core. The NRC staff finds that this could be considered a significant conservatism under the original assumption that all debris would reach the core during a cold-leg break. However, the additional testing that was conducted for the cold-leg case determined separate debris loads for the hot-leg and cold-leg breaks. The method used to determine the plant specific debris amounts that could reach the core during a cold-leg break reduce the debris amounts proportionally to the flow split between the core and the RCS break. Therefore, the flow split cannot be credited to demonstrate additional conservatism. The most recent testing showed that the debris limits for the hot-leg break case are close to the cold-leg break limits. Because less debris reaches the core for the cold-leg case the hot-leg case is limiting for all breaks. However, if licensees perform plant specific evaluations to increase hot-leg debris limits they must evaluate the cold-leg case to ensure that it is not more limiting. *This Condition is addressed further in Section 4.0, Item number 1, of this SE.*

Because many PWRs credit the mixing of the concentrating boric acid in the core with the less concentrated solution in the reactor vessel lower plenum to delay the onset of boric acid precipitation in their cold-leg break core cooling evaluations, and because debris build up at the core inlet could have a significant impact on the mixing capability, the PWROG is developing a separate program to address the effects of debris on boric acid precipitation within the reactor core. The effect of debris on core cooling can be tempered by limiting the quantity of fiber transported to the reactor core. If the hot-leg break fiber load is limited to 15-grams per fuel assembly, the amount of fiber entering the core during a cold-leg break scenario would be small (less than 7.5 grams per fuel assembly), based on the flow split between what enters the core and what exits the break. During fuel-assembly testing at cold leg break flow rates, a significant differential pressure was not detected at fiber loads below 10 grams. Therefore the maximum anticipated fibrous debris load of 7.5 grams for a cold-leg break is acceptable.. Licensees that credit debris limits greater than 15 grams per fuel assembly must evaluate the effects of additional debris on boric acid precipitation. *This Condition is addressed further in Section 4.0, Item number 1, of this SE.*

The response to RAI number 19 in Reference 9 regarding the effects of boiling on the debris bed (which is discussed above for the hot-leg break) states that the RAI response also applies to the cold-leg break. However, for the cold-leg break, the RAI response noted that the debris bed formed at the core inlet or at the first spacer grid where it would not likely be affected by boiling. Therefore, boiling would not affect the head losses. The NRC staff finds this assessment of the effects of boiling for the cold-leg case acceptable. The NRC staff concluded that boiling would not affect head losses for the cold-leg case because the debris bed forms at the core inlet and the fluid entering the core is sub-cooled to the degree that boiling will not likely occur until higher up in the core after adequate energy is transferred to the water to heat it to saturation temperature.

The NRC staff finds that sufficient information has not been provided to support the statement that during a cold-leg break, boiling will cause turbulence in the core that could move debris that is not trapped, and inhibit the deposition of a uniform debris bed within the core. Therefore, the NRC staff has no basis for determining the degree to which the deposits will be non-uniform or how much debris will be inhibited from being trapped. The NRC staff position is that a uniform bed will result in the most limiting head loss and that unless otherwise demonstrated, a uniform debris bed comprised of the total cold-leg debris load should be assumed for the cold-leg break case.

d) Evaluation of Test Program, Additional Testing, and RAI Responses

The NRC staff finds that plants that have debris quantities that are bounded by the debris limits defined by testing and approved by the NRC staff are acceptable. Because the fuel blockage testing demonstrated that adequate flow could be maintained through the debris bed, even with the bed distributed relatively evenly over the cross section of the assembly, the NRC staff finds that adequate flow can be maintained if debris loads are limited to those shown to be acceptable by the testing associated with the topical report (15 grams per fuel assembly). With fibrous debris limited to 15 grams per assembly, the head losses associated with limiting flow rates are significantly lower than the available driving head required to get coolant into the core. Because of the large margin, the NRC staff concluded that plants that maintain fiber loading within the limit will not experience reduced flow to the core.

The NRC staff finds it acceptable that industry take additional actions, including testing, to justify higher debris limits as justified for plant specific cases. However, the staff notes that several

factors will have to be accounted for in plant specific evaluations. For example, if the head losses approach the acceptance criteria, or flow reductions or evaluations are required during testing to maintain adequate driving head, sensitivity and repeatability testing will likely be required.

The NRC staff finds that the tests were usually conducted at constant flow rates and that reducing flow rates through the debris bed would generally result in decreasing the head loss across the debris bed. However, some tests that included greater than 15 grams of fibrous debris required reductions in flow rates to maintain head losses within the test limits. Several tests required the flow rates to be reduced to zero. The NRC staff also noted that the flow rate at which the bed is established has an effect on the resistance of the bed. This is discussed further below. The NRC staff finds that once the bed is formed that reducing flow through the bed will decrease head loss.

The testing did not credit alternate flow paths that bypass the core inlet that may be available in some reactor designs. Licensees that credit alternate flow paths should demonstrate that the flow paths are effective, that the flow holes will not become blocked with debris during a LOCA, and that debris will not deposit in other locations after passing through the alternate flow path such that LTCC would be jeopardized. Also, if credit for alternate flow paths leads to a boil-off condition in the core, boron precipitation issues must be addressed. *This Condition is addressed further in Section 4.0, Item number 3, of this SE.*

An additional alternate flow path discussed in the topical report is the potential for coolant to spill over the steam generator tubes in the intact loop and enter the top of the core. This scenario could occur during a hot-leg break if the head loss across the core exceeds the height of the steam generator tubes. The NRC staff finds that this is a potential flow path for coolant to enter the core, but notes that plant specific evaluations would have to assure that this is a viable flow path. The NRC staff has performed exploratory calculations which model this potential flow path for one new reactor design. The results of the study indicate that coolant preferentially flows through the steam generator tubes on the broken loop and out the break, not to the top of the core through the intact loops. Therefore, this flow path may not provide adequate core cooling.

The NRC staff finds that the amount of particulate included in the tests affect the formation of the debris bed and the resulting head loss. The NRC staff also finds that fiber is the limiting debris variable and is the only type of debris that requires a limit. This is based on NRC staff review of the test results and the fact that many p/f ratios were tested with the 1:1 ratio resulting in the limiting head losses. Industry did not conduct tests at the limiting flow rate with p/f ratios lower than 1:1. The NRC staff determined that because the head losses incurred at 15 grams were well below the acceptance criteria that it was not necessary to perform sensitivity testing at p/f ratios less than 1:1. Additionally, the NRC staff concluded that it is more likely for particulate debris to pass through the strainer than it is for fibrous debris. This effect makes it unlikely for the ratio to be less than 1:1, but the ratio is also dependent on the amount of each type of debris that reaches the strainer. Plants with lower amounts of fiber are more likely to have a higher p/f ratio. Industry has stated that some plants may attempt to show that they will have a relatively high p/f ratio thus allowing the fibrous debris load to be determined at a less limiting ratio. The NRC staff will consider such evaluations, but notes that debris generation evaluations are usually conducted to maximize debris from each source. Much particulate debris comes from unqualified coatings which may not fail or may fail late in an accident sequence. Crediting failed coatings or other potential particulate sources may be non-conservative. The NRC staff noted that as part of the discussion of conservatism, Appendix G of the WCAP used the example that

fiber is recommended to make up 15 percent of the mass of the total estimated inventory. The statement referenced the NEI 04-07 guidance for latent debris. The NRC staff concluded that the use of this statement is misleading because it points only to latent debris which is frequently a small contributor to the total debris that may reach the strainer. The examples cited here indicate that caution must be used when estimating the minimum amount of particulate debris that may be present downstream of the strainer or the potential range of p/f ratios that may occur in the reactor vessel.

For the cold-leg break, the test report stated that the limiting p/f ratio is about 45:1. Based on a review of the tests, the NRC staff concluded that this is correct. The cold-leg break case is currently not limiting because the 15 gram debris limit for the hot-leg break case is low enough to prevent significant blockage in the cold-leg case. Industry testing previously showed that the cold-leg break case would maintain acceptable LTCC if fiber entering the core was limited to 18 grams per fuel assembly. During a cold-leg break, a significant amount of injected coolant spills out the break, while during hot-leg break recovery all coolant flows into the core. Therefore, the amount of debris that reaches the core following a cold-leg break is expected to be less than following a hot-leg break. The reduction of debris for the cold-leg case is a plant specific value. If the limits for the hot-leg case are increased by plant specific evaluation, the potential for the cold-leg case to become limiting should be addressed in the analysis. *This Condition is addressed further in Section 4.0, Item number 1 of this SE.*

Based on a review of the tests conducted under varying flow rates and debris loads, the NRC staff concluded that the hot-leg break represents the limiting condition. The NRC staff finds that the highest flow rate results in the limiting head loss. Although testing was conducted to demonstrate that head losses for lower flow cases were bounded by the high flow cases, the testing was not conducted at the limiting fibrous debris load. The testing was conducted at higher fibrous debris loads (50 grams vs. 15 grams). Tests conducted at the maximum hot-leg flow rates and 50 grams of fiber had significantly higher head losses both before and after chemicals were added than the tests conducted at the lower flow rates with 50 grams of fiber.

The test program generally added all of the particulate to the test loop first followed by batches of fiber. After all fiber was added to the facility and head loss stabilized, chemicals were added. The debris addition sequence was intended to search for a thin bed similarly to strainer testing. Some tests added additional particulate debris at the end of the test, but these were not tests conducted close to limiting conditions. Although the order of debris addition was not fully explored by the test program, the staff concluded that the debris limit of 15 grams has margin to allow significant increases in head loss before LTCC would be adversely affected and that the margin combined with the number and variations of tests run shows reasonable assurance that LTCC will be maintained.

The NRC staff finds that the test facilities were capable of creating repeatable test results if the tests were controlled to an appropriate degree and that small changes in the test loops could result in significant changes in results. The NRC staff considered this an important factor when reviewing the test results and determining which results could be considered reasonable maximum debris loads. The NRC staff considered that it is important to understand why results varied among tests and test facilities. The level of understanding of these issues affects the conservatism the staff considers necessary when evaluating test results.

The NRC staff concluded that the fibrous debris limits determined by the testing and accepted by the staff were adequate, but potentially not overly conservative in all cases as discussed in

this section. In particular, the NRC staff noted that small increases in fiber above the accepted limits can result in rapidly increasing head loss values under some conditions.

The Westinghouse Fuel Assembly test report (Reference 16) made several conclusions regarding the testing. The NRC staff agreed that the tests conducted in the test facility could be used to draw conclusions regarding the impact of debris on in-vessel effects. However, the NRC staff was not able to conclude that tests conducted at the Westinghouse facility were conservative with respect to those conducted at CDI.

The Westinghouse test report (Reference 16) also stated that the Westinghouse test facility provides repeatable test results and that repeat tests are not required. Repeatability appeared to be good at a relatively high debris load. However, the NRC staff was not able to conclude that repeatability was demonstrated for the limiting debris load case because there were no tests conducted at the limiting debris loads at the Westinghouse facility.

The WCAP stated that the test program is conservative and bounds all plant types. The NRC staff finds that a 15 gram fiber limit is a conservative value for all plant types included in the WCAP as discussed above. Other test results may not be conservative.

The NRC staff concluded that the testing conducted to determine the effects of inlet pipe location provided significant information regarding the potential for changes in test results due to seemingly minor changes in test configuration. The NRC staff noted that the sensitivity tests performed to understand these effects were not performed at the debris limit, but with significantly higher fibrous debris loading. However, the NRC staff accepts that the debris limits determined at the CDI facility were the most limiting. The CDI test results provide the basis for 15 gram fiber limit.

The NRC staff finds that the head loss across a debris laden fuel assembly is a function of the debris bed structure and the velocity of the fluid through the debris bed. The report's conclusion that a head loss at a lower flow rate will be lower than that at a higher flow rate with the same debris load may be misleading and the statement that beds formed at lower flow rates have lower resistance is not correct. Based on the evaluation of several tests which included changes in flow to help characterize the debris bed flow resistance, the NRC staff concluded that debris beds formed at lower flow rates have higher resistance to flow. The tests that included flow changes made through a fully formed bed showed that differential pressure varied approximately by the flow velocity to the1.6 power. However, when debris beds were formed at lower flow rates, head losses were higher than would be predicted for beds formed at higher flow rates followed by flow reductions. The NRC staff also noted that for some tests that incurred high differential pressure, reductions in flow rate after the bed was formed did not substantially reduce the head loss across the fuel assembly. The NRC staff postulates that the debris bed characteristics vary with the flow rate at which the debris bed is formed. Therefore, there is no direct correlation between flow velocity and debris quantity that can be applied to calculate a pressure drop associated with a constant debris load and resultant debris beds built at various flow rates. The NRC staff's conclusion is that understanding flow rate, and any potential changes in flow rate are important when evaluating fuel assembly head losses. The NRC staff also concluded that within the population of tests conducted, the high flow tests had been shown to be limiting. Because of the unpredictable behavior of the fuel assembly head loss under various conditions, the NRC staff does not accept the use of any equation or model to predict head loss, but instead accepts the test results from the fuel assembly tests. *This Condition is addressed further in Section 4.0, Item number 4, of this SE.*

The NRC staff concluded that the accepted test program limits are applicable to UPI plants based on the similarity of the fuel design and the lower flow rates associated with the design of the plants. The direct injection into the upper plenum likely has some advantage with respect to debris deposition in the core. However, the NRC staff concluded that the 15 gram per fuel assembly limit should be maintained for UPI plants until plant specific evaluations provide adequate justification for a higher limit.

The NRC staff noted, in RAI number 15 of Reference 9, that for the cold-leg test evaluation, Revision 1 of the WCAP stated that only the head loss across the P-grid or first spacer grid at the bottom of the fuel assembly was compared against the available driving head. The RAI stated that this approach was not acceptable because testing had shown that some head loss occurred at spacer grids above these locations. Because the total head loss from all spacer grids exceeded the allowable cold-leg break head loss, additional testing was conducted to determine acceptable cold-leg debris loading. Related to this area, RAI number 18 in Reference 9 requested additional information to justify that the hot-leg tests conducted bounded the cold-leg tests. The NRC staff noted that the cold-leg break conditions could actually be more limiting than the hot-leg conditions. The PWROG responded by conducting separate tests for the hot and cold-leg conditions to define separate debris loads for each case. The NRC staff review of the later test results found that each case had been treated appropriately. The basis for this conclusion is provided in the NRC staff trip reports documented in References 22 through 28. The NRC staff generally found the response to RAI number 18 to be acceptable, but did not agree with the concept that numerical analyses demonstrated that, even if a large blockage occurs, decay heat removal would continue. The NRC staff position is that if a plant maintains its potential debris load within limits defined by the testing, LTCC will be maintained. Any debris amounts greater than those tested and accepted by the NRC staff should either be mitigated or be justified on a plant-specific basis. The latest tests found that if a plant is within the limit for the hot-leg condition the cold-leg condition will also have adequate LTCC ensured. If plant specific evaluations increase debris limits in the future, cold-leg conditions may become important. *This Condition is addressed further in Section 4.0, Item number 5, of this SE.*

The NRC staff finds that the debris surrogates used in the fuel assembly testing were adequately representative of the debris that could be present in the core following a postulated LOCA. The particulate and microporous debris was sized per strainer head loss test guidance in Reference 29. The fibrous debris was sized to approximate the size distribution of fibers that had passed through strainers during strainer bypass testing. However, the staff did not agree that the debris addition sequences would result in the limiting head losses in all cases. In Reference 9, RAI number 16 requested that the addition of one-half of the microporous insulation at the beginning of the debris addition sequence and one-half of the microporous insulation at the end of the debris addition sequence be justified. In response to the RAI, the PWROG conducted several additional tests with microporous insulation and determined that it behaved similarly to particulate debris. Additional testing added all microporous insulation early in the debris addition sequence. Based on a review of the microporous insulation sensitivity testing conducted by Westinghouse, the NRC staff concluded that for fuel assembly head losses, microporous debris behaves similarly to particulate debris with similar size characteristics. RAIs number 1 and number 7 of Reference 9 and number 5 of Reference 10 also address this issue. The NRC staff generally accepts (with exceptions noted below) the conclusions of these RAI responses which were transmitted in References 11 and 12.

The NRC staff noted that the addition of Cal-Sil to the fuel assembly testing generally resulted in lower head losses than if no Cal-Sil were added. This is not expected based on strainer head loss testing which has shown that Cal-Sil can result in significantly increased head losses. RAI

number 1 of Reference 10 and RAI number 7 of Reference 9 requested additional information regarding this phenomenon. The responses in References 11 and 12 stated that the inclusion of Cal-Sil as a debris source in the testing consistently resulted in reduced head losses for both Westinghouse and AREVA fuel, that the debris beds with and without Cal-Sil appeared to be similar, and that further investigation into the phenomenon was not conducted. The reason for this behavior was not identified. One possible explanation is that the additions of Cal-Sil to the testing increased the particulate-to-fiber ratio which was observed to result in a decrease in head loss during sensitivity testing. Although the behavior is unexpected, the NRC staff accepts that the addition of Cal-Sil to the fuel assembly tests resulted in reduced head loss and, therefore, the results of testing that did not include Cal-Sil are acceptable.

One statement in the response to RAI number 7 of Reference 10 that the NRC staff does not accept is that once a specific pressure differential across a bottom nozzle is reached, flow will pass through the gaps between fuel assemblies. This statement was made in Reference 12 in response to a NRC staff question about a specific fuel inlet filter design that seems likely to incur higher head losses than with similar debris loadings. Some testing has shown that these gaps can become fully blocked. The NRC staff position is that testing should be used to define the maximum allowable debris load for various plant configurations, debris loads, and flow rates. This finding is true for all fuel designs and not just the fuel design that was the subject of the RAI.

The NRC staff concluded that the flow velocities that were used during the fuel assembly testing were bounding with respect to the conditions being tested. Cold-leg flow rates were based on the maximum boil off rate expected at the earliest time of switchover from injection to recirculation. Flow rates were prorated to an average for a single fuel assembly. Westinghouse also conducted tests specifically for plants that had lower hot-leg flow rates. These tests also used an average flow rate based on a single fuel assembly. The NRC staff finds using a flow rate averaged over all of the assemblies to be acceptable even though flow in some portions of the core would be higher than others. The acceptance of the average flow rate is based on the NRC staff's expectation that while flow to different fuel assemblies within the core may vary before debris reaches them, that debris will be deposited within fuel assemblies in proportion to the flow through the assembly. As debris is collected within an assembly, the resistance to flow in that assembly will increase, thus redirecting flow to assemblies that previously received lower flow. The net effect is to equalize the distribution of debris over the core inlet, resulting in a more balanced head loss and flow rate across the core inlet. Under the highest allowable debris loading conditions simulated during testing, flow would be expected to be similar in all the fuel assemblies in the core. If debris loading is within the NRC staff accepted WCAP acceptance criteria, the head losses will be low enough to allow adequate flow to ensure LTCC.

The NRC staff evaluated the fuel protective filters and the other components of the partial fuel assemblies used in the testing. The NRC staff concluded that the mock fuel assemblies adequately represented the fuel assemblies in the plant and that the bottom nozzles tested would result in limiting head losses with respect to those expected for the PWR fleet. The NRC staff's conclusion is based on comparative testing conducted for the various bottom nozzles as documented in References 11 and 12, and review of the physical aspects of the fuel assemblies. Some less common bottom nozzles or fuel inlet filter designs resulted in higher or lower head losses than the more widely used components. These less common fuel filters were evaluated separately by Westinghouse and AREVA. The NRC staff reviewed the test results for the current testing and the results of previous industry testing and concluded that the testing of Westinghouse fuel using the standard P-grid resulted in the definition of debris loads that may be applied to the fuel assemblies with inlet filters discussed in Reference 7, based on the

similarity of the inlet filters. Specifically, the standard P-grid is considered to be the limiting debris filter when compared to the Alternate P-grid and Guardian Grid based on the results of industry testing in support of the WCAP and a physical comparison of the filters.

WCAP-17057-P, Revision 0 (Reference 7), states that fuel designs with no inlet filter are bounded by the testing conducted for other Westinghouse fuel designs with the exception of the UPI plants that are discussed in Section 9 of the WCAP. The NRC staff does not agree that this is necessarily the case based on testing conducted by AREVA for an inlet filter that had a relatively low profile (area that blocks flow). These tests resulted in unexpectedly high head loss values. The NRC staff does not accept that the testing for the remainder of the Westinghouse fuel inlet filter designs bounds the fuel with no inlet filter installed. If Westinghouse supplies fuel having no inlet filters to plants other than UPI plants, the debris acceptance limits should be developed through a program similar to that used to develop the acceptance limits in the WCAP. *This Limitation is discussed further in Section 4, Item number 6, of this SE.* This conclusion is based on the concepts discussed below regarding the differences in behavior between AREVA and Westinghouse fuel.

The test program conducted has been limited to the 17x17 fuel assembly. There are additional fuel assembly designs in use in PWRs, such as 15x15, 16x16, and 14x14 fuel assemblies. The 17x17 fuel assembly design is the most limiting design from the perspective of debris capture and blockage due to the fact that the design has the smallest rod-to-rod gap clearance, thus presenting the smallest gap in which to capture debris. Therefore, the NRC staff accepts that the testing conducted with the 17x17 mock fuel assemblies is representative or conservative with respect to current fuel designs that have lattices with fewer fuel rods. However, new or evolving fuel designs having different inlet fittings or grid straps may exhibit different debris capture characteristic. Therefore, as stated in Reference 13, response to RAI number 22, utilities and vendors will have to evaluate fuel design changes in accordance with 10 CFR 50.59 to ensure that unacceptable debris blockage in the core will not occur. *This limitation is discussed further in Section 4, Item number 6, of this SE.*

The NRC staff evaluated the debris amounts and combinations used during the fuel assembly testing. During the review of the test results, the NRC staff noted that there was a correlation between the head loss and the amount of particulate debris added to the test. RAI number 3 of Reference 9 and number 4 of Reference 10 requested additional information regarding the relationship between fuel assembly head loss and the particulate-to-fiber ratio used in the testing for both the hot and cold-leg conditions. The initial testing was conducted with particulate debris amounts considered to bound the potential maximum particulate load for most PWRs. The NRC staff noted that head losses tended to increase with decreasing particulate loads as well as increasing fibrous debris loads. The NRC staff also noted that some of the tests were terminated prior to the addition of the full amount of fiber because head loss was increasing significantly as fiber was added. RAI number 2 of Reference 9 requested additional information to justify why the original fibrous debris limit was appropriate considering that some tests on Westinghouse fuel appeared likely to exceed the head loss criteria if all of the fibrous debris had been added. RAI number 6 of Reference 10 requested information regarding debris loading combinations for testing of AREVA fuel. The final round of testing was conducted at the limiting p/f ratios and appropriate debris limits were identified.

The PWROG conducted additional testing on Westinghouse and AREVA fuel to better define the appropriate maximum fibrous debris loading and better understand how particulate-to-fiber ratios and other debris combinations affect head loss. Letters from the PWROG dated February 9, 2010, (Reference 11) and June 24, 2010 (Reference 12) discuss these issues as responses

to the corresponding RAIs. The updated testing validated that fuel assembly head loss is related to fibrous debris amount and particulate-to-fiber ratio, and that the debris limits and limiting p/f ratios are different for the hot-leg and cold-leg flow rates. Testing for hot-leg conditions determined that low p/f ratios (about 1:1) result in the most limiting head losses while for the cold-leg condition the limiting ratio is higher (about 45:1). The initial follow-up testing also found that for Westinghouse fuel, the fibrous debris limit for cold-leg conditions is much lower than for the hot-leg conditions. The testing illustrated that chemical precipitates can have a significantly greater effect on head loss when the ratio of particulates to fiber included in the testing is relatively low.

Because of anomalies discovered during testing, the NRC staff requested that additional tests be conducted. The further testing found, as discussed above, that the hot leg case is actually limiting and that all fuel types included in the program have a similar debris limit. The limits described in the RAI responses are not considered valid by the NRC staff. The current NRC staff position is that the fibrous debris limit per fuel assembly is 15 grams. Licensees may perform plant specific evaluations to increase this limit.

The Westinghouse test report (Reference 16) made an additional observation that fiber, by itself does not provide adequate resistance to flow to result in limiting head losses. This conclusion was based on tests conducted at 3 gpm (cold-leg conditions), two of which included particulate debris and one that did not. Because the 3 gpm tests have a limiting p/f ratio of about 45:1, the NRC staff expects that a test conducted with no particulate debris would result in a lower head loss. The NRC staff also noted that the tests were conducted with fiber loads significantly greater than the limiting loads. The test program did not explore p/f ratios below 1:1 for high flow rate test (hot-leg condition) for which 1:1 was identified as the limiting ratio. The assertion that a debris bed with no particulate, and that p/f ratios below 1:1 need not be explored, is accepted by the staff for the fibrous limit of 15 grams per fuel assembly because of the large margin between available driving head and measured head loss. If individual plants attempt to increase fiber loads, p/f ratios below 1:1 should be evaluated. *This Condition is discussed further in Section 4, Item number 1, of this SE.*

The Westinghouse test report (Reference 16) discussed a test performed under a separate program to evaluate boric acid precipitation. The test simulated a reactor vessel after a cold-leg break. The test included heating of the fluid defined by a decay heat curve and introduced debris into the coolant. There was reportedly no debris accumulation at the bottom nozzle or grids. The NRC staff is interested in this type of test, but did not receive adequate information regarding the test to determine whether the test could be considered prototypical. Therefore, the results of the test are not credited by the NRC staff in this SE.

The PWROG test program was based on the hypothesis that the results of the testing would be similar for Westinghouse and AREVA fuel designs. After the bulk of the testing had been completed on Westinghouse fuel, the NRC staff requested that testing be performed on AREVA fuel to validate that the different fuel designs have similar head loss behavior for similar debris loads and flow conditions. The first AREVA fuel assembly validation testing showed that, at cold-leg conditions, the head loss in the Westinghouse fuel tests bounded the AREVA fuel head loss. However, when hot-leg testing was conducted, the AREVA fuel incurred much higher head losses than anticipated. The PWROG performed testing to determine a valid debris limit for AREVA fuel for hot-leg conditions. The NRC staff reviewed the results of the testing and requested that additional hot-leg and cold-leg testing be performed on AREVA fuel to ensure that the behavior of the fuel was well understood. The results of the testing, including hot and cold-leg debris limits, were transmitted via Reference 12. The differences between the behavior

of Westinghouse and AREVA fuel are presented below, including a potential explanation for the behavior. The acceptance of the test results is based on the testing that is evaluated in detail throughout this SE.

The AREVA and Westinghouse fuel behave similarly under cold-leg, or low flow conditions. The fuel types also behaved similarly, during the second round of testing, under hot-leg conditions when the debris contained a high ratio of particulate to fibrous debris. However, when the p/f ratio was reduced, the debris collection within the fuel assembly became dissimilar. Under high flow and low p/f ratios, the Westinghouse fuel collected debris in a manner that resulted in relatively low head losses when compared to head losses observed in AREVA fuel tested under similar conditions. Both fuel types collect debris in a similar manner for cold-leg (low flow) tests.

The results of some of the AREVA hot-leg case testing were significantly different from the results of the Westinghouse test under similar flow conditions. The fuel vendors and the NRC staff theorized that the differences in test results are due to differences in fuel design. However, they recognized that differences between the test facilities could have contributed to the difference in the test results. The NRC staff requested that the PWROG and the fuel vendors resolve this uncertainty. The NRC staff suggested that one or both fuel vendors' test-articles be tested in the other vendor's test facility (cross test) to validate that the differences in behavior are due to fuel design and not due to the test facility. The vendors performed cross tests with the AREVA assembly at the Westinghouse test facility and the Westinghouse assembly at CDI.

The Westinghouse test report (Reference 16) compares tests of Westinghouse fuel tested at Westinghouse (CIB54) and Westinghouse fuel tested at CDI (1-W-FPC-0811). Both tests were conducted with 25 grams of fiber and 25 grams of particulate. The test performed in the CDI test loop resulted in higher head losses after chemicals were added to the loop. It is also noted that the amount of chemical debris required to attain the maximum head loss in the CDI loop was significantly higher. The test report concludes that the CDI test loop results in more conservative fiber limits, but that the testing conducted at Westinghouse is also valid. The conclusion is based on the argument that the test protocols contain adequate conservatism with respect to the plant condition. The NRC staff position is that because the conservatisms have not been demonstrated, the conservative fiber limit should be used.

The AREVA test report (Reference 17) also evaluates tests conducted at the two facilities. This report makes two comparisons, one between two tests and one among three tests. The test report concluded that the two test loops behave differently. The report also concluded that there is no discernable difference between the behavior of Westinghouse and AREVA fuel when tested under similar conditions. The test report stated that the graphs presented were for information only. Additionally, the test report did not identify the test numbers for the tests that were compared or include adequate information regarding the test conditions or results for the staff to determine whether the comparisons were valid, especially for the comparison of the three tests. For example, the NRC staff could not determine whether the debris loads or flow rates were similar. For the comparison of the two test assemblies in a single test facility, the NRC staff was able to determine test conditions relatively well. The conclusions of the test report were not well supported. However, based on other test comparisons, the NRC staff was able to conclude that the test loops behave differently under some of the conditions tested.

Based on the comparisons by both test reports (References 16 and 17), the NRC staff concluded that the CDI test rig provides more conservative results and that a 15 gram fiber limit is appropriate for all fuel types. The basis for the accepted fiber limit is documented throughout this SE.

e) Evaluation of Mixed Cores and Lead Test Assemblies

The debris limits defined for Westinghouse fuel and AREVA fuel are the same. However, in the future some difference may be identified between fuel designs that results in varying fuel dependent limits. Therefore, a means of prorating the debris limits to reactor cores that contain fuel from both vendors is provided by the PWROG in Reference 14 and paragraph 3.0 of Reference 13. As stated in Reference 13 and 14, one-third of the fuel assemblies are replaced during a typical refueling outage. Therefore, it is possible for a mixed core to contain one-third to two-thirds of one vendor's fuel. The guidance in Reference 14 requires that if two-thirds of a mixed core is comprised of fuel assemblies having the lower debris limit of the two fuels, then, the lower debris limit should be applied to the entire core. However, if two-thirds of the mixed core is comprised of the fuel having the higher debris limit, then the debris limit, per fuel assembly, for the entire core should be determined by summing the products of the number of each type of fuel assembly in the core and the debris limit for that assembly, and dividing that sum by the total number of assemblies. The NRC staff accepts this method based on (1) the open-lattice design of the fuel assemblies that allows the flow of coolant across fuel assemblies, (2) the similar head losses for fuels at low fibrous debris loads, absent chemicals, (3) the expectation that most fibrous debris will transport to the core prior to the arrival of significant quantities of chemical precipitates, and (4) the modeling presented in Reference 18 that demonstrates that cross flow can provide adequate cooling. Based on the above, the NRC staff concluded that debris entering the core would collect in fuel assemblies relatively evenly up to the debris limit for the fuel with a lower allowable debris amount. Based on test results, the NRC staff concluded that the fuel with the lower debris limit would likely collect more fibrous debris per assembly than is allowed by its acceptance criterion. When precipitates arrive in the core, the fuel with the lower debris limit could become blocked. However, the fuel with the high debris limit would have less than its allowable debris load. Therefore, flow would be able to enter the core through the high-debris-limit fuel and cross flow into the fuel assemblies having the lower debris limit, thus providing adequate cooling. Any credit for increased debris loads based on mixing of fuel assembly designs with higher allowable debris loads within a single core shall be evaluated as discussed above to ensure that the assumptions above are valid. It should also be noted that the above discussion is for hot-leg breaks and assumes that the cold-leg debris loads are either the same for both fuel types, or are not limiting with respect to debris amounts.

Under some conditions, the reactor may contain a full core comprised almost entirely of one vendor's fuel with a limited number of fuel assemblies, called Lead Test Assemblies (LTAs), from another vendor. (The LTAs are installed to allow a plant to evaluate the performance of a new fuel type.) The AREVA test report (Reference 8) states that the limited number of fuel assemblies installed under these conditions will not adversely affect LTCC. For this case, the staff finds that the limiting debris load should be based on the majority of the fuel assemblies installed in the core, without consideration of the test assemblies. This criterion is applicable only if LTAs are limited to a maximum of eight assemblies per core. The NRC staff's acceptance of the use of the overall core limit when LTAs are installed is based on the small effect that the limited number of assemblies can have on the core considering that cross flow will occur to allow the coolant to flow across fuel assemblies to provide adequate cooling to all fuel assemblies.

f) Evaluation of RAIs Specific to AREVA Testing Not Previously Discussed

RAI number 2 and number 3 of Reference 10 specifically address issues identified with the first AREVA test report (Reference 8). RAI number 2 identified an apparent discrepancy between the graphical representation of test results and tabulated test results in the report. The response to RAI number 2 (Reference 12) explained that the tabular data listed only the final test head loss which was generally less than the maximum head loss observed during the testing. The NRC staff considered that the RAI was resolved based on the clarification provided in the RAI response.

RAI number 3 of Reference 10 was directed at a specific test conducted with the TRAPPER Fine Mesh® inlet filter. The test report attributed head loss decrease when chemical precipitates were added to air entrainment (from the chemical addition) disturbing the debris bed. The RAI requested an evaluation of the head loss behavior had the debris bed not been disturbed. The response (Reference 12) stated that it was expected that the addition of chemical precipitates would likely have had a small impact on head loss based on the results of other testing with similar debris loads. The NRC staff reviewed other similar tests and concluded that the effect of chemical precipitates would likely be small. The NRC staff also considered that this fuel is being phased out, as discussed below.

RAI number 7 of Reference 10 requested additional information on the behavior of the TRAPPER Fine Mesh® inlet filter when loaded with Cal-Sil or microporous debris. The response to RAI 7 (Reference 12) discussed the results of testing on other fuel inlet filter designs and how the results of those tests could be applied to the TRAPPER Fine Mesh® inlet filter. The NRC staff did not agree that the results of testing on other fuel inlet filter designs could be applied to the TRAPPER Fine Mesh® inlet filter because of the significant differences in the designs. The response also stated that if the inlet filter were to become completely blocked, flow could pass through the gaps between fuel assemblies. As discussed above, based on the results of more recent testing, the NRC staff does not agree that the gaps between fuel assemblies have been shown to allow adequate flow to maintain LTCC.

The responses to RAI number 3 and number 7 in Reference 10 further stated that the use of the TRAPPER Fine Mesh® inlet filter is very limited and that it is no longer in production. Two PWR units had one-third of the core fueled with the TRAPPER Fine Mesh® inlet filter as of fall 2010, and these were in the final cycle prior to removal from the core. Based on review of fuel testing using various AREVA inlet filters, including limited testing of the TRAPPER Fine Mesh® inlet filter, the NRC staff concluded that the maximum debris loads defined for the rest of the test population may be applied to the TRAPPER Fine Mesh® inlet filter until those assemblies are removed from the cores of the plants utilizing them at the next refueling outage. This conclusion is based on the limited amount of TRAPPER Fine Mesh® fuel deployed in any plant and the staff's expectation that the TRAPPER Fine Mesh® fuel would accumulate debris at one elevation in the fuel assembly (as did the Fuelguard®), and would therefore exhibit similar head loss behavior. AREVA did not perform an extensive test program for the TRAPPER Fine Mesh® inlet filter because it is being phased out. If additional use of a fuel design using the TRAPPER Fine Mesh® inlet filter be desired, further testing will be needed as discussed in Section 4, Item number 6, of this SE.

3.4.3 Cladding Heat-up Calculations

The PWROG investigated the effects of a potential build-up of debris in the volume between the fuel rod and spacer grid such that the volume would be completely filled with debris and

cladding surfaces could become coated with concentric layers of oxide, crud, and chemical precipitate. The effect of debris build-up on fuel cladding temperature is addressed in Sections 4.3.1 and 4.3.2 of the WCAP. The sections describe evaluations performed to (1) determine the fuel cladding surface temperature within a fuel grid in a post-LOCA recirculation environment when the rod is plated with debris and (2) determine the fuel cladding surface temperature between spacer grids in a post-LOCA recirculation environment when debris is deposited on the cladding surface. Detailed discussions of the evaluation methods are included in Appendices C and D of the WCAP.

Section 4.3.1 of the WCAP describes an ANSYS® finite element model of a single fuel rod that was created to predict fuel cladding heat up within a spacer grid. The model was cut down to a "1-quarter pie piece" to allow for the preservation of symmetry of the fuel rod. To conservatively model convection from the fuel rod surface, the clad was divided into 20 zones. No convection was assumed to occur at the planes of symmetry. Only radial heat transfer was considered. A mesh size of 0.05 inches was used for the model.

A constant heat flux was assigned to the entire inner surface of the cladding, and convection heat transfer, with a constant convection coefficient, was assigned to the entire outer surface of the rod assembly. Four values were used to parametrically simulate the range of thermal conductivities for the postulated deposition on the fuel clad surface. The thermal conductivity values were 0.1, 0.3, 0.5, and 0.9 BTU/(h-ft-°F). These thermal conductivities were applied to a range of deposition thicknesses ranging from 5 mils to 50 mils. The ANSYS® model simulated a 12-foot long, 0.36-inch diameter fuel rod having a cladding thickness of 0.0225 inches. Spacer grids were modeled as 2.25 inches long for the large grids, and 0.475 inch long for the smaller grids.

Section 4.3.1 of the WCAP states that the maximum calculated clad temperature occurs within the spacer grid. A minimum thermal conductivity of 0.1 BTU/(h-ft-°F) and a debris thickness of 0.050 inch were used to calculate the maximum cladding temperature behind a grid as 474 degrees Fahrenheit. This calculated temperature is well below the 800 degrees Fahrenheit LTCC acceptance basis identified in Appendix A. Thus, the clad surface temperature acceptance basis of 800 degrees Fahrenheit identified in Appendix A is satisfied.

Section 4.3.1 of the WCAP states that the temperatures calculated with this model are conservatively high because the calculation assumed no flow through the debris in the grid, contrary to what was observed during fuel assembly testing. Therefore, some coolant flow is expected to pass through the debris buildup within the spacer grid and cool the clad surface.

Section 4.3.2 of the WCAP describes an analysis of fuel rod heat-up between the spacer grids (grid straps) assuming that the cladding is surrounded by concentric layers of oxide, crud and chemical precipitates, with no gaps between them. The source of heat was decay heat in a post-LOCA environment with a two-phase liquid/vapor environment in the core. This analysis used the generic resistance form of the heat transfer equation for a radial coordinate system. The thermal conductivity values used in this evaluation were 0.1, 0.3, 0.5, and 0.9 BTU/(h-ft-°F). In all cases, the maximum clad surface temperatures calculated were less than 560 degrees Fahrenheit. Thus, the clad surface temperature acceptance basis of 800 degrees Fahrenheit is satisfied for debris thickness of up to 50 mils.

The fuel rod diameter used in the calculations described above was 0.36 inches. To demonstrate the applicability of these results to all PWR fuel designs, two sets of sensitivity

calculations were performed using the following fuel rod specifications:

- 0.420 inch outer diameter (OD) fuel rod at 0.388 kilowatt per foot (kW/ft) power value
- 0.416 inch OD fuel rod at 0.383 kW/ft power value

The maximum clad surface temperatures reported in Section 3.2.3.1 of the WCAP is 714 degrees Fahrenheit. Thus, the clad surface temperature acceptance basis of 800 degrees Fahrenheit is satisfied for debris thickness of up to 50 mils for the 0.36-inch, 0.42-inch, and 0.416-inch diameter fuel rods.

NRC Staff Evaluation

During the long-term cooling period following a LOCA, local hot spots might form within a reactor core as a result of boiling. For some reactor safety system designs, boiling might occur for an extended period as a result of relatively low ECCS flows or relatively high ECCS water temperatures. Local hot spots might result from the plate out of material directly on the fuel rods or as a result of material trapped between the spacer grids and the fuel rods.

Mock fuel assembly testing as reported in References 8, 16, and 17 with fiber, particulate and chemical precipitates, demonstrated that debris could accumulate at the fuel inlet nozzles and the spacer grids to form a uniform debris bed. Further, for break scenarios for which coolant flow to the core only replenishes boil-off, liquid flow to flush debris from the core is not available. Therefore, there is a potential for debris to accumulate between the fuel rods on the spacer grids and to deposit onto the fuel cladding surface and, thereby, diminish core cooling. The PWROG addressed these concerns by performing analysis to determine the fuel cladding temperature in a post-LOCA sump circulation environment.

To assess a maximum clad temperature under worst-case debris deposition in a single spacer grid/fuel rod configuration, the following assumptions are made:

1. A uniform debris layer thickness of 50 mils is assumed on the cladding, and,

2. The debris layer is assigned an effective thermal conductivity of 0.1 BTU/(h-ft-°F) for a fibrous debris bed. The NRC staff concludes that this value has been demonstrated to be conservative in the response to RAI number 15 (Reference 4).

Under these conditions, the maximum clad temperature behind a grid would be 474 degrees Fahrenheit (Reference 1, Table 4-2). Further, in response to RAI number 14 in Reference 4, a maximum clad temperature of 738 degrees Fahrenheit was calculated for a uniform debris layer thickness of 110 mils. These are conservative estimates of clad temperature if the gap between a spacer grid and a fuel rod were to become completely filled with debris because:

1. A conservatively small value of conduction through the debris bed identified in the response to RAI number 15 (Reference 4) is used,

2. The calculation does not account for circumferential heat transfer about the debris bed which would form in the spacer grid between the dimples and springs and the corners of the spacer grid, and,

3. Convection of heat by the flow of coolant through the debris bed is neglected.

In addition to conduction through any plated out or accumulated material, heat must also traverse the oxide layer built up on the surface of the cladding during normal operation and during the earlier phases of the LOCA event. The oxide will have a thermal conductivity lower than the original cladding material. Westinghouse described the approach to calculating the oxide layer thickness and heat transfer in Reference 4. Westinghouse proposed that the assumed cladding oxide thickness for input to LOCADM be the peak local oxidation allowed by 10 CFR 50.46, equal to 17 percent of the cladding wall thickness.

The PWROG (Reference 4) then used as an example an analysis based on a cladding thickness of 0.0225 inches with a reduced metal thickness of 0.0187 inches and an oxide layer of 0.006 inches. The calculation presented in the WCAP was based on an oxide thickness of 0.004 inch with an oxide thermal conductivity of 0.1 BTU/(h-ft-°F). Increasing the thickness of the cladding oxide layer to 0.006 inches resulted in a temperature increase of 2 degrees Fahrenheit over the 0.004 inch oxide layer temperature. Based on the information provided in Reference 4, the NRC staff finds that this is a conservative approach to evaluating the effect of the oxide layer on cladding heat-up. Therefore, the NRC staff finds its use acceptable for this application.

The WCAP describes three separate analyses performed to examine the effects of (1) debris buildup underneath grid straps, (2) oxide, crud and chemical precipitate buildup on the fuel rods, and (3) precipitate plate-out and crud deposited on the surface of a fuel rod on fuel cladding temperature. These three methods are:

(1) Cladding Temperature Due to Collection of Debris at Spacer Grids

Section 4.3.1 and Appendix C of the WCAP describe the method used to evaluate hot spots due to debris build-up between the fuel rod and spacer grid. This method uses the ANSYS® Mechanical software to calculate cladding temperatures. The ANSYS® Mechanical software provides solutions for linear and nonlinear and dynamic analysis for a variety of engineering problems. For the cladding heat up calculations, only the thermal solution capabilities were used. The software was used in a manner consistent with standard industry practices for ANSYS® mechanical software. Therefore, the NRC staff finds its use acceptable for this application.

Using the ANSYS® Mechanical software, the WCAP provides sample calculations for debris thicknesses of up to 50 mils which might form between a spacer grid and an enclosed fuel rod. Based on the dimensions (e. g., rod diameter, pitch, and inlet nozzle area) of the various fuel assemblies in service, the distance between a fuel rod and the enclosing spacer grid may be larger than 50 mils. The WCAP stated that the minimum clearance between two adjacent fuel rods, including an allowance for the spacer grid thickness, is greater than 100 mils (two times the gap between rod and spacer grid plus the thickness of spacer grid). Therefore, a 50-mil debris thickness on a single fuel rod is the maximum deposition allowed to preclude touching of the deposition of two adjacent fuel rods with the same thickness of deposition. The 50-mil thickness is the maximum acceptable deposition thickness before bridging of adjacent fuel rods by debris is predicted to occur. In the worst-case scenario (minimum gap and maximum deposition), local bridging might occur but would not be expected to lead to significant localized fuel rod heating due to the conservatisms described in the analysis. These include:

- The heat-up calculation assumes a heat input at the onset of circulation from the ECCS sump, at which time deposits from debris in the coolant have not formed

(WCAP, Appendix C). The heat input is lower by the time a 50-mil deposit could form.

- The analysis assumes that no convection occurs under the grids in the fuel assembly (WCAP, Appendix C). Because some convection would occur, the actual cladding temperature would be lower.
- The analysis neglects longitudinal heat transfer.
- The maximum calculated cladding temperature with 50 mils of deposit and neglecting bridging is 474 degrees Fahrenheit, demonstrating substantial margin to the 800 degrees Fahrenheit acceptance criterion.

For current fuel designs, the minimum clearance between the cladding and the spacer grid is about 40 mils (Reference 18). This condition occurs where the springs and dimples of the grid contact the fuel rod. The maximum clearance between the cladding and the spacer grid occurs along the diagonal of a grid cell and is about 110 mils. Thus, if a spacer grid were to become completely filled, the radial thickness of the debris on the outside clad would vary from about 40 to 110 mils about the circumference of a fuel rod.

In Section 4.3.1 of the WCAP, the ANSYS® code is used to evaluate the resulting fuel cladding temperatures for fuel rods covered with debris having thermal conductivities as low as 0.1 BTU/(h-ft-°F). In Reference 2, RAI number 15, the NRC staff questioned whether insulating fiber beds can have thermal conductivities considerably smaller than this value. In response, the PWROG stated (Reference 4) that they had contacted PCI, the owner of the NUKON® brand low density fiberglass insulation material commonly used in PWR containment buildings, to provide information regarding the effect of wetting on the thermal conductivity of the NUKON® insulation. PCI stated they had no data for the thermal performance of wetted insulation because wetted insulation ceases to perform its insulating function because the thermal conductivity of the wetted insulation approaches the thermal conductivity of water. Further, the PWROG provided an analysis (Reference 4) of the thermal conductivity of wet versus dry fiber and showed that the insulating property of the dry material is five times, or more, as effective as wetted fiber material. The NRC staff has reviewed the submittal and finds this assessment acceptable. The NRC staff believes this conclusion to be valid for all fiber types expected to be in containment as debris. Based on a review of the information presented, the NRC staff accepts this rationale as justification for the assumption that a thermal conductivity for the insulating material equivalent to its dry conductivity is conservative. Therefore, the evaluation is acceptable to the NRC staff.

(2) Cladding Temperature Due to Oxide, Crud and Precipitate Buildup on Fuel Rods

Section 4.3.2 and Appendix D of the WCAP describe the method used to calculate cladding temperature due to debris build-up on the fuel rods. The analysis uses a steady-state cylindrical heat conduction model to calculate fuel rod cladding temperature as a result of crud deposits on the surface of a fuel rod at locations which are not within a spacer grid. Sample calculations are provided for crud thicknesses up to 50 mils and crud thermal conductivities as low as 0.1 BTU/(h-ft-°F). Appendix E to the WCAP is the basis for calculation of thickness, composition, and heat transfer characteristics of crud. The deposition process described is based on work such as that detailed in Reference 30.

Calculation of the heat transfer through the crud deposit is done using LOCADM with either a plant-specific estimate based on fuel examinations that have been performed for the plant fuel, or using limiting values from industry fuel examinations. The default version of the model uses a crud thickness of 100 microns for second and third cycle fuel, and 50 microns for first cycle fuel. The thickest crud measured in PWR fuel is 127 microns.

The limiting crud thermal conductivity quoted for PWR fuel, 0.3 BTU/(h-ft-°F), falls at the lower end of the measured calcium-rich crud on boiler tubes that range from 0.29 to 0.55 BTU/(h-ft-°F). Sodium aluminum silicate based crud deposits have thermal conductivities as low as 0.11 BTU/(h-ft-°F). Thus, the WCAP concludes that using a limiting thermal conductivity of 0.1 BTU/(h-ft -°F) is conservative. As stated in (1) above, based on a review of the information provided, the staff accepts the WCAP conclusion that a thermal conductivity of 0.1 BTU/(h-ft-°F) is conservative. Therefore, the evaluation is acceptable to the NRC staff.

(3) Cladding Temperature Due to Precipitate Plate-Out on Fuel Rod Surface

Section 7 and Appendix E of the WCAP describe the method used to calculate the thickness of deposits on the fuel cladding and the method used to calculate the resulting cladding temperature. The method uses steady-state slab heat conduction model in the LOCADM computer code to calculate fuel rod cladding temperature as a result of crud deposits on the surface of a fuel at locations which are not within a spacer grid. The LOCADM computer code also calculates the amount of suspended and dissolved material in the core, and the plate out of that material on the fuel rods.

In Reference 2, the NRC staff raised questions concerning the calculation of plate out of dissolved and suspended materials including fibrous debris upon the surface of fuel rods undergoing long-term boiling. The PWROG responded (Reference 4) that the deposition of small fibers that do not dissolve but are small enough to be transported through the sump strainer and into the core cannot be ruled out. They also stated that the quantity of transported fines is expected to be small compared to both the total amount of debris and the amount of debris that dissolves or corrodes. The PWROG stated that an estimate of the effect of the fiber on deposit thickness and fuel temperature can be accounted for in LOCADM by use of a "bump-up factor" applied to the initial debris input. Therefore, the evaluation, with bump-up factor, is acceptable to the NRC staff.

The PWROG presented a comparison of the LOCADM slab conduction methodology with the cylindrical conduction methodology of WCAP-16793, Appendix D, showing that similar results are obtained for thin crud layers. In Reference 2, the NRC staff questioned whether for thick crud layers, the inaccuracies of the slab model for modeling cylindrical heat transfer might make this methodology unusable. In response, the PWROG provided results (Reference 4, Table 34-1) of a comparison of a slab geometry heat transfer model versus a cylindrical geometry heat transfer model. The slab geometry resulted in a temperature drop across the deposit that was 61 degrees Fahrenheit greater than for the cylindrical geometry, thus indicating that LOCADM is conservative. The NRC staff accepts this conclusion based on a review of the calculations presented.

The PWROG has selected a cladding temperature of 800 degrees Fahrenheit as the acceptance basis for long-term cooling. The PWROG stated (Reference 4) that autoclave test data has demonstrated that oxidation and hydrogen pickup would be acceptable at and below

the 800 degrees Fahrenheit temperature and that the reduction in cladding mechanical performance would be small. The cladding specimens in the autoclave tests were for fresh material. In Reference 2, the NRC staff requested data for specimens which have undergone prior exposure to LOCA heat-up and quench conditions. The PWROG responded (Reference 4) by referring to autoclave test data and a literature review that indicates that susceptibility to localized accelerated corrosion occurs at temperatures in excess of 800 degrees Fahrenheit. The PWROG stated that it does not expect cladding properties to degrade due to a 30-day exposure to a temperature of 800 degrees Fahrenheit. The NRC staff accepts that the autoclave results are sufficient to justify this temperature, and the NRC staff accepts a temperature limit of 800 degrees Fahrenheit as the long-term cooling limit. If a licensee's final LOCADM calculation shows a cladding temperature that exceeds this value, cladding strength data must be provided for oxidized or pre-hydrided cladding material temperature in excess of that calculated. *This Condition is addressed further in Section 4.0, Item number 7, of this SE.*

3.4.4 Summary of Collection of Debris on Fuel Grids

Section 4.4 of the WCAP states that debris that does not collect at the core inlet will pass through the fuel assembly bottom nozzle and enter the core region where it may lodge in some of the smaller clearances in the spacer grids. However, the debris buildup will not impede LTCC, because the extent of the buildup is limited by the spacer grid design and debris that does collect will have some packing factor that will allow "weeping" flow through the resulting debris bed, as evidenced by a review of the test data that indicated that coolant continued to pass through the debris bed.

The PWROG stated that the collection of debris observed during fuel assembly testing represents an upper bound of the debris accumulation because of conservatisms in the testing process and the buildup of debris at spacer grids in an operating plant would be considerably lower with a low likelihood of blockages at any singular spacer grid. Further, the blockages that do occur can be treated as localized blockages.

For localized blockages, calculations described in Sections 4.3 of the WCAP demonstrate that the maximum surface temperature of cladding between two grids during recirculation from the containment sump following a postulated LOCA would be less than 800 degrees Fahrenheit. For the 0.360 inch diameter fuel rod, the maximum temperature with 50 mils of precipitate on the clad OD is calculated to be less than 560 degrees Fahrenheit. For the 0.416-inch or 0.422-inch rods, the maximum temperature with 50 mils of precipitate deposited on the clad OD is calculated to be less than 715 degrees Fahrenheit. The PWROG stated that these temperatures are conservatively high, as they assume a decay heat level at the time of ECCS switchover to recirculation from the containment sump (20 minutes after initiation of the transient). At this point in the transient, there has been no time to build a layer of precipitate. Chemical products have had little time to form and the concentrations are therefore low, and coolant from the sump is just being introduced into the RV by the ECCS. As decay heat continues to decrease, the calculated clad surface temperatures for a specific thickness of precipitate would also decrease.

NRC Staff Evaluation

The NRC staff finds acceptable the statement that debris that is not deposited at the core inlet can be captured at spacer grids within the core. This conclusion is based on results of fuel assembly testing that showed that, under certain flow conditions, fiber, as well as particulate

and chemical debris was deposited at spacer grids throughout the test assembly. The NRC staff also finds that when the quantity of debris transported to the core is within the acceptance limits specified in Section 10 of the WCAP, as approved by NRC staff in this SE, the flow of coolant required for LTCC can be maintained. This conclusion is based on the results of fuel assembly testing that established the debris limits based on the ability to maintain the required coolant flow rate to ensure LTCC.

The NRC staff finds that the PWROG has adequately demonstrated that deposition of debris between the spacer grids and fuel rods or deposits of 0.050 inch or less precipitates on the fuel rods will not result in the fuel cladding temperature exceeding 800 degrees Fahrenheit. However, as stated in Section 7 of the WCAP, plants will need to prepare a plant-specific LOCADM evaluation using plant-specific design inputs. Each licensee's GL 2004-02 submittal to the NRC should state the peak cladding temperature predicted by the LOCADM analysis. *This Condition is discussed in Section 4.0, Item number 7, of this SE.*

Therefore, plants that satisfy the NRC staff-accepted debris acceptance criteria of 15 grams per fuel assembly as defined in Section 10 the WCAP, criteria contained in other sections of this SE, and the conditions and limitations of this SE can state that they that they have adequately addressed GL 2004-02 with respect to the downstream effects.

3.5 COLLECTION OF FIBROUS MATERIAL ON FUEL CLADDING *(TR WCAP-16793-NP, Revision 2, Section 5)*

The PWROG cited report NEA/CSNI/R (95)11 (Reference 31) in its investigation of testing performed to evaluate the potential for fibrous material to collect on fuel cladding. The following observations from the report are cited:

1. From Section 5.4.2.1 of Reference 31, little adherence of fibrous material to clad surfaces was observed, and the material that did adhere was loose and easily removed. What was observed to adhere to clad surfaces was the binder used to make fiberglass. This binder, however, was observed to carry with it very limited fibrous debris. The report noted that much of the binder is quickly driven off of the fiberglass due to the heat associated with normal operating conditions. These observations were determined to be applicable to both NUKON® and Knauf ET® Panel.

2. Section 5.4.2.3 of Reference 31 provided observations regarding fibrous material collection at fuel grids. It was noted that fibrous debris will collect on grids, but that a pure fibrous bed is porous and water will pass through it.

The PWROG concluded that the above test results indicate that fibrous debris, should it enter the core region, will not tightly adhere to the surface of fuel cladding. Thus, fibrous debris will not form a "blanket" on clad surfaces to restrict heat transfer and cause an increase in cad temperature. Therefore, adherence of fibrous debris to the cladding is not plausible and will not adversely affect core cooling.

<u>NRC Staff Evaluation</u>

Testing of fibrous debris on heated surfaces is described in Reference 32. In these tests, an immersion heater, which was heated electrically or with a torch, was quenched in a slurry mixture of glass wool and associated insulation blanket material that included the adhesive binder. Initial heater temperatures ranged from saturated to 2200 degrees Fahrenheit. In other

tests the electric heater was immersed within the slurry for hours. Fiber-to-metal or fiber-to-fiber adherence was not observed. The small amount of fiber which was present on the heater surface after the test was easily brushed away with a soft brush. Some adherence of the binder material to the heater was observed with the maximum fibrous debris layer two fiber diameters thick.

The tests conducted and reported in Reference 32, referred to in Reference 31, were conducted in three ways: 1) a rod heated to 2200 degrees Fahrenheit was quickly quenched in a slurry of chopped blanket fibers in distilled water, 2) a stainless steel strip was placed in the slurry and heated to nucleate boiling and held for three hours, and 3) a stainless steel strip was placed in the slurry and heated to film boiling and held for three hours. The slurry was made from commercially available (1978 era) high temperature fiberglass insulation with a density of 1 to 2.5 pounds per cubic feet, and a service temperature of up to 1000 degrees Fahrenheit. The tests provided qualitative indications that the fibrous material would not adhere significantly to the heated surface.

The NRC staff concluded in its SE on Reference 32 (the SE is included in Reference 32) that the fiber deposits which occur on rods heated to film or nucleate boiling temperatures will not accumulate in sufficient thickness or quantity to measurably change either the flow of coolant or the heat transfer characteristics of the heated surface.

The WCAP does not address the potential for blockage due to fuel swelling or rupture. Following a large-break LOCA, fuel rods in the core may swell or rupture leaving sharp edges at the rupture locations and diminish channel flow area. Debris may collect in the restricted channels and at the rough edges of the rupture locations. Swelling and rupture of the fuel rod cladding during design basis LOCAs is one of the phenomena which licensees evaluate under 10 CFR 50.46. Therefore, in Reference 2, the NRC staff asked the PWROG to evaluate the possibility of excessive blockage being produced by the combination of swelling and rupture and debris collection. Such blockage might produce the occurrence of the hot spots above the blockage location.

The PWROG stated (Reference 4) that based on work performed and reported in Reference 33, only about 10 percent of the fuel rods in the core would experience cladding rupture. Therefore, the PWROG concluded: "wide-spread blockage due to swelling and rupture would not be expected in a large-break LOCA scenario." Accumulation of significant debris in the balloon and burst region would have a low probability of occurrence. The NRC staff found acceptable the conclusion for CE, B&W, and Westinghouse three-loop and four-loop plants, based on the information provided in Reference 4. Westinghouse two-loop UPI plants have significantly different flow patterns during re-flood. The general flow pattern is for the upper plenum to drain into the lower power regions of the core, while the hotter regions are cooled by a bottom-up flow. Thus, the UPI plants are expected to encounter a circulation pattern with both up flow and down flow. This would make the accumulation of debris in the hotter regions of the core less likely. The NRC staff reviewed the conclusion in Reference 4 that the blockage due to swelled or burst fuel cladding will not cause unacceptable core heat-up (PCT > 800 degrees Fahrenheit). Based on NRC staff experience with confirmatory analyses and the review of the LOCA analyses of a representative number of operating PWRs, NRC staff considers the cited PWROG statement to be correct. Based on information received, the NRC staff concludes that the debris limits accepted in this SE will not cause significant blockage with cladding rupture or swelling.

3.6 PROTECTIVE COATING DEBRIS DEPOSITED ON FUEL CLADDING SURFACES *(TR WCAP-16793-NP, Revision 2, Section 6)*

The PWROG identified zinc-rich primers, epoxies, and non-epoxies (typically applied to small equipment applied by the original equipment manufacturer) as the primary types of coating materials typically contained in PWR containment buildings. These coatings may be transported to the reactor vessel and either deposit directly on cladding surfaces, or collect within fuel grids or behind debris beds within the fuel grids. The PWROG evaluated these coatings to have a negligible effect on LTCC because:

1. The amount of non-epoxy coatings used inside a PWR containment building is small and therefore, has negligible contribution to post-LOCA PWR chemistry effects.

2. PWROG testing in support of WCAP-16530-NP-A (Reference 21) has demonstrated that zinc contributes little to the generation of PWR corrosion products post-LOCA and that any zinc particulate deposited on a fuel rod would have a much higher thermal conductivity than assumed in the LOCADM analysis. Therefore, zinc-rich primers have negligible contribution to post-LOCA PWR containment pool dissolved solids that could precipitate and deposit on the fuel cladding.

3. Chemical resistance testing under simulated design basis accident (DBA) conditions has demonstrated that epoxy coating systems are chemically inert and contribute only a small amount of leachate to the recirculating coolant and therefore, epoxy coatings are evaluated to have negligible contribution to post-LOCA PWR chemistry effects (response to RAI number 2 in Section D of Reference 21).

Based on the above, the PWROG concluded that protective coatings debris would have a negligible effect on the post-LOCA chemistry of a PWR containment pool, and on post-LOCA LTCC. The PWROG also evaluated protective coatings debris (particulate) to have a negligible effect on post LOCA LTCC.

<u>NRC Staff Evaluation</u>

The NRC staff reviewed the information in the WCAP and other sources of relevant information. The NRC staff finds the following:

1. Testing described in response to RAI 2 (Reference 21) demonstrated that the predominant coating of concern (epoxy) retains its structural integrity at temperatures up to 350 degrees Fahrenheit and, therefore, is not expected to fail outside the zone of influence (ZOI). The NRC staff notes that plants may contain non-DBA qualified epoxies that would be expected to fail outside the ZOI. However, due to the limited quantity of non-qualified epoxy (Reference 34), the NRC staff treats this material as non-epoxy for the purpose of this SE.

2. Testing documented in NUREG/CR-6916, "Hydraulic Transport of Coating Debris," (Reference 35), has shown that epoxy paint chips tend to sink rather than transport.

3. When immersed in fluid at temperatures less than 350 degrees Fahrenheit, epoxy does not form adhesive bonds with other materials.

4. Given the conservative assumptions used in the plant-specific chemical-effects evaluations, the staff finds that the amount of non-epoxy coating used inside a PWR containment building is small (Reference 34) and will not significantly contribute to fuel deposits.

5. Testing of zinc rich primers reported in References 21 and 36 has shown that they will not leach a significant amount of chemicals. The staff finds that zinc particulate deposited on the fuel will have thermal conductivity values much higher (65 BTU/(h-ft-°F)) than that assumed for deposits in LOCADM analysis.

6. Although not credited in the WCAP, some coating particulate that reaches the sump strainer will probably be filtered by a sump strainer debris bed.

Therefore, the NRC staff accepts the position of the PWROG that protective coatings will not have a significant effect on clad-to-coolant heat transfer.

3.7 CHEMICAL PRECIPITATES AND DEBRIS DEPOSITED ON FUEL CLAD SURFACES *(TR WCAP-16793-NP, Revision 2, Section 7, Appendices E and F)*

Information regarding how post-LOCA chemicals and containment debris combine to form potential impediments to cooling water flow and heat transfer is found, principally, in WCAP-16530-NP-A (Reference 21) and the WCAP (Reference 18). References to substantiating evidence are provided in each of these documents as well as in the responses to RAIs for WCAP-16530-NP and the WCAP (Reference 18).

In order to best explain how potential chemical effects could occur, the LOCA sequence of events is presented from a chemical perspective. Following the reactor coolant system blow-down during the initial phase of the LOCA, borated water from storage tanks is pumped into the reactor core through the cold-legs (for most plants) and through the UPI points (for a few plants). Simultaneously, some of this water is sprayed into the containment building, thereby cooling the released hot liquid/steam to maintain overall building pressure to less than the plant design limit. The initial water source used for injection into the reactor vessel and for containment spray is of very high purity (stored in tanks after purification) and with appropriate boron concentration (approximately 2,500 ppm boron) for emergency core cooling.

Depending on plant design, there are several ways that chemicals are added to adjust the pH following a LOCA. Plants using sodium hydroxide (NaOH) to adjust pH inject it into the containment spray during the initial spray period. Thus, the containment spray is initially at the high end of the design pH range, and the spray pH decreases once all NaOH has been added to the spray system and the pool volume has diluted the NaOH concentration. The pool that forms on the containment floor will initially be acidic, due to the borated water from the RCS and storage tanks (e.g., refueling water tank) that has spilled out the break. The pH of that pool and the water collected in the sump will increase to greater than 7.0 over time as the higher pH spray collects at the containment floor and adds to the pool volume.

Plants using trisodium phosphate (TSP) to adjust pH store the granular TSP powder in baskets on the floor of the containment building. The TSP dissolves (over approximately a two-hour period) as the water level on the floor of the containment building rises with time. In this case, initial RCS blowdown and containment spray are acidic until the switch from injection phase to the recirculation phase and the dissolution of TSP has adjusted the pH above 7.0. Some plants store sodium tetraborate (STB) in baskets on the containment floor for post-LOCA pH control,

and their overall system pH trend with time would be similar to that for a plant with TSP. Plants designed with ice condensers have STB frozen into the ice to adjust the post-LOCA containment building pool pH. Following a LOCA, STB would be added to the pool as the ice melts, thereby increasing the pool pH to greater than 7.0.

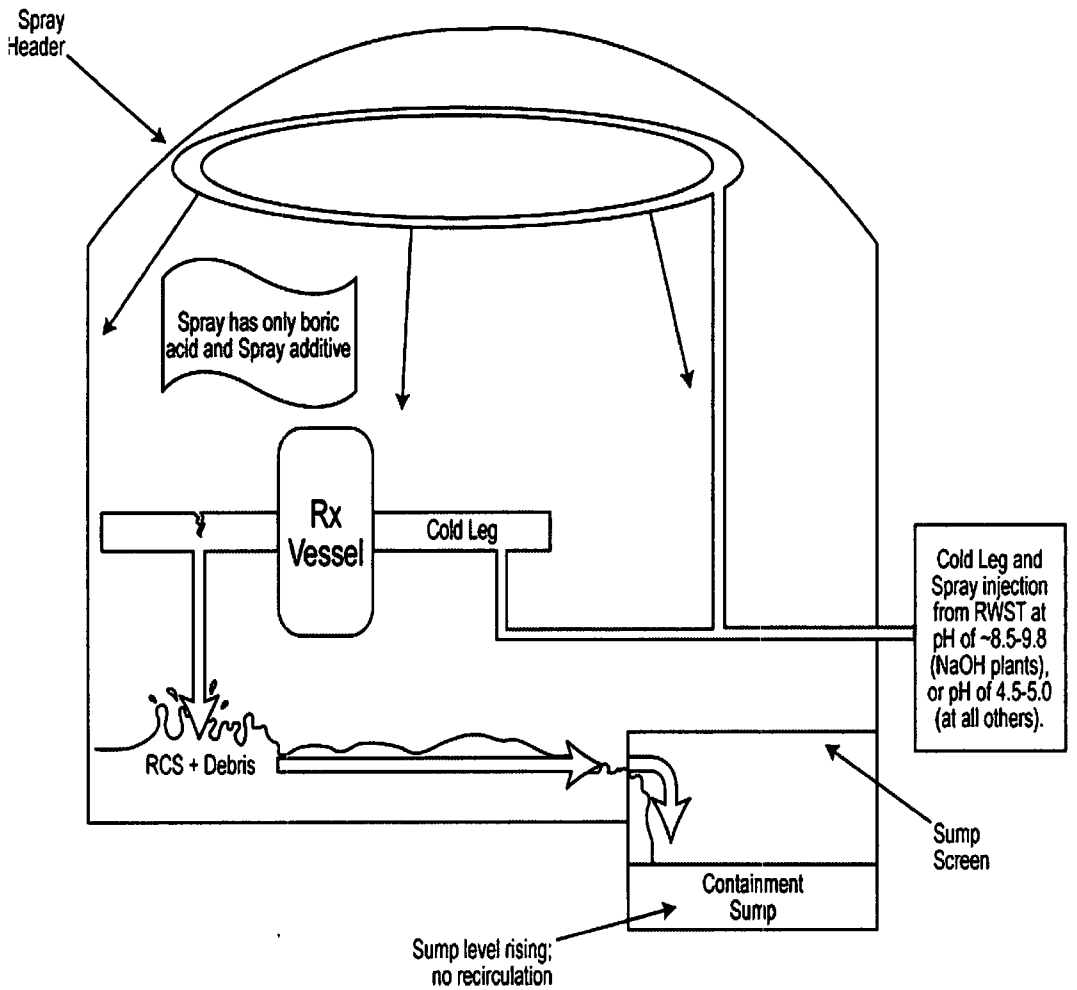

Figure 3.7-1
Initial Injection into RCS Immediately Following Break
(Schematic is representative of NaOH injection plants).

Figure 3.7-1, developed for this SE, shows a schematic of the initial phases of the LOCA with chemicals and flow indicated. This schematic is representative of some plant designs. Over time, the liquid on the containment floor builds up dissolved and suspended chemical concentrations as a result of the reaction between liquid spilling out the break (initially acidic and gradually trending higher to a pH greater than 7.0), plant debris, and wash-down from the spray. Reactions of the containment spray with containment materials take place at a range of pH values depending on the plant design, the chemical used to adjust pH, and whether this chemical is added directly into the spray system or dissolves in the pool.

Potential chemical reactions between the post-LOCA containment building environment and plant materials that may produce chemical products are considered as chemical effects in GSI-191. The Integrated Chemical Effects Tests (ICET) (Reference 36), as well as testing reported in WCAP-16530-NP-A, showed that certain combinations of materials and representative post-LOCA containment environments could produce hydrated, amorphous precipitates that would need to be considered in GSI-191 chemical effect evaluations. Figure 3.7-2, developed for this SE, shows the change in flow path for the core cooling water as well as the source of the contaminants to the core following the switch to recirculation flow.

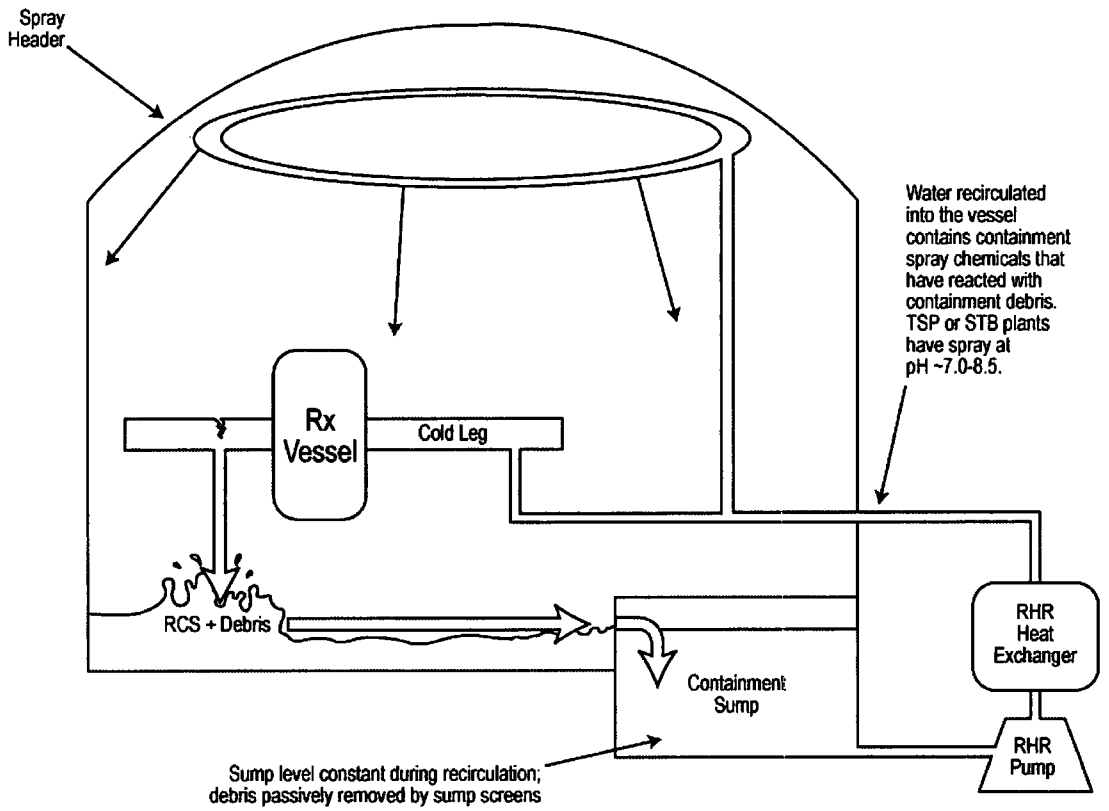

Figure 3.7-2
Injection into Cold-Leg Using Previously Injected Water Captured in the Containment Sump.

(Note: Features such as the presence of a sump pit and strainer design are plant-specific)

Although this SE primarily focuses on the WCAP, there is a link between the WCAP and WCAP-16530-NP-A since the chemical model contained in WCAP-16530-NP-A is used to develop the potential source term of species that may enter the reactor vessel. WCAP-16530-NP-A provides a method for evaluating plant-specific chemical effects in a post-LOCA environment, including guidance for how to prepare surrogate chemical precipitates that may be used in strainer head loss tests. The NRC staff reviewed WCAP-16530-NP-A, and the NRC staff SE is available in ADAMS at Accession No. ML073520891. WCAP-16530-NP-A,

however, does not explicitly address potential chemical effects that may occur in the reactor vessel.

The WCAP evaluates potential chemical effects that may occur downstream of the sump strainer in the reactor vessel. The materials tested in WCAP-16530-NP-A included:

1. commercially pure aluminum and galvanized steel,
2. calcium silicate insulation,
3. NUKON™ fiberglass,
4. other fiberglass - Temp Mat™,
5. Interam™ E-class insulation,
6. powdered concrete,
7. mineral wool insulation,
8. microporous insulation (e.g., Min-K™), and
9. fire-retardant material (e.g., FiberFrax™).

WCAP-16530-NP-A describes a number of dissolution tests conducted to examine the chemical behavior of various materials found in the sump environment. Sampling times for the dissolution test were set at 30 minutes, 60 minutes, and 90 minutes. The results of the WCAP-16530-NP-A test program are consistent with previous work such as the ICET program and show that:

1. The predominant materials leached from containment materials are:
   - aluminum ions
   - silicates
   - calcium ions

2. The predominant chemical precipitates formed are:
   - aluminum (oxy) hydroxide
   - sodium aluminum silicate
   - calcium phosphate (for plants using trisodium phosphate for pH control)

It is possible that other silicate materials may be generated (e.g., calcium aluminum silicate or zinc silicate), but their contribution, based on the referenced studies, will be small (contributing less than five percent of the total mass) relative to the predominant precipitates.

NRC Staff Evaluation

The WCAP-16530-NP-A model considers the release rates of aluminum, calcium and silicate, as these provide the greatest masses of materials that can become insoluble and impacts of other materials are negligible (Reference 21). Given a source term of material from the

WCAP-16530-NP-A model, the NRC staff reviewed the methodology used by the PWROG to determine that these materials:

1. would not deposit on fuel surfaces to the extent that heat transfer is unacceptably low, and
2. would not block flow through the fuel channels should the scale materials deposited become dislodged by spallation during fuel cool down.

In evaluating the potential for plate-out of dissolved or suspended chemical compounds on the fuel surface, the WCAP methodology assumes that all of the dissolved species and compounds resulting from the WCAP-16530-NP-A assessment are transported through the containment sump screen to the reactor vessel. This material represents the source term in the WCAP for evaluating plate-out of scale-forming materials on the fuel cladding. The NRC staff finds this source term assumption to be acceptable since the chemical source term is based on WCAP-16530-NP-A testing, and it is conservative for the reactor vessel fuel analysis to assume that no precipitate settles on the containment floor, no precipitate becomes trapped in a filtering debris bed covering the sump strainer, and material does not deposit in other locations downstream of the strainer (e.g., heat exchangers, reactor vessel lower plenum).

Although the NRC staff finds the use of the chemical model spreadsheet contained in WCAP-16530-NP-A to be acceptable for determining the chemical source term for LOCADM, a Limitation and Condition was provided in the SE for WCAP-16530-NP (ADAMS Accession No. ML073520891) related to the aluminum release rate. The WCAP-16530-NP-A chemical model aluminum release rate is based, in part, on a fit to ICET data using an averaged 30-day release. Actual corrosion of aluminum coupons during the ICET 1 test, which used sodium hydroxide (NaOH), appeared to occur in two stages; active corrosion for the first half of the test followed by passivation of the aluminum during the second half of the test. Therefore, while the 30-day fit to the ICET data is reasonable, the WCAP-16530-NP-A model under-predicts aluminum release by about a factor of two during the active corrosion phase of ICET 1. This is important since the in-core LOCADM chemical deposition rates can be much greater during the initial period following a LOCA, if local conditions predict boiling. As stated in WCAP16530-NP-A, to account for potentially greater amounts of aluminum during the initial days following a LOCA, a licensee's LOCADM input should apply a factor of 2 increase to the WCAP-16530-NP-A spreadsheet predicted aluminum release, not to exceed the total amount of aluminum predicted by the WCAP-16530-NP-A spreadsheet for 30 days. In other words, the total amount of aluminum released equals that predicted by the WCAP-16530-NP-A spreadsheet, but the timing of the release is accelerated. Alternately, licensees may choose to use a different method for determining aluminum release but licensees should not use an aluminum release rate equation that, when adjusted to the ICET 1 pH, under-predicts the aluminum concentrations measured during the initial 15 days of ICET 1. *This Condition is addressed further in Section 4.0, Item number 8, of this SE.*

If a licensee uses plant-specific refinements to reduce the chemical source term calculated by the WCAP-16530-NP-A base model, the licensee should provide technical justification demonstrating that the refined chemical source term adequately bounds the postulated plant chemical product generation. *This Condition is addressed in Section 4.0, Item number 9, of this SE.*

The WCAP uses various heat transfer computer programs (ANSYS® Mechanical Software and WCOBRA/TRAC WCAP-12945-P-A) and a commercially available calculational software package (MATHCAD) for estimating the effects of the plate out of dissolved materials on the increase in fuel clad temperature. The WCAP relies on the LOCADM code for its final assessments since the LOCADM calculations address non-uniform chemical deposition due to variation of core power and boiling.

The starting assumption for the LOCADM model with respect to chemical effects is that all the dissolved and suspended chemicals pass through the containment sump screen and into the reactor core. This is a conservative assumption because it maximizes the amount of chemicals available to cause deleterious effects.

The LOCADM model also assumes that some of the fibrous material from destroyed insulation is not removed by the sump strainer and that this material also passes on to the reactor core area. The mass of fiber passing through the strainer is determined on a plant-specific basis, based on bypass testing. LOCADM assumes instantaneous chemical participation of this fiber. Therefore, in the LOCADM analysis, the fiber bypass quantity is converted to a mass of fiberglass and then to an equivalent mass of elements (calcium + aluminum + silicon) that is immediately available to be deposited. This increase in the mass of dissolved chemicals is compared to the original mass of dissolved chemicals determined by the WCAP-16530-NP-A calculations (calcium + aluminum + silicon) and a percent increase is calculated. This increase is on the order of one to two percent, and is referred to as a "bump-up factor." The staff finds this approach acceptable because the chemical contribution of bypassed debris is considered and because the physical effects of bypassed debris are determined by separate testing.

These two chemical sources are then used in the plant-specific application of LOCADM. Given the potential plant-specific chemical source term in the reactor vessel, LOCADM determines the amount of scale that deposits on the fuel over time and then calculates maximum fuel clad temperature. An assumption that is very important to the LOCADM calculations is the coefficient of thermal conductivity for the chemical deposits. In order to determine an appropriate thermal conductivity coefficient for the LOCADM calculations, two different thermodynamic equilibrium based codes were used to assess the chemical species that may form in the post-LOCA reactor vessel environment. Westinghouse performed an analysis using the HSC program by Outokomptu, and AREVA performed similar analysis with the OLI StreamAnalyzer Version 2.0.43 program by OLI Systems, Inc. These thermodynamic equilibrium codes were used to evaluate potential differences in the predicted species and to support the choice of a limiting thermal conductivity value for a chemical deposit that may form on the fuel. Using the chemical species predicted by these thermodynamic equilibrium analyses, a lower-bound thermal conductivity value was selected for the LOCADM analysis in the WCAP to minimize heat transfer and maximize the temperature rise on the fuel surfaces. A chemical deposit thermal conductivity value of 0.11 BTU/(h-ft-°F) was selected based on the possible formation of a postulated sodium aluminum silicate scale. A thermal conductivity value of 0.11 BTU/(h-ft-°F) is the minimum thermal conductivity value reported for sodium aluminum silicate scale. For comparison, the thermal conductivity of dry fiberglass insulation is approximately 0.05 BTU/(h-ft-°F), and, with eight percent of its mass wetted, it increases to approximately 0.1 BTU/(h-ft-°F). In Reference 2, RAI number 15, the NRC staff questioned if there were any materials from the thermodynamic predictions for fuel clad surface deposits which could have lower thermal conductivity values. The PWROG responded (Reference 4) stating that 0.11 BTU/(h-ft-°F) was a bounding thermal conductivity value reported for any of the postulated species that could form a scale deposit on the fuel clad surface. Information provided by the PWROG in RAI response number 34 (Reference 4) showed thermal conductivity coefficients of representative calcium-based boiler scale deposits that were in the 0.3 to 0.5 BTU/(h-ft-°F) range, and the thermal conductivity of glass was reported as 0.59 BTU/(h-ft-°F).

Since the LOCADM calculations do not consider the presence of large debris, the NRC staff questioned whether small pieces of insulation ("fines") incorporated into a deposit could result in a lower thermal conductivity value than the 0.11 BTU/(h-ft-°F) assumed for a sodium aluminum silicate scale. The PWROG responded (Reference 4) stating that because core temperatures will have decreased by the time the ECCS switches from injection to recirculation mode (which is the time when the first fibrous debris could bypass the sump screens and enter the core) the temperature of the core is insufficient to cause melting of the fiberglass or other fibrous material. Therefore, the presence of fiber fines would not create a different type of scale other than that

predicted by the thermodynamic models. The PWROG further stated (Reference 4) that although dry fiberglass has a lower thermal conductivity than the 0.11 BTU/(h-ft-°F) assumed for the chemical deposit, a fiber deposit would be porous and would allow water to fill in the porosity. Since water has a much higher thermal conductivity than air, the overall thermal conductivity for a deposit containing fiberglass would be bounded by the assumed 0.11 BTU/(h-ft-°F) value. This reasoning is supported by literature (Reference 37) that indicated the fiberglass thermal conductivity constant increases by a factor of two with an eight percent volume of water incorporated into its structure. This is also consistent with insulation manufacturer recommendations to change insulation if it is wetted since the heat conduction through the insulation increases; in other words, it is no longer an effective insulator.

Based on the above discussion, the NRC staff finds that the 0.11 BTU/(h-ft-°F) thermal conductivity value assumed for deposition of scale and particulate represents an acceptably low value to help achieve a conservative prediction of fuel clad temperature increases due to chemical deposits. Since the assumed deposit thermal conductivity has a significant effect on the heat transfer analysis, the use of a value less conservative (greater) than 0.11 BTU/(h-ft-°F) for sodium aluminum silicate scale needs to be justified. *This Condition is discussed further in Section 4.0, Item number 10, of this SE.* If plant-specific calculations use a less conservative thermal conductivity value for scale (i.e., greater than 0.11 BTU/(h-ft-°F)), the NRC staff expects the licensee to provide a technical justification for the plant-specific thermal conductivity to the NRC staff. This justification should demonstrate why it is not possible to form a sodium aluminum silicate scale or other scales with conductivities below the selected plant value.

Given the potential chemical source term and using a conservative value for thermal conductivity, LOCADM calculates deposit growth over time. The default initial oxide and crud thicknesses assumed by LOCADM are based on the fuel age and the limiting values that have been measured at modern PWRs. Since the boiling deposition mechanism results in the most rapid deposit growth and forms the most tenacious deposits, LOCADM assumes that all deposition occurs through the boiling process if conditions at a core node predict any boiling. The amount of scale calculated to be deposited under boiling assumes that 50 percent of the water present at the clad surface boils and all solutes transported into the deposit by boiling are deposited locally, as liquid evaporates, at a rate proportional to the steaming rate. Subsequent plate-out of solids, once boiling subsides, is estimated (from RAI Set number 2, RAI response number 8 of Reference 18, Appendix I) to be 1/80th of the solids deposition rate during boiling based on the temperatures encountered at the fuel. Once formed, deposits are assumed not to thin by flow attrition, dissolution, or spall. The sample LOCADM calculation in the WCAP included a 3188 megawatt thermal power PWR with high fiber (7000 ft^3) and a large quantity of calcium silicate insulation (80 ft^3). In Reference 3, the NRC staff questioned what additional effect the existing clad crud film and oxide scale (from three cycles) would have on the LOCADM calculations. The PWROG responded (Reference 4) that the sample LOCADM calculation, for the conditions stated above, including initial fuel clad oxide and crud, showed the maximum chemical scale thickness calculated over 30 days was 0.010 inches (10 mils). The maximum clad surface temperature after the start of recirculation was 324 degrees Fahrenheit, which meets the acceptance criteria of 800 degrees Fahrenheit discussed in Section 3.1.2 of this SE.

Validation of the LOCADM Code was performed by several comparisons. For example, LOCADM was tested against SKBOR, a safety code used to predict boron build-up in the core for a cold-leg break with cold-leg injection. The LOCADM analysis predicted that the boric acid concentration would increase to 23.53 weight percent, the hot-leg switchover point, in 7.6 hours. SKBOR, version 7, predicted that 8.0 hours would be required to reach the same concentration.

LOCADM was also used to predict laboratory tests (Reference 38) involving an electrically heated rod placed into a calcium sulfate solution. In this test, a calcium sulfate solution near saturation entered a tube at 80 °C and was heated causing precipitation on the heat transfer surface. The temperature of the heat transfer surface was monitored with time as the calcium sulfate precipitated. The heat fluxes were high enough to cause boiling within the deposits (Reference 38). The fouling resistance was calculated and plotted. The LOCADM calculated deposition rate was equal to the highest deposition rate recorded experimentally and was about five times higher than the measured deposition rate for most of the test.

The NRC staff finds the LOCADM Code validation acceptable since LOCADM has been shown to be in reasonable agreement with the SKBOR Code for boric acid concentration and to conservatively predict deposits compared to a laboratory experiment with a heated rod in calcium sulfate solution.

Since LOCADM does not directly account for fiber fines bypassing the sump screen, the NRC staff also questioned (in Reference 3) how possible effects from fibers depositing in the core are assessed. To model potential local hot spots, heat transfer analysis was provided in Appendix D of the WCAP assuming heat transfer in the radial direction only (i.e., ignoring any axial heat transfer) and using a chemical scale thermal conductivity of 0.1 BTU/(h-ft-°F). These calculations showed that for a chemical scale thickness of 0.050 inches (50 mils) that formed "instantaneously" at the start of recirculation, the maximum fuel clad surface temperature for a fuel rod diameter of 0.36 inches is 560 degrees Fahrenheit. Additional analyses were performed for larger diameter fuel rods, 0.416 inch and 0.422 inch OD rods. The predicted peak clad-oxide interface temperature was less than the acceptance basis value of 800 degrees Fahrenheit in each case. The NRC staff finds this analysis to be acceptable since the assumptions of instantaneous chemical precipitate formation, heat transfer only in the horizontal plane (radial direction), and the assumed thermal conductivity for chemical scale are conservative.

The NRC staff also questioned whether blockage of core flow channels might occur from scale initially deposited on the fuel surface that would flake off during the cool down process. The PWROG responded in Reference 4, stating that the thickness of the scale formed is limited by the amount of solids dissolved in the water. Using scale deposition models the PWROG demonstrated that the thickest scale fragment would be insufficient to bridge a fuel rod to fuel rod span to block flow. The NRC staff finds this justification acceptable because the spallation process from the fuel is slow, and experience from spent fuel pool debris generated at PWRs shows these scale materials to be granular and of small size rather than large flakes (Reference 40).

The NRC staff also considered whether sufficient cooling water flow to the core may be compromised by other chemical precipitation effects outside the fuel assemblies. Specifically, the NRC staff inquired in Reference 3 about potential blockage of the residual heat removal (RHR) heat exchanger tubes due to precipitate formation from lowering temperature. In Reference 4, the PWROG presented industry work that had been performed to identify how deposits build up and block flow orifices. In the case of the RHR heat exchangers, the linear velocity of the water (approximately 2.5 to 5.0 ft/sec) and the RHR pump discharge pressure (approximately 300 psi) are relatively high. The chemical flocculent that might form as fluid temperature is reduced would have very little shear strength, and flocculent formation is typically time dependent. Thus, the water velocity and pump output pressure would be sufficient to prevent blockage from deposition of these materials. The flow rate changes considerably once the water is in the lower core area due to a much greater surface area as well as flow lost to the

break. The flow prior to encountering the baffle is downward. Therefore, precipitates, if formed due to temperature decreases in the RHR heat exchanger, will be more likely to be transported to the reactor vessel than to cause blockages in the RHR heat exchanger or piping. Therefore, the NRC staff finds with the PWROG's evaluation that chemical precipitates will not block the RHR heat exchangers.

The NRC staff finds that:

1. The mass of material used to determine the debris and scale loading is conservative based on the source term calculated from the WCAP-16530-NP-A tests, along with the assumption that no precipitates settle on the containment floor, are filtered at the sump screen, or deposited in heat exchangers, piping, or in the reactor vessel outside of the core. The mass of materials includes a "bump up factor" to account for leaching of aluminum, calcium and silicon from pieces of insulation materials that bypass the sump screens. The NRC staff finds this bump up factor to be acceptable for reasons stated in this SE section.

2. The thermal conductivity assumed for chemical scale and debris deposits represents an acceptably low value (0.11 BTU/(h-ft-°F) to help achieve a conservative prediction of fuel clad temperature increase. Wetted insulation allows for better conduction of heat and the thermal conductivity of wetted insulation would be higher. Thus the use of 0.11 BTU/(h-ft-°F) is a conservative assumption.

3. Blockage of the RHR heat exchanger based on chemical deposition is unlikely due to time-dependent formation of precipitates and system flows and pressures being able to overcome the low shear strength of precipitate deposits.

4. Industry-recognized calculation models were used to predict temperature increases at the fuel surface as a result of chemical plate-out, and these models confirm that the limit of 800 degrees Fahrenheit is not exceeded when these models are used in conjunction with the source term assumptions in WCAP-16530-NP-A.

5. Blockage of fuel rod spans by spall of fuel scales is unlikely due to the time dependency for spallation and the small thickness of the scale compared to the space between the fuel rods.

Overall, this is a valid approach to determining potential flow restrictions due to chemical effects of RCS liquid and containment debris and materials, and is both conservative and representative of the post-LOCA conditions based on chemical reactions described in WCAP-16530-NP-A. Therefore, given the acceptance criteria for fiber bypass, the NRC staff concludes the chemical effects on core cooling resulting from debris and scale deposition following a LOCA are insufficient to create a condition resulting in fuel clad temperatures exceeding the temperature limit of 800 degrees Fahrenheit.

3.8 BORIC ACID PRECIPITATION *(TR WCAP-16793-NP, Revision 2, Section 8)*

The WCAP does not address boric acid mixing and transport in the RCS and the potential precipitation mechanisms that may occur during the sump recirculation phase of a LOCA. However, the PWROG is currently funding a program to define, develop and obtain NRC approval of post-LOCA boric acid precipitation analysis scenarios, assumptions and acceptance criteria and resultant methodologies that demonstrate adequate post-LOCA LTCC.

NRC Staff Evaluation

Section 8 of Reference 18 states that the effects of boron precipitation are not addressed in TR WCAP-16793-NP, Revision 2. Effective LTCC involves (1) provision of sufficient coolant flow to the core to remove decay heat without unacceptable fuel clad heat-up, and (2) prevention of boric acid precipitation sufficient to inhibit adequate core cooling. In response to NRC findings made during a technical audit of Westinghouse Topical Report CENPD-254-P (Reference 41), the PWROG initiated a separate program to address staff questions related to boric acid precipitation, including the effect of debris accumulating in the core. The NRC staff finds this approach acceptable if a plant limits the amount of fiber bypassing the strainer to the amount approved by this SE.

Tests performed by the PWROG on fuel assemblies at the low debris limits approved in this SE do not indicate the presence of a fiber bed that would significantly impact the timing of the onset of boric acid precipitation. For a cold-leg break (the break of concern for boron precipitation), the quantity of debris entering the core is generally one half or less of the total amount of debris that passes through the strainer and reaches the RCS (e.g., 15 grams per fuel assembly). In order for debris to reach the core, it must be well mixed within the ECCS fluid; easily transported; and nearly neutrally buoyant. During a cold-leg break, flow into the core is limited by the boiloff rate, so only a portion of the flow passing through the sump strainer enters the core. The remainder of the ECCS flow exits the break, carrying with it a proportionate amount of suspended debris. Fuel assembly testing at cold leg flow rates with low fiber amounts--approximately 10 grams per fuel assembly--did not exhibit a noticeable head loss. This indicates that the fiber beds will be minimal at the maximum expected fibrous debris load of 7.5 grams per fuel assembly and, therefore, no significant impact on boron precipitation is expected. Therefore, NRC staff finds it acceptable for the PWROG to address the effects of fibrous debris on boron precipitation under the separate PWROG program.

Fiber in excess of the acceptance criterion (e.g., greater than 15 grams per assembly) could build debris beds at the core inlet or at spacer grids at cold leg break conditions that inhibit mixing of coolant between the core and the reactor vessel lower plenum regions. This mixing is credited in analyses for the timing of boron precipitation, which forms the bases for operating procedures that respond to the event. Therefore, if a plant intends to justify fiber loads above the limit approved in this SE, the effects of fibrous debris on boric acid precipitation must be considered in the course of resolution of in-vessel downstream effects. (The PWROG has stated that debris is part of the boron precipitation program). *This Condition is addressed further in Section 4.0, Item number 1, of this SE.*

3.9 COOLANT DELIVERED TO THE TOP OF THE CORE *(TR WCAP-16793-NP, Revision 2, Section 9)*

The WCAP describes the two scenarios by which coolant can be delivered to the top of the core as follow:

1. For all PWR plants, except Westinghouse two-loop plants, ECCS flow is initially delivered to the cold-legs. At a predetermined time after sump switchover, coolant may be introduced into the hot-legs (hot-leg recirculation) to act as flushing flow to mitigate the precipitation of boric acid in the vessel.

2. For two-loop Westinghouse PWRs, ECCS flow is initially delivered simultaneously to the cold-legs and to the upper plenum of the RPV. When the RWST coolant source is

exhausted, the coolant flow-path to the cold-legs is secured and coolant flow from the containment sump to the upper plenum is initiated and maintained throughout plant recovery. However, plants maintain the ability to re-initiate flow through the cold-leg(s).

For both conditions described above, debris in the circulated coolant can flow into the core from the top.

For all PWR plants, except Westinghouse two-loop plants, the WCAP states that for a cold-leg break, the amount of debris that enters the core from the top during hot-leg injection is small because much of the debris is depleted by the time (several hours after the initiation of recirculation) hot-leg injection is initiated. The WCAP states that much of the debris in the circulated containment pool is depleted either by capture on the sump screen or by settling in the containment sump or in low-flow locations of the ECCS RV flow path such as the reactor vessel lower plenum. The WCAP cites examples of debris depletion in WCAP-16406-P-A (Reference 39).

For Westinghouse two-loop PWRs, the WCAP states that full ECCS flow to the reactor vessel upper plenum is established at ECCS actuation and is maintained throughout plant recovery. Therefore, this flow path begins to deliver debris to the top of the core when ECCS pump suction is realigned to draw coolant from the containment sump. The quantity of debris that could enter the core is determined by the break location as follows;

1. For a hot-leg break, coolant flow to the core is through the UPI ports with all cold-leg flow initially secured. ECCS flow that enters the upper plenum can either enter the core or exit the break, with most of the flow exiting the vessel through the break. Only the flow required to replenish boil-off enters the core. The PWROG stated that quick turnaround time and turbulent mixing of the upper portion of the core results in a situation where debris that enters the upper plenum with the coolant is either kept in suspension and expelled through the hot-leg piping, or is deposited over a broad area of the core.

2. For a cold-leg break, the same injection points (the UPI ports) apply. The WCAP states that the debris suspended in the coolant pumped to the upper plenum will flow into the core. The flow path is down through the core, up the downcomer, into the cold-legs and out the break.

The WCAP states that considering the above, the debris that may be captured on mixing vanes, fuel grids and the bottom of the fuel is limited. Further, the collection of debris by these features occurs over time. Therefore, the formation of a debris bed takes time and the collected debris would have some packing factor that would allow "weeping" flow through particulate debris buildup and into the core. This allows for coolant to pass through a debris bed that might form. The PWROG stated that because coolant is introduced into the RV above the core, the debris would accumulate on top of the fuel but would not result in complete blockage in those plants that have debris loads meeting the acceptance criteria as was demonstrated by the testing described in Reference 7.

The WCAP states that the turbulent mixing of the upper portion of the core causes the debris that enters the upper plenum to either be kept in suspension and expelled through the hot-leg piping (for a hot-leg break scenario), or to be deposited over a broad area of the core. Collection of debris at grid locations is covered by the analysis performed in Appendix C of the WCAP. The analysis demonstrates that adequate cooling in such locations will be maintained.

Further, to demonstrate that adequate flow can be maintained through the core of a UPI plant with debris entrained in the circulated coolant, the PWROG conducted a test on a 17 x 17 mock fuel assembly where debris laden water was introduced at the top of the fuel. The test was conducted with the maximum debris loads that were tested in the Westinghouse hot-leg test. The PWROG stated in the WCAP that the loss of static head was well below that required to maintain core flow for UPI plants. Therefore, the PWROG concluded that the test results demonstrate that sufficient flow will reach the core to remove core decay heat and the acceptance criteria developed at hot-leg conditions is bounding and applicable to all UPI plants. Appendix G of the WCAP and Reference 7 contain additional information about the UPI test and the applicability of the debris acceptance criteria to UPI plants.

The WCAP states that if the coolant flow is sufficiently restricted through a debris bed for the clad temperature to increase to about 15 to 20 degrees Fahrenheit above the coolant temperature, the coolant will begin to boil. The steam formed will be about 40 to 50 times the volume of the water, and will cause the debris bed to be displaced, allowing for coolant to flow to and cool the cladding surface. This process will provide for cooling of the clad.

The conservative clad heat-up calculations documented in Appendix D of the WCAP demonstrate that acceptably low clad temperatures are calculated with as much as 50 mils of solid precipitate applied to the outside surface of a fuel rod. The PWROG stated that these calculations provide further assurance that, with weeping flow through a debris bed collected on fuel elements, LTCC for UPI plants will be maintained.

The evaluation of the effects of chemicals dissolved in the UPI flow for a hot-leg break is performed on a plant-specific basis using the LOCADM calculation tool described in WCAP Section 7 and Appendix E. To account for deposition on fuel cladding in the core, a bump-up factor is used in the LOCADM calculation to deposit fiber material according to the core boiling and heat flux distribution.

NRC Staff Evaluation

The NRC staff finds that for all plants, except UPI plants, the quantity of debris that enters the core during hot-leg injection will be significantly less than that which enters through the cold leg in the event of a hot-leg break. Therefore, the potential for core blockage during hot-leg injection is bounded by the cold-leg injection phase of containment pool circulation. The NRC staff's acceptance of this argument is based on the depletion of debris due to settling and filtering prior to the initiation of hot-leg injection.

For reasons previously cited, the NRC staff accepts the assertion that during a hot-leg break in a UPI plant, much of the debris entering the upper plenum will remain suspended and be ejected from the vessel through the hot-leg break. This conclusion is based on the expected turbulence in the upper plenum caused by boiling in the core, the expectation that debris is uniformly distributed in the coolant, and the fact that all flow except that required to replenish boil off from the core exits the break.

The NRC staff does not agree that the packing factor of debris within the core will be limited to 60 percent such that LTCC will be ensured. The NRC staff also disagrees that a 60 percent packing factor can be conservatively considered to equate to 60 percent blockage of the core because the statement is not supported and would require the consideration of several variables. The NRC staff finds that if debris amounts are maintained within accepted limits

listed in Section 10 of the WCAP, within the conditions stated in Section 3.10 of this SE, flow to the core will not be compromised by debris suspended in the circulated containment pool.

Although the testing program included only a single UPI-related test, the results are acceptable since the testing was performed under single-phase flow conditions and did not include the effects of boiling that would be expected post-LOCA. Core quench post-LOCA means the core is no longer in the film-boiling regime, but nucleate boiling still continues for an extended period of time. The turbulence that results from this two-phase flow, along with counter-current flow conditions at the top of the core, will inhibit the formation of a uniform debris bed that could block flow to the core. Additionally, the debris limits derived from tests configured to represent other plants are applied to UPI plants. These limits have significant margin between the head loss values measured and the available driving head. As described above the UPI plants are less likely to form a debris bed due to the turbulent flow within the core. Therefore, the NRC staff finds the results of the single-phase water test for UPI plants, in conjunction with the other tests performed by Westinghouse and AREVA, to be an acceptable basis for defining the fibrous debris limit for UPI plants to demonstrate that adequate water is supplied to the core for cooling. However, this does not resolve the LTCC issue for assuming boric acid precipitation is prevented. That will be addressed in a separate program undertaken by the PWROG.

Some UPI plants may utilize cold-leg injection to prevent boric acid concentration within the core. Cold-leg injection would be initiated several hours after the initiation of a LOCA to flush boric acid from the core. Plants other than the UPI design use hot-leg injection to prevent boric acid accumulation. Because cold-leg injection is initiated several hours after recirculation begins, the debris in the recirculation pool available for transport to the cold-leg (bottom of the core) will be depleted due to filtering on the sump strainer, deposition on the upper core, and settling in the sump pool. If UPI plant specific evaluations are conducted to increase the accepted debris limits, the ability to ensure any required cold-leg flushing should be demonstrated. *This Condition is discussed in Section 4.0, Item number 1, of this SE.*

Based on the results of the testing and for reasons previously cited in this SE, the NRC staff accepts the PWROG's statement that for those plants that do not exceed the debris limits stated in the WCAP, as approved by the NRC staff, adequate coolant flow will reach the core. The NRC staff also finds that boiling in the core will tend to disrupt the formation of a debris bed and thus will allow coolant flow to the cladding. The NRC staff position is based on the results of the fuel assembly testing and observations of debris bed behavior, as discussed in this section.

3.10 SUMMARY *(TR WCAP-16793-NP, Revision 2, Section 10)*

The WCAP provides guidance and acceptance criteria for licensees of PWR plants to use to demonstrate that post-LOCA LTCC can be maintained in the presence of debris in the circulated coolant downstream of the sump strainer, and provides evaluations to demonstrate that after the initial quench of the core, the fuel cladding temperature will not exceed a temperature of 800 degrees Fahrenheit (a temperature that has been demonstrated by autoclave testing not to cause cladding to become brittle). These evaluations address cladding heat-up due to collection of debris between the fuel rod and spacer grid and the collection of debris, including containment protective coatings and fiber glass, on fuel grids and at the core inlet. Additionally, the WCAP provides numerical analyses to bolster the assertion that adequate coolant will continue to flow to the core and decay heat will continue to be removed, even with debris from the sump reaching the RCS and reactor core.

The acceptance criteria for demonstrating adequate core cooling, as defined in the WCAP, Section 2.2, is as follows:

1. The maximum clad temperature shall not exceed 800 degrees Fahrenheit during the sump recirculation phase of a LOCA.

2. The thickness of cladding oxide and fuel deposits shall not exceed 0.050 inches in any fuel region.

Key technical points in the WCAP Summary Section include the following:

1. Adequate flow to remove decay heat will continue to reach the core even with debris from the sump reaching the RCS and core. Plants that operate at or below the debris load acceptance limits identified in the report can state that debris that bypasses the strainer will not build an impenetrable blockage at the core inlet. While any debris that collects at the core inlet will provide some resistance at the core inlet, in the extreme case that a large blockage does occur, numerical analyses have demonstrated that core decay heat removal will continue.

2. Decay heat will continue to be removed even with debris collection at the fuel assembly spacer grids. Plants that operate at the debris loads identified in the report can state that debris that bypasses the strainer will not build an impenetrable blockage at the fuel spacer grids. The report further states that in the event that a large blockage does occur that numerical and first principle analyses have demonstrated that decay heat removal will continue.

3. Fibrous debris will not adhere tightly to the surface of the fuel cladding. Therefore, core cooling will not be adversely affected by a blanket of fiber on the fuel.

4. Protective coating debris, should it enter the core region, will not restrict heat transfer and cause an increase in clad temperature.

5. Using the chemical effects source term method developed in WCAP-16530-NP-A, the WCAP developed a method to predict chemical deposition on fuel cladding. The calculation tool, LOCADM, will be used by each utility to perform a plant-specific evaluation.

6. New insights and methodologies are required to address the potential for boric acid mixing and transport within the RCS. These issues are being addressed by a separate PWROG program.

7. All PWR plants (including UPI plants) that operate at the debris loads identified in the report can state that debris that bypasses the strainer will not build an impenetrable blockage within the core region.

The WCAP stated that the purpose of the testing described in the report was to develop bounding acceptance criteria for debris reaching the RCS while ensuring LTCC. The WCAP stated that fiber was found to be the limiting variable and is the only type of debris that requires a limit. The WCAP stated that due to the conservative methods used in the testing that bounding guidelines were developed by the program. Regarding the debris limits, the WCAP concluded:

1. All PWR plants that demonstrate that less than 15 grams of fiber per fuel assembly reaches the core can demonstrate that LTCC is not impeded.

2. Because the 15 gram limit results in very low head loss leaving significant head loss margin with respect to the available driving head, utilities may conduct plant specific tests to increase this fiber limit.

3. Westinghouse plants that have available driving head value greater than the head loss measured during test number 1-W-FPC-0811 and which operate under similar conditions may demonstrate adequate LTCC with 25 grams of fiber per fuel assembly.

4. Westinghouse plants that can maintain high sump water temperatures can decrease the debris head loss at specific fiber loadings which results in the ability to increase the allowable fiber load.

5. Plants that can successfully demonstrate that the precipitation of chemicals will be delayed until after hot-leg switchover may be able to demonstrate higher fiber limits because flow through the core will be reduced prior to chemical precipitation resulting in lower head losses.

6. Plants that can demonstrate that chemical precipitates do not form can use the pressure drop (dP) values recorded with just particulate and fiber in the test loop, in conjunction with the dP available, to make a determination on the amount of allowable fiber.

The WCAP also stated that the following actions are required of licensees to demonstrate LTCC with debris and chemical products in the circulating fluid.

1. Perform plant-specific LOCADM evaluations (WCAP, Section 7 and Appendix E) and confirm that their plant-specific evaluations are bounded by the 800 degrees Fahrenheit acceptance criterion.

2. Demonstrate that the quantity of debris passing through the strainer and transported to the core inlet is less than or equal to debris limit specified in Section 10 of the WCAP, as supported by the proprietary fuel assembly test reports (References 8, 16, and 17).

In References 11 and 12, the PWROG made reference to a calculation tool (Margin Calculator) to enable licensees to perform their evaluations. However, in RAI response number 18 (Reference 13), the PWROG withdrew the Margin Calculator, replacing it with a method that utilities can follow to calculate the plant-specific available driving head. The method uses the Darcy equation and, in all cases, the flow losses in the core are neglected based on their assumed relatively small contribution to the total head loss.

Section 10 of the WCAP concludes by stating that several actions are available to plants whose debris loads are outside of the limits tested. These actions include, but are not limited to, reduction of problematic debris sources by removing or restraining the debris source, plant-specific fuel assembly testing, eliminating or reducing chemical effects, evaluating debris bypass and transport, or other engineering evaluations.

NRC Staff Evaluation

The NRC staff finds acceptable the acceptance bases of 800 degrees Fahrenheit for maximum cladding temperature and maximum cladding oxide and deposit thickness of 0.050 inches as discussed in Section 3.2 of this SE. Each plant evaluation of compliance with these limits should be performed using the LOCADM model discussed in Section 7 of the WCAP and Section 3.7 of this SE. Also, each licensee's GL 2004-02 submittal to the NRC should state the peak cladding temperature predicted by the LOCADM analysis. *This condition is discussed further in Section 4, Item number 7, of this SE.*

The NRC staff finds that plants that demonstrate that the quantity of debris transported to the core inlet during ECCS recirculation is less than or equal to 15 grams per fuel assembly can state that adequate coolant flow to the core can be maintained to acceptably close GL 2004-02. The NRC staff has not accepted that some PWRs can maintain adequate core cooling with greater than 15 grams fiber per fuel assembly entering the core. Additional testing and/or evaluations are required to justify values above this limit. *This condition is discussed further in Section 4, Item number 11, of this SE.*

The NRC staff concluded that observations of recent fuel assembly tests do not support the WCAP assertion that numerical analyses have demonstrated that adequate core decay heat removal will continue in the case of a large blockage. The behavior of debris in the FA testing does not follow the modeling used in the numerical analyses. Therefore, adherence to the test-supported debris limit described above should be the basis for a licensee to conclude that adequate flow to the core to remove decay heat will continue. However, the NRC staff finds that the numerical analyses can be used to bolster the argument that once coolant reaches the core, decay heat will continue to be removed, even with debris collection at the fuel assembly spacer grids. *This Condition is addressed further in Section 4.0, Item number 5, of this SE.*

The NRC staff finds that fibrous debris that enters the core will not adhere to the fuel cladding. The basis for this conclusion is discussed in Section 3.5 of this SE.

The NRC staff finds that protective coating debris that enters the core will not adhere to the cladding to restrict heat transfer and cause an increase in clad temperature. The basis for this conclusion is discussed in Section 3.6 of this SE.

The NRC staff finds that the chemical effects source term method developed in WCAP-16530-NP-A is an acceptable input to the LOCADM tool and that LOCADM may be used by licensees to determine the cladding deposition thickness. Also, the NRC staff finds that each plant should perform a plant-specific evaluation based on their calculated chemical, fiber and particulate debris loads to show that decay heat will be removed and acceptable fuel clad temperatures will be maintained. The basis for this conclusion is discussed in Section 3.7 of this SE.

The NRC staff finds that boric acid precipitation in the core is a broader issue that may be aggravated by debris intrusion into the core. Therefore, a separate PWROG program to address questions related to boric acid precipitation is warranted and is currently underway. The NRC staff understands that this program includes the effects of debris in the coolant, and the staff finds this acceptable. This subject is discussed further in Section 3.8 of this SE.

The NRC staff accepts the debris limits defined for UPI plants to be the same as other PWRs. This item is discussed further in Section 3.9 of this SE.

Regarding the debris limits, the NRC staff concluded the following:

1. Fiber is the limiting variable because other debris types were included in quantities that produced bounding test results. Therefore, fiber is the only debris type that requires a limit.

2. An upper limit of 15 grams of fiber per fuel assembly has been demonstrated to not result in core inlet blockage that would compromise flow to the core.

3. Utilities may conduct plant specific tests to increase the fiber limit. However, the NRC staff expects plant specific tests to ensure margins adequate to address concerns with the test program discussed throughout this SE. Margins may be reduced if uncertainties are addressed, conservatisms are better defined, and repeatability is adequately demonstrated.

4. PWR plants that use Westinghouse fuel are subject to the same fibrous debris limit as PWR plants that use AREVA fuel. Although test number 1-W-FPC-0811 demonstrated that a Westinghouse fuel assembly could ingest 25 grams of fibrous debris at a head loss close to the expected limit for some plants, the flow rate during the test had to be reduced. This indicates that there was little margin at the tested debris load. There was no repeatability test performed for this condition. Additionally, the NRC staff expects there to be considerable margin between the available driving head and debris head loss values. The expectation for margin is based on issues with the test program discussed throughout this SE and the need for assurance that core cooling will be maintained. Additional testing and evaluation may allow the NRC staff to re-evaluate their expectations for repeatability and retained margin.

5. The NRC staff finds that maintaining water temperature at higher values will decrease head loss across any debris bed within the core. In the case of in-vessel effects, the temperature of concern may be the core inlet temperature. It may not be sufficient for a plant to maintain an elevated sump temperature if the ECCS flow is significantly cooled before it reaches the core. Maintaining higher water temperature may allow the fiber loading to be increased; however, significant sensitivity testing is required to determine the potential beneficial effects of maintaining a higher sump temperature.

6. Plants that can demonstrate that chemical precipitates will not occur until after hot-leg switchover may be able to justify higher fibrous debris limits since lower head losses may result from precipitate addition at lower flow rates. However, the limiting p/f ratio should be considered before and after chemical precipitates form.

7. The NRC staff finds that plants that show that chemical precipitates or other chemical effects will not adversely affect head loss can use head loss values that do not include chemical effects. The NRC staff cautions, however, that demonstrating that no chemical precipitates form will be very difficult because it will involve an extensive number of long-term, complex, tests that consider additional factors compared to previous conservative chemical effects test methodologies. In addition, if a plant is able to demonstrate chemical precipitates will not form, a debris limit will require the determination of a different limiting p/f ratio than that defined by the limiting tests that included chemicals.

Evaluations and tests intended to justify debris limits greater than 15 grams per fuel assembly should include consideration of the issues discussed in this SE. Testing should ensure that repeatability has been established. If hot-leg break debris limits are increased, the cold-leg break location may become limiting. If debris limits are increased for the hot-leg break, the cold-leg break should be re-evaluated. The potential for debris to affect boric acid precipitation analyses should be included in any evaluation that increases debris loading.

The NRC staff concluded that plants need to demonstrate that the quantity of fibrous debris that transports to the fuel inlet is less than or equal to 15 grams per fuel assembly or as otherwise justified on a plant-specific basis. As previously communicated to stakeholders in public meetings, the following methods for determining the quantity of debris that passes through the strainer are acceptable to the NRC staff:

1. Performing strainer bypass testing using the plant strainer design, plant-specific debris loads, and plant-specific flow velocities.

2. Relying on strainer bypass values developed through strainer bypass testing of the same strainer manufacturer and design, and same perforation size, extrapolated to the licensee's plant-specific strainer area; approach velocity; debris types, and debris quantities. To perform such an extrapolation, adequate testing and evaluation is required to show understanding of how each variable affects bypass.

3. Assuming that the entire quantity of fiber that is transported to the sump strainer passes through the sump strainer.

Licensee's submittals to the NRC regarding in-vessel downstream effects should include:

1. The means used to determine the amount of debris that bypasses the ECCS strainer and the fiber loading expected, per fuel assembly, for the cold-leg and hot-leg break scenarios

2. The peak clad temperature calculated using LOCADM

3. The available driving head used in the hot-leg evaluations

4. The licensee's planned and/or completed actions if the acceptance criteria stated herein are exceeded (e.g., plans to reduce fiber loads)

5. A description and justification for any deviations taken from the topical report as accepted and modified by the Conditions and Limitations in Section 4.0 of this SE

The "Margin Calculator," referenced in References 11 and 12, has not been submitted to the NRC under formal letter, and NRC staff has not performed a detailed review of the document. Therefore, NRC staff expects licensees to base their GL 2004-02 in-vessel effects evaluations on the information provided in the proprietary test reports and associated RAI responses (References 8, 16, 17, 11, and 12), including the conditions and limitations stated in this SE, and existing plant design-basis calculations and analyses. *This Condition is addressed further in Section 4.0, Item number 14, of this SE.*

The NRC staff finds that licensees will have to demonstrate that the available driving head for a hot-leg break is equal to or greater than the driving head derived from the fuel assembly tests

which defined the debris limits discussed in the topical report and this SE. RAI response number 18 in Reference 13 provides a method acceptable to NRC staff for determining the available driving head. Plants that demonstrate an acceptable fibrous debris load greater than that approved in this SE (15 grams per fuel assembly) shall also demonstrate that their available driving head under cold-leg break conditions is greater than that used in the fuel assembly testing (1.5 psi). The calculation of available driving head for the cold-leg break scenario shall include a 2-phase pressure drop multiplier to account for 2-phase flow, and the pressure required to clear the loop-seal (cold leg break) as discussed in Section 3.1. The NRC staff has determined that it is acceptable to calculate the core pressure drop using staff-approved LOCA methods, but adjusted to the recirculation or boil-off flow conditions, as appropriate. *This Condition is addressed further in Section 4.0, Item number 1, of this SE.*

The available head and flow rate values used when determining the available driving head for the post-LOCA clean-loop flow resistance should be taken from the plant LOCA analysis. The evaluation for head loss across the core should not take credit for reduced ECCS flow rate due to core blockage or reduced pump capacity, other than that supported by testing with debris beds formed at the reduced flow rate. *This Condition is addressed further in Section 4.0, Item number 2, of this SE.*

4.0 LIMITATIONS AND CONDITIONS

1. Licensees should confirm that their plants are covered by the PWROG sponsored fuel assembly tests by confirming that the plant available hot-leg break driving head is equal to or greater than that determined as limiting in the proprietary fuel assembly tests and that flow rate is bounded by the testing. Licensees should validate that the fuel types and inlet filters in use at the plant are covered by the test program (with the exception of LTAs). Licensees should limit the amount of fibrous debris reaching the fuel inlet to that stated in Section 10 of the WCAP (15 grams per fuel assembly for a hot-leg break scenario).

   Alternately, licensees may perform plant specific testing and/or evaluations to increase the debris limits on a site-specific basis. The available driving head should be calculated based on the core exit void fraction and loop flow resistance values contained in their plant design basis calculations, considering clean loop flow resistance and a range of break locations. Calculations of available driving head should account for the potential for voiding in the steam generator tubes. These tests shall evaluate the effects of increased fiber on flow to the core, and precipitation of boron during a postulated cold-leg break, and the effect of p/f ratios below 1:1. The NRC staff will review plant specific evaluations, including hot- and cold-leg break scenarios, to ensure that acceptable justification for higher debris limits is provided. (Sections 3.1.2 (c), 3.1.2 (e), 3.3.1, 3.4.2, 3.8, 3.9 and 3.10 of this SE).

2. Each licensee's GL 2004-02 submittal to the NRC should state the available driving head used in the evaluation of the hot-leg break scenario, the ECCS flow rates, and the results of the LOCADM calculations. Licensees should provide the type(s) of fuel and inlet filters installed in their plants, as well as the amount of fiber (gram per fuel assembly) that reaches the core. (Section 3.3.1 and 3.10 of this SE)

3. Section 3.1.4.3 of the WCAP states that alternate flow paths in the RPV were not credited. The section also states that plants may be able to credit alternate flow paths for demonstrating adequate LTCC. If a licensee chooses to take credit for alternate flow

paths, such as core baffle plate holes, to justify greater than 15 grams of bypassed fiber per fuel assembly, the licensee should demonstrate, by testing or analysis, that the flow paths would be effective, that the flow holes will not become blocked with debris during a LOCA, that boron precipitation is considered, and that debris will not deposit in other locations after passing through the alternate flow path such that LTCC would be jeopardized. (Sections 3.3.1 and 3.4.2 of this SE)

4. Sections 3.2 and 3.3 of the WCAP provide evaluations to show that even with large blockages at the core inlet, adequate flow will enter the core to maintain LTCC. The staff recognizes that these calculations show that significant head loss can occur while maintaining adequate flow. However, the analyses have not been correlated with debris amounts. Therefore, the analyses cannot be relied upon to demonstrate adequate LTCC. (Sections 3.3.3 and 3.4 of this SE)

5. In RAI Response number 18 in Reference 13, the PWROG states that numerical analyses demonstrated that, even if a large blockage occurs, decay heat removal will continue. The NRC staff's position is that a plant must maintain its debris load within the limits defined by the testing (e.g., 15 grams per assembly). Any debris amounts greater than those justified by generic testing in this WCAP must be justified on a plant-specific basis. (Sections 3.4.2 and 3.10 of this SE)

6. The fibrous debris acceptance criteria contained in the WCAP may be applied to fuel designs evaluated in the WCAP. Because new or evolving fuel designs may have different inlet fittings or grid straps that could exhibit different debris capture characteristics, licensees should evaluate fuel design changes in accordance with 10 CFR 50.59 to ensure that new designs do not impact adequate long term core cooling following a LOCA. (Section 3.4.2 of this SE)

7. Sections 2 and 4.3 of the WCAP establish 800 degrees Fahrenheit as the acceptance limit for fuel cladding temperature after the core has been re-flooded. The NRC staff accepts a cladding temperature limit of 800 degrees Fahrenheit as the long-term cooling acceptance basis for GSI-191 considerations. Each licensee's GL 2004-02 submittal to the NRC should state the peak cladding temperature predicted by the LOCADM analysis. If a licensee calculates a temperature that exceeds 800 degrees Fahrenheit, the licensee must submit data to justify the acceptability of the higher clad temperature. (Sections 3.2, 3.4.3, 3.4.4, and 3.10 of this SE)

8. As described in the Limitations and Conditions for WCAP-16530-NP (ADAMS Accession No. ML073520891) (Reference 21)[5], the aluminum release rate equation used in TR WCAP-16530-NP provides a reasonable fit to the total aluminum release for the 30-day ICET tests but under-predicts the aluminum concentrations during the initial active corrosion portion of the test. Actual corrosion of aluminum coupons during the ICET 1 test, which used sodium hydroxide (NaOH), appeared to occur in two stages; active corrosion for the first half of the test followed by passivation of the aluminum during the second half of the test. Therefore, while the 30-day fit to the ICET data is reasonable, the WCAP-16530-NP-A model under-predicts aluminum release by about a factor of two during the active corrosion phase of ICET 1. This is important since the in-core LOCADM chemical deposition rates can be much greater during the initial period

[5] The NRC SE on WCAP-16530-NP-A, including the NRC staff limits and conditions, is included in WCAP-16530-NP-A.

following a LOCA, if local conditions predict boiling. As stated in WCAP16530-NP-A, to account for potentially greater amounts of aluminum during the initial days following a LOCA, a licensee's LOCADM input should apply a factor of 2 increase to the WCAP-16530-NP-A spreadsheet predicted aluminum release, not to exceed the total amount of aluminum predicted by the WCAP-16530-NP-A spreadsheet for 30 days. In other words, the total amount of aluminum released equals that predicted by the WCAP-16530-NP-A spreadsheet, but the timing of the release is accelerated. Alternately, licensees may choose to use a different method for determining aluminum release but licensees should not use an aluminum release rate equation that, when adjusted to the ICET 1 pH, under-predicts the aluminum concentrations measured during the initial 15 days of ICET 1. (Section 3.7 of this SE)

9. In the response to NRC staff RAIs, the PWROG indicated that if plant-specific refinements are made to the WCAP LOCADM base model to reduce conservatisms, the user should demonstrate that the results still adequately bound chemical product generation. If a licensee uses plant-specific refinements to the WCAP-16530-NP-A base model that reduces the chemical source term considered in the downstream analysis, the licensee should provide a technical justification that demonstrates that the refined chemical source term adequately bounds chemical product generation. This will provide the basis that the reactor vessel deposition calculations are also bounding. (Section 3.7 of this SE)

10. The WCAP states that the material with the highest insulating value that could deposit from post-LOCA coolant impurities would be sodium aluminum silicate. The WCAP recommends that a thermal conductivity of 0.11 BTU/(h-ft-°F) be used for the sodium aluminum silicate scale and for bounding calculations when there is uncertainty in the type of scale that may form. If plant-specific calculations use a less conservative thermal conductivity value for scale (i.e., greater than 0.11 BTU/(h-ft-°F)), the licensee should provide a technical justification for the plant-specific thermal conductivity value. This justification should demonstrate why it is not possible to form sodium aluminum silicate or other scales with thermal conductivities less than the selected value. (Section 3.7 of this SE)

11. Licensees should demonstrate that the quantity of fibrous debris transported to the fuel inlet is less than or equal to the fibrous debris limit specified in the proprietary fuel assembly test reports and approved by this SE. Fiber quantities in excess of 15 grams per fuel assembly must be justified by the licensee. Licensees may determine the quantity of debris that passes through their strainers by (1) performing strainer bypass testing using the plant strainer design, plant-specific debris loads, and plant-specific flow velocities, (2) relying on strainer bypass values developed through strainer bypass testing of the same vendor and same perforation size, prorated to the licensee's plant-specific strainer area; approach velocity; debris types, and debris quantities, or (3) assuming that the entire quantity of fiber transported to the sump strainer passes through the sump strainer. The licensee's submittals should include the means used to determine the amount of debris that bypasses the ECCS strainer and the fiber loading expected, per fuel assembly, for the cold-leg and hot-leg break scenarios. Licensees of all operating PWRs should provide the debris loads, calculated on a fuel assembly basis, for both the hot-leg and cold-leg break cases in their GL 2004-02 responses. (Section 3.10 of this SE)

12. Plants that can qualify a higher fiber load based on the absence of chemical deposits should ensure that tests for their conditions determine limiting head losses using particulate and fiber loads that maximize the head loss with no chemical precipitates included in the tests. (Section 3.3.1 of this SE) Note that in this case, licensees must also evaluate the other considerations discussed in Item 1 above.

13. Licensees should verify that the size distribution of fibrous debris used in the fuel assembly testing referenced by their plant is representative of the size distribution of fibrous debris expected downstream of the plant's ECCS strainer(s). (Section 3.4.2.1 of this SE)

14. The "Margin Calculator," referenced in References 11 and 12, has not been submitted to the NRC under formal letter, and NRC staff has not performed a detailed review of the document. Therefore, NRC staff expects licensees to base their GL 2004-02 in-vessel effects evaluations on the information provided in the proprietary test reports and associated RAI responses (References 8, 16, 17, 11 and 12), including the conditions and limitations stated in this SE, and existing plant design-basis calculations and analyses.

5.0 CONCLUSIONS

The NRC staff has reviewed WCAP-16793-NP, Revision 2, which describes a methodology for consideration of particulates, fibrous and chemical debris affecting the long-term cooling at operating PWRs following a LOCA. Based on this review, the NRC staff finds that application of the procedures and methods described in the WCAP, as qualified by the limitations and conditions stated in Section 4.0 of this SE, provides an acceptable, plant-specific evaluation method for demonstrating that adequate coolant flow reaches the core to maintain fuel clad temperature within acceptable limits. However, this TR does not evaluate the potential for debris in the core to change flow patterns or otherwise inhibit the mixing of boric acid that could result in earlier boric acid precipitation. Ongoing PWROG efforts are addressing boric acid precipitation in a separate program. Therefore, plants may apply the WCAP-16793-NP, Revision 2, evaluation methods and acceptance criteria, as accepted and modified by this SE, to evaluate the effects of debris blockage in the core to resolve the issues identified in GL 2004-02.

6.0 REFERENCES

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Attachment: Resolution of Comments

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Date: April 8, 2013

RESOLUTION OF COMMENTS BY THE OFFICE OF NUCLEAR REACTOR REGULATION REGARDING THE DRAFT SAFETY EVALUATION FOR TOPICAL REPORT WCAP-16793-NP, REVISION 2, "EVALUATION OF LONG-TERM COOLING CONSIDERING PARTICULATE, FIBROUS AND CHEMICAL DEBRIS IN THE RECIRCULATING FLUID"

PRESSURIZED WATER REACTOR OWNERS GROUP

PROJECT NO. 694

This Attachment provides the U.S. Nuclear Regulatory Commission (NRC) staff's review and disposition of the comments made by the Pressurized Water Reactor Owners Group (PWROG) on the draft safety evaluation for Topical Report WCAP-16793-NP, Revision 2, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid" (Agencywide Documents and Management System (ADAMS) Accession No. ML12115A304). The PWROG provided its comments by a letter dated March 6, 2013 (ADAMS Accession No. ML13093A082).

Reference	PWROG Comment	NRC Staff Resolution
Page 4, Lines 18-20	For clarity, the following text change is recommended: *Initially, the source for this water is from stored locations, e.g., the refueling-water storage tank (RWST) at ~~CE PWRs and~~ Westinghouse PWRs, the refueling water tank (RWT) at CE PWRs, or the borated water storage tank (BWST) at B&W PWRs.*	The proposed change is accepted. The text has been modified to read: *Initially, the source for this water is from stored locations, e.g., the refueling-water storage tank (RWST) at Westinghouse PWRs, the refueling water tank (RWT) at CE PWRs, or the borated water storage tank (BWST) at B&W PWRs.*
Page 7, Line 31	The clad temperature limit, as expressed in WCAP-16793-NP Revision 2, is 800 degrees F during the 30-day period following initial quench of the core (as first correctly stated on Page 6, lines 42-44 of the draft SE). For consistency, the text should be revised accordingly. Throughout the entirety of the draft SE, this change also applies to the following: Page 9, line 32 Page 10, line 7 Page 10, line 45 Page 11, line 11 Page 12, line 1 Page 48, line 46 Page 49, line 7 Page 50, line 7 Page 61, line 43 Page 65, line 42 Page 68, line 4 Page 72 (Limitation and Condition #7), lines 28 and 32	The proposed change is not accepted. This section of the SE (3.1.3) states the acceptance bases defined in the Executive Summary of the WCAP. Further, the last paragraph in Section 2.0 of the SE provides a definition of adequate core cooling during the ECCS mission time and subparagraph (b) of the NRC staff evaluation of the executive summary statements clearly accepts the 800 degree Fahrenheit cladding temperature limit after the initial quench. The SE is sufficiently clear that the temperature limit is for LTCC. Therefore, there is no need to specify, in the cited paragraphs, that the limit applies to the 30 days following the initial quench of the fuel.

ATTACHMENT

Page 8, Line 5	Suggested text change: *...through engineering evaluations of plant- or group-specific conditions and/or plant- or group-specific testing.*	The proposed change is not accepted: The term "plant-specific" does not preclude plants from collaborating on developing alternate solutions. However, as required for plants using WCAP-16793-NP, each plant participating in the collaboration shall demonstrate that the criteria developed bound their plant conditions.
Page 8, Lines 39-41	For clarity, the following text change is suggested: *~~It is expected that e~~Each plant will be able to use this tool to show that decay heat would be removed and acceptable fuel clad temperatures would be maintained.*	The proposed change is not accepted The wording in the SE is taken from the executive summary of the WCAP
Page 9, Lines 6-9	Suggested text change: *Licensees will have to perform plant-specific LOCADM evaluations (Section 7 and Appendix E of Reference 18) and ~~prove~~ confirm the plant-specific conditions are bounded by the debris load acceptance criteria (Sections 3 and 10, and Appendix G of Reference 18).*	The proposed change is accepted. The text has been modified to read: *Licensees will have to perform plant-specific LOCADM evaluations (Section 7 and Appendix E of Reference 18) and confirm the plant-specific conditions are bounded by the debris load acceptance criteria (Sections 3 and 10, and Appendix G of Reference 18).*
Page 9, Lines 21-27	To remove addressing boric acid precipitation from the consideration of higher fiber limits, the suggested text modification is as follows: *Further, NRC staff finds the position stated in the WCAP that post-LOCA boric acid precipitation analysis scenarios, assumptions and acceptance criteria and resultant methodologies that demonstrate adequate post-LOCA LTCC can be addressed in a separate PWROG program acceptable. ~~if debris limits approved by the staff in this SE are not exceeded. Larger debris loads require the potential for boric acid precipitation to be addressed in conjunction with the resolution of in-vessel downstream effects~~.*	The proposed change is not accepted. Boric acid precipitation (BAP) is an important aspect of LTCC. The testing that supports WCAP-16793 NP, Rev. 2, indicates that at high debris levels, debris beds could form in a way that could lead to premature BAP. There is no basis for removing BAP from the consideration of higher debris limits.

Page 9, Lines 39-42	Suggested text change: *Also, the NRC staff finds that plants with debris loads above the debris load acceptance criteria may perform engineering evaluations and/or tests of plant- or group-specific conditions to demonstrate adequate LTCC capability.*	The proposed change is not accepted. See response to Page 8, Line 5 for basis
Page 9 - footnote	To remove addressing boric acid precipitation from the consideration of higher fiber limits, the suggested text modification is as follows: *In the context of GSI-191 in-vessel downstream effects evaluations, the WCAP references to LTCC generally refer to the capability to maintain adequate core flow in the presence of debris and the absence of deposits on fuel rods and in grid straps that would result in fuel clad temperatures exceeding 800 °F. The staff considers LTCC to include all phenomena needed to satisfy the requirements of 10 CFR 50.46(b)(4) and (b)(5).* ~~*One Phenomenon that must be considered for LTCC is boric acid precipitation. This TR does not evaluate the potential for boric acid precipitation.*~~	The proposed change is not accepted. See the response to Page 9, Lines 21-27.
Page 10, Lines 5 and 16	Page 10 lines 5 and 16 make a reference to Paragraph 3.1.3. There is no paragraph denoted as 3.1.3 in the draft SE.	The proposed change is accepted. The text has been modified by adding a heading – *3.1.3 Executive Summary* – near the top of page 7, immediately above the paragraph beginning with "The Executive Summary of the WCAP".
Page 10, Lines 24-29	Suggested text change: *The NRC staff finds the description of actions that are required of utilities to demonstrate acceptable LTCC with debris and chemical products present in the circulating fluid acceptable because it calls for licensees to perform plant- or group-specific evaluations to demonstrate that they satisfy the debris limits, debris and oxide deposition limits and cladding temperature limits of the WCAP, as qualified by the Limits and Conditions stated in this SE.*	The proposed change is not accepted. See response to Page 8, Line 5 for basis

Page 13, Line 27	(Grammatical) The following text change is suggested: 2. *The ~~F~~flow rate associated with a hot-leg break represented the limiting head loss condition.*	The proposed change is accepted. The text has been modified to read: *The flow rate associated with a hot-leg break represented the limiting head loss condition.*
Page 13, Lines 37-39	For completeness, the following text change is suggested: *The section concludes that plants that have in core debris loadings that are within the limits of the debris masses successfully tested are bounded by the test program and that plants with debris amounts greater than those successfully tested can take other actions to ensure LTCC, including, but not limited to, reducing problematic debris sources by removing or restraining the affected debris source, plant-specific FA testing, engineering evaluations of plant-specific conditions, removal or reduction of chemical precipitate formation, and evaluation of debris transport/bypass calculations.*	The proposed change is not accepted. The safety evaluation is not intended to provide a list of potential options for resolving issues associated with LTCC. Licensees may take actions that they determine to be appropriate to address situations where debris limits are exceeded.
Pages 14 Line 14 through Page 18 line 10	In the NRC Staff Evaluation, the write-up is silent on the conservative success criteria for hot leg break testing, which was to maintain the same flow into the simulated fuel assembly as was observed prior to development of a debris bed. That is, the hot-leg break testing took no credit for excess flow provided to the core. This is potentially a significant margin.	The proposed change is not accepted. The purpose of the SE is to determine acceptable debris limits. The staff recognizes that the highest flow value resulted in the lowest debris limits. The relationships between flow (and other variables) and debris limits is discussed throughout the SE. Although the SE does not recognize the use of the limiting flow rate as margin, it does recognize how flow affects head loss and debris limits.
Page 15 Line 10	(Typographical) The word "head" should be deleted.	The proposed change is accepted. The text has been modified to read: *The NRC staff does not fully accept the method described in response to RAI 18 of Reference 13 (WCAP Reference 19, Section 2.18) for calculating the available driving head for cold-leg or hot-leg breaks because*

		it attempts to envelope the entire fleet of PWRs.
Page 15, Line 13	(Typographical) The statement *Based on the very low fiber bypass limits stated in Section 10 of this SE,...* Should be changed to *Based on the very low fiber ~~bypass~~ limits stated in Section 10 of WCAP-16793-NP, Revision 2,..."*	The proposed change is accepted. The text has been modified to read: *Based on the very low fiber limits stated in Section 10 of Reference 18, the quantity of debris reaching the core under cold-leg break conditions is very low, as explained in Section 3.8 of this SE, and, therefore, the verification of available driving head under cold-leg break conditions may be addressed in the boric acid precipitation evaluation program.*
Page 15, Lines 31 and 32	Suggested text change: *If debris limits for the hot-leg break are increased through plant- or group-specific testing, the cold-leg break may become limiting.*	The proposed change is not accepted. See response to Page 8, Line 5 for basis
Page 16, Line 16; Page 27, Line 14; Page 37, Line 9	(Typographical): Change "*p-grid*" to "*P-grid*"	The proposed change is accepted. Text in 3 locations has been changed from "p-grid" to "*P-grid*".
Page 16, Lines 47-49	Suggested text change: *However, if debris limits for a hot-leg break scenario are increased through additional plant- or group-specific testing, the cold-leg break scenario debris loads should be re-evaluated.*	The proposed change is not accepted. See response to Page 8, Line 5 for basis
Page 17, Lines 22-24	Suggested text change: *The NRC staff finds that plants that have debris loadings within those defined by acceptable tests are bounded by the test program and that plants not bounded by the program can perform plant- or group-specific evaluations to demonstrate acceptable LTCC,*	The proposed change is not accepted. See response to Page 8, Line 5 for basis

Page 17, Lines 46-47	Suggested text change: *If credit for settling in the lower plenum is used in later plant- or group-specific evaluations, it should be ~~adequately~~ justified via analysis and/or testing.*	The proposed change is not accepted. Regarding the "plant-specific" comment, see response to Page 8, Line 5 for basis. Regarding the comment "justified via analysis and/or testing", the SE is not intended to provide a list of options or limitations for dealing with scenarios outside the scope of the topical report.
Page 18, Lines 1-4	Suggested text change: *The NRC staff reviewed the position stated in Section 3.1.4.3 of the WCAP and finds that alternate flow paths into the core have not been credited when determining debris loads limits and that some licensees could potentially credit these alternate flow paths via plant- or group-specific evaluations.*	The proposed change is not accepted. See response to Page 8, Line 5 for basis
Page 18, Lines 5-10	To remove addressing boric acid precipitation from the consideration of higher fiber limits, and to clarify alternate flow path expectations, the suggested text modification is as follows: *If a licensee elects to take credit for alternate flow paths, such as core baffle plate holes, the licensee would need to demonstrate that the flow paths would be effective; i.e., that the flow holes would not become blocked with debris during a LOCA, and that debris ~~would not deposit in other locations after passing through the alternate flow path~~ that does pass through the alternate flow path does not adversely affect core cooling. ~~and that any changes to the flow patterns do not adversely impact boron precipitation~~.*	The proposed change regarding alternate flow paths is accepted. The text has been revised to read: *If a licensee elects to take credit for alternate flow paths, such as core baffle plate holes, the licensee would need to demonstrate that the flow paths would be effective; i.e., that the flow holes would not become blocked with debris during a LOCA, and that debris that does pass through the alternate flow path does not adversely affect core cooling and that any changes to the flow patterns do not adversely impact boron precipitation.* The proposed change regarding BAP is not accepted. Taking credit

		for alternate flow paths inherently credits new flow patterns in the core; therefore, standard assumptions regarding BAP timing may not be applicable. There is no justification for removing consideration of BAP if alternate flow paths are relied upon to demonstrate higher debris limits.
page 23, lines 20-34	All of the gpm values presented are on a per fuel assembly basis. The following text changes are suggested: *Line 24: ...44.5 gpm per fuel assembly and 3.0 gpm per fuel assembly...* *Line 26: ...11.0 gpm per fuel assembly for plants with AREVA fuel and 6.25 gpm per fuel assembly...* *Lines 28/29: ...3.0 gpm per fuel assembly...* *Lines 33/34: ...15.5 gpm per fuel assembly...*	The proposed change is accepted. A sentence that reads "*All flow rates discussed in this section are based on the flow through a single fuel assembly*" has been added near the beginning of the paragraph.
Page 23, Line 36	Suggested text change: *For Westinghouse-designed fuel,...*	The proposed change is accepted. The text has been revised to read: *For Westinghouse designed fuel, the cross section of the test assembly was prototypical of a 17x17 fuel assembly.*
Page 29, Line 29	(Typographical): Change "AIs" to "RAIs"	The proposed change is accepted. The text has been revised to read: *The majority of the RAIs issued regarding information presented in fuel assembly test reports (References 8, 16, and 17) are discussed in this section.*

Page 29, Lines 33-35	Suggested text change: *The NRC staff also concluded that debris loads above 15 grams may be acceptable, but should be evaluated on a plant- or group-specific basis.*	The proposed change is not accepted. See response to Page 8, Line 5 for basis
Page 31, Lines 26-30	To remove addressing boric acid precipitation from the consideration of higher fiber limits, the suggested text modification is as follows: *The NRC staff notes that the evaluations conducted in the topical report and the testing did not account for the potential for boric acid precipitation.~~ and that this issue could affect LTCC in some cases. The NRC staff concluded that for a hot-leg break scenario at a fibrous debris limit of 15 grams per fuel assembly, LTCC would not be challenged because adequate coolant can flow through the core to maintain boric acid concentrations below the saturation limit. For the cold-leg break or hot-leg break scenarios where the licensee wishes to justify a higher fibrous debris limit such that flow through the core is decreased, the NRC staff concluded that boric acid concentration may affect LTCC. These effects should be addressed by industry as described in Section 8 of the WCAP.~~*	The proposed change is not accepted. See the response to Page 9, Lines 21-27.
Page 32, Lines 47-49	Suggested text change: *However, if licensees perform plant- or group-specific evaluations to increase hot-leg debris limits they must evaluate the cold-leg case to ensure that it is not more limiting.*	The proposed change is not accepted. See response to Page 8, Line 5 for basis
Page 33, Line 9	Regarding the parenthetical phrase *(less than 7.5 grams per fuel assembly, as stated by the WCAP)* The 7.5 gram figure is not stated in WCAP-16793 Revision 2. It is thus recommended that the parenthetical phrase be deleted.	The proposed change is accepted with modifications. The parenthetical text has been revised to read: *(less than 7.5 grams per fuel assembly)* Basis: The 7.5 gram value will be retained because, with hot-leg fibrous debris limited to 15 grams per assembly, it is the largest value that the staff expects to reach the core inlet for any cold-leg break.

Page 33, Lines 10-12	Lines 10-12: The statement *During fuel-assembly testing at cold leg break flow rates, a significant debris bed was not detectable (via differential pressure) at fiber loads below 7.5 grams.* is not correct. All cold leg break tests were performed with the first fiber addition equal to 10 grams, not 7.5 grams (WCAP-17057-P, Revision 2). There are no references to 7.5 gram additions in WCAP-16793-NP, Revision 2. It is suggested that the statement be revised to say *During fuel-assembly testing at cold leg break flow rates, a significant debris bed was not detectable (via differential pressure) at fiber loads below 10 grams.*	The proposed change is accepted with modifications. The text has been revised and a sentence has been added as follows: *During fuel-assembly testing at cold leg break flow rates, a significant differential pressure was not detected at fiber loads below 10 grams. Therefore the maximum anticipated fibrous debris load of 7.5 grams for a cold-leg break is acceptable.* See the comment for page 33, line 9.
Page 33, Lines 13-14	To remove addressing boric acid precipitation from the consideration of higher fiber limits, the suggested text modification is as follows: ~~*Licensees that credit debris limits greater than 15 grams per fuel assembly must evaluate the effects of additional debris on boric acid precipitation.*~~	The proposed change is not accepted. See the response to Page 9, Lines 21-27.
Page 33, Lines 50-51	Suggested text change: *The NRC staff finds it acceptable that industry take additional actions, including testing, to justify higher debris limits as justified for plant- or group-specific cases.*	The proposed change is not accepted. See response to Page 8, Line 5 for basis
Page 34, Lines 16-19	Suggested text change: *Licensees that credit alternate flow paths should demonstrate that the flow paths are effective; i.e, that the flow holes will not become blocked with debris during a LOCA, and that debris will not deposit in other locations after passing through the alternate flow path such that LTCC would be jeopardized.*	The proposed change is not accepted. The proposed wording changes the intent of the sentence as it limits the actions necessary (e.g., hydraulic analyses)

Page 34, Lines 19-20	To remove addressing boric acid precipitation from the consideration of higher fiber limits, it is recommended that the following sentence be deleted: ~~*Also, if credit for alternate flow paths leads to a boil-off condition in the core, boron precipitation issues must be addressed.*~~	The proposed change is not accepted. See the response to Page 18, Lines 5-10.
Page 34, Lines 26-28	Suggested text change: *The NRC staff finds that this is a potential flow path for coolant to enter the core, but notes that plant- or group-specific evaluations would have to assure that this is a viable flow path.*	The proposed change is not accepted. See response to Page 8, Line 5 for basis
Page 35, Lines 18-19	Suggested text change: *If the limits for the hot-leg case are increased by plant- or group-specific evaluation, the potential for the cold-leg case to become limiting should be addressed in the analysis.*	The proposed change is not accepted. See response to Page 8, Line 5 for basis
Page 36, Lines 10-14	Regarding the statement *"The AREVA test report (Reference 17) also states that the Westinghouse test facility provides repeatable test results…"* There is no such statement in Ref. 17 (AREVA 51-9170258-000). It is not clear from whence this statement originated, nor what is intended in regard to that report. The closest statement within the AREVA 51-9170258-000 that can be could find is: *"The AREVA and Westinghouse Fuel Assemblies behaved virtually identically in the RTU/Westinghouse loop. This is based on Figure 2-1, by comparing the AREVA and Westinghouse test assembly results in the RTU loop".* This is not a remark on test repeatability, but a remark on the fact that the two assemblies, when put in the same loop, give the same results. It is therefore suggested that the sentence in the draft SE be deleted.	The proposed change is accepted. The issue is a result of a typographical error. The text has been revised to read: *The Westinghouse test report (Reference 16) also stated that the Westinghouse test facility provides repeatable test results and that repeat tests are not required.* The SE inadvertently referenced the AREVA test report instead of the Westinghouse test report.

Page 37 Lines 4-6	Suggested text change: *However, the NRC staff concluded that the 15 gram per fuel assembly limit should be maintained for UPI plants until plant- or group-specific evaluations provide adequate justification for a higher limit.*	The proposed change is not accepted. See response to Page 8, Line 5 for basis
Page 37, Lines 25-26	Suggested text change: *Any debris amounts greater than those tested and accepted by the NRC staff should either be mitigated or be justified on a plant- or group-specific basis.*	The proposed change is not accepted. See response to Page 8, Line 5 for basis
Page 37, Lines 28-29	Suggested text change: *If plant- or group-specific evaluations increase debris limits in the future, cold-leg conditions may become important.*	The proposed change is not accepted. See response to Page 8, Line 5 for basis
Page 40, Lines 15-16	Suggested text change: *Licensees may perform plant- or group-specific evaluations to increase this limit.*	The proposed change is not accepted. See response to Page 8, Line 5 for basis
Page 41, Lines 13-14	(Typographical) - Suggested text change: ~~*Because t*~~*The results of some of the AREVA hot-leg case testing were significantly different from the results of the Westinghouse test under similar flow conditions.*	The proposed change is accepted. The text has been revised to read. *The results of some of the AREVA hot-leg case testing were significantly different from the results of the Westinghouse tests under similar flow conditions.*

Page 43, Lines 21-30	This paragraph discusses flow past the Trapper Fine Mesh® filter. While AREVA no longer manufactures this filter, the PWROG believes that the statement made by the NRC is not correct: *"As discussed above, based on the results of more recent testing, the NRC staff does not agree that the gaps between fuel assemblies have been shown to allow adequate flow to maintain LTCC."* While it is understood that this statement applies to gaps around the spacer grids, the gaps around the Trapper Fine Mesh® filter are different. The Trapper is a 3-D body that collects debris. As explained in Ref. 8 (AREVA 51-9102685-000), debris will collect to a certain dP, at which point all flow and debris goes through the gap around the Trapper fine mesh filter. The testing has demonstrated this. It is therefore suggested that the above sentence be deleted.	The proposed change is not accepted. The discussion in the SE is addressing the response to RAI 7 of reference 10 which discusses the ability for flow to pass through the gaps between fuel assemblies. Based on test results, the staff has concluded that the gaps between fuel assemblies can become blocked such that adequate flow cannot pass through the gaps. The discussion is not directed specifically at the fine mesh filter, but for all fuel inlet types.
Page 50, Lines 21-23	The first sentence can be interpreted to say that the PWROG did testing to determine adherence of fiber to fuel cladding. For clarity, it is recommended that the following change be made to the text: *The PWROG cited report ~~investigated the potential for fibrous material to collect on the fuel cladding by performing testing. The results of the tests are discussed and evaluated in~~ NEA/CNSI/R (95)11 (Reference 31) in its investigation of testing performed to evaluate the potential for fibrous material to collect on the fuel cladding.*	The proposed change is accepted. The subject text has been revised to read: *The PWROG cited report NEA/CSNI/R (95)11 (Reference 31) in its investigation of testing performed to evaluate the potential for fibrous material to collect on fuel cladding.*
Page 57, Lines 17-31	For consistency with PWROG comments on Limitation and Condition #8 (herein), the following test changes are suggested: *...Actual corrosion of aluminum coupons during ~~the~~ ICET 1, which used sodium hydroxide (NaOH), ~~test~~ appeared to occur in two stages; active corrosion for the first half of the test followed by passivation of the aluminum during the second half of the test. (NaOH is known to actively react with aluminum. The ICET tests that used tri-sodium phosphate (TSP) or sodium tetraborate did not have such an active early corrosion of aluminum.) Therefore, while the 30-day fit to the ICET data is reasonable, the WCAP-16530-NP-A model under-predicts aluminum release by about a factor of two during the active corrosion phase of ICET 1. This is important since the in-core LOCADM chemical deposition rates can be much greater during the initial period following a LOCA, if local conditions predict boiling. As stated in WCAP-16530-NP-A, to account for potentially*	The proposed change is accepted with modifications: Text has been modified to read as follows: *Actual corrosion of aluminum coupons during the ICET 1 test, which used sodium hydroxide (NaOH), appeared to occur in two stages; active corrosion for the first half of the test followed by passivation of the aluminum during the second half of the test. Therefore, while the 30-day fit to the ICET data is reasonable, the WCAP-16530-NP-A model under-predicts aluminum release by about a factor of two during the active*

	greater amounts of aluminum during the initial days following a LOCA, a licensee's LOCADM input should apply a factor of 2 increase to the WCAP-16530-NP-A spreadsheet predicted aluminum release, not to exceed the total amount of aluminum predicted by the WCAP-16530-NP-A spreadsheet for 30 days. In other words, the total amount of aluminum released equals that predicted by the WCAP-16530-NP-A spreadsheet, but the timing of the release is accelerated. This applies to plants that use NaOH as a buffer. Alternately, licensees may choose to use a different method for determining aluminum release, including the use of ICET tests run with TSP or sodium tetraborate buffers.~~but~~ I Licensees that use NAOH as a buffer should not use an aluminum release rate equation that under-predicts the aluminum concentrations measured during the initial 15 days of ICET 1	*corrosion phase of ICET 1. This is important since the in-core LOCADM chemical deposition rates can be much greater during the initial period following a LOCA, if local conditions predict boiling. As stated in WCAP16530-NP-A, to account for potentially greater amounts of aluminum during the initial days following a LOCA, a licensee's LOCADM input should apply a factor of 2 increase to the WCAP-16530-NP-A spreadsheet predicted aluminum release, not to exceed the total amount of aluminum predicted by the WCAP-16530-NP-A spreadsheet for 30 days. In other words, the total amount of aluminum released equals that predicted by the WCAP-16530-NP-A spreadsheet, but the timing of the release is accelerated. Alternately, licensees may choose to use a different method for determining aluminum release but licensees should not use an aluminum release rate equation that, when adjusted to the ICET 1 pH, under-predicts the aluminum concentrations measured during the initial 15 days of ICET 1.* Basis for staff action: Limitation and Condition #8, related to the aluminum release rate, is not limited to NaOH plants only. While the aluminum corrosion rate for the TSP and sodium tetraborate (STB) ICET tests were much lower than the ICET 1 test (with NaOH), the observed difference is strongly influenced by the significantly lower pH in those tests compared to ICET 1. The aluminum release rate equation has

		terms that already consider pH, so plants with TSP or STB buffer that have lower post-LOCA pool pH should predict less aluminum release. In addition, relying solely on the ICET tests to exclude a more active corrosion of aluminum in TSP and STB buffered fluid doesn't account for the fact that much of the aluminum corrosion occurs at temperatures greater than the constant 140^{0}F ICET test temperature.
Page 57, Lines 34-37	Suggested text change: *If a licensee uses <u>plant- or group-specific</u> refinements to reduce the chemical source term calculated by the WCAP-16530-NP-A base model, the licensees should provide technical justification demonstrating that the refined chemical source term adequately bounds the postulated plant chemical product generation.*	The proposed change is not accepted. See response to Page 8, Line 5 for basis.
Page 59, lines 15-17	The sentence *Since the assumed deposit thermal conductivity has a significant effect on the heat transfer analysis, the use of a 0.11 BTU/(h-ft-°F) value for sodium aluminum silicate scale needs to be justified.* is inconsistent with lines 18-20 of this same paragraph; *If plant-specific calculations use a less conservative thermal conductivity value for scale (i.e., greater than 0.11 BTU/(h-ft-°F)), the NRC staff expects the licensee to provide a technical justification for the plant-specific thermal conductivity to the NRC staff.* and with Limitation and Condition #10 (page 73, lines 12-17); *. . . The WCAP recommends that a thermal conductivity of 0.11 BTU/(h-ft-°F) be used for the sodium aluminum silicate scale and for bounding calculations when there is uncertainty in the type of scale that may form. If plant-specific calculations use a less conservative thermal conductivity value for scale (i.e., greater than 0.11 BTU/(h-ft-°F)), the licensee should provide a technical justification for the plant-specific thermal conductivity. . ."*	The proposed change is accepted, correcting an editing error in the SE. The SE has been revised to replace the existing text with the following: *Since the assumed deposit thermal conductivity has a significant effect on the heat transfer analysis, the use of a value less conservative (greater) than 0.11 BTU/(h-ft-°F) for sodium aluminum silicate scale needs to be justified.*

	Therefore, it is suggested that lines 15-17 of page 59 be amended to read as follows: *Since the assumed deposit thermal conductivity has a significant effect on the heat transfer analysis, the use of a value less conservative (greater) than 0.11 BTU/(h-ft-°F) ~~value~~ for sodium aluminum silicate scale needs to be justified. "*	
Page 61, Lines 47-48	Regarding the statement *"The WCAP states that the effect of debris that passes through the ECCS sump strainer during a LOCA is not addressed in the WCAP."* WCAP-16793-NP, Revision 2 was written to address the effect of debris that passes through the ECCS sump strainer post-LOCA. This introduction should say *"The WCAP does not address boric acid mixing and transport in the RCS and potential precipitation mechanisms that may occur during the sump recirculation phase of a LOCA".*	The proposed change is accepted. The text has been revised to read: *The WCAP does not address boric acid mixing and transport in the RCS and potential precipitation mechanisms that may occur during the sump recirculation phase of a LOCA.*
Page 62, Lines 3-11	Section 8 of WCAP-16793-NP Rev 2 only acknowledges that "additional insights and new methodologies are needed to answer fundamental questions about boric acid mixing and transport in the RCS and potential precipitation mechanisms that may occur both during the ECCS injection phase and the sump recirculation phase after a LOCA", and makes no statement on LTCC as presented in the draft SE. It is suggested that the text be modified as follows: *Section 8 of Reference 18 states that the effects of boron precipitation are not addressed in TR WCAP-16793-NP, Revision 2. ~~This statement is in acknowledgement that effective LTCC involves (1) provision of sufficient coolant flow to the core to remove decay heat without unacceptable fuel clad heat-up, and (2) prevention of boric acid precipitation sufficient to inhibit adequate core cooling. In response to NRC findings made during a technical audit of Westinghouse Topical Report CENPD-254-P (Reference 41), the PWROG initiated a separate program to address staff questions related to boric acid precipitation, including the effect of debris accumulating in the core. NRC staff finds this approach acceptable if a plant limits the amount of fiber bypassing the strainer to the amount approved by this SE.~~*	The proposed change is accepted with modifications. The section has been revised to read: *Section 8 of Reference 18 states that the effects of boron precipitation are not addressed in TR WCAP-16793-NP, Revision 2. Effective LTCC involves (1) provision of sufficient coolant flow to the core to remove decay heat without unacceptable fuel clad heat-up, and (2) prevention of boric acid precipitation sufficient to inhibit adequate core cooling. In response to NRC findings made during a technical audit of Westinghouse Topical Report CENPD-254-P (Reference 41), the PWROG initiated a separate program to address staff questions related to boric acid precipitation, including the effect of debris accumulating in the core. The*

		NRC staff finds this approach acceptable if a plant limits the amount of fiber bypassing the strainer to the amount approved by this SE.
Page 62, Lines 22-24	Regarding the statement, *Fuel assembly testing for cold leg conditions at low fiber amounts of approximately 7 grams (e.g., after the first batches of fiber addition) do not exhibit noticeable head loss.* There are no references to fiber amounts of 7 grams in WCAP-16793-NP, Revision 2. All cold leg break tests were performed with the first fiber addition equal to 10 grams (WCAP-17057-P, Revision 2) and should be stated as such in the text above. The suggested text revision is: *Fuel assembly testing for cold leg conditions at low fiber amounts ~~of approximately 7 grams (e.g., after the first batches of fiber addition)~~ does not exhibit noticeable head loss.*	The proposed change is accepted with modifications. The sentence has been revised to read: *Fuel assembly testing at cold leg flow rates with low fiber amounts--approximately 10 grams per fuel assembly--did not exhibit a noticeable head loss. This indicates that the fiber beds will be minimal at the maximum expected fibrous debris load of 7.5 grams per fuel assembly and, therefore, no significant impact on boron precipitation is expected.* See the response to comment on page 33, line 9 and page 33, lines 10-12.
Page 62, Lines 28-36	To remove addressing boric acid precipitation from the consideration of higher fiber limits, it is recommended that this paragraph be deleted in its entirety.	The proposed change is not accepted. See the response to Page 9, Lines 21-27.
Page 65, Lines 26-28	Suggested text change: *If UPI plant- or group-specific evaluations are conducted to increase the accepted debris limits, the ability to ensure any required cold-leg flushing should be demonstrated.*	The proposed change is not accepted. See response to Page 8, Line 5 for basis
Page 67, Lines 30-32	Suggested text change: 3. *Perform plant-specific LOCADM evaluations (WCAP, Section 7 and Appendix E) and ~~prove~~ confirm that their plant-specific evaluations are bounded by the 800 degrees Fahrenheit acceptance criterion.*	The proposed change is accepted. The text has been revised to read: *Perform plant-specific LOCADM evaluations (WCAP, Section 7 and Appendix E) and confirm that their plant-specific evaluations are bounded by the 800 degrees Fahrenheit acceptance criterion.*

Page 69, Lines 10-12	Suggested text change: *3. Utilities may conduct plant- or group-specific tests to increase the fiber limit. However, the NRC staff expects plant- or group-specific tests to ensure margins adequate to address concerns with the test program discussed throughout this SE*	The proposed change is not accepted. See response to Page 8, Line 5 for basis
Page 70, Lines 3-4	To remove addressing boric acid precipitation from the consideration of higher fiber limits, it is recommended that the following sentence be deleted: ~~*The potential for debris to affect boric acid precipitation analyses should be included in any evaluation that increases debris loading.*~~	The proposed change is not accepted. See the response to Page 9, Lines 21-27.
Page 70, Lines 6-7	The following change is suggested: *The NRC staff concluded that plants need to demonstrate that the quantity of fibrous debris that ~~bypasses the strainer and~~ transports to the ~~core~~fuel inlet is less than or equal to 15 grams per fuel assembly or as otherwise justified on a plant- or group-specific basis.*	The proposed change is accepted with modifications. The text has been revised to read: *The NRC staff concluded that plants need to demonstrate that the quantity of fibrous debris that transports to the fuel inlet is less than or equal to 15 grams per fuel assembly or as otherwise justified on a plant-specific basis.* The comment regarding group specific testing is not accepted. See comment for page 8, line 5.
Page 71 Limitation and Condition 1, Lines 25-26	The following change is suggested: *Licensees should limit the amount of fibrous debris reaching the fuel inlet ~~passing through the ECCS sump strainer~~ to that stated in section 10 of the WCAP (15 grams per fuel assembly).*	The proposed change is accepted with modifications. The text has been revised to read: *Licensees should limit the amount of fibrous debris reaching the fuel inlet to that stated in Section 10 of the WCAP (15 grams per fuel assembly for a hot-leg break scenario).*

Page 71 Limitation and Condition 1, Lines 28-29	Suggested text change: *Alternately, licensees may perform plant- or group- specific testing and/or evaluations to increase the fibrous debris limits…*	The proposed change is not accepted. See comment for page 8 for discussion on group-specific testing. It is not necessary to specify fibrous in all cases where debris limits are referenced because fibrous debris is the only type that has a limit associated with it.
Page 71, Limitation and Condition 1 Lines 29-32	The phrase, "*and also consider loop seal clearing*" represents an increase in the scope of GSI-191 that runs counter to the direction of the Commission. Loop seal clearing concerns require very specific conditions that are not consistent with GSI-191 conditions for limiting debris bed formation and resulting head loss. Furthermore, the topic of loop seal clearing is the subject of another PWR Owners Group program. Therefore, it is recommended that this phrase be stricken from Limitation and Condition #1.	The reason for the statement is to alert licensees that they may need to more accurately calculate the driving head if higher debris loads are pursued. Loop seal clearing is an important consideration for some break locations. The proposed change is accepted with modifications. The text has been revised to read: *The available driving head should be calculated based on the core exit void fraction and loop flow resistance values contained in their plant design basis calculations, considering clean loop flow resistance and a range of break locations.*
Page 71, Limitation and Condition 1 Lines 33-35	To remove addressing boric acid precipitation from the consideration of higher fiber limits, the suggested text modification is as follows: *These tests shall evaluate the effects of increased fiber on flow to the core~~, and precipitation of boron during a postulated cold-leg break,~~ and the effect of p/f ratios below 1:1*	The proposed change is not accepted. See the response to Page 9, Lines 21-27.
Page 71 Limitation and Condition 1, Lines 35-37	Suggested text change: *The NRC staff will review plant- or group- specific evaluations, including hot- and cold-leg break scenarios, to ensure that acceptable justification for higher fibrous debris limits is provided.*	The proposed change is not accepted. See comment for page 71, lines 28-29.

Page 71, Lines 48-51 and Page 72, Lines 1-3 Limitation and Condition 3	To remove addressing boric acid precipitation from the consideration of higher fiber limits, the suggested text modification is as follows: *If a licensee chooses to take credit for alternate flow paths, such as core baffle plate holes, to justify greater than 15 grams of bypassed fiber per fuel assembly, the licensee should demonstrate, by testing or analysis, that the flow paths would be effective, that the flow holes will not become blocked with debris during a LOCA, ~~that boron precipitation is considered,~~ and that debris will not deposit in other locations after passing through the alternate flow path such that LTCC would be jeopardized.*	The proposed change is not accepted. See the response to Page 18, Lines 5-10.
Page 72 Limitations and Conditions #4	Limitation and Condition #4 is silent on the calculations with an increased head loss coefficient uniformly applied to the bottom of the core (i.e., no unblocked or "open" fuel area) that are documented in Appendix B of the WCAP and are also discussed in Section 3.3.3. Furthermore, this Limitation and Condition incorrectly implies that the increase in loss coefficients included specific head loss information regarding debris bed formation but did not include chemical effects. From Appendix B of WCAP 16793-NP Revision 2, it is clear that the loss coefficients were increased as a parametric study; the increase was not ascribed to either the development of a fiber bed with debris and or post-accident chemical products. Therefore, it is respectfully concluded that the underlined statement in the Limit and Condition incorrectly characterizes the analysis approach. The amended Limit and Condition #4 given below is respectfully suggested as an accurate and technically correct replacement for the Limit and Condition 4 in the draft SE. *Sections 3.2 and 3.3 and Appendix B of the WCAP provide evaluations to show that even with large blockages at the core inlet or with a large loss coefficient applied uniformly at the core inlet, adequate flow will enter the core to maintain LTCC. In all cases, the blockage assumptions are implemented by increasing the loss coefficient at the entrance to the fuel region. This increase in loss coefficient is not ascribed to or associated with the formation of debris beds or deposition of post-accident chemical products on and or within debris beds. The use of these analyses to demonstrate adequate LTCC requires the licensee to equate loss coefficients used in the calculation with a loss coefficient obtained from fuel assembly testing.*	The proposed change is accepted. The text has been revised to read: *Sections 3.2 and 3.3 of the WCAP provide evaluations to show that even with large blockages at the core inlet, adequate flow will enter the core to maintain LTCC. The staff recognizes that these calculations show that significant head loss can occur while maintaining adequate flow. However, the analyses have not been correlated with debris amounts. Therefore, the analyses cannot be relied upon to demonstrate adequate LTCC. (Sections 3.3.3 and 3.4 of this SE)*

Page 72 Limitations and Conditions #5, Lines 16-18	The following change is recommended: *Any debris amounts greater than those justified by generic testing in this WCAP must be justified on a plant- or group-specific basis.*	The proposed change is not accepted. See response to Page 8, Line 5 for basis
Page 72 Limitations and Conditions # 6 Lines 20-21	The following change is recommended: *The fibrous debris acceptance criteria contained in the WCAP may be applied to fuel designs evaluated in the WCAP."*	The proposed change is accepted. The text has been revised to read: *The fibrous debris acceptance criteria contained in the WCAP may be applied to fuel designs evaluated in the WCAP.*
Page 72 Limitation and Condition 8	Limitation and Condition #8 requires licensees to use the aluminum release rate observed in ICET 1. It is noted that ICET 1 utilized sodium hydroxide (NaOH) as a buffer. Aluminum is known to be very reactive In a NaOH solution. The requirement that all plants use aluminum release associated with ICET 1, rather than the aluminum release rate associated with their specific buffer places an unnecessary and an inappropriate burden on plants not using NaOH as a buffer agent. It is thus suggested that the text in Limitation and Condition #8 be revised as follows: *15. As described in the Limitations and Conditions for WCAP-16530-NP (ADAMS Accession No. ML073520891) (Reference 21)[5], the aluminum release rate equation used in TR WCAP-16530-NP provides a reasonable fit to the total aluminum release for the 30-day ICET tests but under-predicts the aluminum concentrations during the initial active corrosion portion of ~~the~~ test ICET 1. ICET 1 used sodium hydroxide (NaOH) as a buffer agent, which is known to actively react with aluminum. To provide more appropriate levels of aluminum for the LOCADM analysis for in-vessel effects in the initial days following a LOCA, licensees using NaOH as a buffer should apply a factor of two to the aluminum release rate as determined by the TR WCAP-16530-NP-A spreadsheet. The total aluminum considered does not need to exceed the total predicted by the WCAP-16530-NP-A spreadsheet for 30 days. Alternately, if a licensee chooses to use a different method for determining the aluminum release, it should demonstrate that the method does not under-predict the aluminum concentrations measured during the initial 15 days of the ICET ~~1~~ test*	The proposed change is accepted with modifications. The following paragraph has been added: *Actual corrosion of aluminum coupons during the ICET 1 test, which used sodium hydroxide (NaOH), appeared to occur in two stages; active corrosion for the first half of the test followed by passivation of the aluminum during the second half of the test. Therefore, while the 30-day fit to the ICET data is reasonable, the WCAP-16530-NP-A model under-predicts aluminum release by about a factor of two during the active corrosion phase of ICET 1. This is important since the in-core LOCADM chemical deposition rates can be much greater during the initial period following a LOCA, if local conditions predict boiling. As stated in WCAP16530-NP-A, to account for potentially greater amounts of aluminum during the initial days following a LOCA, a licensee's LOCADM input should apply a factor of 2 increase to the WCAP-16530-NP-A spreadsheet predicted aluminum release, not to exceed the total amount*

	run with the same buffer agent as is used in the plant. (Section 3.7 of this SE)	*of aluminum predicted by the WCAP-16530-NP-A spreadsheet for 30 days. In other words, the total amount of aluminum released equals that predicted by the WCAP-16530-NP-A spreadsheet, but the timing of the release is accelerated. Alternately, licensees may choose to use a different method for determining aluminum release but Licensees should not use an aluminum release rate equation that, when adjusted to the ICET 1 pH, under-predicts the aluminum concentrations measured during the initial 15 days of ICET 1.* Basis for staff action: Limitation and Condition #8, related to the aluminum release rate, is not limited to NaOH plants only. While the aluminum corrosion rate for the TSP and sodium tetraborate (STB) ICET tests were much lower than the ICET 1 test (with NaOH), the observed difference is strongly influenced by the significantly lower pH in those tests compared to ICET 1. The aluminum release rate equation has terms that already consider pH, so plants with TSP or STB buffer that have lower post-LOCA pool pH should predict less aluminum release. In addition, relying solely on the ICET tests to exclude a more active corrosion of aluminum in TSP and STB buffered fluid doesn't account for the fact that much of the aluminum corrosion occurs at temperatures greater than the constant 140^{0}F ICET test temperature.

Page 73 Limitation and Conditions #10 Line 78	Suggested text change: *...technical justification for the plant-specific thermal conductivity value.*	The proposed change is accepted. The text has been revised to read: *If plant-specific calculations use a less conservative thermal conductivity value for scale (i.e., greater than 0.11 BTU/(h-ft-°F)), the licensee should provide a technical justification for the plant-specific thermal conductivity value.*
Page 73 Limitations and Conditions # 11 Lines 22-28	Condition 11 should be revised to say *Licensees should demonstrate that the quantity of ~~strainer-bypassed~~ fibrous debris transported to the ~~core~~ fuel inlet is less than or equal to the fibrous debris limit specified in the proprietary fuel assembly test reports and approved by this SE. Fiber quantities in excess of 15 grams per fuel assembly must be justified by the licensee.*	The proposed change is accepted. The text has been revised to read: *Licensees should demonstrate that the quantity of fibrous debris transported to the fuel inlet is less than or equal to the fibrous debris limit specified in the proprietary fuel assembly test reports and approved by this SE.*
Page 73 Limitation and Condition 11 Lines 32-35	It is not clear whether the expected fiber loading in the core for the cold-leg and hot-leg break scenarios is required to be provided for all plants, or for only those plants that do not meet the 15g/FA limit. Please clarify.	The proposed change is accepted. The text has been revised to read as follows: *Licensees should demonstrate that the quantity of fibrous debris transported to the fuel inlet is less than or equal to the fibrous debris limit specified in the proprietary fuel assembly test reports and approved by this SE. Fiber quantities in excess of 15 grams per fuel assembly must be justified by the licensee. Licensees may determine the quantity of debris that passes through their strainers by (1) performing strainer bypass testing using the plant strainer design, plant-specific debris loads, and plant-specific flow velocities, (2) relying on strainer bypass values developed through strainer bypass testing of the same vendor and same perforation size,*

		prorated to the licensee's plant-specific strainer area; approach velocity; debris types, and debris quantities, or (3) assuming that the entire quantity of fiber transported to the sump strainer passes through the sump strainer. The licensee's submittals should include the means used to determine the amount of debris that bypasses the ECCS strainer and the fiber loading expected, per fuel assembly, for the cold-leg and hot-leg break scenarios. Licensees of all operating PWRs should provide the debris loads, calculated on a fuel assembly basis, for both the hot-leg and cold-leg break cases in their GL 2004-02 responses.
Page 73 Limitation and Condition 12 (Lines 37-40)	Suggested text change: *Plants or groups of plants that can qualify a higher fiber load based on the absence of chemical deposits should ensure that tests for their plant- or group-specific conditions ~~search for~~ determine limiting head losses using particulate and fiber loads that maximize the head loss with no chemical precipitates included in the tests.*	The proposed change is accepted. The text has been revised to read: *Plants that can qualify a higher fiber load based on the absence of chemical deposits should ensure that tests for their conditions determine limiting head losses using particulate and fiber loads that maximize the head loss with no chemical precipitates included in the tests.* The "groups of plants" clarification is not accepted. See comment for Page 8, Line 5

Page 73 Limitation and Condition 13	Consistent with discussions at the May, 2012 ACRS meeting, it is recommended that Limitation and Condition 13 be revised as follows: *"Licensees should verify that the size distribution of fibrous debris used in the fuel assembly testing referenced by their plant trends with the size distribution of fibrous debris expected downstream of the plant's ECCS strainer(s). (Section 3.4.2.1 of this SE)"*	The proposed change is accepted with modifications. The text has been revised to read: *Licensees should verify that the size distribution of fibrous debris used in the fuel assembly testing referenced by their plant is representative of the size distribution of fibrous debris expected downstream of the plant's ECCS strainer(s).* Basis for staff action: The staff considered the ACRS comments during the referenced meeting and believes that the wording above better represents the intent of the committee. That is, licensees should verify that the fiber size distribution used in testing is relatively close to that expected for their plant. The use of the term "trends with" does not convey this concept well. Therefore, the staff decided to use the term "is representative of."
Page 74 Conclusions Lines 10-12	Suggested text change: *... in Section 4.0 of this SE, provides acceptable, plant- or group-specific evaluation methods for demonstrating that adequate coolant flow reaches the core to maintain fuel clad temperature within acceptable limits.*	The proposed change is not accepted. See response to Page 8, Line 5 for basis.

PWR Owners Group
Member Participation* for Project PA-SEE-0312

Utility Member	Plant Site(s)	Participant	
		Yes	No
Ameren Missouri	Callaway	X	
American Electric Power Co.	D.C. Cook 1&2	X	
Arizona Public Service Co.	Palo Verde Unit 1, 2, & 3	X	
Constellation Energy Group	Calvert Cliffs 1 & 2	X	
Constellation Energy Group	Ginna	X	
Dominion Kewaunee	Kewaunee	X	
Dominion Nuclear Connecticut	Millstone 2 & 3	X	
Dominion VA Power	North Anna 1 & 2, Surry 1 & 2	X	
Duke Energy	Catawba 1 & 2, McGuire 1 & 2, Oconee 1, 2, 3	X	
Entergy Nuclear Northeast	Indian Point 2 & 3	X	
Entergy South	Arkansas 1 & 2, Waterford 3	X	
Entergy – Palisades	Palisades	X	
Exelon Generation Co. LLC	Braidwood 1 & 2, Byron 1 & 2, TMI 1	X	
FirstEnergy Nuclear Operating Co.	Beaver Valley 1 & 2, Davis-Besse	X	
FPL / NextEra	St. Lucie 1 & 2, Turkey Point 3 & 4, Seabrook, Pt. Beach 1 & 2	X	
Luminant	Comanche Peak 1 & 2	X	
XCEL Energy	Prairie Island 1 & 2	X	
Omaha Public Power District	Fort Calhoun	X	
Pacific Gas & Electric	Diablo Canyon 1 & 2	X	
Progress Energy	Robinson 2, Shearon Harris, Crystal River 3	X	
PSEG - Nuclear	Salem 1 & 2	X	
Southern California Edison Co.	SONGS 2 & 3	X	
South Carolina Electric & Gas	V.C. Summer	X	
So. Texas Project Nuclear Operating Co.	South Texas Project 1 & 2	X	
Southern Nuclear Operating Co.	Farley 1 & 2, Vogtle 1 & 2	X	
Tennessee Valley Authority	Sequoyah 1 & 2, Watts Bar	X	
Wolf Creek Nuclear Operating Co.	Wolf Creek	X	

* Project participants as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending this document to participants not listed above.

PWR Owners Group
International Member Participation* for Project PA-SEE-0312

Utility Member	Plant Site(s)	Participant	
		Yes	No
AXPO AG	Beznau 1 & 2	X	
British Energy Ltd.	Sizewell B	X	
Electrabel (Belgian Utilities)	Doel 1, 2 & 4, Tihange 1 & 3	X	
Electricite de France	58 Units	X	
Electronuclear ETN	ANGRA 1	X	
Hokkaido	Tomari 1 & 2	X	
Japan Atomic Power Company	Tsuruga 2	X	
Kansai Electric Co., Ltd	Mihama 1, 2, & 3, Ohi 1, 2, 3 & 4, Takahama 1, 2, 3 & 4	X	
Korea Hydro and Nuclear Power Corp.	Kori 1, 2, 3, & 4 Yonggwang 1 & 2	X	
Korea Hydro and Nuclear Power Corp.	Yonggwang 3, 4, 5 & 6; Ulchin 3, 4, 5, & 6	X	
Kyushu	Genkai 1, 2, 3 & 4, Sendai 1 & 2	X	
Nuklearna Elektrarna KRSKO	Krsko	X	
Ringhals AB	Ringhals 2, 3 & 4	X	
Shikoku	Ikata 1, 2 & 3	X	
Spanish Utilities	Asco 1 & 2, Vandellos 2, Almaraz 1 & 2	X	
Taiwan Power Co.	Maanshan 1 & 2	X	
AXPO AG	Beznau 1 & 2	X	

* This is a list of participants in this project as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending documents to participants not listed above.

RECORD OF REVISIONS

Revision	Date	Description
0	May 2007	Original
1	April 2009	Rev bars are not included in this document because this revision required a total reorganization of the original document. The text is organized as follows: 1 Introduction • Section 1.0 from original WCAP 2 Long-Term Core Cooling Acceptance Basis • Section 3, a few details from Appendix A from original WCAP and applicable RAIs 3 Blockage at the Core Inlet • Sections 2.1 & 6 from original WCAP, add testing w/ prototypical fuel filters, and applicable RAIs 4 Collection of Debris on Fuel Grids • Sections 2.2 & 4 from original WCAP, add testing w/ prototypical fuel filters, and applicable RAIs 5 Collection of Fibrous Material on Fuel Cladding • Section 2.3 from original WCAP and applicable RAIs 6 Protective Coating Debris Deposited on Fuel Clad Surfaces • Section 2.5 from original WCAP 7 Chemical Precipitates and Debris Deposited on Fuel Clad Surfaces • Sections 2.4 & 5, and applicable RAIs 8 Boric Acid Precipitation • Section 2.6 from original WCAP 9 Coolant Delivered to the Top of the Core • Section 2.7 from original WCAP and applicable RAIs 10 Summary • Sections 2.8 & 7 from original WCAP and addition of acceptance criteria for fuel debris loading during long-term recirculation from containment sump. 11 References Appendix A • Keep and add applicable RAIs Appendix B • Keep and add applicable RAIs Appendix C • Keep and add applicable RAIs Appendix D • Keep and add applicable RAIs Appendix E • Keep and add applicable RAIs Appendix F • Keep and add applicable RAIs

Revision	Date	Description
		Appendix G – Description of FA Testing • New – Add new appendix to describe the fuel assembly testing (in general). The references will provide a hook to the actual test descriptions and data. Appendix H • New – RAI Set #1 Appendix I • New – RAI Set #2 Appendix J • New – RAI Set #3
2-A	September 2011	Revision bars are used to highlight changes. General • References (Section 11) were updated from alpha-numeric to numeric. These changes were reflected throughout the text. Executive Summary • Updated to include results from new FA testing. Fiber limit = 15 g/FA. Section 2 • Updated to include results from new FA testing. Fiber limit = 15 g/FA. Section 3 • Updated to reflect most recent FA testing goals. That is, testing was conducted to address RAIs and define a maximum allowable fiber limit. Section 4 • Updated to include new observations regarding debris collection at spacer grids in Section 4.2. Section 10 • Updated to include results from new FA testing. Fiber limit = 15 g/FA. Appendix A • Updated to include results from new FA testing. Fiber limit = 15 g/FA. Appendix G • Updated to include results from new FA testing. Included detailed discussion on loop conservatisms. Appendix K • New. Most recent RAI responses.
2	See EDMS	Incorporated comments from participants of PA-SEE-0312, including: • Expansion of fiber limit options. Clarified there are plant-specific options that can be pursued to allow for additional fibrous debris. • Editorial changes. • Edits of Section 10 and Appendix G.

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LIST OF ACRONYMS

B&W	Babcock & Wilcox Co.
B/B	barrel/baffle
BELOCA	best estimate LOCA
BOA	boron-induced offset anomaly
BWST	borated water storage tank
CCFL	counter current flow limitation
CE	Combustion Engineering
CL	cold leg
CLL	collapsed liquid level
CSS	containment spray system
DC	downcomer
dP or ΔP	differential pressure
ECCS	emergency core cooling system
ECR	equivalent clad reacted
FA	fuel assembly
GL	generic letter
GSI	generic safety issue
HA	hot assembly
HL	hot leg
HLSO	hot leg switchover
HPSI	high pressure safety injection
LPSI	low pressure safety injection
LBB	leak before break
LOCA	loss-of-coolant accident
LOADM	LOCA deposition analysis model
LP	lower power
LTCC	long-term core cooling
NRC	Nuclear Regulatory Commission
OD	outer diameter
OEM	original equipment manufacturer
P:F	particulate-to-fiber
PCT	peak cladding temperature
PWR	pressurized water reactor
PWROG	PWR Owners Group
RAI	request for additional information
RCS	reactor coolant system
RHR	residual heat removal
RV	reactor vessel
RVI	reactor vessel internals
RVVV	reactor vessel vent valves
RWST	refueling water storage tank
SG	steam generator
SIRWT	safety injection and refueling water tank
UP	upper plenum
UPI	upper plenum injection
WC/T	WCOBRA/TRAC

EXECUTIVE SUMMARY

Pressurized water reactor (PWR) containment buildings are designed both to contain radioactive materials releases and to facilitate core cooling in the event of a postulated loss-of-coolant-accident (LOCA). The cooling process requires water discharged from the break and containment spray to be collected in a sump for recirculation by the emergency core cooling system (ECCS) and the containment spray system (CSS). Typically, a containment sump contains one or more screens in series that protect the components of the ECCS from debris that could be washed into the sump. Fibrous debris could form a mat on the screen that would collect particulates, keeping them from being ingested into the ECCS and CSS flow paths. However, as the fiber bed forms, particulates and some fibrous material may be ingested into the ECCS and subsequently, into the reactor coolant system (RCS).

Concerns have been raised about the potential for debris ingested into the ECCS to affect long term core cooling (LTCC) when recirculating coolant from the containment sump. The fuel assembly (FA) bottom nozzles are designed with flow passages that provide coolant flow from the reactor vessel lower plenum into the region of the fuel rods. During operation of the ECCS to recirculate coolant from the containment sump, debris in the recirculating fluid that passes through the sump strainer may collect on the bottom surface of the FA bottom nozzle, causing resistance to flow through this path. The collection of sufficient debris on the FA bottom nozzle is postulated to impede flow into the FA and core. Other concerns have been raised with respect to the collection of debris and post accident chemical products within the core itself. Specifically, the debris has been postulated to either form blockages or adhere to the cladding, thereby reducing the ability of the coolant to remove decay heat from the core. Similarly, chemical precipitates have been postulated to plate out on fuel cladding, again resulting in a reduction of the ability of the coolant to remove decay heat from the core.

Guidance provided to the industry in the following documents has been used as the framework for analyses that address these concerns.

- WCAP-16406-P-A: Section 9.0 (cold leg injection, hot leg injection, fiber, particulates, etc.), including addenda
- NEI 04-07, Volume 1: Section 7.3
- NEI 04-07, Volume 2: Section 7.3
- Draft NRC Staff Review Guidance for Evaluation of Downstream Effects of Debris Ingress into the PWR RCS on Long Term Core Cooling Following a LOCA, dated November 22, 2005.

The Pressurized Water Reactor Owners Group (PWROG) undertook a program to provide additional analyses and information on the effect of debris and chemical products on core cooling for PWRs when the ECCS is realigned to recirculate coolant from the containment sump. The objective of the program was to demonstrate reasonable assurance that sufficient LTCC is achieved for PWRs to satisfy the requirements of 10 CFR 50.46 with debris and chemical products that might be transported to the reactor vessel and core by the coolant recirculating from the containment sump. This program supersedes the efforts documented in WCAP-16406-P-A, Section 9. The debris composition includes particulate and fiber debris, as well as post-accident chemical products. The program was performed such that the results of this program are bounding and apply to the fleet of PWRs, regardless of the design of the plant (Westinghouse, Combustion Engineering [CE], or Babcock & Wilcox [B&W]) or fuel vendor (Westinghouse or AREVA).

This evaluation considered the design of the PWR, the design of the open-lattice fuel, the design and tested performance of replacement containment sump strainers, the tested performance of materials inside containment, and the tested performance of fuel assemblies in the presence of debris. Specific areas addressed in this evaluation include:

- Blockage at the core inlet
- Collection of debris on fuel grids
- Collection of fibrous material on fuel cladding
- Protective coating debris deposited on fuel clad surfaces
- Production and deposition of chemical precipitants
- Coolant delivered from the top of the core

The following acceptance bases were selected for the evaluation of the topical areas identified above:

1. The maximum clad temperature shall not exceed 800°F.
2. The thickness of the cladding oxide and the fuel deposits shall not exceed 0.050 inch in any fuel region.

These acceptance bases were applied after the initial quench of the core and are consistent with the LTCC requirements stated in 10 CFR 50.46 (b)(4) and 10 CFR 50.46 (b)(5). They do not represent, nor are they intended to be, new or additional LTCC requirements. These acceptance bases provide for demonstrating that local temperatures in the core are stable or continuously decreasing and that debris entrained in the cooling water supply will not affect decay heat removal. The 800°F temperature was selected based on autoclave data that demonstrated oxidation and hydrogen pickup to be well behaved at and below the 800°F temperature and the reduction in cladding small. Therefore, there would be minimal reduction in post-LOCA load carrying capability. A discussion of the technical basis for the 800°F temperature is given in Appendix A. The 0.050 inch limit for oxide plus deposits was selected so as to preclude the formation of deposits that would bridge the space between adjacent rods and block flow between fuel channels.

In addition to these acceptance criteria, utilities must evaluate site-specific fiber loading against the debris load acceptance criteria provided in this document. Plants with debris loads above the debris load acceptance criteria may demonstrate adequate LTCC capability through engineering evaluations of plant-specific conditions and/or plant-specific testing.

In order to demonstrate reasonable assurance of LTCC, all plants must evaluate the areas identified above, demonstrate they are bounded by the maximum fuel cladding temperature and maximum deposit thickness requirements and evaluate the site-specific fiber loading against the developed debris load acceptance criteria. Specifically,

- Adequate flow to remove decay heat will continue to reach the core even with debris from the sump reaching the RCS and core. Plants that follow the guidance provided in Section 10 can state that debris that bypasses the strainer will not build an impenetrable blockage at the core inlet. While any debris that collects at the core inlet will provide some resistance to flow, in the extreme case that a large blockage does occur, numerical analyses have demonstrated that core decay heat removal will continue. The details of this evaluation are provided in Section 3.

- Decay heat will continue to be removed even with debris collection at the FA spacer grids. Plants that follow the guidance provided in Section 10 can state that debris that bypasses the screen will not build an impenetrable blockage at the fuel spacer grids. This assertion is bolstered by numerical and first principle analyses. The details of this evaluation are provided in Section 4.

- Fibrous debris, should it enter the core region, will not tightly adhere to the surface of fuel cladding. Thus, fibrous debris will not form a "blanket" on clad surfaces to restrict heat transfer and cause an increase in clad temperature. Therefore, adherence of fibrous debris to the cladding is not plausible and will not adversely affect core cooling. The details of this evaluation are provided in Section 5.

- Protective coating debris, should it enter the core region, will not restrict heat transfer and cause an increase in clad temperature. Therefore, adherence of protective coating debris to the cladding is not plausible and will not adversely affect core cooling. The details of this evaluation are provided in Section 6.

- The chemical effects method developed in WCAP-16530-NP-A was extended to develop a method to predict chemical deposition of fuel cladding. The calculational tool, LOCADM, can be used by each utility to perform a plant-specific evaluation. It is expected that each plant will be able to use this tool to show that decay heat would be removed and acceptable fuel clad temperatures would be maintained. The details of this evaluation are provided in Section 7.

- PWRs use boron as a core reactivity control method and are subject to concerns regarding potential post-LOCA boric acid precipitation in the core. In light of NRC staff and ACRS challenges to the simplified methods commonly used, it has recently become clear that additional insights and new methodologies are needed to answer fundamental questions about boric acid mixing and transport in the RCS and potential precipitation mechanisms that may occur both during the ECCS injection phase and the sump recirculation phase after a LOCA. This will be addressed in a separate PWROG program. This program is discussed in Section 8.

- The PWROG FA test results demonstrated that sufficient flow will reach the core to remove core decay heat. The debris load acceptance criteria developed is bounding and applicable to all PWR plants, including UPI plants. The details of this evaluation are provided in Section 9.

Actions are required of utilities to demonstrate acceptable LTCC with debris and chemical products in the recirculating fluid. Plants will have to perform plant-specific LOCADM evaluations (Section 7 and Appendix E) and prove the plant conditions are bounded by the debris load acceptance criteria (Section 3, Section 10 and Appendix G). Plants with debris loads above the debris load acceptance criteria may demonstrate adequate LTCC capability through engineering evaluations of plant-specific conditions and/or plant-specific testing.

These actions along with reference to this report provide the basis for demonstrating LTCC will not be compromised following a LOCA as a consequence of debris ingestion to the RCS and core.

1 INTRODUCTION

The scope of Generic Safety Issue 191 (GSI-191) (Reference 1) addresses a variety of concerns associated with the operation of the emergency core cooling system (ECCS) and the containment spray system (CSS) in the recirculation mode. These concerns include debris generation associated with a postulated high energy line break, debris transport to the containment sump when the ECCS is realigned to operate in the recirculation mode, and the effects of debris that might pass through the sump strainers on downstream components and fuel. In addition to debris resulting from the action of the jet from the postulated pipe break, there is also the potential for generation of chemical products from the reaction of containment materials and coolant that may also be transported to and through the sump strainer.

During operation of the ECCS to recirculate coolant from the containment sump, debris in the recirculating fluid that passes through the sump strainer may collect throughout the fuel assembly (FA), causing resistance to flow through this path. The collection of sufficient debris throughout the FA is postulated to impede flow into the fuel assemblies and core. Other concerns have been raised with respect to the collection of debris and post accident chemical products within the core itself. Specifically, the debris has been postulated to either form blockages at spacer grids or adhere to the cladding, thereby reducing the ability of the coolant to remove decay heat from the core. Similarly, chemical precipitates have been postulated to plate out on fuel cladding, again resulting in a reduction of the ability of the coolant to remove decay heat from the core.

Guidance provided to the industry in the following documents has been used to provide the framework for analyses that address these concerns.

- WCAP-16406-P-A: Section 9.0 (cold leg injection, hot leg injection, fiber, particulates, etc.) including addenda (Reference 2)
- NEI 04-07, Volume 1: Section 7.3 (Reference 3)
- NEI 04-07, Volume 2: Section 7.3 (Reference 4)
- Draft NRC Staff Review Guidance for Evaluation of Downstream Effects of Debris Ingress into the PWR RCS on Long Term Core Cooling Following a LOCA, dated November 22, 2005. (Reference 5)

The Pressurized Water Reactor Owners Group (PWROG) undertook a program to provide additional analyses, test data, and information on the effect of debris and chemical products on core cooling for pressurized water reactors (PWRs) when the ECCS is realigned to recirculate coolant from the containment sump. The objective of the program is to enable each plant to demonstrate that there is reasonable assurance that sufficient long term core cooling (LTCC) is achieved for PWRs to satisfy the requirements of 10 CFR 50.46 with debris and chemical products that are postulated to be transported to the reactor vessel. This program supersedes the efforts documented in WCAP-16406-P-A, Section 9 (Reference 2). For the purposes of this work, "long-term core cooling" is defined to be when the ECCS and CSS are realigned to recirculate coolant from the containment sump. The program was performed such that the results of this program are bounding and apply to the fleet of PWRs, regardless of the design of the plant (Westinghouse, Combustion Engineering [CE], or Babcock & Wilcox [B&W]) or fuel vendor (Westinghouse or AREVA).

2 LONG-TERM CORE COOLING ACCEPTANCE BASIS

2.1 INTRODUCTION

Part of the resolution of GSI-191 involves defining the relevant LTCC bases. This section describes the criteria that will be used in determining GSI-191 acceptance of the debris effects on fuel. These LTCC acceptance criteria are based on the requirements of Title 10 of the Code of Federal Regulations, Part 50.46 (10 CFR 50.46). The criteria are to be used with engineering evaluations that demonstrate acceptable LTCC, once established following the initial recovery of the core post LOCA, is successfully maintained. Successful LTCC is defined as meeting the criteria highlighted in this section. A detailed discussion of the criteria can be found in Appendix A.

2.2 GSI-191 LONG-TERM CORE COOLING ACCEPTANCE BASES

The LTCC acceptance bases defined for GSI-191 are listed below. These acceptance bases are consistent with 10 CFR 50.46 (b)(4) and 10 CFR 50.46 (b)(5) and demonstrate that the local temperatures are stable or continuously decreasing and that debris entrained in the cooling water supply will not affect decay heat removal.

1. Decay Heat Removal/Fuel Clad Oxidation

   The cladding temperature during recirculation from the containment sump will not exceed 800°F.

   Cladding temperatures at or below 800°F maintain the clad within the temperature range where additional corrosion and hydrogen pickup over a 30 day period will not have a significant effect on cladding properties. At temperatures greater than 800°F, there are occurrences of rapid nodular corrosion and higher hydrogen pickup rates that can reduce cladding mechanical performance. Long term autoclave testing has been performed to demonstrate that no significant degradation in cladding mechanical properties would be expected due to a localized hot spot. This information is proprietary to the fuel vendors but could be made available upon request. This testing demonstrated that the increase in oxide thickness and hydrogen loading was limited at temperatures of less than 800°F for periods of 30 days. With limited corrosion and hydrogen pickup, the impact on cladding mechanical performance is not significant. Therefore, no significant degradation in cladding properties would occur due to 30-day exposure at 800°F and there would not be any adverse impact on core coolability. The autoclave results justify a maximum clad temperature 800°F as an LTCC acceptance basis.

2. Deposition Thickness

   For current fuel designs, regardless of vendor, the minimum clearance between two adjacent fuel rods, including an allowance for the spacer grid thickness, is greater than 100 mils. Therefore, a 50-mil debris thickness on a single fuel rod is the maximum deposition to preclude touching of the deposition of two adjacent fuel rods with the same deposition. The 50 mil thickness is the maximum acceptable deposition thickness before bridging of adjacent fuel rods by debris is predicted to occur. The 50 mils of solid precipitation described here include the clad oxide, crud layer and debris deposition.

2.3 SUMMARY

These LTCC bases applicable to GSI-191 have been defined based on the requirements of 10 CFR 50.46 as clarified by the NRC (Reference 6). They are summarized as follows:

1. The cladding temperature during recirculation from the containment sump will not exceed 800°F.

2. The deposition of debris and/or chemical precipitates will not exceed 50 mils on any fuel rod.

These bases will facilitate the demonstration of acceptable core cooling following a postulated large break LOCA.

3 BLOCKAGE AT THE CORE INLET

During operation of the ECCS to recirculate coolant from the containment sump, debris in the recirculating fluid that passes through the sump strainer may collect on the bottom surface of the FA bottom nozzle, causing resistance to flow through this path. The collection of sufficient debris at this location is postulated to impede flow into the fuel assemblies and core. In order to address this concern, a prototypical FA testing program was initiated to establish guidance on the debris mass that could bypass the reactor containment building sump strainer and not impede core inlet flow and challenge LTCC.

Additionally, this section provides an overview of WCOBRA/TRAC (WC/T) evaluation, which examines the extreme case of almost complete blockage at the core inlet in order to provide additional assurance that LTCC will not be challenged. This calculation provides additional "defense in depth" assurance that LTCC will be maintained.

3.1 PROTOTYPICAL FUEL ASSEMBLY TESTING

The prototypical FA testing program was designed to establish a bounding, conservative analysis on the debris mass that could bypass the reactor containment building sump strainer and not impede core inlet flow and challenge LTCC. 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. A detailed discussion of the FA test program can be found in Appendix G. The following sections summarize the program and pertinent results. The results from these FA tests are discussed in the proprietary test reports (References 7, 8 and 21).

3.1.1 Pressure Drop Considerations for Testing

The FA testing program undertaken by the PWROG is designed to provide reasonable assurance that sufficient flow will reach the core to remove core decay heat. To that end, it must be demonstrated that the head available to drive flow into the core is greater than the head loss (also referred to as pressure drop) across the core due to possible debris blockage. The following relationship must be true to ensure sufficient flow is available to maintain LTCC:

$$dP_{avail} > dP_{debris}$$

The available driving head (dP_{avail}) is a plant-specific value and the pressure drop due to debris (dP_{debris}) is determined by the FA test program.

3.1.1.1 Available Driving Head

At the time of sump switchover, the core has been fully recovered and the fluid inventory in the RCS is above the top of the core. The core decay heat is being removed by ECCS injection. Core flow is only possible if the manometric balance between the downcomer and the core is sufficient to overcome the flow losses in the reactor vessel (RV) downcomer, RV lower plenum, core, and loops, or reactor vessel vent valves (RVVVs)[1] 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 or RVVVs

The manometric differences are determined considering plant geometry, system water levels, core void fractions, and flow path resistances. The flow losses are calculated using the following form of the Darcy equation.

$$\Delta P_{flow} = \frac{k}{A^2} \cdot \frac{\omega^2}{288 \cdot \rho_g \cdot g_c}$$

where:

ΔP_{flow}	=	differential pressure (psid)
k	=	form-loss coefficient
A	=	area upon which the form-loss coefficient is based (ft^2)
ω	=	flow rate (lbm/s)
ρ_g	=	liquid density (lbm/ft^3)
g_c	=	gravitational constant (32.2 lbm-ft/lbf-sec^2)

The driving head at the core inlet is dependent on the break location. In either case, core heatup will not occur until there is sufficient debris accumulation to limit the core flow rate to the point where the fluid is exactly saturated steam at the core exit. Therefore, for either the hot or cold leg break, the calculation of allowed pressure drop for debris should not consider any liquid associated with entrainment or bubbly, frothy flow downstream of the core at the limiting condition.

1. The B&W plant designs have RVVVs that short-circuit the steam path to the break for CL break scenarios. These passive valves provide a path between the RV outlet plenum and the RV upper downcomer region. They open on a small differential pressure and provide a path for steam to vent from the RV upper plenum directly to the break in the CL. Therefore, steam flow through the loops is not expected for B&W-designed plants.

For postulated cold leg (CL) breaks, the ECCS liquid from each CL runs to the break, ensuring that the downcomer is full to at least the bottom of the CL 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 or RVVVs. The net ECCS flow to the core is only what is required to make up for core boiling to remove the decay heat. Most of the ECCS liquid spills directly out of the break. The flow downstream of the core at recirculation would be two-phase with entrained liquid or a bubbly flow wherein the flow and elevation heads balance the downcomer elevation head to produce a flow rate matching decay heat. As debris builds up, the flow of liquid into the core reduces and the level of liquid downstream of the core lowers or the entrained liquid is not replaced until the critical condition is reached with just sufficient flow to match decay heat with no remaining liquid downstream of the core. This condition is commensurate with the maximum allowed blockage at the core inlet.

For a break in the HL, the ECCS liquid must pass through the core to exit the break. The driving force is the manometric balance between the liquid in the downcomer and the core. Should a debris bed begin to build up in the core, the liquid level will begin to build in the CLs and flow will spill back through the reactor coolant pumps into the pump suction piping, steam generator (SG) inlet plenum, and SG tubes. 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 tubes for Westinghouse- and CE-designed plants has been filled or reaches the HL spillover elevation for B&W-designed plants. Once the ECCS flow reaches the elevation of the shortest tubes, the flow area of the shortest tubes or HL piping are large enough that no increase in water level to the higher tubes is achieved. This is conservative, as it provides for the minimum static head available. The core mixture level will be at least to the HL nozzle elevation, and the core flow rate will equal the ECCS flow rate. The flow downstream at recirculation would be liquid that has been heated in the core, but not likely boiled, and is being pushed out the break (Appendix J, RAI #14). As debris builds up, the flow of liquid is reduced until boiling initiates and the break flow becomes two-phase. Increased accumulation of debris further slows the flow until the critical condition is reached with just sufficient flow to match decay heat and no liquid downstream of the core. This condition is commensurate with the maximum allowed blockage at the core inlet.

The methodology to calculate the plant-specific dP_{avail} value is presented in Section 2.18 of Reference 19.

3.1.2 Pressure Drop Due to Debris (dP_{debris})

Testing was conducted to define dP_{debris} values corresponding to specific fiber loads. This testing was designed to measure the pressure drop resulting from a specified debris loading and this value was defined as dP_{debris} at this debris load. A high-level summary of the testing is provided here and additional details are provided in Appendix G.

1. The test facility is a closed-loop system that continually recirculates fluid and debris through a single test assembly.

2. The test chamber is formed by walls that are sized to match the FA pitch. The distance from the end of the test FA to the chamber walls is half the distance between adjacent FAs.

3. The flow entering the bottom of the FA is uniform and constant.

4. All debris is available to form debris beds at either the simulated core inlet of at the intermediate spacer grids.

All these design features contribute to the promotion of debris capture in the test loop and provide a conservative representation of the debris capture in an actual accident scenario.

In either a HL or CL break case, core heatup will not occur until there is sufficient debris accumulation to limit the core flow rate to the point where the fluid is exactly saturated steam at the core exit. Therefore, for either the HL or CL break, the calculation of allowed pressure drop for debris should not consider any liquid associated with entrainment or bubbly, frothy flow downstream of the core at the limiting condition.

As long as the pressure drop due to debris (defined by FA testing) is less than the available driving head (defined by plant-specific calculation), flow will pass through the core and reach the break.

3.1.3 Description of Tests

The PWROG developed a common test protocol to ensure that testing for all of the PWROG members was consistent among test sites. The protocol is described in Reference 9. The test matrix, acceptance criteria, and test procedures were developed based on this protocol. The details of the test program are provided in Appendix G.

3.1.4 Discussion of Test Results

Testing was performed at hot and cold leg break flow rates.

- The test matrices used for this program are provided in Table G-2 and G-3 and the results are provided in References 7, 8 and 21.
- The HL break flow rate (i.e. the highest flow rate) represented the limiting head loss test condition.
- The amount of particulate tested affects the formation of the debris bed and the resulting head loss across the FA. Testing was conducted at the limiting particulate-to-fiber (p:f) mass ratio which produced the limiting result. Tests conducted at this condition experienced a significant increase in head loss upon the introduction of chemical surrogate to the test loop.
- Fiber was the greatest driver for increasing head loss at the core inlet. The FA test program evaluated the impact of various debris types (particulate, microporous insulation, cal-sil insulation, chemical precipitates and fiber) on head loss. Testing demonstrated fiber is the limiting variable and, due to the behavior of the other debris types, is the only debris variable that requires a limit.

Plants that have bypass debris loadings that are within the limits of the debris masses tested are bounded by the test program. The specific acceptance criteria are listed in Section 10. Several courses or actions have been identified for plants whose debris loads are outside the limits tested including, but not limited to, reducing problematic debris sources by removing or restraining the affected debris source, plant-specific FA testing, engineering evaluations of plant-specific conditions, removal or reduction of chemical precipitate formation, and evaluation of debris transport/bypass calculations.

Additional FA topics are discussed in the following sections.

3.1.4.1 Impact of Thin Bed on Head Loss

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 (Reference 10). All tests followed this guidance, with the NRC staff observing a few of the tests. In all cases, a thin bed was not observed, even with very small quantities of fibrous debris. That is, a large head loss was not observed only with fiber and particulates in the loop. However, as previously mentioned, the p:f mass ratio has a direct impact on the head loss.

3.1.4.2 Debris Settling in Lower Plenum

Credit for settling in the lower plenum is not being considered as part of the demonstration of LTCC. 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.

3.1.4.3 Alternate Flow Paths

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 falls within the limits of the debris loads tested, the core inlet will not become fully blocked with debris. Therefore, flow paths into the core other than through the core inlet or exit (i.e., alternate flow paths) are not considered in applying the debris mass acceptance criteria and are not credited or utilized in establishing acceptable debris loading conditions for LTCC.

In the event that a plant should choose to credit alternate flow paths for LTCC, the plant would be expected to identify the number, size, flow capability, and potential for blockage of the flow paths the plant is crediting.

3.2 WCOBRA/TRAC EVALUATION OF BLOCKAGE AT THE CORE INLET

To further bolster the assertion that core cooling flow will be maintained, WC/T analyses were performed to demonstrate that adequate flow is provided and redistributed within the core to maintain adequate LTCC. This computer code is used for evaluating best estimate large break LOCA methodology and is described in detail in Reference 11. A bounding evaluation was performed, using limiting assumptions, to evaluate the consequences of core inlet blockage on LTCC. The blockage was assumed to deterministically occur and is not representative of actual plant conditions. The objective of the calculation was to demonstrate that, should blockage at the core inlet occur, sufficient liquid could enter the core to remove core decay heat once the plant had switched to sump recirculation with up to 99.4 percent core blockage to assure acceptable cladding temperatures. Presented here is a summary of the evaluation performed. Appendix B contains a more detailed description of the evaluation performed.

3.2.1 Approach

The effects of blockage at the core inlet were simulated by ramping the dimensionless friction factor (C_D) at the core inlet to a large number, simulating a postulated debris buildup that results in a reduction of

flow. A modified version of WC/T was created to allow the friction factor at the core inlet to be ramped. Code simulations were performed using standard input for a problem time of 20 minutes. The 20 minute time was taken to be representative of the earliest time of realignment of the ECCS to operate in the recirculation mode. Starting at 20 minutes, the friction factor at the core inlet was ramped to its terminal value over the next 30 seconds. The core inlet flow blockage occurring in 30 seconds from the start of recirculation is not physical and does not represent any plant condition. The postulated core blockage was modeled in this manner to perform a bounding calculation. After the core inlet resistance was ramped to its terminal value of about $C_D = 10^9$ (which essentially eliminates all flow through the path), the code simulations were run out to 40 minutes to show the flow rate supplied to the core would be sufficient to remove decay heat and maintain a coolable core geometry.

3.2.2 Selection of Limiting Reactor Vessel Design

The core inlet blockage simulations were designed to bound the U.S. PWR fleet. To ensure a bounding calculation, the limiting break type and the limiting vessel design were taken into consideration before selecting a plant model for the simulation.

The selection of the limiting break for modeling purposes combines the conditions from a double-ended CL and a double-ended HL break to create a bounding scenario. During a double-ended CL break, the ECCS liquid will spill into containment, which decreases the driving head of core flow to a minimum. However, the debris that reaches the RCS lower plenum and core inlet for a CL break will be substantially lower than the debris that reaches the core for a HL break. During a double-ended HL break, no spilling of ECCS liquid occurs. Therefore, an additional driving head from the build-up of liquid level in the downcomer and in the steam generator tubes to the spillover elevation is present. However, the higher flow rates also result in faster debris build-up, and because there is more debris available to accumulate, the HL break represents the conservative case in terms of debris load. To create the worst possible scenario, the limiting break case for modeling purposes will be a modified double-ended CL break, i.e., limiting flow at the core inlet, combined with faster debris build-up time that occurs for a high flow HL break.

Similarly, Westinghouse, CE, and B&W vessel designs were considered and a limiting design was chosen based upon which vessel design would be most limiting with respect to the condition of core inlet flow blockage. Three general Westinghouse vessel designs were considered: designed barrel/baffle (B/B) upflow, converted B/B upflow, and B/B downflow. For Westinghouse designed plants, the most limiting design is downflow plants since the only means for the flow to enter the core is through the lower core plate. As described in Appendix B, this design was also determined to bound both the B&W and CE plants. Thus, a Westinghouse downflow plant was used for this WC/T evaluation.

3.2.3 Model Inputs

A plant with an existing WC/T model, downflow plant configuration, and high core power density is desired for the core blockage simulations. A three-loop downflow model plant rated at 2900 MWt was chosen. The axial power shape used high enthalpy rise peaking factor ($F_{\Delta H} = 1.73$), a skewed to the top power distribution (13 percent axial offset), and a relatively high total peak factor ($F_Q = 2.3$). The top-skewed power shape, shown in Figure 3-1, is limiting compared to base load or bottom skewed power

shapes due to the longer time for the quench front to approach the elevations with the highest power and its susceptibility to heatup if the core becomes uncovered due to inlet blockage.

The radial power distributions between the four core channels are listed in Table 3-1. The radial power distribution in the core is flat with the exception of the periphery assemblies and the hot assembly. The hot assembly power is conservatively modeled to a high normalized power of 1.66.

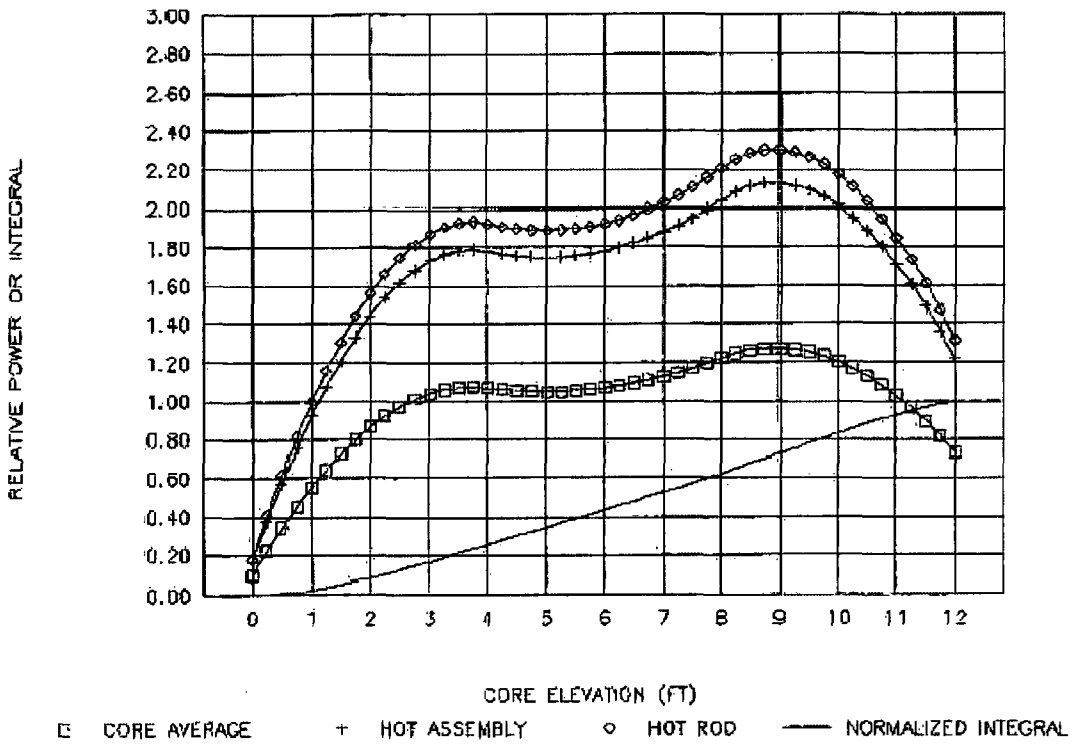

Figure 3-1 Plant Transient Power Shape

Table 3-1 Core Channel Radial Power Distribution

Channel Description	Channel Number	Normalized Power	Number of Assemblies
Hot Assembly Channel (HA)	13	1.66	1
Guide Tube Channel (GT)	12	1.17	53
Non-Guide Tube Channel (AVG)	11	1.17	75
Low Power Periphery Channel (LP)	10	0.20	28

Additional information about the plant chosen for the core inlet blockage simulations, including schematics and WC/T noding diagrams, is provided in Appendix B.

At 20 minutes, in addition to the ramping of the loss coefficient at the core inlet of the model, the ECCS liquid temperature was increased. The increase in the ECCS liquid better simulates the recirculating coolant temperature and is representative of residual heat removal (RHR) heat exchanger outlet temperature following switchover to sump recirculation. The temperature of the injected water was set to be 190°F, which is typical for Westinghouse designs and is expected to bound B&W designs. CE plant designs do not have RHR heat exchangers, and after switchover to recirculation, high pressure safety injection flow is pumped directly from sump to the RCS. As described in Appendix B, the increase in sump ECCS injection temperature is assessed to be a non-factor in core inlet blockage simulations. Prior to recirculation, termination of extensive downcomer boiling and cooling of vessel internals has already occurred. Therefore, the increase in injection temperature should not lead to boiling and only a small decrease in flow rate supplied to the core will ensue due to the density effects.

3.2.4 Results

Two simulations were run with no changes to the standard noding scheme but with different amounts of core blockage. The first case modeled 82 percent core flow blockage and allowed flow through the periphery fuel assemblies as shown in Figure 3-2. The second case modeled 99.4 percent core flow blockage and allowed flow only through the hot assembly (HA) channel. The cross-sectional core noding schemes for Case 1 and Case 2 are shown in Figure 3-2 and Figure 3-3, respectively.

As shown in Figure 3-4, a comparison between the calculated flow rates for Cases 1 and 2, and the flow rate needed to match core boil-off shows there is ample flow into the core to replace boil-off after the simulated core blockage occurred. Also, as shown in Figure 3-5, the calculated peak clad temperature (PCT) history plot of the hot rod is predicted to occur in traditional early time frame (within ~200 seconds) for a postulated large-break LOCA analysis. Figure 3-5 also shows that after roughly 300 seconds the core is quenched and no significant heatup occurs thereafter. Because no heat up occurs during the sump recirculation phase of the event, the maximum local and core wide oxidation calculations for traditional LOCA analyses are still considered applicable. It is therefore concluded that sufficient liquid can enter the core to remove core decay heat once the plant has switched to sump recirculation with up to 99.4 percent blockage at the core inlet.

The evaluation documented in Appendix B considered the Case 2 modeling approach of leaving the hot assembly unblocked due to core cross-flow. The void fraction in the HA channel was shown to reach higher values, demonstrating that much of the flow exits the HA channel via cross-flow to adjacent lower-power assemblies in the core. It was therefore concluded that there was no non-conservatism in the calculations due to the modeling approach.

The containment back pressure was modeled by a containment pressure vs. time table input for each of the broken loop CL components. The containment backpressures used in both cases were based on the existing pressure vs. time tables used in the best estimate LOCA (BELOCA) analysis. The BELOCA table was extrapolated down to atmospheric pressure and held at atmospheric conditions for the remainder of the simulation. Consistent with the objective of this evaluation, the applicability of this evaluation to sub-atmospheric containments was also evaluated. As stated in Appendix B, it was determined that the sub-atmospheric containment pressure plant designs are bounded by the atmospheric containment simulations performed to examine the effects of core inlet blockage.

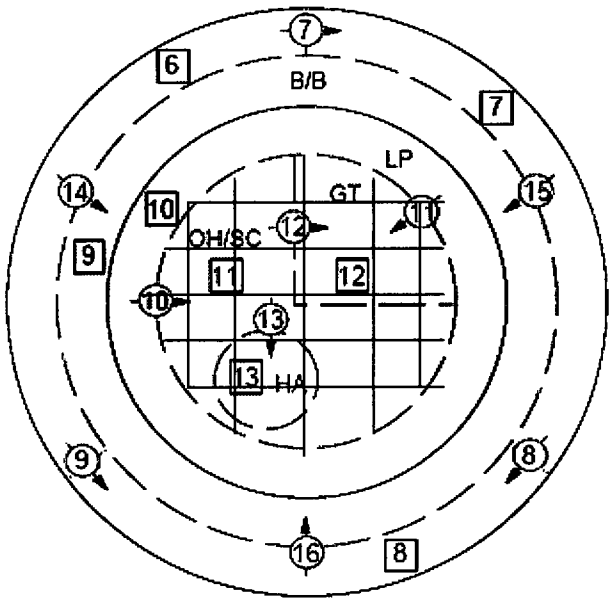

Figure 3-2 Case 1 – 82% Core Blockage Modeling Approach

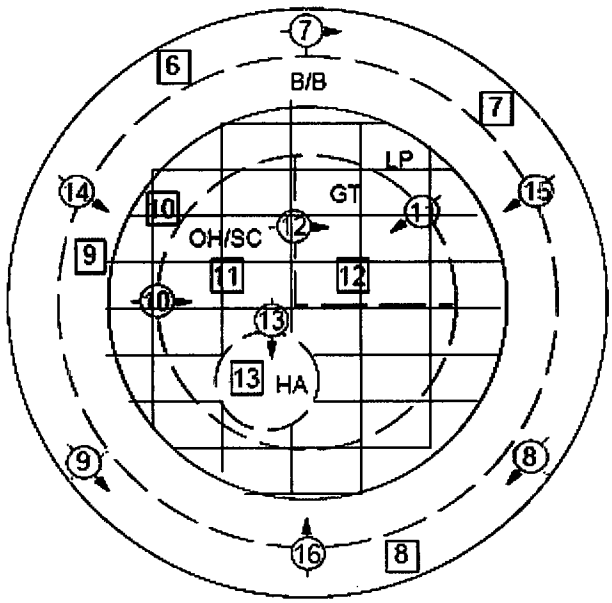

Figure 3-3 Case 2 – 99.4% Core Blockage Modeling Approach

Note: Regions 6, 7, 8, and 9 are the downcomer and downflow B/B regions.

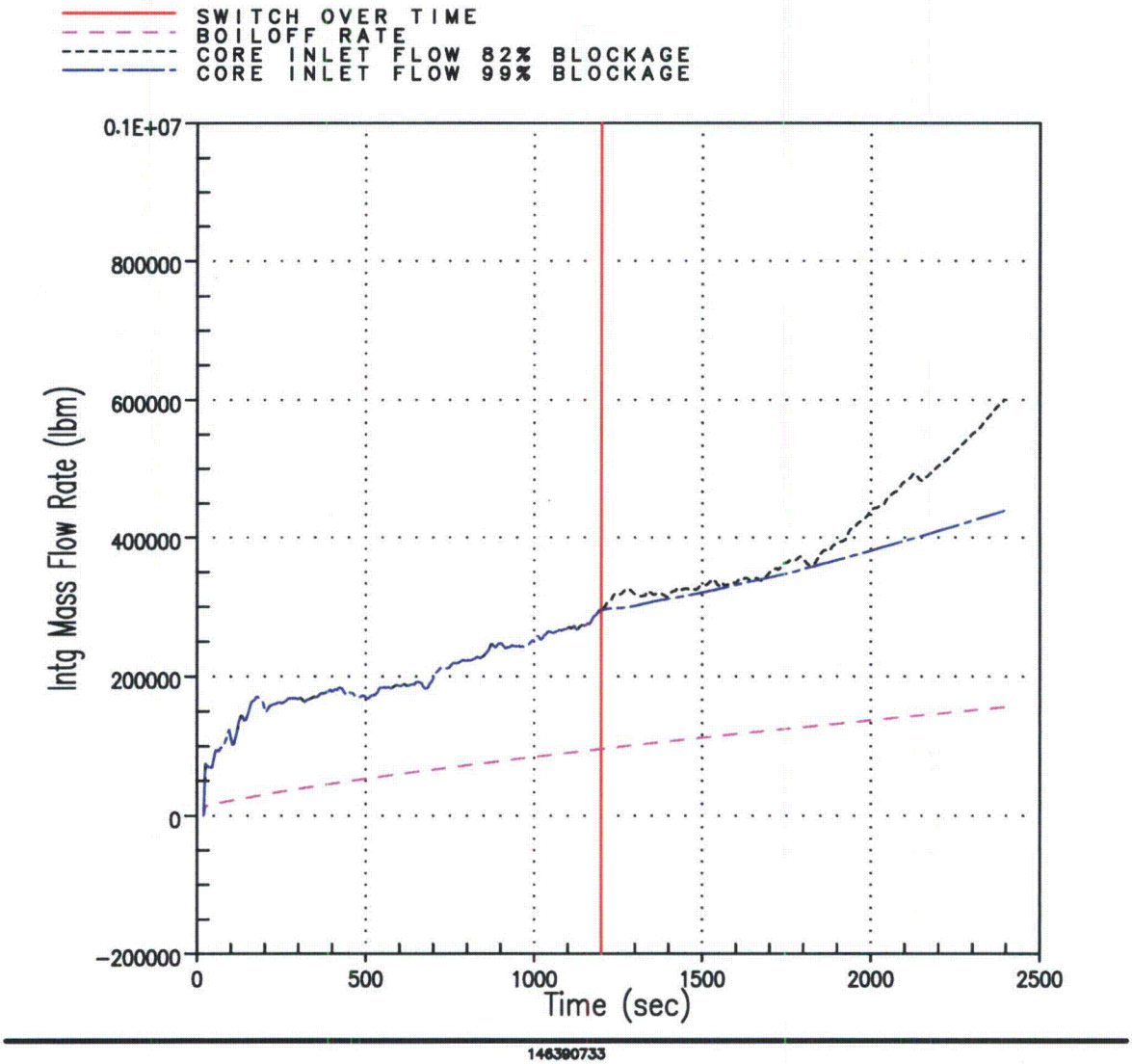

Figure 3-4 Integrated Core Flow vs. Core Boil-off for Case 1 and Case 2

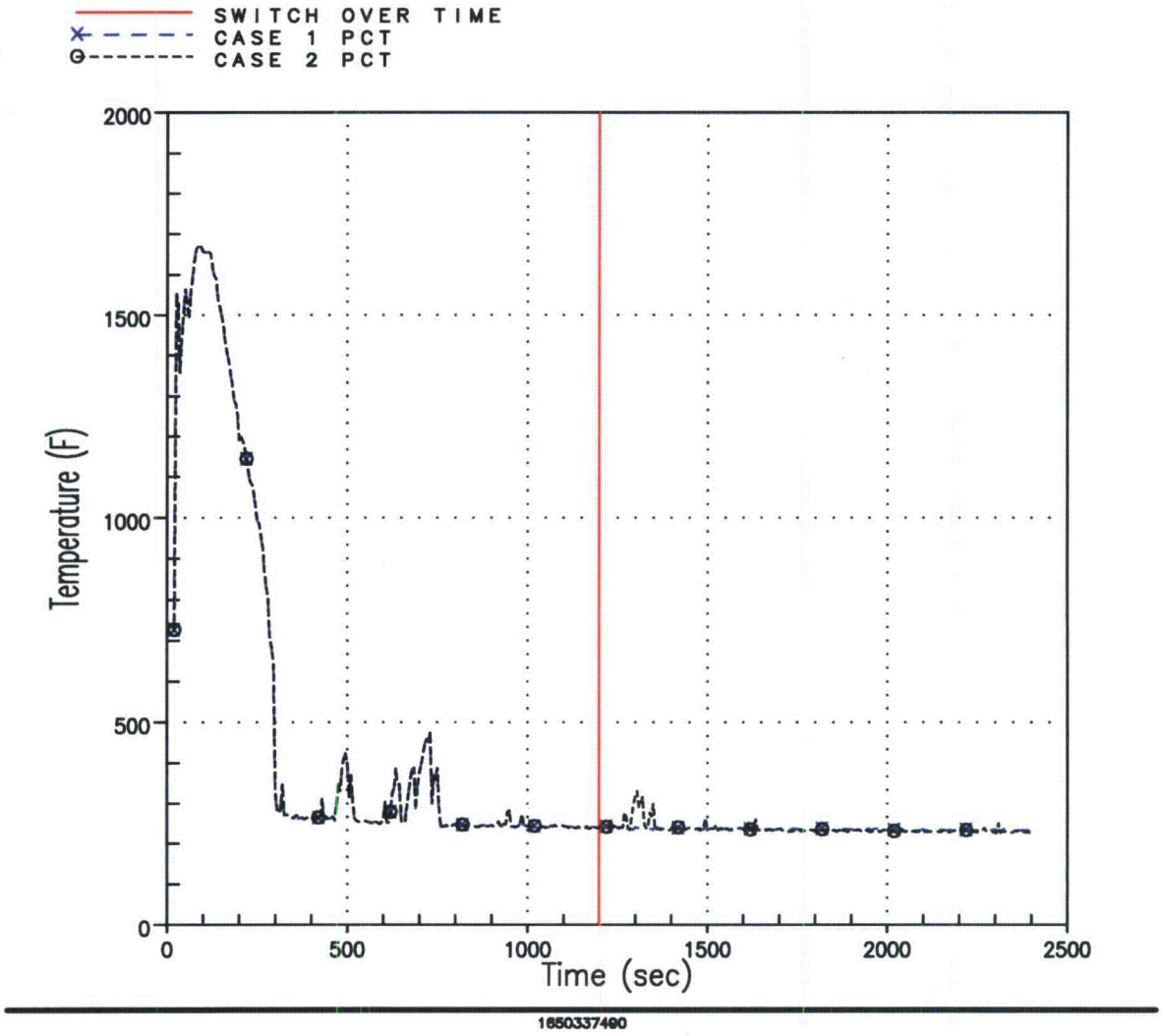

Figure 3-5 Case 1 and Case 2 Hot Rod Peak Clad Temperature History

3.3 ADDITIONAL WC/T CALCULATIONS

Several additional WC/T analyses were performed in support of the effort documented by this report. These 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. The detailed documentation for these additional calculations is presented in Appendix B and includes time history plots 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.

3.3.1 Method Discussion & Input

The base case for the calculation results presented in here is Case 2, or the more restricted flow area case, from Section 3.2. 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 LTCC. 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.

1. 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. 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.

2. 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.

3.3.2 Results from Flow Area Reduction Runs

The first flow reduction run performed reduced the hot channel (Channel 13) flow area of Case 2 by 50%, which yields a total core inlet flow reduction of 99.7% compared to an unblocked core. Figure 3-6 shows a comparison of the integrated core inlet flow and the core boil-off rate, starting at 1200 seconds, the time that switchover from injection to recirculation from the containment sump is simulated. As shown, even with the increase in core blockage, the flow that enters the core is still in excess of the boil-off rate. The peak cladding temperature (PCT) is shown in Figure 3-7. There are no significant PCT excursions after the core blockage is simulated.

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%. Figure 3-8 shows a comparison of the integrated core inlet flow and boil-off rate, again starting at 1200 seconds. For this increase in core blockage, the flow that enters the core cannot match the boil-off rate. In addition, Figure 3-9 shows that the PCT increases for the remainder of the calculation.

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 LTCC.

3.3.3 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. Figure 3-10 shows a comparison of the integrated core inlet flow and boil-off rate, again starting at the time of switchover from injection to recirculation from the sump. 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 3-10 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.) The PCT is shown in Figure 3-11. There are no significant PCT excursions after the core inlet loss coefficient is increased.

The second uniform loss coefficient run performed applied a uniform C_D of 100,000 at the core inlet. Figure 3-12 shows a comparison 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 3-12 shows a similar behavior as was shown in Figure B-36. 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 time.) The PCT is shown in Figure 3-13. There are no significant PCT excursions after the core inlet loss coefficient is increased.

The next uniform loss coefficient run performed applied a uniform C_D of 1,000,000 at the core inlet. Figure 3-14 shows a comparison of the integrated core inlet flow and boil-off rate. With the increased resistance to flow into the core specified for this case, the flow that enters the core can not match the boil-off rate. As a consequence, as shown in Figure 3-15, the PCT increases until the end of the transient calculation.

The results indicate that an increase in the form loss coefficient at the core inlet of up to C_D = 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 LTCC.

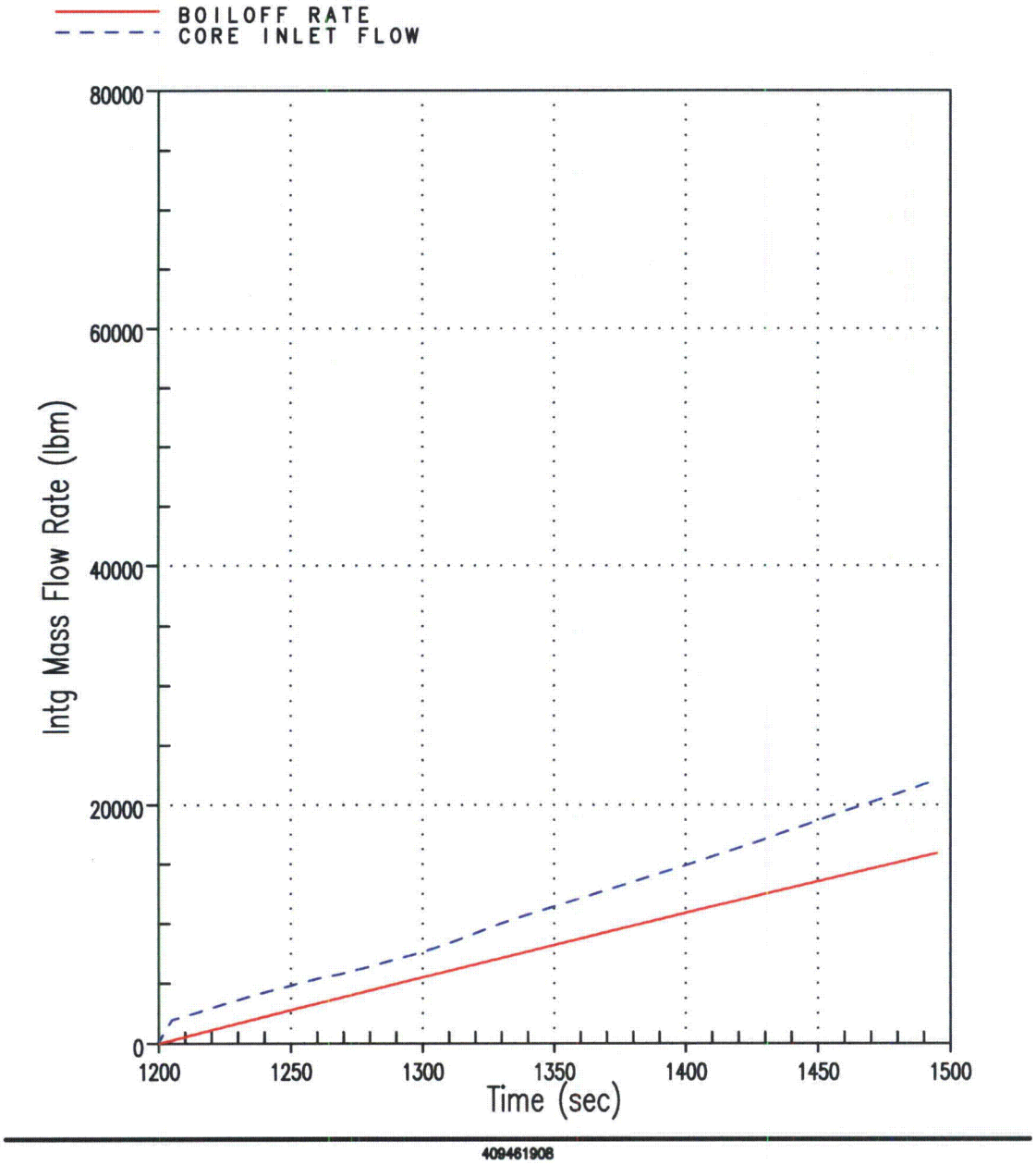

Figure 3-6 Integrated Core Flow vs. Core Boil-off for Channel 13 Flow Reduction 50% Case (Shifted Scale)

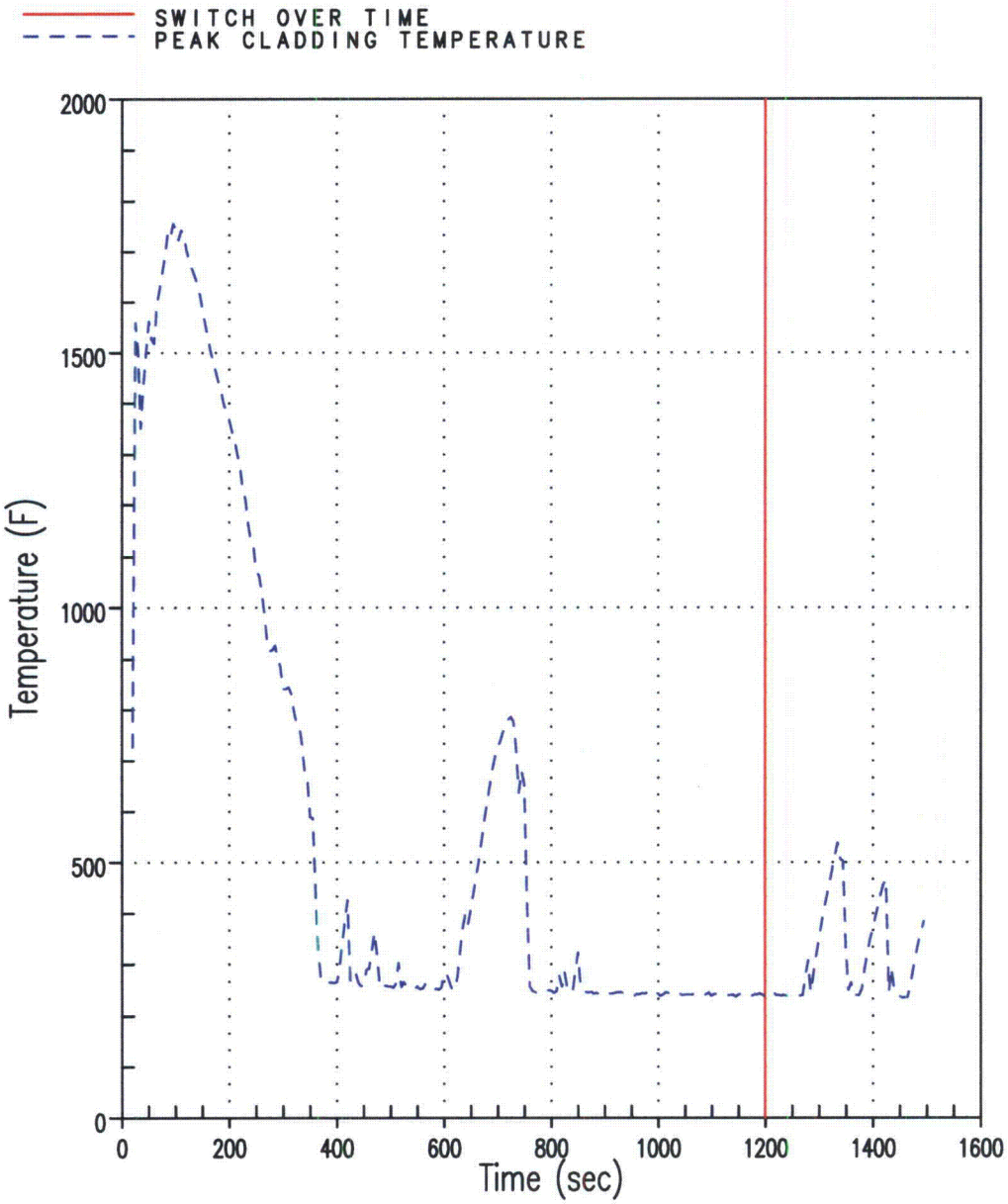

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Figure 3-7 Hot Rod PCT for Channel 13 Flow Reduction 50% Case

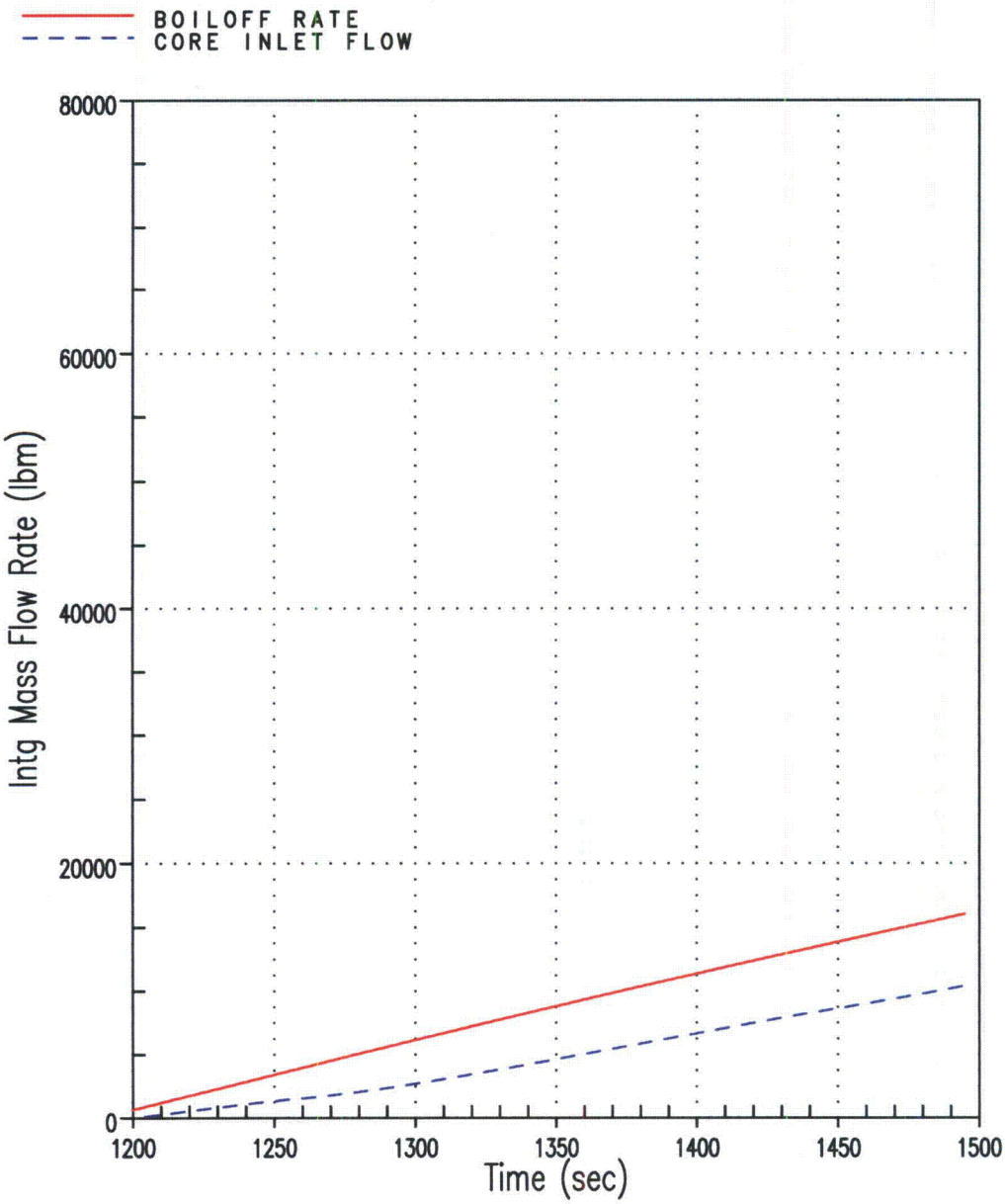

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Figure 3-8 Integrated Core Flow vs. Core Boil-off for Channel 13 Flow Reduction 80% Case (Shifted Scale)

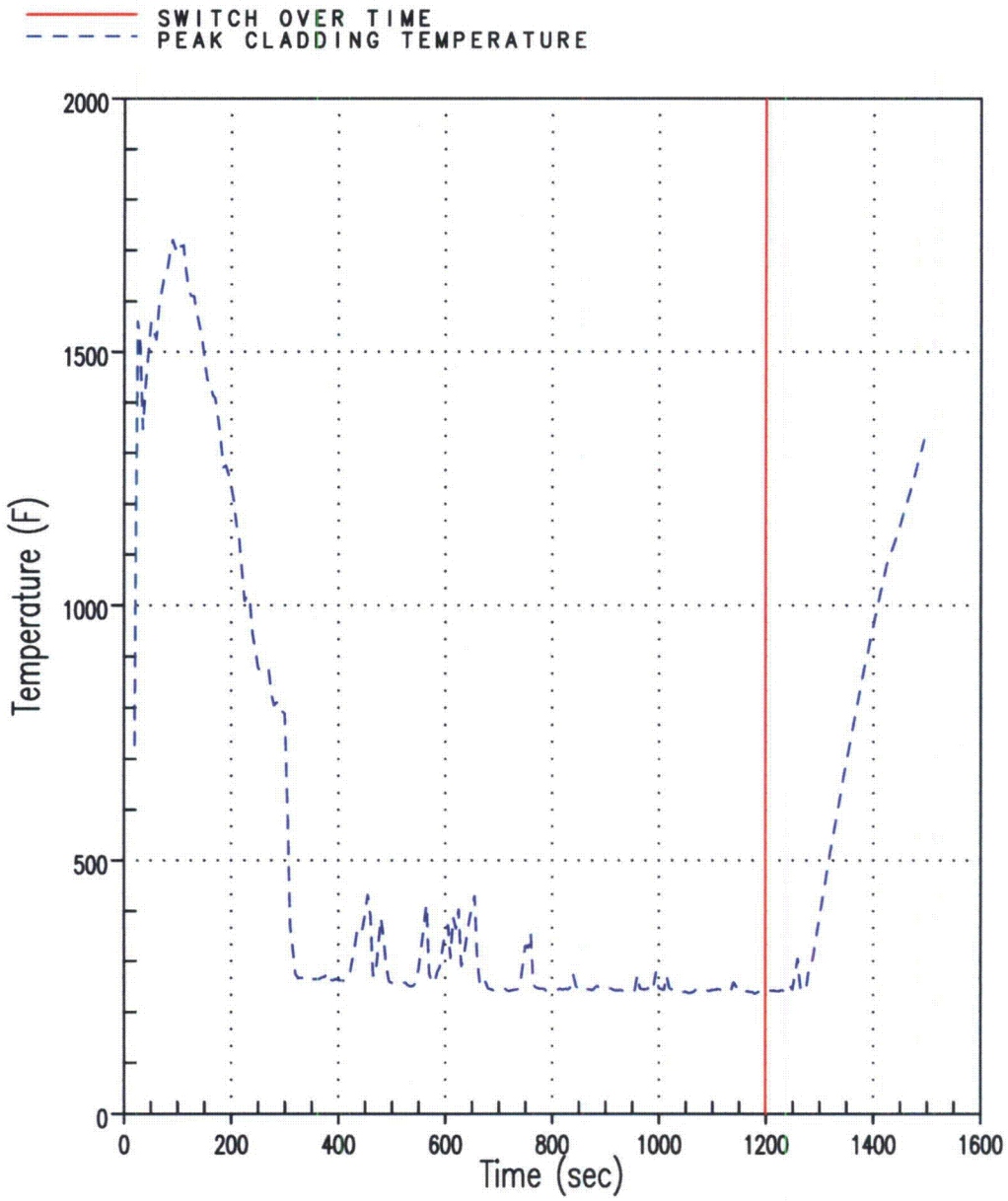

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Figure 3-9 Hot Rod PCT for Channel 13 Flow Reduction 80% Case

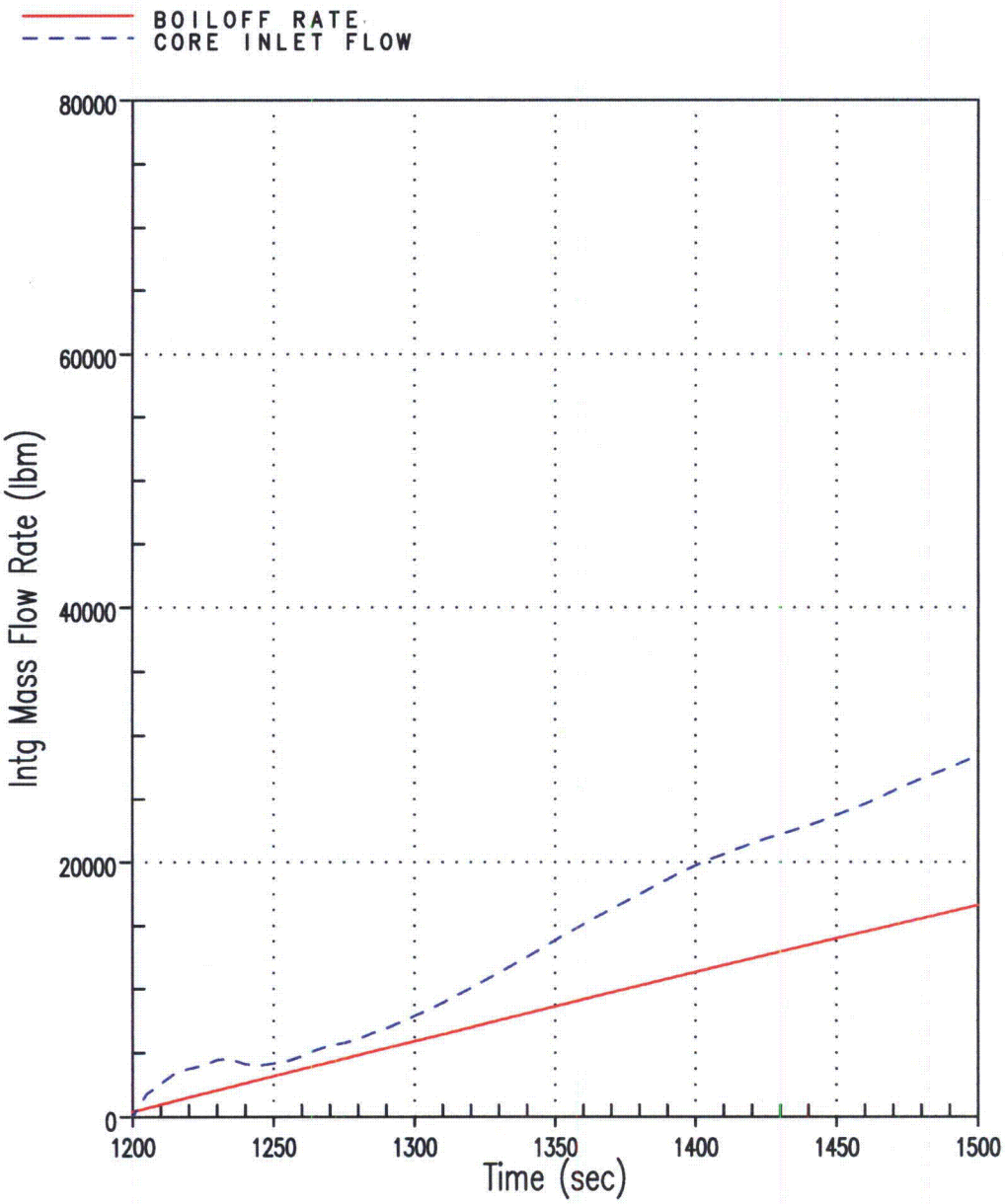

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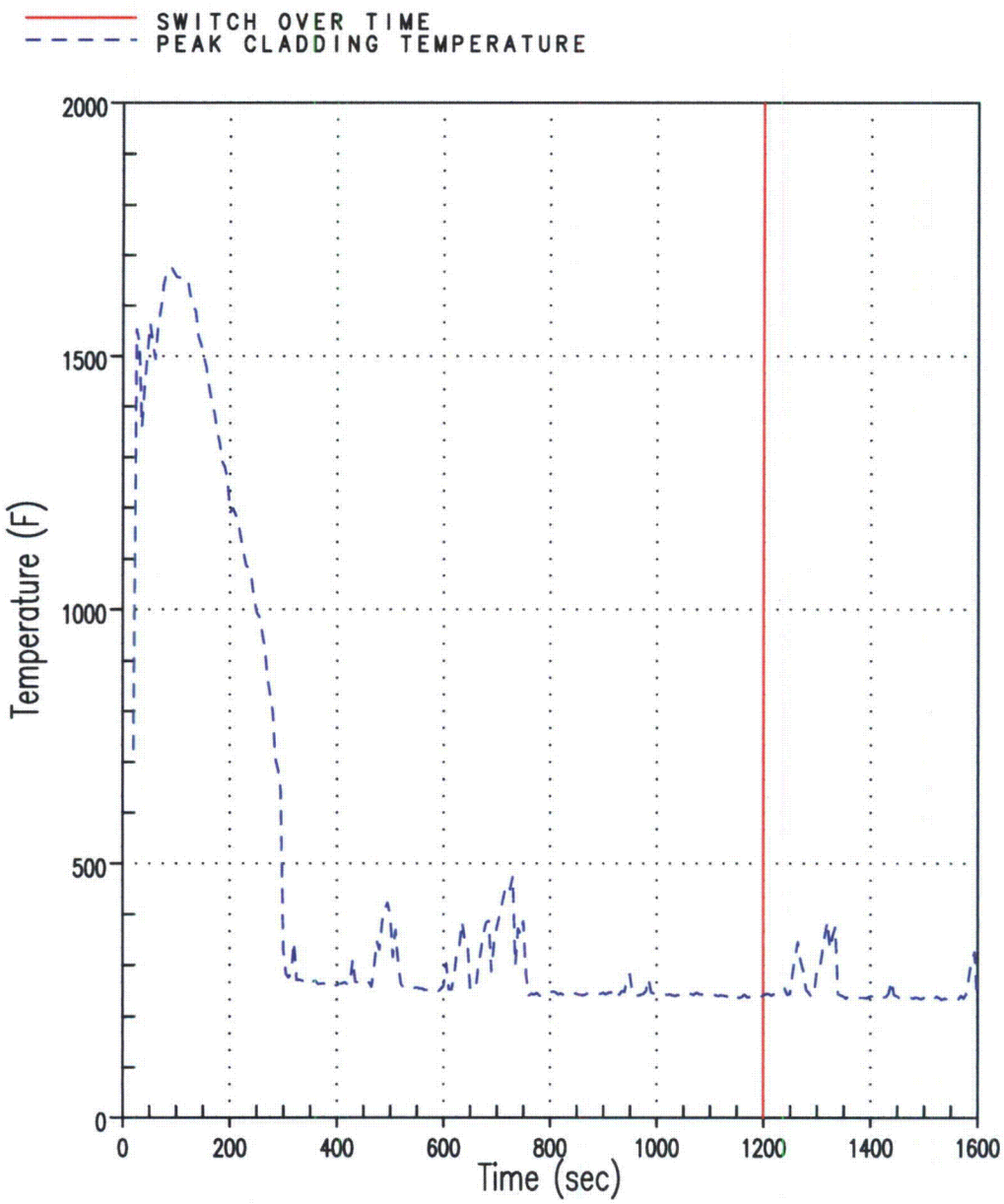
Figure 3-10 Integrated Core Flow versus Core Boil-off for Channel for Uniform C_D = 50,000 (Shifted Scale)



1912580093

Figure 3-11 Hot Rod PCT for Uniform C_D = 50,000

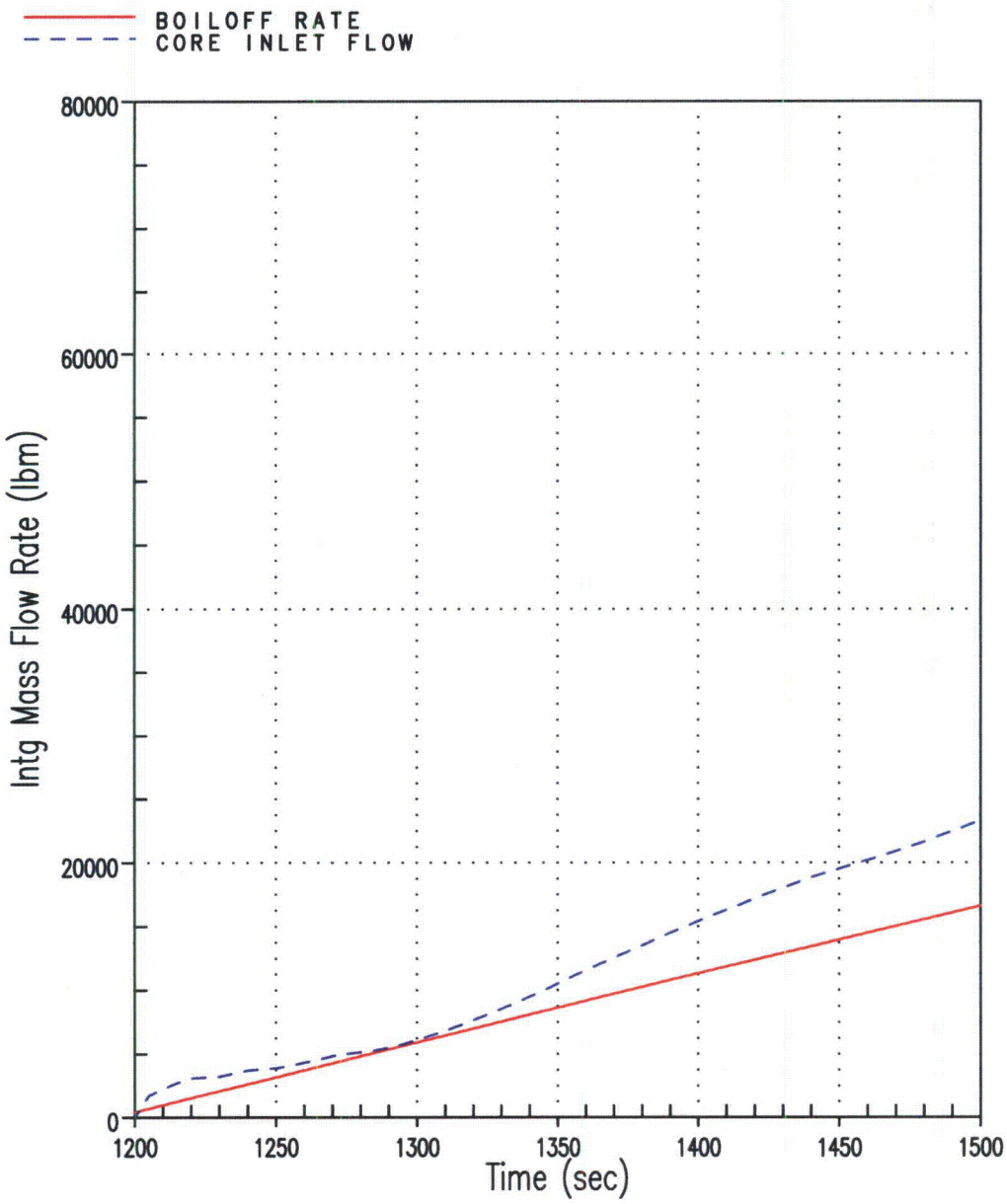


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Figure 3-12 Integrated Core Flow vs. Boil-off for Uniform C_D = 100,000 Case (Shifted Scale)

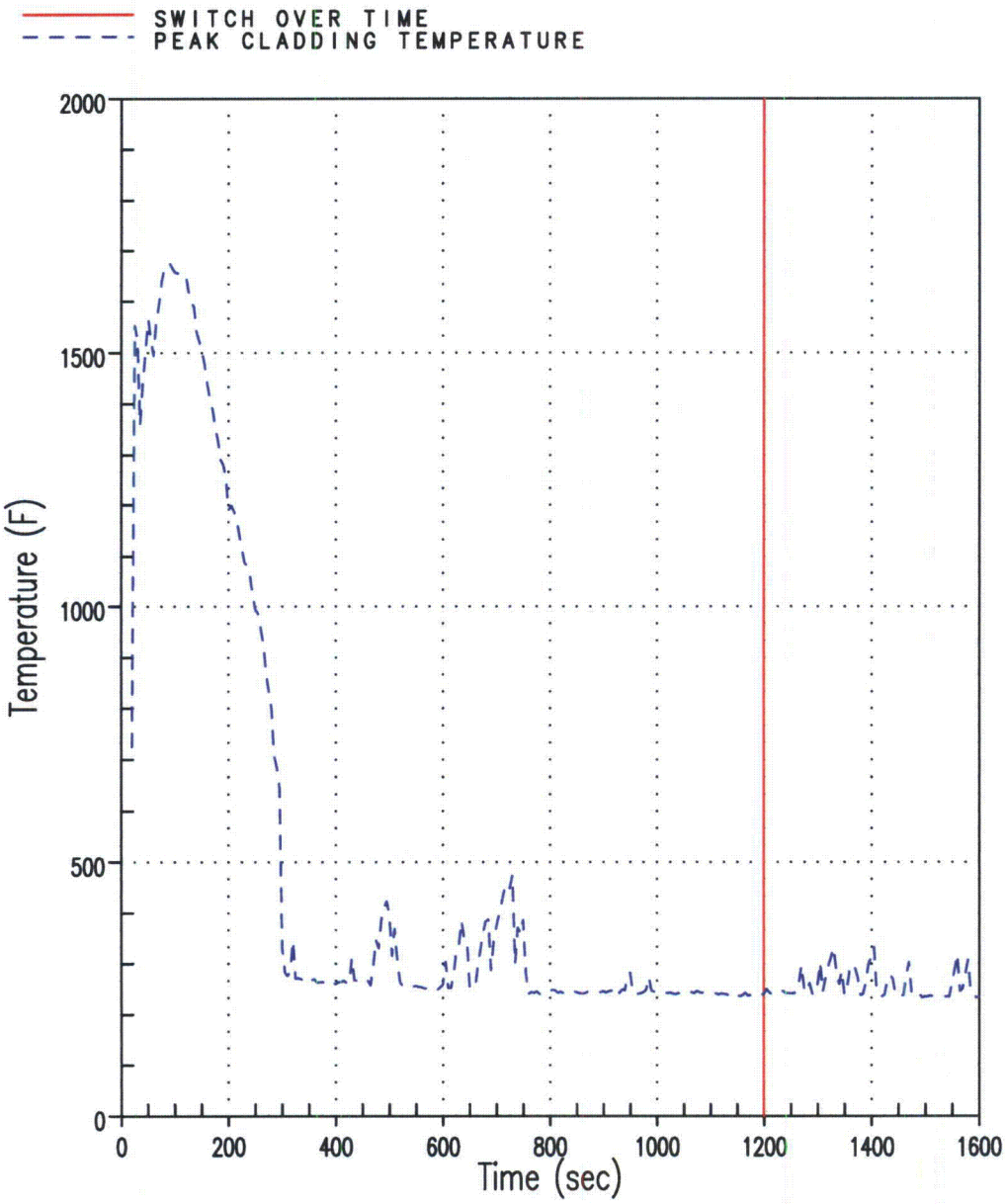

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Figure 3-13 Hot Rod PCT for Uniform C_D = 100,000 Case

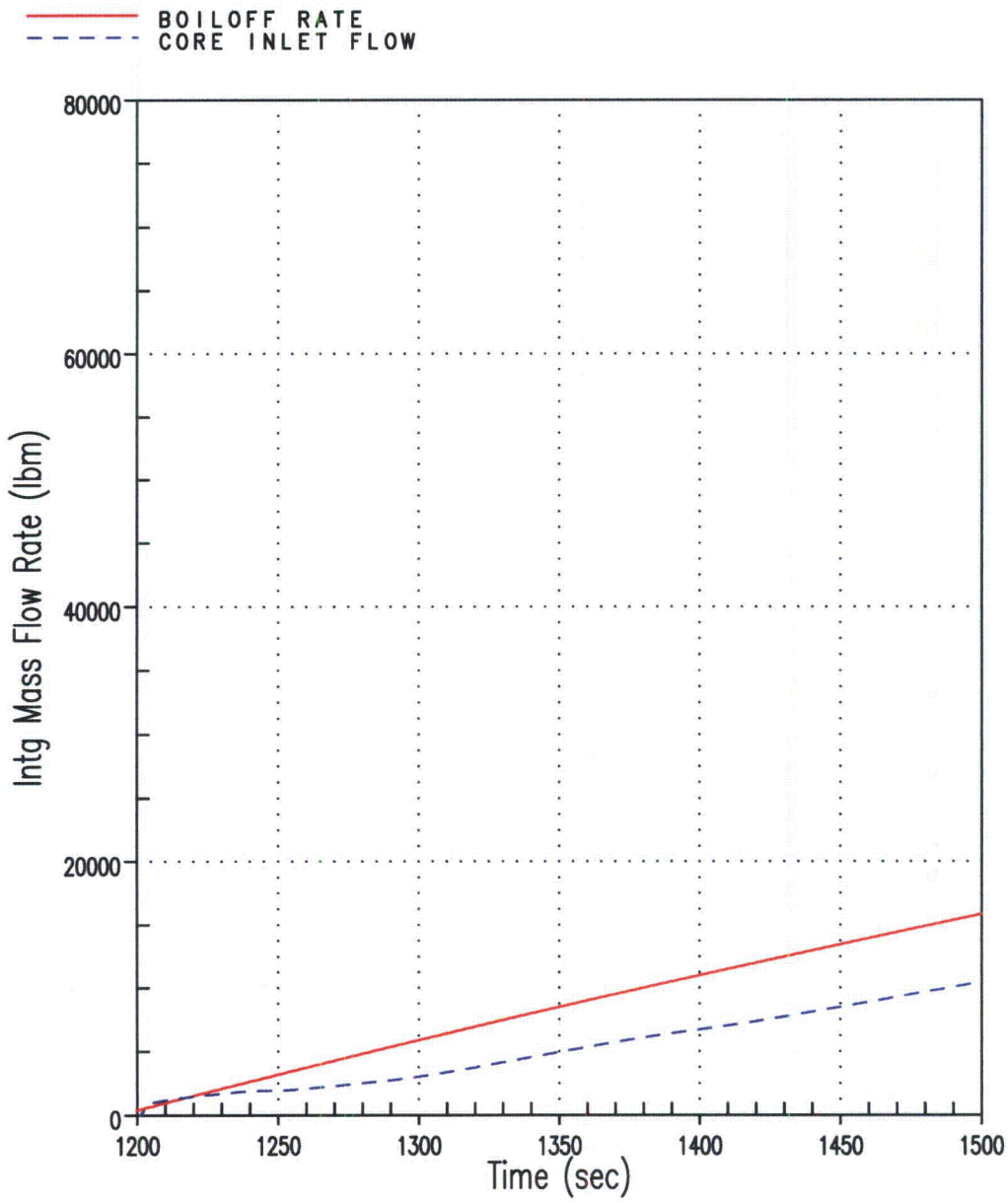

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Figure 3-14 Integrated Core Flow vs. Boil-off for Uniform C_D = 1,000,000 Case (Shifted Scale)

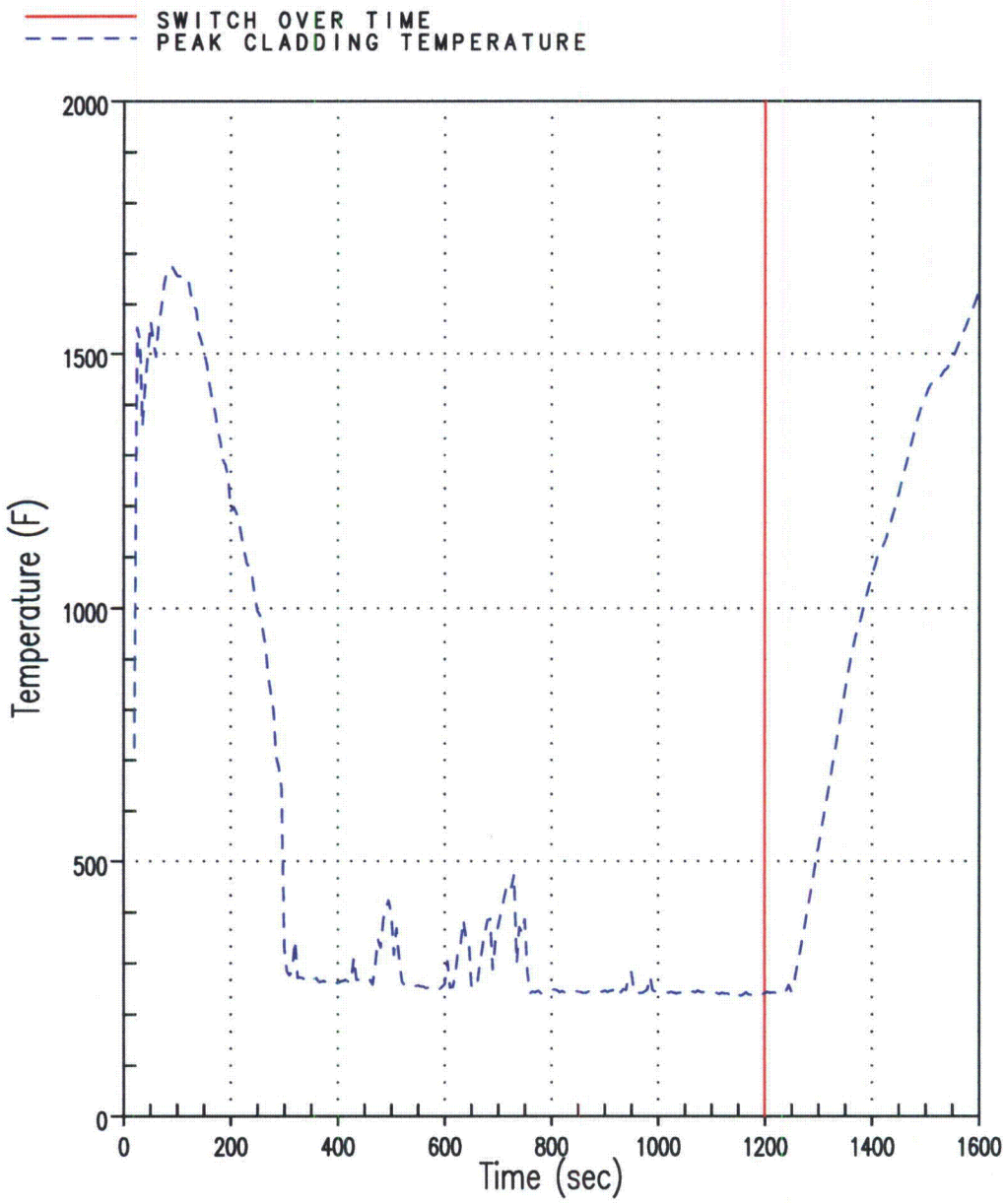

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Figure 3-15 Hot Rod PCT for Uniform C_D = 1,000,000 Case

3.4 SUMMARY

The FA testing program demonstrated that, at the debris load acceptance criteria, no potential blockage is expected that restricts flow into or through the core such that removal of decay heat and maintaining of decay heat is compromised. Therefore, plants that have bypass debris loadings that are within the limits of the debris masses tested are bounded by the test. The specific acceptance criteria are listed in Section 10.

In addition to the FA testing program, WC/T examined the cases of 82% and 99.4% blockage of the core inlet flow area. Additional sensitivity calculations performed with WC/T demonstrate there is margin in these two cases. These WC/T calculations provide defense-in-depth that LTCC will not be compromised with a debris blockage at the core inlet. It was concluded that sufficient liquid can enter the core to remove core decay heat once the plant has switched to sump recirculation with up to 99.4 percent blockage at the core inlet.

4 COLLECTION OF DEBRIS ON FUEL GRIDS

Debris that does not collect at the core inlet will pass through the FA bottom nozzle and enter the core region. It is possible that this debris may lodge in some of the smaller clearances in the fuel grids. Three supporting analyses are presented in this section to demonstrate that blockage at spacer grids will not impede LTCC. First, a general discussion of debris build up is presented along with an evaluation of the effect on LTCC. Second, the FA test data is reviewed. Finally, ANSYS® and first principle calculations are presented to demonstrate that the fuel rod will continue to be cooled even for extreme cases with significant blockages around the fuel grids.

4.1 GENERAL DISCUSSION

Each FA has a number of spacer grids. These grids are designed to support the fuel rods. Following a LOCA, they provide the most likely location for debris accumulation within the core region. Spacer grid designs commonly used have hard and soft stops, which are small "springs" in the middle of the grids. These "springs" and the leading edge of the grids are the most likely locations for debris to build up, although flow diversion will limit the buildup at this location

The size of particulate debris that may pass through the replacement sump strainers is dependent upon the hole size of the replacement sump strainer. This dimension is 0.11 in. or less. The maximum debris size that may be passed by sump strainers is of the magnitude of the maximum clearance between fuel rods and grid.

The design of a fuel grid allows for cross flow through the grid between adjacent fuel rods. That is, the stops are punched out of the grid such that a flow path exists from one fuel rod to the next near the middle of the spacer grid. This will limit both the extent of the debris build up and its consequences. Should debris collect and form a resistance to the flow of coolant along the fuel rod, both coolant and debris carried by the coolant will be diverted to adjacent "cleaner" locations. A similar phenomenon will occur for fuel designs without hard or soft stops, albeit at the leading edge of the grid. As debris builds up at the leading edge, the flow will divert around it to open channels, limiting the debris build up.

Debris that does collect will have some packing factor that will allow "weeping" flow through debris buildup to cool the cladding. Complete compaction of the debris will not occur and the packing density of the debris is limited to less than unity or perfect compaction. From Reference 12, the packing will most likely be less than ~60 percent. Thus, any debris buildup will not become impenetrable. Boiling in the area of the blockage will occur with less than a 10 to 15°F increase in the clad temperature over the adjacent coolant temperature. Even a small amount of fluid flow through the debris bed will provide sufficient heat removal via convection to maintain the fuel rod a few degrees below the liquid saturation temperature.

This general discussion provides solid arguments for asserting that blockages at the spacer grids will not adversely affect LTCC. Additional arguments and analyses are further developed in the following sections.

4.2 PROTOTYPICAL FUEL ASSEMBLY TESTING

The PWROG sponsored a test program to justify acceptance criteria for the mass of debris that can be deposited at the core entrance and not impede LTCC flows to the core. By testing a prototypical FA with spacer grids, additional information was obtained regarding the buildup of debris at the spacer grids. A detailed discussion of the test can be found in Appendix G. The results from these FA tests are discussed in the proprietary test reports (References 7, 8 and 21).

During the FA tests, debris was observed to accumulate at spacer grids. In some cases the accumulation seemed to be extensive. However, a review of the test data indicated that coolant continued to pass through the debris bed, verifying the "weeping" flow postulated in Section 4.1. Furthermore, the blockage at the spacer grids observed during the testing is conservative as described below.

During the FA tests, debris was observed to accumulate at spacer grids during tests performed at higher p:f ratios. While some buildup is expected, the observations from the tests represent an upper bound of the debris accumulation because of the following conservatisms in the testing process.

- Once the debris-laden fluid exits the break, it is returned to the sump where it can settle or at least be filtered again before it can return to the RCS. As the debris bed builds up on the sump strainer, less debris reaches the RCS. In the test loop, the debris-laden water was continuously circulated without filtration, allowing the debris multiple opportunities to be captured on the fuel filter or spacer grid.

- While the entire ECCS volume must pass through the core to reach the break, core boiling may not be suppressed following a HL break. This is more likely if one train of ECCS is lost to a failure. With boiling, additional turbulence is present in the core region, which will tend to remove debris from the spacer grids and confine blockages to isolated regions. Boiling was not simulated in the test loop.

- In the event of a CL break, the core flow will be multidimensional. Boiling at high-power locations will push liquid and steam to the top of the core where the steam will escape. The liquid will flow down the lower-power regions of the core. This results in a vigorously mixed boiling pot of liquid that will continuously move any debris that is not trapped. The additional spacer grids will provide additional locations for debris to accumulate. The result is that there will not be coplanar blockage of the core that could lead to unacceptable core cooling. There may be unique flow patterns related to potential local debris formation but core cooling is maintained. Any local blockages will not result in significant fuel pin heatup because they will be well dispersed in regions with limited size.

- Following a LOCA, rod and assembly bow will occur as a result of the thermal transient on the fuel rods. As a consequence, flow channels between fuel assemblies will become larger in some locations and smaller in others. These channels will allow flow around blockages at spacer grids, should they form. Rod and assembly bowing were not modeled in the test loop.

For tests conducted at high p:f ratios, debris was seen to accumulate at the spacer grids. While some buildup is expected, the observations from the tests represent an upper bound of the debris accumulation

as discussed in the previous sections. Further, these buildups, as extensive as some of them were, did not form an impenetrable blockage to flow. Therefore, "weeping" flow was confirmed such that flow continued near the fuel rod such that decay heat could continue to be removed.

Additionally, the PWROG FA tests were performed to define the limits on the mass of debris that may bypass the sump strainer and still provide for an acceptable pressure drop across the FA such that sufficient flow is provided to assure LTCC requirements are satisfied. It is worth noting that significant debris bed formation at spacer grids was not observed in tests conducted at the limiting p:f ratio (i.e., the tests that defined the limiting fiber load.)

4.3 CLADDING HEATUP CALCULATIONS

4.3.1 Clad Heatup Underneath Fuel Grids

In an extreme case, it has been postulated that the volume between the fuel rod and spacer grid could completely fill with debris. An evaluation was performed to determine the cladding surface temperature of a fuel rod within a fuel grid when the rod is plated with debris in a post LOCA recirculation environment. A parametric study was performed to show the effects on the maximum temperature of the fuel rod underneath a grid strap caused by varying debris thickness and the thermal conductivity of the debris. The following sections summarize this analysis. Appendix C contains a detailed discussion of this calculation, including a discussion of assumptions and boundary conditions.

4.3.1.1 Method Discussion

An ANSYS® finite element model of a single fuel rod was created to predict fuel cladding heat up within a spacer grid. The model was cut down to a "1 quarter pie piece." This allowed for the preservation of symmetry of the fuel rod.

To conservatively model convection from the fuel rod surface, the clad was divided into 20 zones. No convection was assumed to occur at the planes of symmetry. A mesh size of 0.05 in. was used for the model.

A constant heat flux was assigned to the entire inner surface of the cladding, and convection heat transfer, with a constant convection coefficient, assigned to the entire outer surface of the rod assembly. Four values were used to parametrically simulate the range of thermal conductivities for the postulated deposition on the fuel clad surface. The thermal conductivity values were 0.1, 0.3, 0.5, and $0.9\left(\frac{\text{BTU}}{\text{hr}*\text{ft}*°\text{F}}\right)$. These thermal conductivities were applied to a range of deposition thicknesses ranging from 5 mils to 50 mils.

4.3.1.2 Fuel Rod Model

The ANSYS model simulated a 12 ft., 0.36-in. diameter fuel rod. The cladding thickness was 0.0225 in. Spacer grids were modeled as 2.25 in. for the large grids, and 0.475 in. for the smaller grids. Table 4-1 lists the elevations of the fuel grids, relative to the bottom of the fuel.

Table 4-1 Grid Locations	
Grid Type	**Elevation from Base (in)**
Standard	24.57
Standard	45.07
Standard	65.67
Mixing Vane	76.77
Standard	86.17
Mixing Vane	97.37
Standard	106.77
Mixing Vane	117.87
Standard	127.27

The material thermal properties for the cladding material were also taken from the WC/T model described in Appendix B. Table C-5 of Appendix C contains the specific values used for this model.

4.3.1.3 Results

The calculated maximum clad temperatures are summarized in Table 4-2 and are shown graphically in Figure 4-1.

The calculated maximum clad temperatures calculated with this model all occur within the spacer grid. Assuming the minimum thermal conductivity of the debris collected in the grid and assuming a debris thickness of 50 mils, a maximum cladding temperature behind a grid of 474°F is calculated. This calculated temperature is well below the 800°F LTCC acceptance basis identified in Appendix A. Thus, the clad surface temperature acceptance basis of 800°F identified in Appendix A is satisfied.

The temperatures calculated with this model are conservatively high. The calculation assumed no flow through the debris in the grid. As observed in the PWROG testing, in the presence of debris, flow continued through the debris buildup. Thus, some coolant flow is expected to pass through the debris buildup within the spacer grid, cooling the clad surface. Not accounting for this flow through the debris, captured between the grid and the fuel rod, provides for a conservatively large cladding temperature.

Table 4-2 Maximum Clad Temperatures (T_{MAX})				
	Debris Thermal Conductivity $\left(\frac{BTU}{hr * ft * °F}\right)$			
	0.1	0.3	0.5	0.9
Debris Thickness (mils)	T_{MAX}	T_{MAX}	T_{MAX}	T_{MAX}
0	260°F	260°F	260°F	260°F
5	–	269°F	266°F	264°F
10	305°F	277°F	271°F	266°F
15	–	284°F	275°F	269°F
20	–	291°F	280°F	271°F
25	–	297°F	284°F	274°F
30	386°F	303°F	288°F	276°F
35	–	310°F	291°F	278°F
40	–	316°F	295°F	281°F
45	–	322°F	299°F	283°F
50	474°F	327°F	302°F	285°F

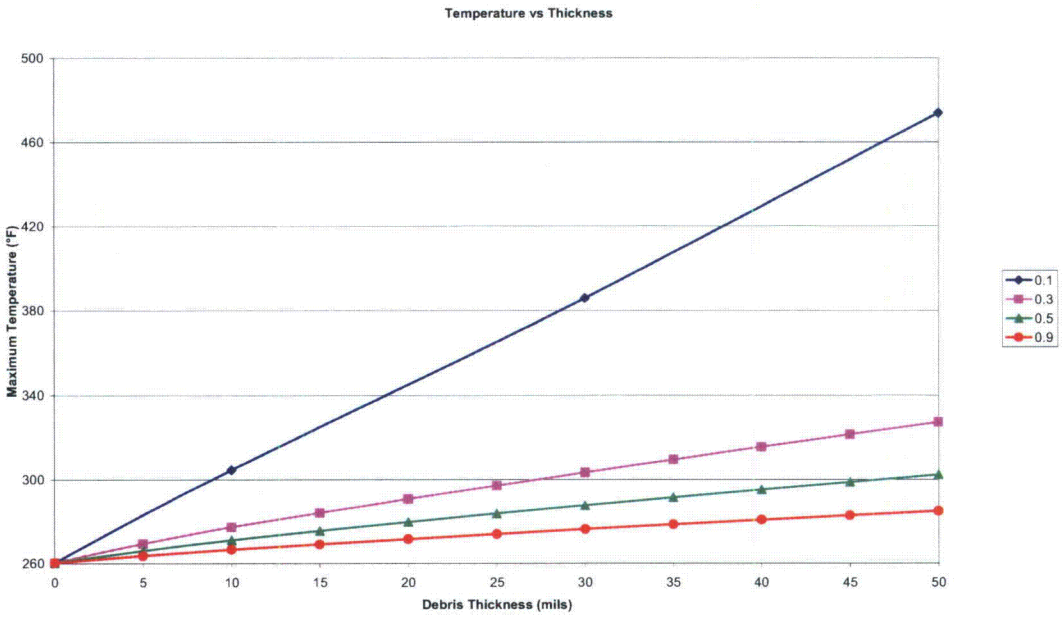

Figure 4-1 Temperature vs. Deposition Thickness and Thermal Conductivity

4.3.2 Cladding Heatup between Grids

The purpose of this analysis was to determine the cladding temperature of a fuel rod between spacer grids with debris deposited on the clad surface in a post-LOCA recirculation environment. While this section discusses blockages at spacer grids, this analysis provides additional information on core cooling when the debris accumulation is allowed to occur without the spacer grid impeding the buildup. A parametric study was performed to show the effects on the maximum temperature of the fuel rod due to deposited debris by varying debris thickness and thermal conductivity. The following sections summarize this analysis. A detailed discussion of the methodology can be found in Appendix D.

4.3.2.1 Methodology

This analysis considered the cladding as being surrounded by concentric layers of oxide, crud, and chemical precipitate, with no gaps between them. The source of heat was decay heat in a post-LOCA environment, and the section of rod analyzed was assumed to be fully exposed to a two-phase liquid/vapor environment in the core. This analysis used the generic resistance form of the heat transfer equation, for a radial coordinate system. A figure of the model is included in Figure 4-2.

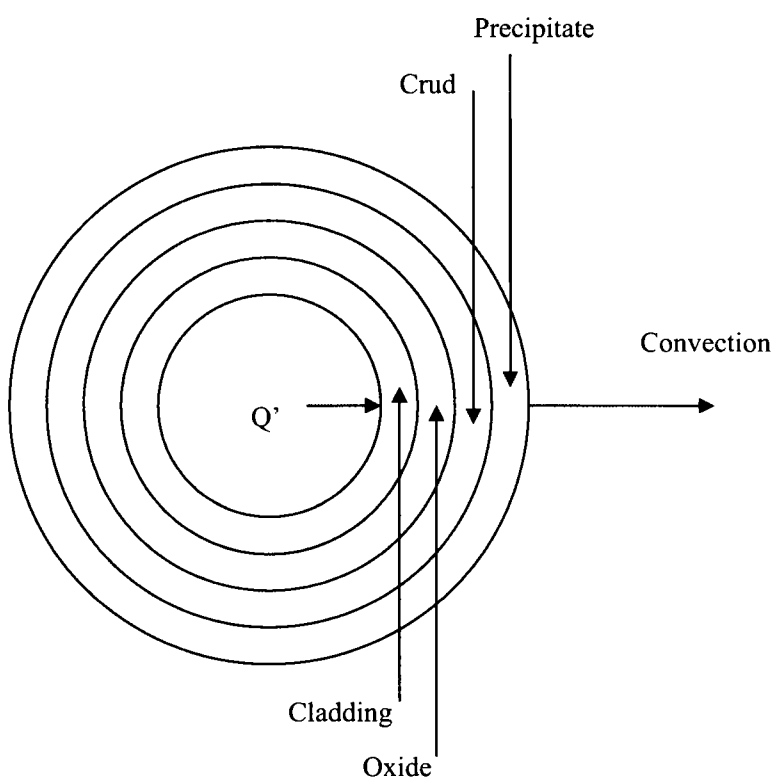

Figure 4-2 Heat Transfer Model (not to scale)

4.3.2.2 Results

Table 4-3 lists the clad/oxide interface temperatures for each of the analyzed cases.

In all cases, the maximum clad surface temperatures calculated between fuel grids under conservatively applied LTCC conditions were less than 560°F. Thus, the clad surface temperature acceptance basis of 800°F is satisfied for debris thickness of up to 50 mils.

Table 4-3 Clad/Oxide Interface Temperature vs. Chemical Precipitate Thickness

Chemical Precipitate Thickness (mils)	$k_{precipitate}$ BTU/hr-ft-°F			
	0.1	0.3	0.5	0.9
0	273°F	273°F	273°F	273°F
10	336°F	293°F	285°F	279°F
20	396°F	313°F	296°F	286°F
30	453°F	331°F	308°F	291°F
40	508°F	350°F	318°F	297°F
50	560°F	367°F	328°F	302°F

Figure 4-3 plots the clad/oxide interface temperature as a function of chemical precipitate thickness for four values of precipitate thermal conductivity.

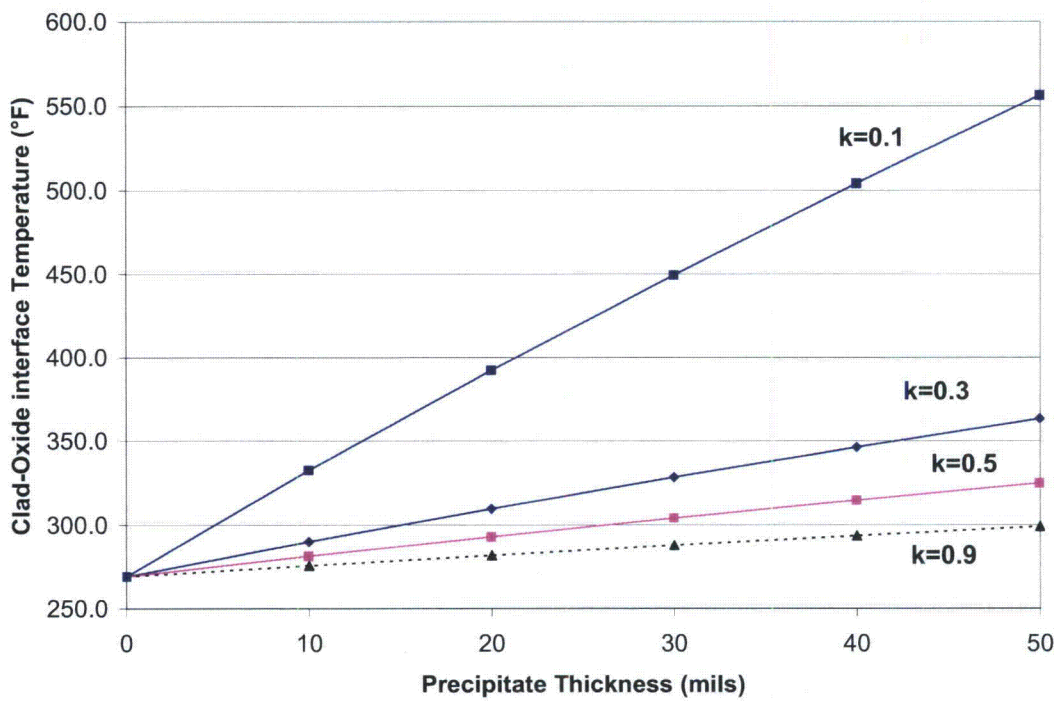

Figure 4-3 Clad-Oxide Interface Temperature vs. Chemical Precipitate Thickness

4.3.2.3 Sensitivity Calculations for Other PWR Fuel Designs

The fuel rod diameter used in the calculations 0.36 in. To demonstrate the applicability of these results to all PWR fuel designs, two sets of sensitivity calculations were performed using the following fuel rod specifications:

- 0.42 in. outer diameter (OD) fuel rod at 0.388 kW/ft power value
- 0.416 in. OD fuel rod at 0.383 kW/ft power value

These two cases, along with the calculations for the 0.360 in. fuel rod, are expected to bound all PWR fuel types.

Table 4-4 lists the clad/oxide interface temperatures for these two sensitivity calculations. The calculations used a bounding low value for thermal conductivity of precipitate.

Table 4-4 Clad/Oxide Interface Temperature vs. Chemical Precipitate Thickness

Chemical Precipitate Thickness (mils)	$k_{precipitate}$ = 0.1 BTU/hr-ft-°F	
	0.422" OD rod	0.416" OD rod
0	284°F	284°F
10	377°F	377°F
20	466°F	466°F
30	552°F	552°F
40	634°F	634°F
50	714°F	713°F

4.4 SUMMARY

Debris that does not collect at the core inlet will pass through the FA bottom nozzle and enter the core region. It is possible that this debris may lodge in some of the smaller clearances in the spacer grids. The debris buildup at these locations will not impede LTCC, because the extent of the buildup is limited by the spacer grid design and debris that does collect will have some packing factor that will allow "weeping" flow through the resulting debris bed.

While FA testing demonstrated that debris did collect at the spacer grids, these observations represent an upper bound of the debris accumulation because of conservatisms in the testing process. Instead, the debris buildup at spacer grids in an operating plant will be considerably lower with a low likelihood of blockages at any singular spacer grid. The blockages that do occur can be treated as localized blockages. Further, a review of the test data indicated that coolant continued to pass through the debris bed, verifying the "weeping" flow asserted above.

For localized blockages, the maximum surface temperature calculated for cladding between two grids, using conservative boundary conditions representative of those during recirculation from the containment sump following a postulated LOCA is less than 800°F. For the 0.360 in. diameter fuel rod, the maximum temperature with 50 mils of precipitate on the clad OD is calculated to be less that 560°F. For the 0.416 in. or 0.422 in. rods, the maximum temperature with 50 mils of precipitate on the clad OD is calculated to be less that 715°F.

These temperatures are conservatively large, as they assume a decay heat level at the time of ECCS switchover to recirculation from the containment sump (20 minutes after initiation of the transient). At this time in the transient, there has been no time to build a layer of precipitate. Chemical products have had little time to form and the concentrations are therefore low, and coolant from the sump is just being introduced into the RV by the ECCS. As decay heat continues to decrease, the calculated clad surface temperatures for a specific thickness of precipitate would also decrease.

Decay heat will continue to be removed even with debris collection at the FA spacer grids. Plants that follow the guidance provided in Section 10 can state that debris that bypasses the strainer will not build an impenetrable blockage at the fuel spacer grids.

5 COLLECTION OF FIBROUS MATERIAL ON FUEL CLADDING

It has been postulated that debris that reaches the core can adhere to the cladding surface. Adherence of fibrous debris is discussed here.

Testing was performed to assess the collection of fibrous debris on fuel cladding surfaces. The results are discussed and evaluated in NEA/CNSI/R (95)11 (Reference 13). The following observations were recorded:

1. From Section 5.4.2.1 of the report, there was little adherence noted of fibrous material to clad surfaces, and the material that did adhere was loose and easily removed. What was observed to adhere to clad surfaces was the binder used to make fiberglass. This binder, however, was observed to carry with it very limited fibrous debris. The report noted that much of the binder is quickly driven off of the fiberglass due to the heat associated with normal operating conditions. These observations were determined to be applicable to both NUKON and Knauf ET Panel.

2. Section 5.4.2.3 of the report provided observations regarding fibrous collection on fuel grids. It was noted that fibrous debris will collect on grids, but that a pure fibrous bed is porous and water will pass through a pure fiber bed.

These test results indicate that fibrous debris, should it enter the core region, will not tightly adhere to the surface of fuel cladding. Thus, fibrous debris will not form a "blanket" on clad surfaces to restrict heat transfer and cause an increase in clad temperature. Finally, during FA testing, recorded in References 7, 8 and 21, fibrous material was not observed to adhere to the fuel cladding. Therefore, adherence of fibrous debris to the cladding is not plausible and will not adversely affect core cooling.

6 PROTECTIVE COATING DEBRIS DEPOSITED ON FUEL CLAD SURFACES

6.1 INTRODUCTION

A concern has been raised regarding the melting of material, particularly protective coatings (paint) that may have been either deposited directly on cladding surfaces, or collected within fuel grids or behind debris beds within the fuel grids. This section discusses both of these occurrences for protective coatings.

6.2 PROTECTIVE COATINGS BEHAVIOR

Protective coatings used inside a PWR containment building may generally be grouped into three categories:

1. Zinc-rich primers

2. Epoxies – either applied directly to the surface of a substrate or to a primer or surfacer that has already been applied to a substrate

3. Non-epoxies – typically applied to small equipment by original equipment manufacturers (OEM's)

The potential for each of these categories of coatings to challenge LTCC is evaluated.

Zinc-rich primers may release elemental zinc to the post-LOCA sump in a powder-like form. The PWROG chemical effects test program described in WCAP-16530-NP-A (Reference 14) has demonstrated that, in general, there is very little zinc reaction with the post-accident sump fluid chemistry. Therefore, zinc-rich primers are evaluated to have negligible effect on post-LOCA chemical precipitate production. If zinc powder were carried into the core and deposited directly onto fuel cladding surfaces or collected within fuel grids, the powder would behave materially and thermally as zinc. The thermal conductivity for zinc is relatively high (approximately 65 Btu/hr-ft-°F). Thus, zinc powder, if it were to be deposited directly onto fuel cladding surfaces or collected behind fuel grids, would not act to insulate the clad surface. Therefore, zinc from zinc-rich primers is not a concern for and does not present a challenge to LTCC.

The non-epoxy coatings are alkyds, urethanes, and acrylics. The amount of these coatings inside containment is generally limited to selected OEM-supplied equipment, such as electrical junction boxes, and represents a small amount of material on the order of a few thousand square feet or less. Thus, these coatings do not represent a significant debris load in the sump. Furthermore, these coatings are, as a class, chemically benign and do not react to the post-LOCA sump fluid. In the case of alkyds, the coating would break down into oligomeric carboxylate salts and glycol. The oligomeric carboxylate salts would tend to inhibit the formation of precipitates. However, since the amount of alkyds inside containments is small, and the salts are expected to be altered by radiolysis, no credit is taken for their presence inside containment. For these reasons, these non-epoxy coatings are evaluated to have a negligible effect on post-LOCA chemical precipitant production and therefore, are not a concern with respect to LTCC.

Most PWR containment buildings have a significant amount of epoxy coatings. Epoxy coatings will retain their structural integrity at temperatures up to about 350°F. When immersed in fluids at temperatures less than 350°F, epoxy coating debris is not sticky or tacky and has no propensity to adhere to the surface of fuel cladding. Therefore, these coatings will not, on their own, attach themselves to cladding.

Testing of epoxy coating systems in both acidic and basic solutions has demonstrated that epoxy coating systems are chemically inert and contribute only a small amount of leachate. From the response to RAI#2 of Section D to Reference 14, the total maximum contribution of leachates from epoxy coatings was conservatively estimated to result in a concentration in the recirculating coolant of less than 16 ppb (parts per billion) for a Westinghouse large four-loop PWR. This value was calculated using the conservative assumption that all leachable material from submerged coatings goes into solution. Considering the small amount of leachates released by epoxy coatings, even under the most conservative assumption that all leachates are released to the sump fluid inventory, epoxy coatings are evaluated to be chemically inert in the post-LOCA chemical environment and therefore have a negligible effect on post-LOCA precipitant production. Thus, epoxy coatings are evaluated to not present a concern with respect to LTCC.

To summarize, protective coatings are generally considered to have minimal impact on the post-LOCA chemistry of the containment sump due to either the small amount of material (non-epoxies) or the demonstrated chemical inertness of the coating itself (zinc-rich primers and epoxy coatings).

6.3 PREDICTED CLADDING TEMPERATURES AND TEMPERATURE-DRIVEN DEBRIS CAPTURE

The WC/T calculations presented in Section 3 and Appendix B simulate the postulated LOCA transient starting with the initial blowdown and extending into the LTCC portion of the event where coolant is recirculated from the containment building sump. Two base case simulations are reported: a case with 82 percent of the core inlet blocked, and a second case with 99.4 percent of the core inlet blocked. In both cases, recirculation from the containment sump is initiated at 1200 seconds (20 minutes). The maximum cladding temperatures calculated for anywhere on the cladding are shown in Figure 3-5.

The temperature history plot of Figure 3-5 demonstrates several important behaviors associated with post-LOCA LTCC and the potential for collecting and melting coatings debris. The predicted clad surface temperature history is evaluated relative to a 350°F temperature value, which is the value at which epoxy coatings begin to lose their structural integrity and become pliable and possibly tacky.

1. Prior to 1200 seconds into the transient, coolant is drawn from the refueling water storage tank (RWST).

2. By 1200 seconds into the transient, the time that recirculation from the containment sump is initiated, the maximum cladding temperatures in the core are about 260°F.

3. After 1200 seconds into the transient, the maximum temperature of either the cladding directly exposed to the recirculating coolant, or the precipitate surface directly exposed to the recirculating coolant, is calculated to be less than 275°F. This temperature is well below the 350°F value at which epoxy coatings begin to be affected by temperature.

During the initial recovery period, and before beginning to recirculate coolant from the sump, all flow to the core originates from the accumulators, the RWST, or the borated water storage tank (BWST). Since there is no coatings debris in the RWST or BWST fluid inventory, no coatings debris is introduced to the fuel while coolant is provided by the accumulators or drawn from the RWST or BWST. By the time that the RWST or BWST inventory is depleted, the core is "recovered" with clad temperatures well below the 350°F temperature at which epoxy coatings are affected by temperature. Figure 3-5 demonstrates that, even with 99.4 percent of the fuel entrance blocked, sufficient water is provided to maintain cladding temperatures at about 250°F.

Additional WC/T sensitivity calculations, described in Section B.5 of Appendix B, were performed for the purpose of determining the amount of blockage necessary to reduce core flow below that necessary to match core boil-off. These calculations represent extreme conditions that are precluded by a plant maintaining the debris loading on the fuel within the limits identified in References 7, 8 and 21. Therefore, the results of those WC/T sensitivity calculations described in Section B.5 do not apply to the discussion on coatings presented here.

Parametric cladding heat-up calculations described in Appendix D were performed for both a blocked grid and for a debris-covered fuel rod. These parametric calculations show that for a precipitate with a sufficiently small value for thermal conductivity and a sufficiently large value of deposited thickness, clad surface temperatures in excess of 350°F may be predicted. However, these same calculations also demonstrate that the temperature of the precipitate surface at the boundary of the coolant, where coatings debris might be expected to collect should they become sticky or tacky, is within about 15°F of the adjacent coolant temperature at the time of switchover. From the fuel rod heat-up calculations described in Appendix B, the surface temperature of the precipitate surface is calculated to be less than 270°F at the time of switchover. The results of these calculations are summarized in Table 6-1. Surface temperatures in the range of 270°F are sufficiently cool so that the material properties of the epoxy coatings will not be affected.

Table 6-1 Precipitate Surface Temperatures vs. Precipitate Thickness

Coolant Temperature, T_∞ (°F)	Precipitate Thickness (mils)	Maximum Precipitate OD Surface Temperature (°F)
250	0	268
250	10	267
250	20	266
250	30	266
250	40	265
250	50	264

Thus, due to the low surface temperatures of either the cladding material before precipitates might collect on the clad surface, or the surface of the precipitate deposited on fuel cladding, the potential for collection, retaining, and melting protective coatings on cladding surfaces or within fuel grids during the LTCC phase of a postulated LOCA is not considered credible.

6.4 SUMMARY

There are three general categories of protective coatings used inside a PWR containment building: zinc-rich primers, epoxy coatings, and non-epoxy coatings. These three categories of coatings have been evaluated to have negligible effect on the generation of precipitate.

1. The amount of non-epoxy coatings used inside a PWR containment building is small and therefore, has negligible contribution to post-LOCA PWR chemistry effects.

2. PWROG testing (Reference 14) has demonstrated that zinc contributes little to the generation of corrosion products post-LOCA and therefore, zinc-rich primers have negligible contribution to post-LOCA PWR chemistry effects.

3. Chemical resistance testing has demonstrated that epoxy coating systems are chemically inert and contribute only a small amount of leachate to the recirculating coolant and therefore, epoxy coatings are evaluated to have negligible contribution to post-LOCA PWR chemistry effects (response to RAI#2 in Section D of Reference 14).

Furthermore, conservative calculations of clad temperatures with deposited precipitate on the cladding surface demonstrate that, for the expected range of deposited precipitate, the precipitate surface temperatures are predicted to remain well below the value that would result in the melting of epoxy coatings debris that may be transported to the core region.

Therefore, protective coatings debris is evaluated to have a negligible effect on the post-LOCA chemistry of a PWR and on post-LOCA LTCC. Also, protective coatings debris has been evaluated to have negligible effect on post LOCA LTCC.

7 CHEMICAL PRECIPITATES AND DEBRIS DEPOSITED ON FUEL CLAD SURFACES

After a LOCA, the chemical makeup of the containment sump and core provides the potential for chemical interactions that may lead to precipitate formation and plate-out on the fuel rods. Consequently, core cooling may be compromised. A method to calculate the amount of these chemical products that might be generated was developed in WCAP-16530-NP-A (Reference 14). Additional work was performed by the PWROG to address excessive margins in the calculations through the use of plant-specific inputs to the calculations. These plant-specific inputs include, but are not limited to plant-specific initial pH values, plant-specific sump fluid temperature histories, and plant-specific alloys of reactant materials.

The chemical precipitates that may form may be transported to the core and influence the pressure drop of debris accumulation at the core inlet or spacer grids. This effect on LTCC is addressed in Section 3 and 4 and Appendix G.

Chemicals may also deposit on the hot fuel rods and possibly insulate them and inhibit decay heat removal. The method developed in WCAP-16530-NP-A (Reference 14) was extended to predict chemical deposition on fuel cladding due to the transport of debris and chemical products into the RCS and the core region by the coolant recirculated from the containment sump. The new method is called the LOCA deposition model (LOCADM).

7.1 DESCRIPTION OF LOCADM

LOCADM is a calculation tool that can be used to conservatively predict the build-up of chemical deposits on fuel cladding after a LOCA. The source of the chemical products is the interaction of the fluid inventory in the reactor containment building sump with debris and other materials exposed to and submerged in the sump fluid or containment spray fluid. LOCADM predicts both the deposit thickness and cladding surface temperature as a function of time at a number of core locations or "nodes." The deposit thickness and maximum surface temperature within the core are listed in the output for each time period so that the user can compare these values to the acceptance basis for long term cooling.

A complete description and qualification of LOCADM is presented in Appendix E. A summary is provided here.

The chemical inputs into LOCADM are the volumes of different debris sources such as fiberglass and calcium silicate (cal-sil) insulation. The surface areas of uncoated concrete, aluminum submerged in the sump, and aluminum exposed to spray are also required. The sump and spray pH are specified as a function of time, as are the inputs of sodium hydroxide, trisodium phosphate, sodium tetraborate, lithium hydroxide and boric acid as appropriate.

Chemical product transport into the core is assumed to occur by the following process:

1. Containment materials corrode or dissolve, forming solvated molecules and ions.

2. Some of the dissolved material precipitates, but the precipitates remain in solution as small particles that do not settle.

3. The dissolved material and suspended particles pass through the sump strainer and into the core during recirculation. For the purpose of adding conservatism, it is assumed that none of the precipitates are retained by the sump strainer or any other non-fuel surfaces.

Note that the transport of small fibers that do not dissolve but are small enough to be transported through the sump strainer and into the core is not considered explicitly in LOCADM. The quantity of transported fines is expected to be small compared to both the total amount of debris and the amount of debris that dissolves or corrodes. Fiber can be accounted for in LOCADM in cases where it is significant by use of a "bump-up factor" applied to the initial debris inputs. The bump-up factor is set such that total mass of deposits on the core after 30 days is increased by the best estimate of the mass of the fiber that bypasses the sump strainer.

Coolant flow rates into the reactor mixing volume as a function of time must be provided by the user and are obtained from a plant's safety analysis for LTCC. The relative amounts of steam and liquid flow out of the reactor mixing volume are calculated by LOCADM. The core input is generalized. The coolant flow could be coming from the CL, the HL, or from upper plenum injection. Various operational modes are accounted for by varying the rate of flow into the mixing volume and the source of the flow (safety injection or recirculated coolant.) Values for generically applicable mixing volumes have been identified and will be provided to users. The temperature of the sump and reactor coolant as a function of time must also be entered by the user.

Within the mixing volume, the coolant is assumed to be perfectly mixed. Coolant chemical products entering the reactor are distributed evenly between all core nodes before deposition calculations are performed. The entire mixing volume is also assumed to be at the same temperature. Pressure is determined by the upper plenum pressure and the hydrostatic pressure at different elevations in the core. No attempt was made to model flow within the mixing volume and variations in that flow that might be caused by grids and flow obstructions. Since flow was not modeled, a heat transfer coefficient of 400 W/m^2-°K (70 BTU/ft^2 °F) was assumed for transfer of heat between bulk coolant with the fuel channels and the surface of the deposits since this is a typical heat transfer coefficient for convective flow within natural circulation systems.

LOCADM deposits chemical products that are dissolved or suspended in solution throughout the core in proportion to the amount of boiling in each core node. It is assumed that deposition rate is equal to the steaming rate multiplied by the chemical product concentration at each node. If there is no boiling, the chemical products are distributed according to heat flux, at an empirically derived rate that is $1/80^{th}$ of the deposition that would have occurred if all of the heat had gone into the boiling process.

The deposition algorithm does not rely on solubility or any other chemical characteristics of the chemical products to determine the deposition rate. All chemical material that is transported to the fuel surface by boiling is assumed to deposit. LOCADM uses a default deposit thermal conductivity for the deposited material of 0.1 Btu/(hr-ft-°F), which is low enough to bound expected core deposits. Likewise, the default deposit density is low enough (e.g., 35 lbm Ca/ft^2) to bound expected deposits including those that incorporate absorbed boron or boron bonded to chemical product elements. Consistent with current

licensing basis calculations for PWRs that demonstrate that the boric acid concentration in the core is limited to values below the solubility limit, the LOCADM does not precipitate boric acid. The same is true for sodium phosphate, sodium borates, and sodium hydroxide, which are also highly soluble.

The core noding within LOCADM can be adjusted by the user. Appendix E provides guidance to the LOCADM user for node selection for different types of cores.

LOCADM runs within Microsoft Excel® and should be easy to use for those familiar with Excel. The first sheet of the workbook instructs the user on how to enter the chemical and flow inputs into worksheets in tabular form. A macro written in Visual Basic for Applications is then run. The macro reads the input, looks for input errors, calculates core conditions in one second intervals, and then outputs the results within the same workbook.

7.2 USE OF LOCADM

Each plant must perform a LOCADM analysis in order to demonstrate the plant is operating within the acceptance criteria defined in Section 2. This section provides a brief overview of how to perform a LOCADM calculation. Appendix E and References 15, 16, 17, and 18 must be consulted for additional guidance.

7.2.1 Overview

7.2.1.1 Inputs

There are 5 input worksheets: 1) Time Input, 2) Materials Input, 3) Materials Conversion, 4) Core Data Input and 5) Switches.

Time Input

The Time Input worksheet contains inputs for time, pH, temperature, flows, pressure and the LOCA mode. Generally, higher pH and temperature values are conservative. Spray pH values should not be entered after the containment spray is terminated. The guidance in Reference 17 should be followed when addressing the flow data. The pressure column contains an equation to calculate the saturation pressure of the RV coolant temperature. Reference 17 also provides guidance for pressure inputs.

The LOCA mode is defined specifically for LOCADM and reflects the times at which changes take place in the ECCS operations. The modes are defined as follows:

- Mode 1: Blowdown/Refill phase (blowdown of water from RCS immediately after the LOCA and refill from accumulators and RWST).
- Mode 2: After reactor vessel refill but before recirculation begins.
- Mode 3: Recirculation from the sump (assumed water is injected into the CL).
- Mode 4: HL injection (still recirculating water from the sump and injecting into HL).

Materials Input

The Materials Input worksheet contains the masses of debris that would be present in the post-LOCA sump that could create deposits. These inputs are relatively straightforward but care must be taken to ensure the units are consistent. The other inputs in this worksheet are the initial sump liquid volume and the initial RV liquid mass.

The initial sump liquid volume must equal the volume of water present in the sump at the start of recirculation (after blowdown and refill have occurred) if the Pre-Filled Sump Option, described in Appendix E, is being used. If the Pre-Filled Sump Option is not being used, this value is zero. It is good practice to run two analyses, one with a minimum sump volume and the other with a maximum sump volume to ensure the most conservative volume is being used.

Refer to Reference 15 for the recommended initial RV liquid mass. These values are based upon plant design.

Materials Conversion

The Materials Conversion worksheet is used to convert the inputs from the Materials Input worksheet to masses in kilograms. Densities in this worksheet are typical but any density can be changed to reflect plant-specific conditions.

Core Date Input

The Core Date Input worksheet contains data about the reactor core. Values for the majority of the variables can be obtained from Appendix E and References 15, 16, 17, and 18. The plant fuel vendor must be consulted to assure appropriate inputs for the core peaking.

Switches

This worksheet can be used to impose certain additional criteria on the analysis. In most cases, the guidance is to retain the default inputs.

7.2.1.2 Outputs

The results of the LOCADM analysis are provided in three worksheets: 1) Out, 2) Releases by Material and 3) Scale Thickness. The Out worksheet contains the majority of the results of the LOCADM analysis. It is a good practice to make sure the final out mass is equal to the input mass plus the total mass of all materials released into the sump water (this mass is the sum of materials in the 'Releases by Material worksheet). Care with units needs to be taken when performing this calculation.

The acceptance criteria results are found in the 'Maximum LOCA scale thickness' and 'Fuel Cladding Temp at Max Thickness' columns of the Out worksheet. Additional calculations are required in order to calculate the total deposition on the fuel rod:

- The total deposition is comprised of crud, oxide and the LOCA scale.
- Maximum LOCA Scale Thickness: The last value in the "Maximum LOCA scale thickness" column.
- Crud Thickness: Assumed to be 140 microns.
- Oxide Thickness: Assumed to be 152 microns
- Add these three values (in microns) and convert to mils (25.4 microns per 1 mil).
- Compare this value to the acceptance criteria of 50 mils

7.2.1.3 Additional Steps

Aluminum Release Rate

In order to provide more appropriate levels of aluminum release for the LOCADM analysis in the initial days following a LOCA, licensees shall apply a factor of two to the aluminum release. The recommended procedure for modifying the aluminum release rate is described in Reference 18.

Bump-Up Factor

LOCADM does not contain an input for debris which bypasses the sump strainer and is available for deposition in the core. Only material released from corrosion or dissolution processes is considered. However, some debris fines may bypass the sump strainer and enter the core area where it could be deposited. A quantitative estimate of the effect of the fiber on deposit thickness and fuel temperature must be accounted for in LOCADM by use of a "bump-up factor" applied to the initial debris inputs. The bump-up factor is set such that total release of chemical products after 30 days is increased by the best estimate of the mass of the fiber that bypasses the sump strainer. This allows the bypassed material to be deposited in the same manner as a chemical reaction product. The recommended procedure for including fiber bypass in the LOCADM deposition calculations is illustrated in Reference 17.

7.2.2 Summary

The methodology presented here is intended to provide a plant specific method to evaluate core deposition, which meets the NRC requirements for predicting post LOCA deposit formation on the core. The recommended modeling approach assumes that all material transported to the fuel surface by boiling will deposit. This conservative approach diminishes the importance of impurity chemical or radiochemical reactions since these reactions could not increase the amount of core deposition beyond what was already measured. Organic coating materials are not expected to experience radiation levels which would cause degradation and subsequent transfer onto heat transfer surfaces. Also, it is expected that most plants using this methodology will be able to demonstrate acceptable LTCC in the presence of core deposits.

8 BORIC ACID PRECIPITATION

All three US PWR designs (B&W, CE, or Westinghouse) use boron as a core reactivity control method and are subject to concerns regarding potential post LOCA boric acid precipitation in the core. All three plant designs have procedures that instruct the operators to realign the ECCS to prevent the core region boric acid concentration from reaching the precipitation point. The common approach for demonstrating adequate boric acid dilution in a post LOCA scenario includes the use of simplified methods with conservative boundary conditions and assumptions. These simplified methods are used with limiting scenarios in calculations to show that boric acid precipitation will not occur or to determine the time at which appropriate operator action must be taken to initiate an active boric acid dilution flow path. In light of NRC staff and ACRS challenges to the simplified methods commonly used, it has recently become clear that additional insights and new methodologies are needed to answer fundamental questions about boric acid mixing and transport in the RCS and potential precipitation mechanisms that may occur both during the ECCS injection phase and the sump recirculation phase after a LOCA. In response to this need, the PWROG is currently funding a program to define, develop and obtain NRC approval of post LOCA-boric acid precipitation analysis scenarios, assumptions and acceptance criteria and resultant methodologies that demonstrate that adequate post-LOCA LTCC.

9 COOLANT DELIVERED TO THE TOP OF THE CORE

There are two scenarios by which coolant can be delivered to the top of the core.

1. For a break in the CL piping, plants may introduce recirculating coolant into HLs to act as flushing flows to mitigate the potential for boric acid precipitation.

2. The ECCS for Westinghouse two-loop PWRs provide for the delivery of coolant directly to the upper plenum through injection nozzles in the RV upper plenum (Upper Plenum Injection or UPI). This flow path is established at the initiation of the ECCS actuation and is maintained throughout plant recovery.

When the ECCS is recirculating coolant from the containment sump, debris in the recirculating coolant can flow into the core.

9.1 HOT LEG RECIRCULATION

HL recirculation is typically initiated several hours after the postulated large LOCA. At this time, the containment sump inventory typically has been recirculated through the ECCS and RCS several times. This provides for particulate and fibrous debris generated by the initial break and carried in the recirculating coolant to be depleted either by capture on the sump strainer, fuel assemblies, or by settle-out in the containment sump or in low-flow locations of the ECCS RV flow path such as the RV lower plenum. Thus, the amount of particulates and fibrous debris in the recirculating flow at the time of initiation of HL recirculation is small. Examples of debris depletion are given in WCAP-16406-P-A (Reference 2).

9.2 UPPER PLENUM INJECTION PLANTS

The ECCS for Westinghouse two-loop PWRs provide for the delivery of coolant directly to the upper plenum through injection nozzles in the RV UPI. This flow path is established at the initiation of the ECCS actuation and is maintained throughout plant recovery. This flow path may provide for the delivery of debris in the recirculating coolant from the initiation of recirculation from the containment sump.

The sump strainer will limit both the size and the amount of the particulate and fibrous debris to the reactor.

1. For a HL break, upon switchover from injection from the RWST, coolant flow to the core is through the UPI ports with all CL flow initially secured. The amount of debris that reaches the core depends on the flow patterns in the upper plenum and is discussed in detail in Section 9.3.

2. For a CL break, the debris introduced by the UPI flow to the RV will flow into the core.

9.3 UPPER PLENUM DEBRIS TRANSPORT FOR HOT-LEG BREAK SCENARIO

ECCS that enters the upper plenum following a HL break for UPI plants can either enter the core or exit the break. There is some flow to the core to make up for steam produced by the decay heat removal process. However, the majority of the flow will exit the break. This assertion is supported by the following discussion.

The UPI nozzle for a Westinghouse 2-loop PWR has an inside diameter of about 4 inches. These nozzles are located approximately 180° opposite of each other. Assuming a minimum total UPI flow of 1200 gpm and an equal flow distribution between the two UPI nozzles, the flow rate through each nozzle is 600 gpm or approximately 1.34 ft^3/sec. Thus, the minimum velocity of the UPI flow through each UPI nozzle is calculated to be approximately 15.3 ft/sec. At these jet velocities, the upper plenum coolant inventory is not stagnant. Rather, the UPI jet flow, in conjunction with impingement of the jets on upper internals structures, generates turbulent mixing of the UPI flow with the coolant inventory in the upper plenum.

The volume between the top of the active fuel and the bottom of the HL for a Westinghouse two-loop PWR is about 190 ft^3. For a UPI flow of 1200 gpm, the equivalent volumetric flow is about 2.68 ft^3/sec. Neglecting any water level above the bottom of the HL, which would be small for a double-ended guillotine HL break, and assuming a constant volume of water in the upper plenum, approximately 71 seconds are required to "turn over" the entire fluid inventory of the upper plenum. This quick turn-over time further supports that the upper plenum is well mixed by the UPI flow.

The turbulent mixing of the upper portion of the core will result in a situation where debris that enters the upper plenum with the coolant will either be kept in suspension and expelled through the HL piping, or will be deposited over a broad area of the core.

9.4 COLLECTION OF DEBRIS ON FUEL

Considering the above, the debris that may be captured on fuel features such as mixing vanes, fuel grids and on debris capturing features at the bottom of the fuel is limited. The collection of debris by these features will also occur over time; that is, the formation of a debris bed will take time to develop. As noted in Section 4.1, the debris that is collected will have some packing factor that will allow "weeping" flow through particulate debris buildup and into the core. That is, complete compaction of the debris will not occur and the packing density of the debris is limited to less than unity or perfect compaction. Again, from Reference 12, the packing will most likely be less than ~60 percent. This will allow for coolant to pass through a debris bed that might form.

The 60% packing factor can be conservatively thought of as a 60% blockage of the core. This would present a bounding or maximum resistance to flow through the debris bed. The WC/T evaluations described in Section 3.2 demonstrate that adequate flow is maintained with a deterministically assigned blockage of 82% to provide for LTCC. Thus, conservatively taking the 60% packing factor to be representative of a 60% blockage, adequate LTCC will be provided for.

Westinghouse 2-loop plants with UPI do not maintain flow into CLs once the switchover of the ECCS from injecting from the RWST to recirculating coolant from the reactor containment building sump is accomplished; the recirculating flow is ducted to the RV through the UPI penetrations in the reactor upper

plenum. For CL breaks, coolant is introduced into the RV from the UPI nozzles and flows down though the core and out the break. If blockage due to the accumulation of debris were to occur, it would occur at the top of the fuel. As was the case with bottom-up flooding of the core, and as demonstrated in the data presented in Section 3.1, a complete blockage is not expected of plants that are within the debris load acceptance criteria. This was demonstrated by testing as described in Section 9.5.

As described in Section 9.3, the turbulent mixing of the upper portion of the core will cause the debris that enters the upper plenum with the coolant to either be kept in suspension and expelled through the HL piping (for a HL break scenario), or will be deposited over a broad area of the core. Should the fiber collect preferentially at grid locations, the analysis performed in Appendix D of this report applies to UPI plants and this analysis demonstrates that adequate cooling in such locations will be maintained. The testing described in Section 9.5 demonstrates sufficient flow will be maintained with debris in the fluid delivered to a FA that LTCC is not challenged. Thus, in case of either a HL or a CL break, the formation of a debris bed on the bottom of the fuel is not considered credible.

If the coolant flow is sufficiently restricted through a debris bed that clad temperatures increase to about 15°F to 20°F above the coolant temperature, the coolant would begin to boil. The steam formed would be about 40 to 50 times the volume of the water, and would cause the debris bed to be displaced, allowing for coolant to flow to and cool the cladding surface. This process would provide for cooling of the clad.

The conservative clad heat-up calculations documented in Appendix D demonstrate that acceptably low clad temperatures are calculated with as much as 50 mils of solid precipitate applied to the outside surface of a fuel rod. These calculations provide further assurance that, with weeping flow through a debris bed collected on fuel elements, LTCC for UPI plants will be maintained.

The evaluation of effect of chemicals dissolved in the UPI flow for a HL break are performed on a plant-specific basis using the LOCADM calculation tool described in Section 7 and Appendix E. To account for deposition on fuel cladding in the core, a bump-up factor is used in the LOCADM calculation to deposit fiber material according to the core boiling and heat flux distribution.

9.5 TEST FOR UPI-DESIGNED PLANT

The purpose of the UPI test was to perform testing to justify the applicability of the debris load acceptance criteria defined by HL break conditions to UPI-designed plants. To simulate the limiting break, the UPI CL break (analogous to the previously discussed HL break) was tested. This test was conducted with the maximum debris loads that were tested in the Westinghouse HL test. The pressure drop was well below what is required to maintain core flow for UPI plants. Therefore, the test results demonstrated that sufficient flow will reach the core to remove core decay heat and the acceptance criteria developed at HL conditions is bounding and applicable to UPI plants. That is, the guidance provided in Section 10 is applicable to all plant designs, including UPI plants. Appendix G and Reference 8 contain additional information about the UPI test and the applicability of the debris acceptance criteria to UPI plants.

10 SUMMARY

10.1 DISCUSSION

PWR containment buildings are designed to facilitate core cooling during a postulated LOCA event. In some LOCA scenarios, the cooling process requires water discharged from the break, ECCS, and CSS to be collected in a sump for recirculation by these systems. The discharged coolant water in the sump will contain chemical impurities and debris as the result of interaction with containment materials.

There has been concern that following a LOCA, the chemical precipitate, fibrous and particulate debris within the sump could collect on the sump strainer and block the flow of cooling water into the core. There is also concern about the effects of the debris that passes through the sump strainer. This debris could be ingested into the ECCS and flow into the RCS.

The PWROG sponsored a program to analyze the effects of debris and precipitates on core cooling for PWRs when the ECCS is realigned to recirculate coolant from the containment sump. The intent was to demonstrate adequate heat-removal capability for all plant scenarios. Additionally, the PWROG initiated prototypical, bounding FA testing to establish limits on the debris mass (particulate, fibrous, and chemical) that could bypass the reactor containment building sump strainer. These debris limits will not cause unacceptable head loss that would impede core inlet flow and challenge LTCC. These limits will be referred to as the debris load acceptance criteria and are intended to demonstrate that adequate flow for long-term decay heat removal exists at these levels.

This evaluation considered the design of the PWR, the design of the open-lattice fuel, the design and tested performance of replacement containment sump strainers, the tested performance of materials inside containment, and the tested performance of fuel assemblies in the presence of debris. Specific areas addressed in this evaluation included:

- Blockage at the core inlet
- Collection of debris on fuel grids
- Collection of fibrous material on fuel cladding
- Protective coating debris deposited on fuel clad surfaces
- Production and deposition of chemical precipitants
- Coolant delivered from the top of the core

The following acceptance criteria were selected for the evaluation of the topical areas identified above:

1. The maximum clad temperature shall not exceed 800°F.
2. The thickness of the cladding oxide and the fuel deposits shall not exceed 0.050 inch in any fuel region.

These acceptance bases were applied after the initial quench of the core and are consistent with the LTCC requirements stated in 10 CFR 50.46 (b)(4) and 10 CFR 50.46 (b)(5). They do not represent, nor are they intended to be, new or additional LTCC requirements. These acceptance bases provide for demonstrating that local temperatures in the core are stable or continuously decreasing and that debris entrained in the cooling water supply will not affect decay heat removal.

In addition to these acceptance criteria, utilities must evaluate site-specific fiber loading against the debris load acceptance criteria provided in this document. (The debris load was defined through a conservative FA test program; conservatisms of this program are discussed in Appendix G.) Plants with debris loads above the debris load acceptance criteria may demonstrate adequate LTCC capability through engineering evaluations of plant-specific conditions and/or plant-specific testing. This revision has been updated with revised debris loads. The maximum allowable debris loads published in Revision 1 of this document are no longer valid. Subsequent to the publication of Revision 1 of this document, RAIs were received (Reference 20) and additional testing was conducted to address these issues. The additional testing is summarized in References 7, 8 and 21.

In order to demonstrate reasonable assurance of LTCC, all plants must evaluate the areas identified above and demonstrate they are bounded by the debris load acceptance criteria, maximum fuel cladding temperature, and maximum deposit thickness requirements. Specifically,

- Adequate flow to remove decay heat will continue to reach the core even with debris from the sump reaching the RCS and core. Plants that follow the guidance provided in Section 10.2 can state that debris that bypasses the strainer will not build an impenetrable blockage at the core inlet. While any debris that collects at the core inlet will provide some resistance to flow, in the extreme case that a large blockage does occur, numerical analyses have demonstrated that core decay heat removal will continue. The details supporting this evaluation are provided in Section 3.
- Decay heat will continue to be removed even with debris collection at the FA spacer grids. Plants that follow the guidance provided in Section 10.2 can state that debris that bypasses the screen will not build an impenetrable blockage at the fuel spacer grid. In the extreme case that a large blockage does occur, numerical and first principle analyses have demonstrated that core decay heat removal will continue. The details supporting this evaluation are provided in Section 4.
- Fibrous debris, should it enter the core region, will not tightly adhere to the surface of fuel cladding. Thus, fibrous debris will not form a "blanket" on clad surfaces to restrict heat transfer and cause an increase in clad temperature. Therefore, adherence of fibrous debris to the cladding is not plausible and will not adversely affect core cooling. The details supporting this evaluation are provided in Section 5.
- Protective coating debris, should it enter the core region, will not restrict heat transfer and cause an increase in clad temperature. Therefore, adherence of protective coating debris to the cladding is not plausible and will not adversely affect core cooling. The details supporting this evaluation are provided in Section 6.
- The chemical effects method developed in WCAP-16530-NP-A was extended to develop a method to predict chemical deposition of fuel cladding. The calculational tool, LOCADM, will be used by each utility to perform a plant-specific evaluation. It is expected that each plant will be able to use this tool to show that decay heat would be removed and acceptable fuel clad temperatures would be maintained. The details for using LOCADM are provided in Section 7 and Appendix E.
- The commonly used approach for demonstrating adequate boric acid dilution in a post-LOCA scenario includes the use of simplified methods with conservative boundary conditions and assumptions. In light of NRC staff and ACRS challenges to the simplified methods commonly

used, it has recently become clear that additional insights and new methodologies are needed to answer fundamental questions about boric acid mixing and transport in the RCS and potential precipitation mechanisms that may occur both during the ECCS injection phase and the sump recirculation phase after a LOCA. This will be addressed in a separate PWROG program. This program is discussed in Section 8.

- The PWROG FA test results demonstrated that sufficient flow will reach the core to remove core decay heat for all PWR plant designs. The guidance provided in Section 10.2 is applicable to all PWR plant designs, including UPI plants. The UPI plants do not have separate guidance. The details supporting this evaluation are provided in Section 9.

10.2 DEBRIS LOAD LIMITS

The purpose of the FA testing described in this report and the supporting test reports (References 7, 8 and 21) was to develop a bounding acceptance criteria for the mass of debris that can reach the RCS and not impede long-term core cooling flows to the core. The testing demonstrated that fiber is the limiting variable and is the only debris type requiring a limit.

Due to the conservative test design used to define fiber limits, bounding guidelines have been developed with which plants can use to determine the maximum allowable fiber load that can reach the core and not impede core cooling. Details on the conservatisms of testing are provided in Appendix G.

- The AREVA testing conducted in support of this program demonstrated that 15 g of fiber/FA does not cause a blockage that will challenge LTCC, the maximum dP due to debris (dP_{debris}) was very small (Reference 21) and all plants have an available driving head (dP_{avail}) that is considerably greater. Therefore, all PWROG plants can demonstrate LTCC is not impeded if the plant-specific fibrous debris load is less than or equal to 15 g of fiber/FA.
- Due to the low dP_{debris} value recorded with 15 g of fiber/FA, utilities could conduct a plant-specific test with test parameters representative of their site to increase this fiber limit. If a plant-specific available driving head value were needed, the methodology is presented in Section 2.18 of Reference 19. Since PWROG testing demonstrated the HL break is limiting, the calculation of HL available driving head is the relevant value. That value could be compared to the dP value recorded from the test conducted with 15 g of fiber (Reference 21) to demonstrate significant margin exists between the expected pressure loss due to a debris bed and the expected driving head available to support core flow. Additionally, this value could be used to develop an engineering evaluation and/or plant-specific test to define an increased allowable fiber loading.
- The test conducted with Westinghouse fuel at CDI to evaluate test facilities, 1-W-FPC-0811, was conducted with 25 g fiber/FA. This test demonstrated flow was able to continue to enter the core, even though the flow rate had to be reduced during the test (Reference 8). Therefore, plants with Westinghouse fuel that have a driving head greater than or equal to this dP_{debris} value, and operate at conditions similar to tested conditions, can withstand 25 g fiber/FA.
- As demonstrated by CIB54, Westinghouse-fueled plants that can maintain high sump water temperatures can decrease the dP_{debris} at a specific fiber loading (Reference 8). This results in the capability of increasing allowable fiber load.

- A test, CIB53, sucessfully demonstrated that if plants can delay the formation of chemical precipitates until after HLSO, a greater amount of fiber will be able to enter the core without impeding LTCC (Reference 8).
- All tests conducted at the limiting p:f ratio conditions, see the largest increase in head loss when chemical precipitates are added to the test loop. If a plant can demonstrate chemical precipitates do not form, the dP_{debris} values recorded with just particulate and fiber in the test loop can be used in conjunction with the dP_{avail} to make a determination on the amount of allowable fiber (References 7, 8 and 21).

The allowable fiber limit defined for a plant will be used in combination with the analyses presented in this document to demonstrate adequate flow for long-term decay heat removal.

10.3 GUIDANCE TO LICENSEES CONCERNING EVALUATION OF DEBRIS

Actions are required of utilities to prove acceptable LTCC with debris and chemical products in the recirculating fluid. Plants will have to perform plant-specific LOCADM evaluations and prove the plant-specific debris loads do not impede LTCC. These actions along with reference to this report provide the basis for demonstrating that LTCC will not be compromised following a LOCA as a consequence of debris ingestion to the RCS and core.

10.3.1 LOCADM

Plants will have to perform a LOCADM evaluation (Section 7 and Appendix E) based on plant-specific debris inputs and prove they are within the acceptance criteria.

10.3.2 Debris Acceptance Criteria

The FA testing was reported in proprietary submittals that support this document. The results from these FA tests are discussed in the proprietary test reports (References 7, 8 and 21). As part of the effort to invoke this WCAP in the plant licensing basis, each plant will evaluate their plant-specific fiber debris load using the guidance provided in subsection 10.2 of this document. It is the evaluation of plant-specific fiber debris loads in combination with the analyses presented in this document utilities will use to demonstrate adequate flow for long-term decay heat removal.

Plants that are within the limits of the parameters tested are bounded by the tests and meet the long-term core cooling requirements. Several courses or actions have been identified for plants whose debris loads are outside the limits tested. These options include, but are not limited to, reducing problematic debris sources by removing or restraining the affected debris source, conducting plant-specific FA testing, performing engineering evaluations of plant-specific conditions, developing a technical basis for the removal or reduction of chemical precipitate formation, and evaluating debris transport/bypass calculations.

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10. Revised Guidance for Review of Final Licensee Responses to Generic Letter 2004-02, "Potential Impact of Debris Blockages on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors," March 2008. [ML080230112]

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15. OG-07-419, "Transmittal of LOCADM Software in Support of WCAP-16793-P, 'Evaluation of Long-Term Cooling Associated with Sump Debris Effects' (PA-SEE-0312)," September 2007.

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20. NRC Document, "Request for Additional Information RE: Pressurized Water Reactor Owners Group Topical Report WCAP-16793-NP, Revision 1, 'Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid' (TAC No. ME1234)," January 2010. [ADAMS Accession Number: ML101800087]

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APPENDIX A
GSI-191 LTCC ACCEPTANCE BASIS

A.1 INTRODUCTION

The PWROG is leading an industry effort to resolve the issues associated with GSI-191 as they pertain to the core. Part of that resolution involves defining the relevant LTCC bases. This appendix describes the acceptance criteria that will be used in determining GSI-191 acceptance of the debris effects on fuel. These LTCC acceptance criteria are based on the requirements of Title 10 of the Code of Federal Regulations, Part 50.46 (10 CFR 50.46). The criteria are to be used with engineering evaluations that demonstrate acceptable LTCC, once established following the initial recovery of the core post-LOCA, is successfully maintained. Successful LTCC is defined as meeting the criteria defined in this appendix.

A.2 REQUEST FOR LONG-TERM CORE COOLING REQUIREMENT CLARIFICATION

On April 12, 2006, NRC staff met with representatives from industry and Westinghouse to discuss acceptance criteria for nuclear plant licensees to employ for evaluating potential effects of debris that may be ingested into the RV following the transition to sump recirculation following a postulated large-break LOCA. The purpose of the criteria is to assist licensees in addressing issues associated with GSI-191 PWR sump performance.

By letter dated July 14, 2006, Westinghouse requested the NRC clarify its LTCC requirements under 10 CFR 50.46 (Reference A-1). The requests were specified as follows:

1. It is requested that NRC provide clarification of the requirements and acceptance criteria for LTCC once the core has quenched and reflooded. This clarification will be used by PWROG in developing the GSI-191 debris ingestion evaluation method for reactor fuel.

2. The standard mission time employed for GSI-191 is 30 days. This mission time may not be appropriate for evaluation of nuclear fuel issues. The NRC staff is requested to provide clarification on this requirement and how it applies to evaluation of debris ingestion effects on reactor fuel. The PWROG will use this clarification in developing the GSI-191 debris ingestion evaluation method for reactor fuel.

By letter dated August 16, 2006, the NRC responded to the request for clarification (Reference A-2). The NRC letter provides the basis for defining LTCC requirements that may be used to address issues associated with GSI-191.

A.3 NRC CLARIFICATION OF LONG-TERM CORE COOLING REQUIREMENTS

With respect to Item 1, the NRC response identified that the 10 CFR 50.46 rule was constructed in two parts as follows:

> The first part governs the performance of the emergency core cooling system (ECCS) during the initial phases of blow down, quench and re flood. During this period, the ECCS is injecting water from the refueling water storage tank (RWST) into the reactor in an effort to ensure that fuel damage is minimized. The criteria used to conclude that fuel damage is minimized are the temperature criteria for the cladding and the oxidation and hydrogen generation values.
>
> The rule then establishes a criterion for long term cooling during any recirculation phase (whether natural or forced recirculation). The acceptance criterion is simply that the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core.
>
> The NRC staff has typically considered the criteria in paragraph (b)(5) to be satisfied when the fuel in the core is quenched, the switch from injection to recirculation phases is complete, and the recirculation flow is large enough to match the boil-off rate. The staff is concerned about the potential for loss of long term cooling capability from chemical effects (boron precipitation) or physical effects (debris). For example, the staff's standard position is that a core flushing flow path should be established well before boron concentrations reach the precipitation limit (Ref. Information Notice 93 66). Similarly, analysis should demonstrate that no significant increase in calculated peak clad temperature (PCT) occurs by demonstrating that the bulk temperature at the core exit is maintained essentially constant at the temperature achieved at the initiation of recirculation or is continuing to decrease. The following paragraph provides further qualification of the NRC concerns with respect to increases in fuel temperature during the recirculation phase.
>
> While the current staff position is conservative with respect to protection of the fuel, other options may be available that provide protection of the fuel, assure a coolable geometry, and could be used to demonstrate compliance with paragraph (b)(5). The staff notes that fuel qualification testing has been restricted to heating the fuel cladding to the regulatory limit and then quenching the material to examine the ductility and strength remaining. The staff is not aware of any testing done to examine the subsequent reheating of fuel to the 10 CFR 50.46 limit with a subsequent second quench (either slow or fast). Situations showing a localized moderate (on the order of 100 to 200 degrees C) PCT increase could be considered as acceptably low if properly justified. The staff would expect any such justifications to consider degradation of the cladding oxide layer, hydrogen embrittlement of the cladding, and accumulated diffusion of oxygen within the cladding microstructure. Duration of time at elevated temperature and peak temperature experienced by the clad should also be limited and justified. The staff would expect the justifications to be supported by test data, where possible.
>
> The submitted information would form the basis for any determination that the calculated core temperatures remain acceptably low as required by the rule. The second clause of 10 CFR 50.46(b)(5), "decay heat removed for the extended period of time required by the long lived radioactivity remaining in the core" was not identified as an issue needing clarification in

Westinghouse letter LTR-NRC-06-46, or at the meeting with Westinghouse on April 12, 2006. The Westinghouse representatives in attendance at the meeting agreed with the staff on the definition of this clause and had no questions on its meaning. Based on this, the staff expects that this clause needs no further clarification.

With respect to Item 2, the NRC response notes the following;

For GSI-191, the 30-day criterion was originally intended for evaluation of operability of equipment. For analysis of core cooling following debris ingestion into the RV, the staff believes that an adequate post-LOCA evaluation duration would be demonstrated when bulk and local temperatures are shown to be stable or continuously decreasing with the additional assurance that any debris entrained in the cooling water supply would not be capable of affecting the stable heat removal mechanism due to sump strainer clogging or downstream effects.

A.4 GSI-191 LONG-TERM CORE COOLING ACCEPTANCE BASES

The LTCC acceptance bases defined for GSI-191 are listed below. These acceptance bases are applied after the initial quench of the core and consistent with the LTCC requirements stated in 10 CFR 50.46 (b)(4) and 10 CFR 50.46 (b)(5). They do not represent, nor are they intended to be new or additional LTCC requirements. These acceptance bases provide for demonstrating that local temperatures in the core are stable or continuously decreasing and that debris entrained in the cooling water supply will not affect decay heat removal.

- Decay Heat Removal/Fuel Clad Oxidation
  Maximum cladding temperatures maintained during periods when the core is covered will not exceed a core average clad temperature of 800°F.

  Cladding temperatures at or below 800°F maintain the clad within the temperature range where additional corrosion and hydrogen pickup over a 30 day period will not have a significant effect on cladding properties. At temperatures greater than 800°F, there are occurrences of rapid nodular corrosion and higher hydrogen pickup rates that can reduce cladding mechanical performance. Long-term autoclave testing has been performed to demonstrate that no significant degradation in cladding mechanical properties would be expected due to a localized hot spot. This information is proprietary to the fuel vendors but could be made available upon request. This testing demonstrated that the increase in oxide thickness and hydrogen loading was limited at temperatures of less than 800°F for periods of 30 days. With limited corrosion and hydrogen pickup, the impact on cladding mechanical performance is not significant. Therefore no significant degradation in cladding properties would occur due to 30-day exposure at 800°F, and there would not be any adverse impact on core coolability. Based on the autoclave results, the data is sufficient to justify a maximum clad temperature of 800°F as an LTCC acceptance basis.

- Deposition Thickness
  For current fuel designs, regardless of vendor, the minimum clearance between two adjacent fuel rods, including an allowance for the spacer grid thickness, is greater than 100 mils. Therefore, a 50-mil debris thickness on a single fuel rod is maximum deposition to preclude touching of the deposition of two adjacent fuel rods with the same deposition. The 50 mil thickness is the maximum acceptable deposition thickness before bridging of adjacent fuel rods by debris is predicted to occur.

A.5 DISCUSSION

Decay Heat Removal and Required Coolant Flow

LOCA ECCS analyses consider the LOCA transient behavior to the point in time at which fuel temperatures are decreasing, the mixture level in the core is rising, and the peak clad temperature has been captured. At the start of sump recirculation, the core has been quenched and is covered by a two-phase liquid/steam mixture and/or single phase liquid. After the start of sump recirculation, LTCC is demonstrated by showing that there is sufficient flow to replace core boil-off, thus keeping the core covered and preventing additional fuel clad heat-up.

For some post-LOCA scenarios, precipitation of boric acid in the core region is prevented by core flow above the core boil-off rate. In these cases, the required core flow to provide for boric acid dilution is usually represented as a multiplier on core flow.

Flow rates required to match boil-off become small quickly following the postulated event. The required flow rate to match boil-off for a large Westinghouse four-loop PWR is taken from the emergency operating procedures (EOP) and shown in Figure A-1. While the actual values are dependent on the initial core power level, these values are representative of the PWR fleet. Within four hours following a postulated LOCA, the required flow to match boil-off is about 250 gallons per minute. At 10 hours, the flow required to match boil-off is about 200 gallons per minute, and at 30 hours, the flow required to match boil-off is about 150 gallons per minute.

The PWROG has used multiple methods to demonstrate that the minimal flow required to remove core decay can be maintained. Testing of a FA in the presence of debris has established the maximum mass of fiber that would not cause total blockage of the flow into the FA (References A-3, A-4 and A-5). Analyses with large system codes (Sections 3 and 4) show that substantial blockage at the core inlet can be tolerated and still maintain the necessary flow rate to maintain acceptable low fuel cladding temperatures.

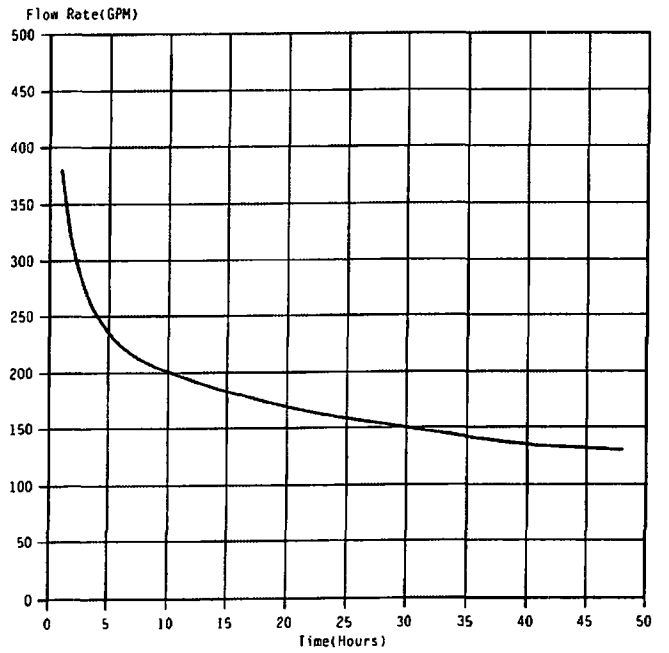


Figure A-1 Boil-off Curve for a Westinghouse Four-loop PWR

Spacing Between Fuel Rods

The minimum clearance between two adjacent fuel rods, including an allowance for the spacer grid thickness, is greater than 100 mils. Therefore, a 50-mil debris thickness on a single fuel rod is maximum deposition to preclude touching of the deposition of two adjacent fuel rods with the same deposition. The 50 mil thickness is the maximum acceptable deposition thickness before bridging of adjacent fuel rods by debris is predicted to occur. The 50 mils of solid precipitation described here include the clad oxide, crud layer and debris deposition.

The example chemical product deposition calculation documented in Appendix E was performed with inputs intended to maximize chemical deposition. That deposition calculated for the sample case was less than 30 mils. Thus, although the chemical deposition of fuel is a plant-specific calculation, plants are not expected to calculate deposition thicknesses in excess of 30 mils.

The formation of a chemical deposition layer followed by the collection of fibrous debris in the remaining open channel will not challenge the cooling of the clad. As was shown in the response to RAI #15, Appendix H, the effective thermal conductivity of a fibrous debris bed is at least 5 times greater than the minimum thermal conductivity of 0.1 Btu/(hr-ft-°F) used in the cladding heat up calculations in Appendix D and with LOCADM in Appendix E.

Thus, for chemical deposition, the range of cladding heat up calculations between spacer grids considering up to a 50 mil buildup presented in Tables D-1 and D-2 of Appendix D are bounding. The maximum calculated clad temperature listed in these tables for up to 50 mils of deposition is below 800°F.

Therefore, a maximum debris layer buildup of 50 mils is an appropriate acceptance criterion for the span between grids.

Spacing Between Fuel Rods and Grids

For current fuel designs, the minimum clearance between the cladding and the spacer grid is about 40 mils. This occurs where the springs and dimples of the grid contact the fuel rod. The maximum clearance between the cladding and the spacer grid occurs along the diagonal of the of a grid cell and is about 110 mils. Thus, if a spacer grid were to become completely filled by either a fibrous debris bed or a chemical deposition, the radial thickness of the debris on the clad would vary from about 40 mils to about 110 mils about the circumference of a fuel rod.

Calculations documented in Appendix C assess the clad temperature under a debris bed in a single spacer grid/fuel rod configuration. The results of these calculations are summarized in Table C-7 of Appendix C. To use these results to assess a maximum clad temperature under worst case debris or chemical deposition under a spacer grid/fuel rod configuration, the following assumptions are made:

- A uniform debris layer thickness of 110 mils is assumed on the cladding.
- The debris layer is assigned the conservative effective thermal conductivity for a fibrous debris bed or chemical deposition layer of 0.1 Btu/(hr-ft-°F).

Under these limiting assumptions, the clad temperature is estimated to be less than 738°F by extrapolating the calculated clad temperatures listed in Table C-7 for the effective thermal conductivity of 0.1 Btu/(hr-ft-°F). This temperature value is an extremely conservative estimate of the clad temperature under worst case debris or chemical deposition beneath a spacer grid/fuel rod configuration for the following reasons:

- A conservatively small value of conduction through the debris bed is used. (As was shown in the response to RAI #15, Appendix H, the effective thermal conductivity of a fibrous debris bed is at least 5 times greater than the minimum thermal conductivity of 0.1 Btu/(hr-ft-°F) used in the cladding heat up calculations in Appendix C and with LOCADM in Appendix E.)
- The calculation does not account for circumferential heat transfer about the debris bed which would form in the spacer grid between the dimples and springs and the corners of the spacer grid.
- In the case of a fibrous debris bed, convection of heat by the flow of coolant through the debris bed is neglected (The ability of coolant to pass through a fibrous and particulate debris bed under PWR LTCC flow conditions was demonstrated in the response by testing).

The formation of a deposition, either fibrous or chemical, under a clad and followed by the collection of fibrous debris in the remaining open channel will not challenge the cooling of the clad.

Based on observations from testing of fibrous debris collection on debris capturing grids, a complete blockage of a spacer grid with fibrous and particulate debris will not occur for the limits of fibrous debris ingestion reported in Section 10. The test data shows that, for the allowed fibrous, particulate, and chemical precipitate debris loads, flow through the resulting debris bed is maintained.

Industry Experience with At-Power Clad Oxidation

As noted previously, long-term autoclave testing has been performed to demonstrate that no significant degradation in cladding mechanical properties would be expected due to a localized hot spot. This testing demonstrated that the increase in oxide thickness and hydrogen loading was limited at temperatures of less than 800°F for periods of 30 days. It is noted that a there was an at-power experience at the Calvert Cliffs Nuclear Power Plant during the late 1970's in which clad temperatures increased to 800°F so that operation for several weeks caused the oxide layer to build on the cladding. This at-power operating experience is not applicable to the post-LOCA LTCC conditions the acceptance basis addresses as discussed below.

- The core conditions that resulted in the clad oxidation at Calvert Cliffs during the late 1970's would not exist in the core post-LOCA. At-power clad corrosion is driven by temperature, fast neutron flux, and thermal feedback through an oxide layer. During long term cooling post-LOCA, the fast neutron flux is negligible and the heat flux is low. Thus, for post-LOCA conditions, only the temperature is directly applicable to corrosion and autoclave data is more representative of the temperature-driven corrosion that would be experienced by cladding. Evaluation of autoclave data for cladding at temperatures of 800°F and below shows only small increases in the corrosion thickness and hydrogen loading compared to the post-LOCA transient conditions immediately following the postulated break that occur prior to long-term cooling.

- Local increases in corrosion due to local hot spots will not impact long term cooling. The impact of corrosion on the clad material properties is small and the heat load continues to decrease with time. The 17 percent equivalent clad reacted (ECR) criteria apply to the LOCA event only. If the local conditions immediately post-LOCA were close to the 17 percent ECR limit (pre-transient corrosion and transient ECR), then the small amount of additional corrosion from a hot spot which resulted in approaching 800°F for 30 days could reach or marginally exceed 17 percent ECR. However, based on the sample deposition calculation, the conservative core blockage calculations and the parametric clad heat-up calculations presented in Section 4, cladding temperatures approaching 800°F for post-LOCA LTCC are not expected.

- Also, the peak ECR region on the rod is not expected to be the same region where a local hot spot would occur. Local hot spots would be expected to occur lower in the core and at or just below a spacer grid. Pre-transient corrosion is suppressed at the spacer grid locations.

- In addition, much of the reduction in ductility from high temperature oxidation (> 1832°F) is due to oxygen diffusion ahead of the oxide layer. At temperatures of < 930°F, there is no observation of oxygen diffusion ahead of the oxide layer.

In summary, the PWR industry at-power experience with cladding oxidation is not applicable to the post-LOCA LTCC environment.

Impacts of Local Hot Spots

The ingestion of debris through the sump strainers and the potential chemical effects from the generation of chemical by-products from the reaction of containment material and coolant following a LOCA create

the possibility of local "hot spots" occurring in the reactor core. Based on the designs and flow hole sizes of the replacement sump strainers, and test data obtained using those designs, the passing of debris in sufficient quantity or size to result in a hot spot is considered small and will not challenge overall LTCC of the fuel. However, the consequences of the formation of hot spots should be evaluated.

Local "hot spots" could occur as a result of debris catching and accumulating on the various nozzles and grids of an FA or by chemical by-products plating out on parts of the fuel. The potential effects of these local "hot spots" can be assessed against the ECCS criteria (10 CFR 50.46) and for their potential impact on the health and safety of the public above those considered for a LOCA.

The current regulatory criteria for LTCC is identified in 10 CFR 50.46 (b)(5), "Long-term cooling. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core."

- "...temperature shall be maintained at an acceptably low value" is interpreted to mean less than 800°F (427°C). (Note: A value of 800°F is cited as the maximum acceptable clad temperature to be consistent with the acceptance basis presented in Section A.4, GSI-191 LTCC Acceptance Basis.)

- "...extended period of time" is interpreted to mean showing that the local temperatures are stable or continuously decreasing and that debris entrained in the cooling water supply will not affect decay heat removal.

As noted previously, based on the testing of replacement sump strainers, the passing of debris in sufficient quantity or size to result in a hot spot is considered small and will not challenge overall LTCC of the fuel. However, assuming a "hot spot" occurs during LTCC following a LOCA, the following should be considered:

- For dose considerations, all fuel is considered to have failed. Therefore, "hot spots" do not contribute additional dose.

- Given a sustainable quench and the replacement of boil-off, any fuel cladding "hot spot" would remain underwater.

- Transitioning the ECCS from a clean water source to recirculation from the reactor containment building sump is addressed under the current licensing basis of PWRs. It is also noted that, during HL switchover or, for B&W plants, the establishment of a core flushing flow, there is no interruption of coolant to the core. Therefore, there is no clad heat-up transient during this operation.

  Once the transition of the ECCS from a clean water source to recirculation of coolant from the reactor containment building sump has occurred, there is limited interruption (termination) of coolant flow to the core due to system realignments such as initiation of HL recirculation. For plants that have a reduction in flow associated with systems realignments, the supplied flow remains above the core boil-off rate and will not result in a reheat of the cladding. Therefore, for

long term cooling, the appropriate acceptance basis for clad temperature is 800°F. This acceptance basis is based on the results of long term autoclave testing that used clean water at temperatures up to about 700°F, and steam at temperatures ranging from 700°F to 900°F.

- A coolable core geometry must be maintained during LTCC.
- The fuel will not be reused.

The source of heat post-LOCA is from decay heat in the fuel rod. This source is limited to the fuel in the rod and decreases with time. "Hot spots" can arise only if the local flow is severely restricted. Local temperature increases would be mitigated by the boil off in the region. Also, the grids act as a radiator and there will be conductive heat removal axially along the fuel rod. If quench is sustained and the boil-off is replaced, the ability of the "hot spot" to obtain significant temperatures (approaching 2200°F (1204°C)) is severely limited.

However, should localized temperatures at a "hot spot" reach sufficient levels to further degrade or damage the fuel cladding, the impact of the temperature increase on LTCC is minimal. Since all rods are assumed to have failed during the LOCA, no additional impact to dose is expected. In addition, if a buildup of chemical deposits or debris were to form such that the buildup would cause an increase in cladding temperature, there are two possible outcomes:

1. The deposit goes back into solution as the cladding temperature increases and the "hot spot" is subsequently cooled.
2. The deposit is fixed and remains on the surface (it does not go back into solution) and the "hot spot" remains.

For the first case, the "hot spot" is self-limiting. For the second case, if the temperature at the "hot spot" were to increase to a level that damage to the fuel cladding would result, the remainder of the fuel rods, fuel skeleton, and other fuel assemblies, would serve to contain the fuel and maintain structural spacing to provide geometry for LTCC. Thus, the fixed deposit would not further impact the coolability of the fuel.

The ability to maintain an average fuel clad temperature below 2200°F during LTCC can be demonstrated. Regardless of the actual temperatures obtained at a localized "hot spot" and the localized damage to fuel cladding during LTCC, the requirements of 10 CFR50.46 will continue to be met.

To summarize, given that the fuel will not be reused following a LOCA, localized "hot spots" during LTCC do not increase the risk to the health and safety of the public and does not jeopardize core coolability as long as the core remains covered and boil-off is replaced.

Impacts of Boric Acid Concentration

The impacts of boric acid concentration will be addressed in a separate PWROG program.

A.6 SUMMARY

The LTCC criteria identified here and proposed for use to address GSI-191 are consistent with the requirements of 10 CFR 50.46. Furthermore, the criteria are conservative and, when used in conjunction with engineering calculations performed considering GSI-191 concerns, provide reasonable assurance that LTCC is successfully maintained.

LTCC bases applicable to GSI-191 have been defined based on the clarification offered by the NRC (Reference A-2). They are summarized as follows:

1. The cladding temperature during recirculation from the containment sump will not exceed 800°F.

2. The deposition of debris and/or chemical precipitates will not exceed 50 mils on any fuel rod.

Properly applied, these bases will facilitate the demonstration of acceptable core cooling following a postulated large break LOCA.

A.7 REFERENCES

A-1 LTR-NRC-06-46, "Requested NRC Action from Meeting with Westinghouse on April 12, 2006; Acceptance Criteria for Long-Term Core Cooling following Quenching and Reflooding of the Core; PWR Containment Sump Downstream Effects Resolution of GSI-191," July 14, 2006.

A-2 Nuclear Regulatory Commission Response to Westinghouse Letter LTR-NRC-06-46 Dated July 14, 2006, Regarding Pressurized Water Reactor (PWR) Containment Sump Downstream Effects," August 16, 2006, ADAMS No. ML0620704511.

A-3 AREVA Document 51-9102685-000, "GSI-191 FA Test Report for PWROG," March 2009.

A-4 WCAP-17057-P, Revision 1, "GSI-191 Fuel Assembly Test Report for PWROG," September 2011.

A-5 AREVA Document 51-9170258-000, "GSI-191 FA Test Report for PWROG – Low Particulate-to-Fiber Ratio Tests," October 2011.

APPENDIX B
EVALUATION OF BLOCKAGE AT THE CORE INLET

B.1 OBJECTIVE

The purpose of this task is to demonstrate that sufficient LTCC is achieved to satisfy the requirements of 10 CFR 50.46 considering the effects of debris ingested into the RCS and core during post-accident operation when safety systems are realigned to recirculate inventory from the containment sump. The flow at the core inlet could be suppressed due to the build-up of sump debris at the lower core plate and bottom nozzle. To show LTCC would be maintained in this situation WC/T simulations were run blocking the inlet at the core entrance with an increased k-factor. This calculation provides additional "defense in depth" to the FA testing to assure that LTCC will be maintained.

B.2 APPROACH

To evaluate the effects of blockage at the core inlet, the dimensionless friction factor (C_D) was ramped at the core inlet to simulate blockage due to debris buildup. A modified version of WC/T was created to allow the ramping of the friction factor at the core inlet. Code simulations were run to the beginning of recirculation (conservatively assumed to be 20 minutes) at which point the ramping of the friction factor took place over 30 seconds. Note that the core inlet flow blockage occurring in 30 seconds from the start of recirculation is non-physical and was modeled in such a manner to perform a bounding calculation. After the core inlet resistance was increased, the code simulations were run out to 40 minutes to show the flow rate supplied to the core would be sufficient to remove decay heat and maintain LTCC.

B.3 MODEL DESCRIPTION AND ASSUMPTIONS

The core inlet blockage simulations were meant to bound the U.S. PWR fleet. To ensure a bounding calculation, the limiting break type and the limiting vessel design were taken into consideration before selecting a plant model for the simulation.

B.3.1 Plant Type Selection Criteria

The selection of the limiting break combines the conditions from a double-ended CL and a double-ended HL break to create a bounding scenario. During a double-ended CL break, the ECCS liquid will spill into containment, decreasing the driving head of core flow to a minimum. However, because of the low flow rate, a slow debris build-up at the core inlet ensues, which is non-limiting. During a double-ended HL break no spilling of ECCS liquid occurs, therefore an additional driving head from the build-up of liquid level in the downcomer and in the steam generator tubes to the spillover elevation is present. However, the higher flow rates also result in faster debris build-up. To create the worst possible scenario, the limiting break case will be a double-ended CL break (i.e., limiting driving head at the core inlet) combined with faster debris build-up time that occurs for a high flow HL break.

The limiting vessel design was chosen based upon which core design would be most limiting under the condition of core inlet flow blockage. Three general vessel designs were considered: designed B/B upflow, converted B/B upflow, and B/B downflow. Designed B/B upflow is the least limiting due to the

numerous large pressure relief holes in the baffle wall. The relief holes allow flow to bypass a blocked core inlet but still enter the core. Converted upflow plants are considered more limiting than designed upflow plants due to the absence of the pressure relief holes, such that only limited bypass flow may enter near the top of the core. The most limiting design is downflow plants since the only means for the flow to enter the core is through the lower core plate.

Other PWR vessel designs were considered including B&W and CE designs. B&W plants are similar to the Westinghouse upflow design with the numerous pressure relief holes in the baffle wall. Therefore, the design is non-limiting with respect to core inlet flow. Other differences in the B&W design, such as the RVVV, were concluded to have no impact on this issue. CE plants are similar to the Westinghouse converted upflow plant in that they have no pressure relief holes, but limited flow may enter near the top of the core. Therefore the design is non-limiting with respect to core inlet flow. CE plant designs lack RHR heat exchangers, and after switchover to recirculation high pressure safety injection flow goes directly from sump to the RCS. The higher ECCS injection temperature is considered to have a small effect on core inlet blockage simulations. Prior to recirculation, termination of extensive downcomer boiling and cooling of vessel internals has already occurred. Therefore, the increase in injection temperature should not lead to boiling and only a small decrease in flow rate supplied to the core will result.

Therefore, it is concluded that the Westinghouse downflow design is bounding for this analysis.

B.3.2 Description of and Basis for Model Inputs

A plant with an existing WC/T model, downflow plant configuration, and high core power density is desired for the core blockage simulations. A three-loop downflow model plant rated at 2900 MWt was chosen. The power shape of the plant's BELOCA reference transient used for these simulations is shown in Figure B-1. (Figures use squares to designate vertical flow paths and circles to designate horizontal flow paths.)

The axial power shape uses a high enthalpy rise peaking factor ($F_{\Delta H}$ = 1.73), a skewed to the top power distribution (13 percent axial offset), and a relatively high total peak factor (F_Q = 2.3). The top-skewed power shape shown in Figure B-1 is limiting compared to base load or bottom skewed power shapes due to the longer time for the quench front to approach the elevations with the highest power, and its susceptibility to heatup if the core becomes uncovered due to inlet blockage. The total peaking factor is on the order of 20 percent higher than a normal base load power shape would exhibit. F_Q higher than 2.3 will only occur in rare transient conditions, where such an F_Q would be temporary and not indicative of the long-term axial decay heat power distribution of interest for LTCC. Therefore, the inputs represent reasonably bounding values for the PWR fleet.

Figure B-1 represents the axial power shape and Table B-1 displays the radial power distribution of the modeled plant.

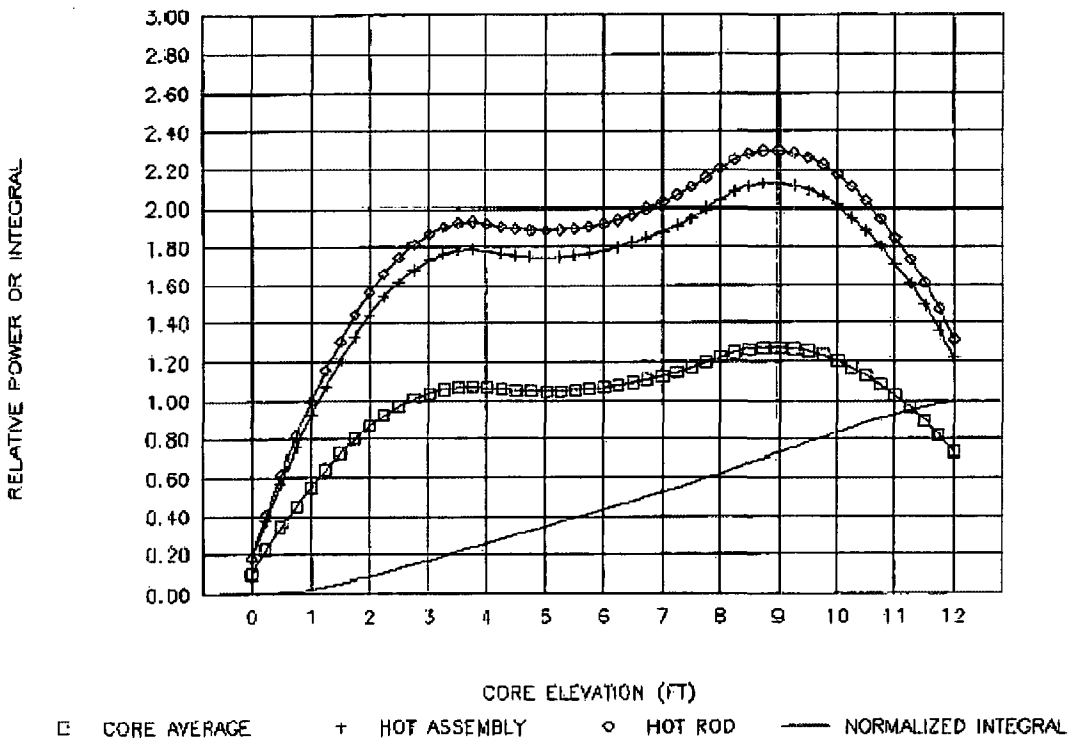


Figure B-1 Plant Transient Power Shape

Table B-1 Core Channel Radial Power Distribution

Channel Description	Channel Number	Normalized Power	Number of Assemblies
Hot Assembly Channel (HA)	13	1.66	1
Guide Tube Channel (GT)	12	1.17	53
Non-Guide Tube Channel (AVG)	11	1.17	75
Low Power Periphery Channel (LP)	10	0.20	28

The radial power distribution in the core is flat other than in the periphery assemblies and the hot assembly. The hot assembly power is conservatively modeled to a high normalized power of 1.66.

Additional information on the plant chosen for the core inlet blockage simulations is given in the schematics and noding diagrams shown in Figures B-2 through B-5.

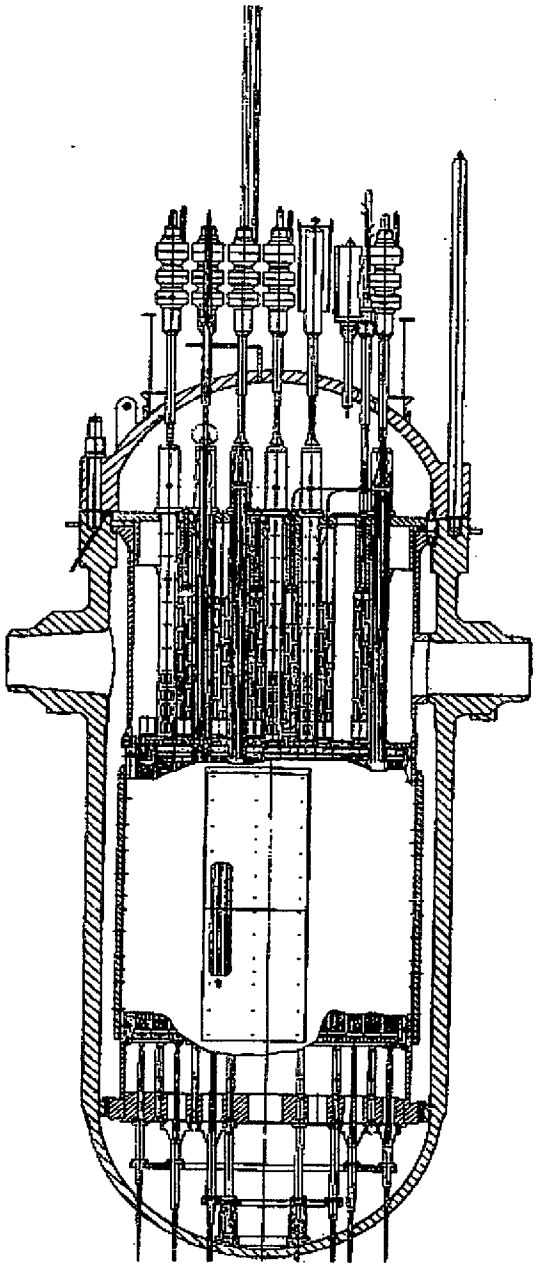


Figure B-2 Plant Vessel Profile

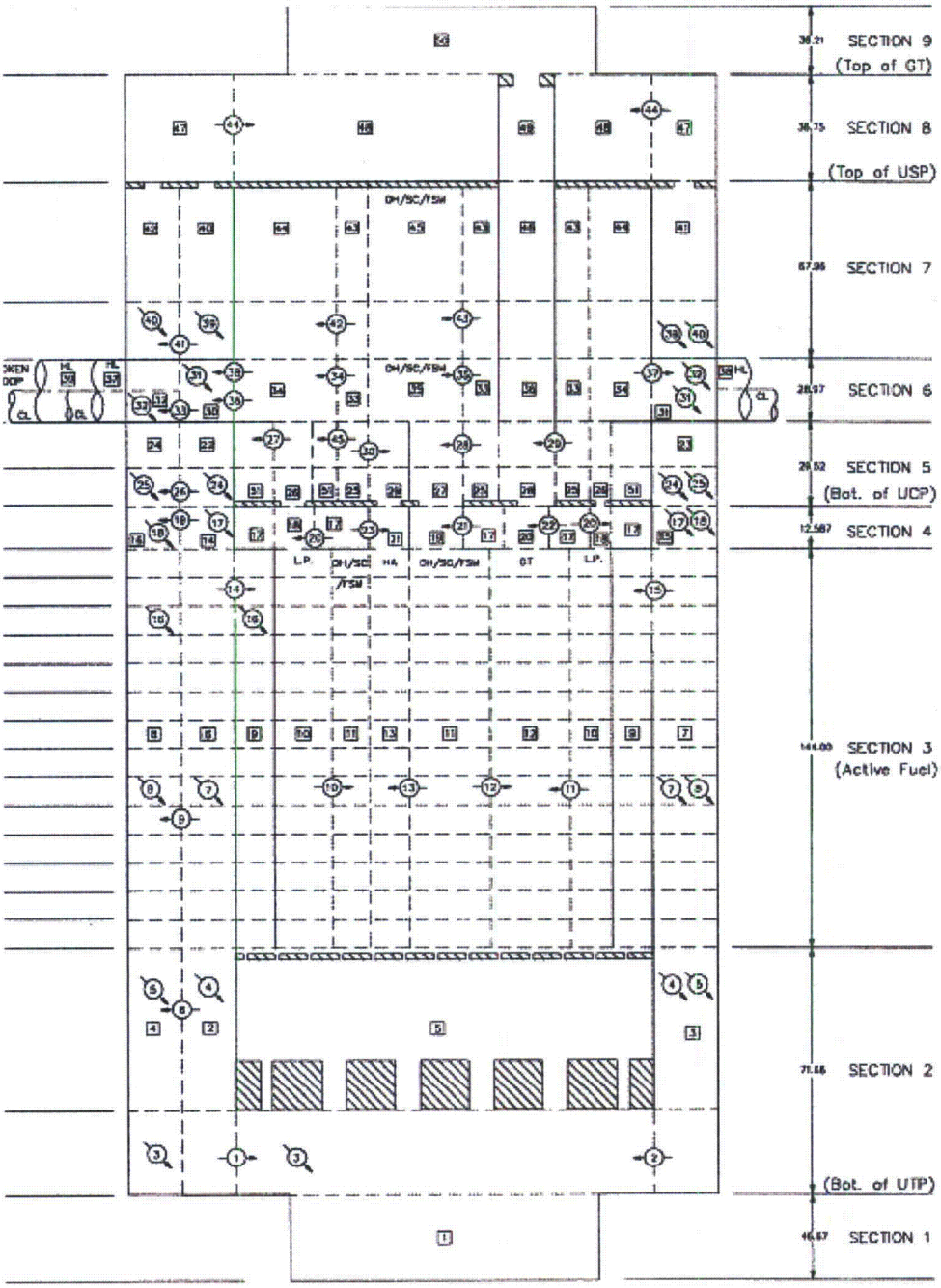
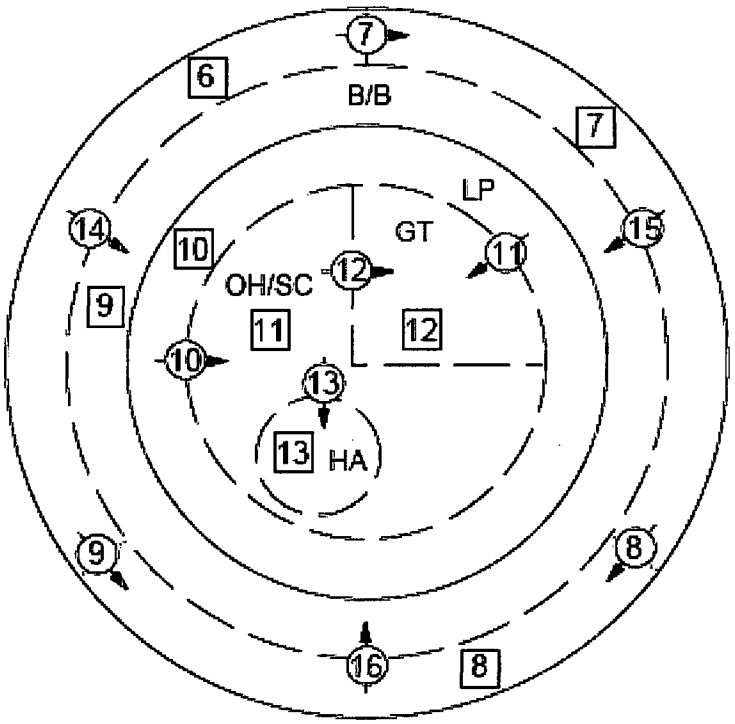


Figure B-3 Plant Vessel Model Noding Diagram



□ – channel
♂ – gap

Figure B-4 Plant Core Channel Modeling

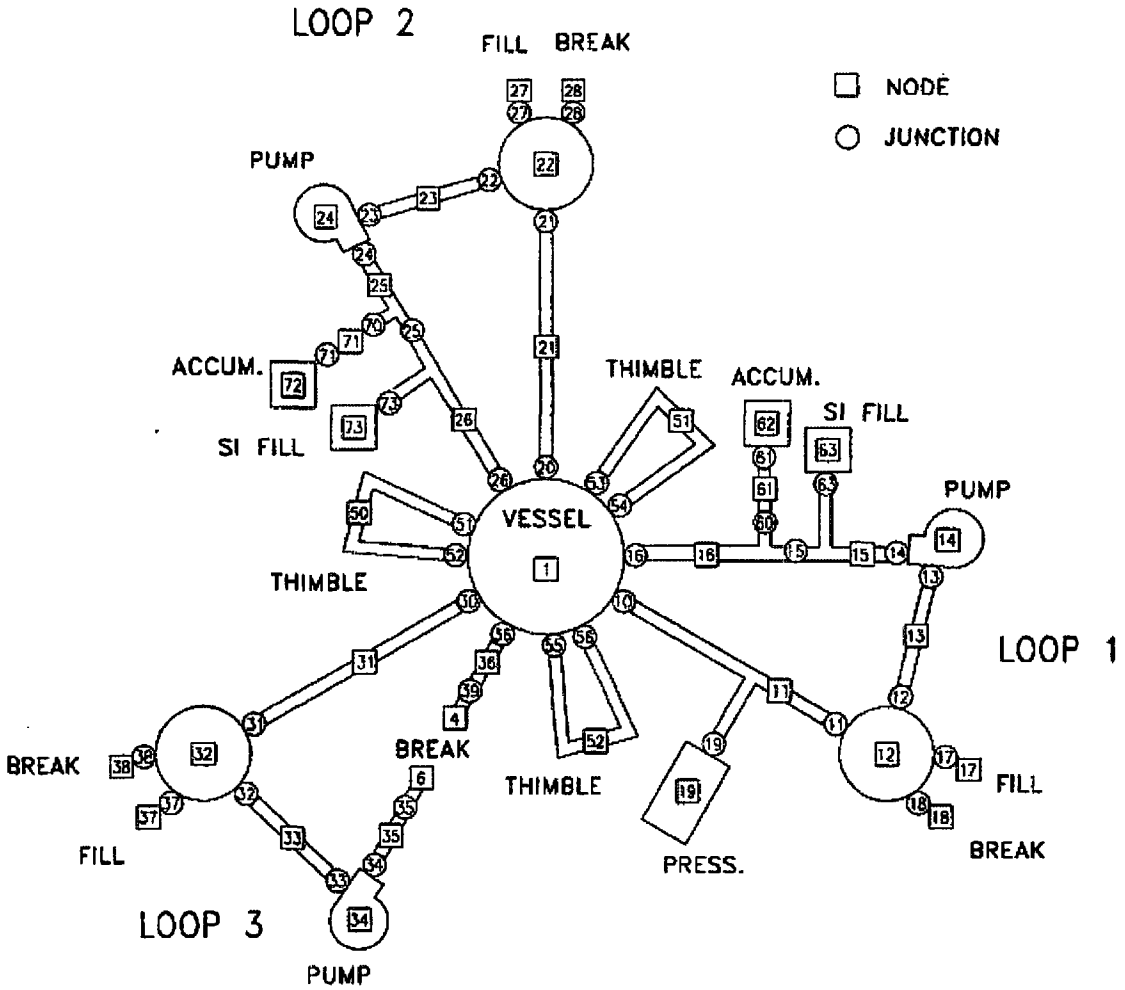


Figure B-5 Plant Loop Model Noding Diagram

The modified WC/T code version and the plant model shown in Figure B-3 were used to study the effects of core blockage on LTCC. The calculations modeled the ramp-up of the dimensionless loss coefficient (C_D) at the core inlet and the increase of the temperature of the ECCS injection water at 20 minutes, which is the modeled beginning of sump recirculation. It should be noted that the increase in the value of C_D, simulating the debris build-up over 30 seconds, is non-physical and was modeled to occur over such a short time period to perform a bounding calculation. The increase in the injection temperature represents a representative RHR heat exchanger outlet temperature following switchover to sump recirculation. The temperature of the injected water was set to be 190°F, which is typical for Westinghouse designs and is expected to bound B&W plant designs. While CE plants may have slightly higher temperatures, the effect on the transient will be minimal. Following preliminary test calculations, it was decided to ramp C_D to extremely large values to completely block the chosen core channels to simulate a percentage of core blockage.

Initially, two simulations were run with no changes to the standard noding but with different amounts of simulated core blockage,. The first case modeled 82% core flow blockage by ramping the value for C_D up to 10^9 in all core channels except for the Lower Power (LP) periphery channel (representing 28 of 157 assemblies). Figure B-6 displays the core channel modeling used for Case 1. Channels 11, 12, and 13 are crossed out to represent total blockage at the inlet of the channels. For this modeling approach flow will only enter the core through channel 10.

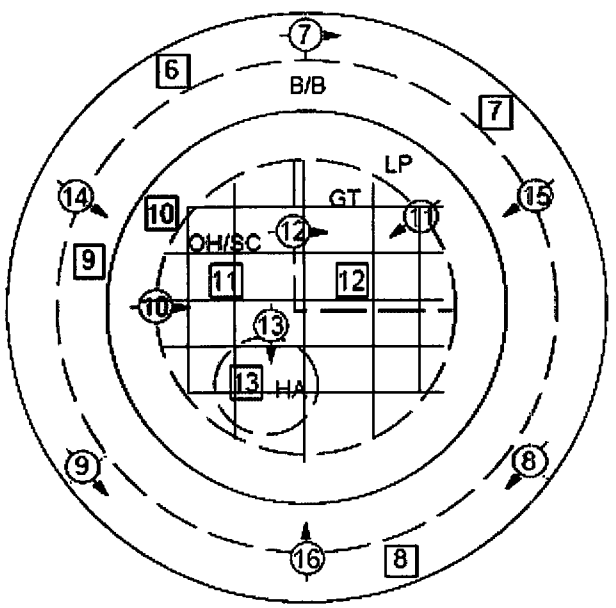


Figure B-6 Case 1 Core Blockage Modeling Approach

The second case modeled 99.4% core flow blockage by ramping the value for C_D up to 10^9 in all core channels except for the Hot Assembly (HA) channel. Figure B-7 displays the core channel modeling used for Case 2. Channels 10, 11, and 12 are crossed out to represent total blockage at the inlet of the channels. For this modeling approach flow will only enter the core through channel 13.

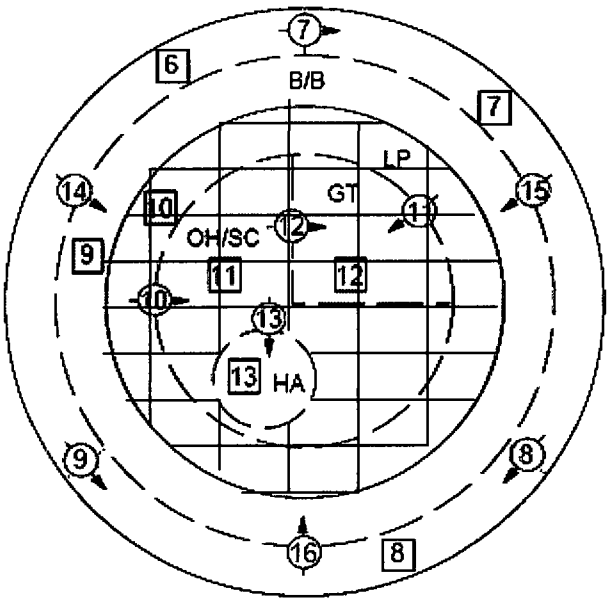


Figure B-7 Case 2 Core Blockage Modeling Approach

The Case 2 modeling approach of leaving the hot assembly unblocked is justified as acceptable due to core cross-flow. The top-skewed power shape used in the simulations creates limiting core conditions in the top half of the core. The flow supplied to the core distributes throughout the core channels, and by the higher power elevations, the flow will be well allocated to all the core channels. This statement can be supported by examining the void fraction of the LP periphery channel and the HA channel. Figure B-8 plots the LP void fraction and the HA void fraction from 2000 to 2400 seconds, near the top of the core. The figure shows the void fraction in the HA channel reaching higher values, demonstrating much of the flow exits the HA channel via cross-flow through the core and suggesting no non-conservatism due to the modeling approach.

The containment back pressure was modeled by a containment pressure vs. time table input for each of the broken loop CL break components. The containment backpressures used in both cases were based on the existing pressure vs. time tables used in the BELOCA analysis. The BELOCA table was extrapolated down to atmospheric pressure and held there. As a result, the containment pressure is assumed to be at atmospheric conditions by switchover to sump recirculation.

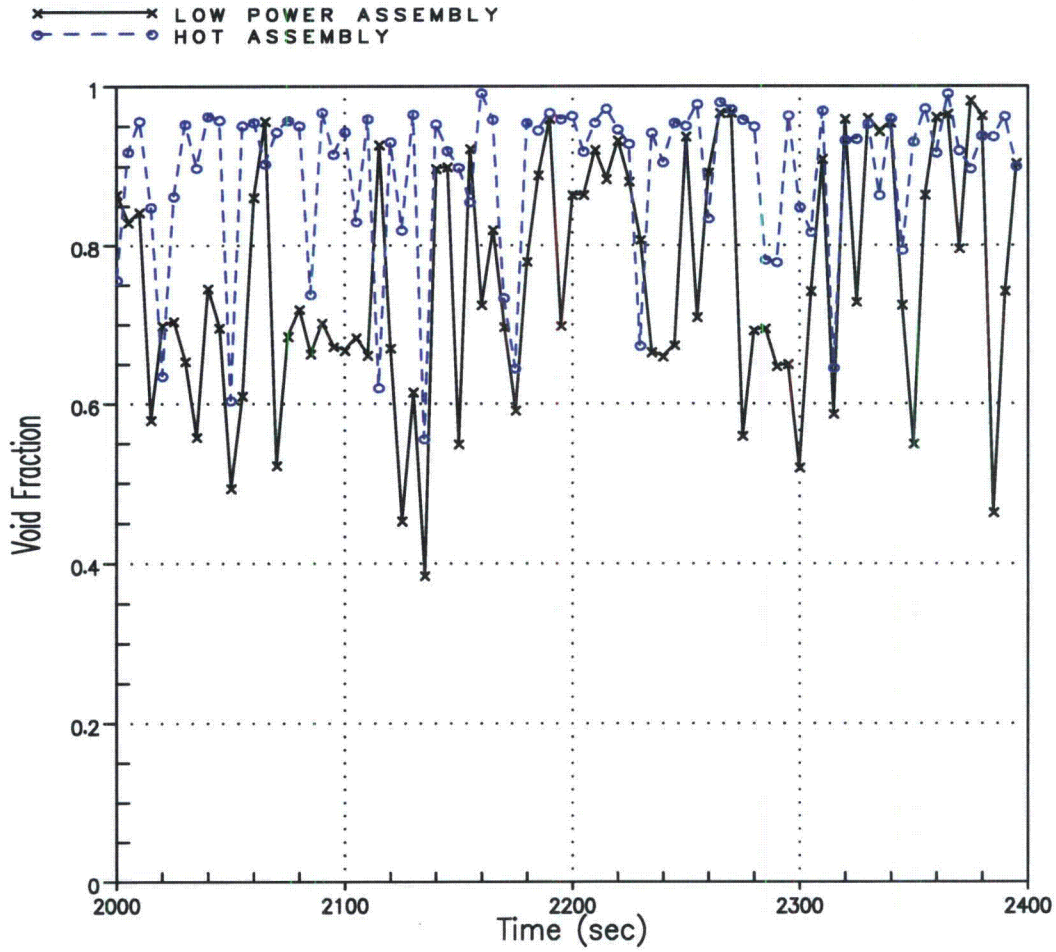


Figure B-8 Low Power Channel and HA Channel Void Fraction, Case 2 - Unblocked HA Test

The additional effects of sub-atmospheric containment plants on the transient have been considered to show the calculations performed apply to the entire US PWR fleet. The containment pressure and pressure drop through the RCS loops have a direct affect on the boil-off rate for a postulated CL break. The decrease in the containment pressure for a subatmospheric plant could cause an increase in loop pressure drop, which could lead to higher core exit pressures and higher boil-off rates. The increased core exit pressure corresponds to higher boil-off rates through the inversely proportional relationship between the boil-off rate ($\omega_{boiloff}$) and the latent heat of vaporization, i.e.,

$$\omega_{\text{boiloff}} = \frac{Q_{DH}}{h_{fg}} \tag{B-1}$$

The relationship between the pressure drop and the containment break pressure, i.e., steam density, is shown in Darcy's equation.

$$\Delta P_{loop} = \frac{k}{A^2} \frac{\omega_{boiloff}^2}{288 \cdot \rho_g \cdot g_c} \tag{B-2}$$

Hand calculations were performed to estimate the loop pressure drop using subatmospheric containment pressures to examine whether the core exit pressure would increase. The hand calculations compared the ratio of the loop pressure drop between atmospheric and subatmospheric containment conditions. Combining Equations (B-1) and (B-2) to create a pressure drop ratio yields,

$$\frac{\Delta P_{sub}}{\Delta P_{dry}} = \left(\frac{h_{fg\_dry}}{h_{fg\_sub}} \right)^2 \left(\frac{\rho_{g\_dry}}{\rho_{g\_sub}} \right) \tag{B-3}$$

Loop pressure drop values were taken from calculations performed for atmospheric containment simulations. The largest loop pressure drop after switchover to sump recirculation was found to be approximately be (18.0–14.7) = 3.3 psia. First, assuming the pressure drop (ΔP) would be the same for the subatmospheric containment conditions, iterative calculations were performed to calculate the loop pressure drop for subatmospheric containment conditions. The calculation was performed twice using containment pressures of 10 psia and 12 psia to address potential non-linearity of the density changes in the pressure range of interest. The calculations found that the loop ΔP would increase, however, the core exit pressure would be lower (tables representing the values used in the final iterations are shown in Tables B-2 and B-3 below). Therefore, the subatmospheric containment pressure plant designs are bounded by the atmospheric containment simulations performed to examine the effects of core inlet blockage.

The core inlet blockage simulations were performed using 'typical' Westinghouse RHR heat exchanger outlet temperature for sump ECCS injection (190°F). It has been acknowledged that CE plant designs do not have RHR heat exchangers, and after switchover to recirculation High Pressure Safety Injection flow goes directly from sump to the RCS. The increase in sump ECCS injection temperature is assessed to be a non-factor in core inlet blockage simulations. The flow rate required to replace boil-off at 20 minutes is less than 60 lbm/sec. The highest injection temperature of concern is estimated to be 250°F, i.e., an increase of 60°F. Prior to recirculation, termination of extensive downcomer boiling and cooling of vessel internals has already occurred, therefore the increase in injection temperature should not lead to boiling and only a small decrease in flow rate supplied to the core will ensue due to the density effects. It is therefore assessed that an increase of 60°F to the ECCS injection should not affect core inlet blockage simulations.

Table B-2 Subatmospheric Loop Pressure Drop for Containment Pressure=10 psia					
			Second Iteration		
Pressure (psia)	**Density (lbm/ft^3)**	**Enthalpy (Btu/lbm)**	**Density (lbm/ft^3)**	**Enthalpy (Btu/lbm)**	**h_{fg} (Btu/lbm)**
Sub cont.					
14.4	59.854	179.25	0.036607	1150.7	971.45
10	60.281	161.36	0.026027	1143.8	982.44
average	60.0675	170.305	0.031317	1147.25	976.945
Dry					
18	59.565	190.79	0.045103	1154.9	964.11
14.7	59.829	180.3	0.03732	1151	970.7
average	59.697	185.545	0.041212	1152.95	967.405
Pressure Ratio					
1.290371					
New P core (psia)	**Sub-atm DP (psia)**				
14.3	4.3				
New P core less than 18 psia					

Table B-3 Subatmospheric Loop Pressure Drop for Containment Pressure=12 psia					
			Second Iteration		
Pressure (psia)	**Density (lbm/ft^3)**	**Enthalpy (Btu/lbm)**	**Density (lbm/ft^3)**	**Enthalpy (Btu/lbm)**	**h_{fg} (Btu/lbm)**
Sub cont.					
15.9	59.729	184.32	0.040162	1152.5	968.18
12	60.074	170.16	0.030868	1147.2	977.04
average	59.9015	177.24	0.035515	1149.85	972.61
Dry					
18	59.565	190.79	0.045103	1154.9	964.11
14.7	59.829	180.3	0.03732	1151	970.7
average	59.697	185.545	0.041212	1152.95	967.405
Pressure Ratio					
1.14801					
New P core (psia)	**Sub-atm DP (psia)**				
15.8	3.8				
New P core less than 18 psia					

(Note: All properties used in Table B-2 and Table B-3 were taken from NIST Thermophycial Properties of Fluid (Reference B-1). Pressure drop ratios evaluated using average properties of core exit and containment pressures)

B.3.3 Assessment of Blockage at Core Inlet

This section discusses the results of the core inlet blockage WC/T simulations. The effects of core blockage on PCT and rod quench are examined through examination of hydraulic results.

The flow distribution results were examined to determine if the flow through channels where the resistance was ramped was indeed blocked and if the flow from the blocked channels enters the unblocked channel, i.e., where the blocked core channel flow was diverted. The C_D ramp of 10^9 at sump switchover time completely blocks all flow into the blocked channels as expected. Also, a large increase in the flow into the unblocked channels occurs at switchover time due to the flow diversion from the blocked channels.

Next, the core inventory is examined. Figure B-9 displays the collapsed liquid level of an average assembly core channel (channel 11 on Figure B-3). The figure shows a slight increase in the collapsed liquid level occurring after the core is blocked for both Case 1 and Case 2. Similar to the core collapsed liquid level, the total vessel liquid mass plotted in Figure B-10 shows that the vessel continues to increase in liquid mass even after the core channels are blocked. The increase in the core liquid mass can be attributed to the flow supplied to the core being in excess of the boil-off rate or from liquid inventory in the UP entering the core. The UP global channel (area above upper core plate) collapsed liquid level plotted in Figure B-11 shows UP inventory is available, however, countercurrent flow limitation (CCFL) at the upper elevations of the core may restrict the UP inventory from entering the core region. Predicted flow results are difficult to assess due to oscillations. In order to clearly present the results, the flow figures plot the integral of the flow rates and examine the slopes of the figures to draw conclusions. Figure B-12 plots the integral of liquid flow rate at the exit of the core. The core outlet flow represents the HA, AVG, and GT core channels and the core inlet flow represents the LP core channel. The lower vapor velocity in the LP channel allows the liquid from the UP to drain into the core increasing the core liquid mass, aiding LTCC.

Whether the increase in core liquid inventory is also aided by the flow supplied at the core inlet can be identified by a comparison between the total core flow and the flow rate needed to match core boil-off. The flow rate to match the core boil-off rate was calculated by dividing the core power by the core average h_{fg}. Figure B-13 compares the integrated inlet flow rate vs. the integrated boil-off rate. The figure shows the flow supplied to the core is larger than the boil-off rate after the time that switchover to recirculate from the containment sump has occurred for both cases. The increase in core liquid mass can therefore be partially attributed to the inlet flow.

The different inlet flow between Case 1 and Case 2 shown in Figure B-13 can be explained due to the difference in the resistance at the core inlet. The resistance ramp, which simulates debris build up, effectively decreases the core inlet flow and causes the core liquid inventory to increase at a lower rate. The core inlet flow rate is governed by the driving head in the downcomer and the amount of flow resistance. The collapsed liquid level (CLL) for each downcomer (DC) channel is plotted in Figure B-14 through B-16. In Case 2, the DC CLL increases well before the DC CCL calculated for Case 1. The increase in the downcomer CLL for Case 2 is calculated to occur due to the large resistance at the core inlet.

In Case 1, the DC CLL increases at a slower rate then Case 2. Although the resistance is ramped in Case 1 to block 82% of the core flow area, significant flow is still able to enter the core without the additional build-up of driving head in the downcomer. The higher core inlet flow rate for Case 1 shown in Figure B-13 increases the UP liquid inventory and eventually increases the liquid flow rate in the loops. The integral of the HL liquid flow rates for each loop are compared in Figures B-17 through B-19. The increased loop liquid flow in Case 1 allows for some liquid inventory to flow through the steam generators and eventually make it back into the downcomer, increasing the downcomer liquid level starting around 1600 seconds (as shown in Figures B-14 through B-16) and increasing the driving head. The build up of driving head explains the increase in the core inlet flow rate for Case 1 shown in Figure B-13.

Finally, the PCT of the hot rod is examined. The hot rod PCT vs. time is displayed in Figure B-20. WC/T predicts the PCT to occur for both cases within the traditional LOCA analysis space. After roughly 300 seconds the core is quenched and no significant heatup occurs thereafter. Because no late heat up occurs, the local maximum and core-wide oxidation calculations for traditional LOCA analyses are still considered applicable. It is concluded that sufficient liquid can enter the core to remove core decay heat once the plant has switched to sump recirculation with up to 99.4% core blockage.

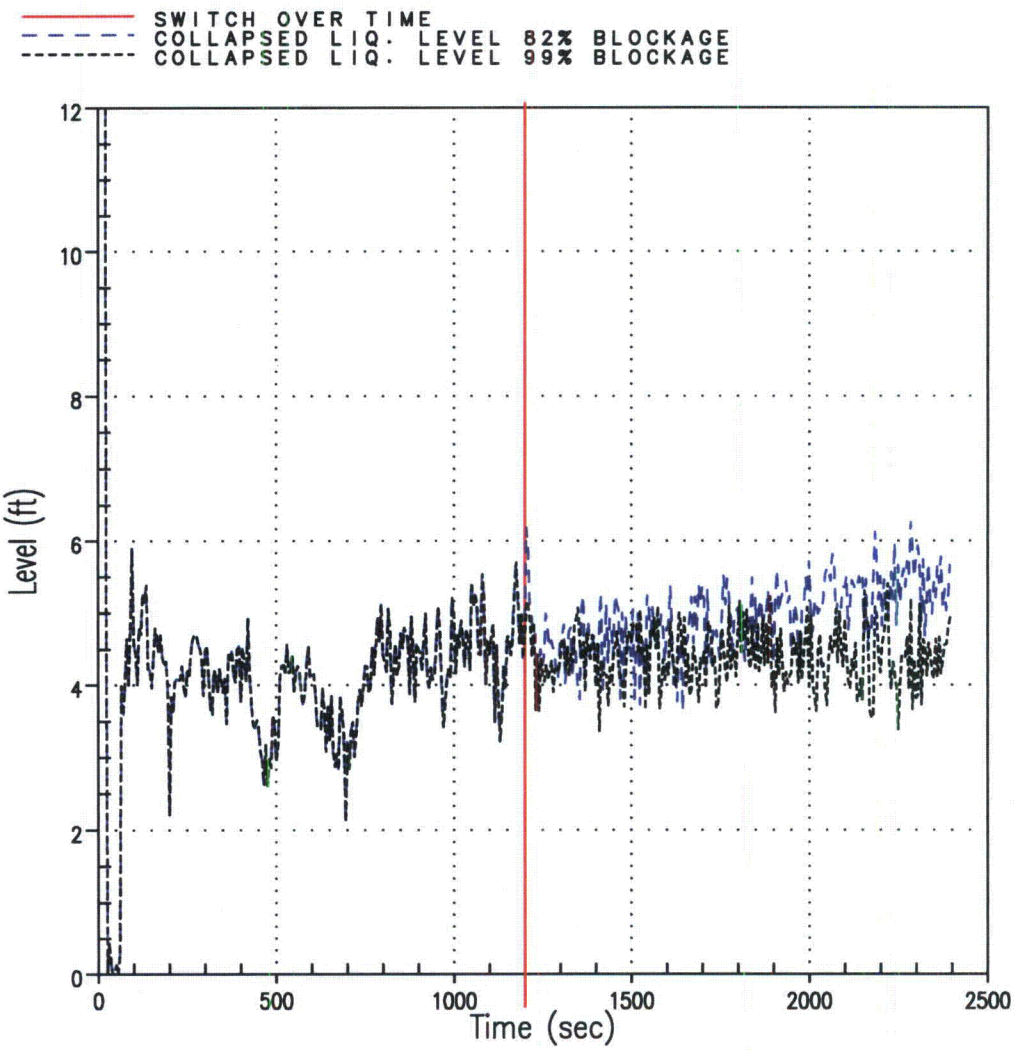


Figure B-9 Average Core Channel Collapsed Liquid Level for Case 1 and Case 2

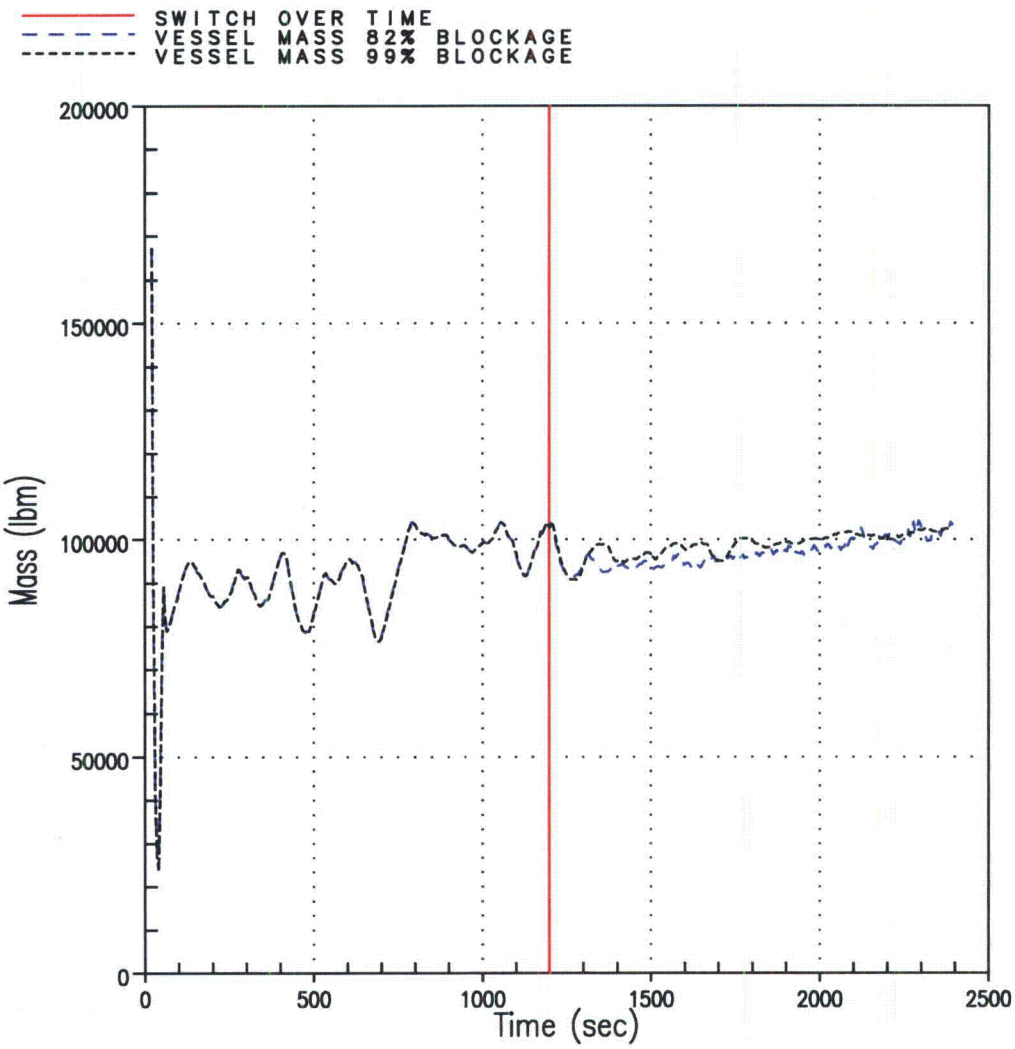


Figure B-10 Total Vessel Liquid Mass for Case 1 and Case 2

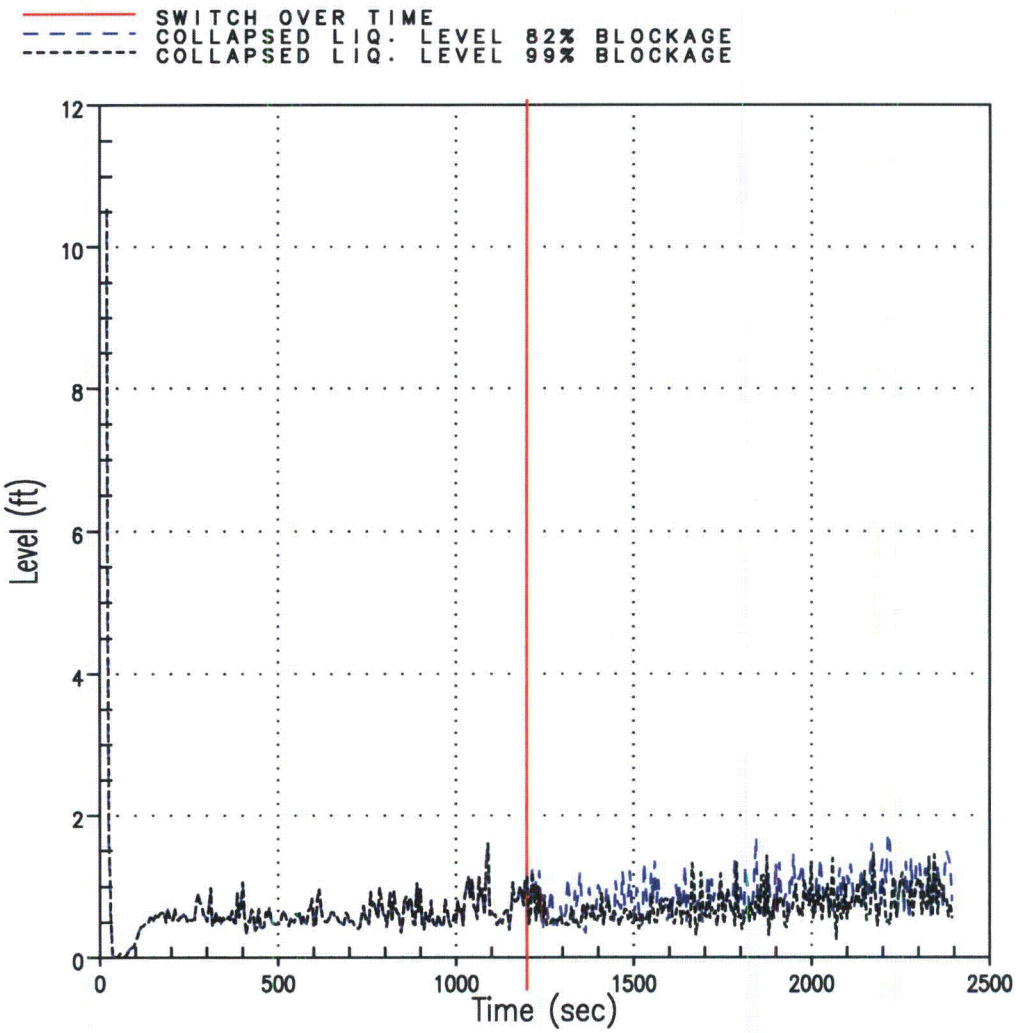


Figure B-11 Upper Plenum Global Channel Collapsed Liquid Level for Case 1 and Case 2

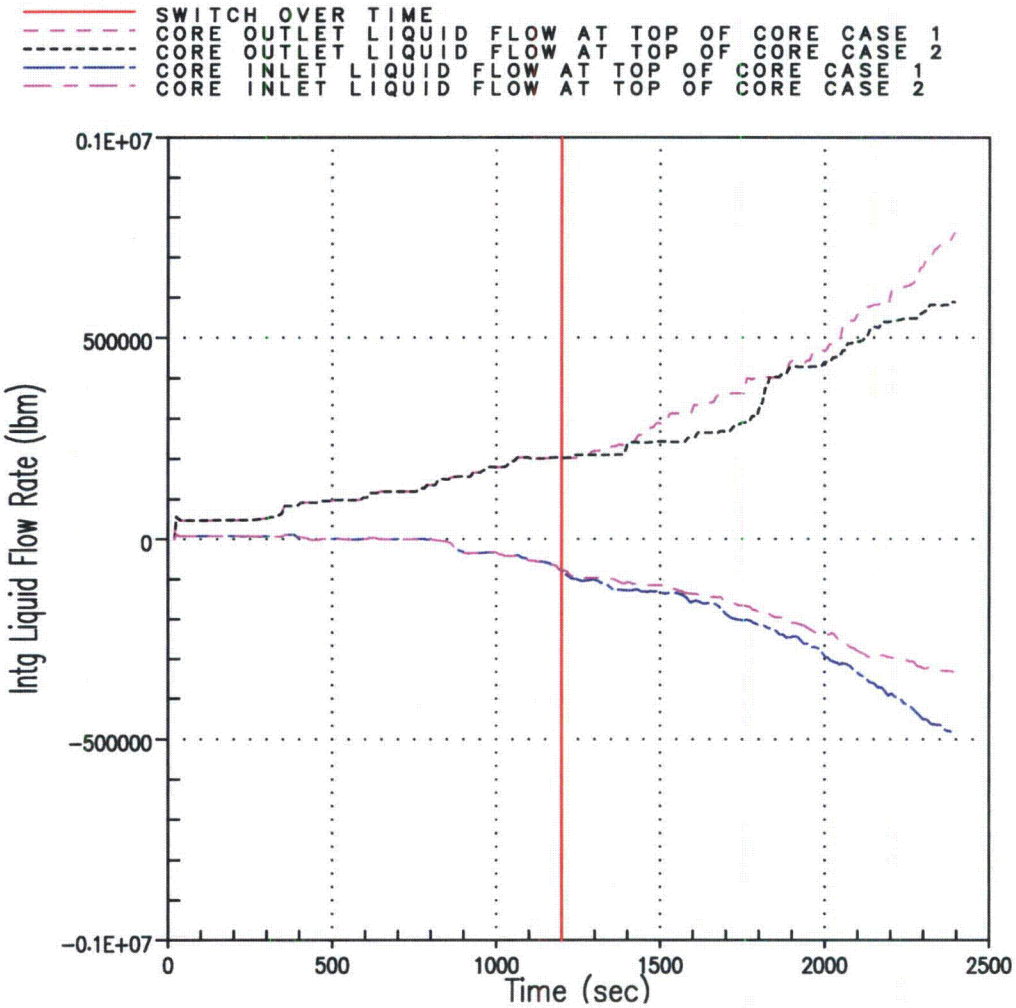


Figure B-12 Total Integrated Liquid Flow at the Top of the Core for Case 1 and Case 2 (Positive/Outlet flow represents HA, GT, AVG channels; Negative/Inlet flow represents LP channel)

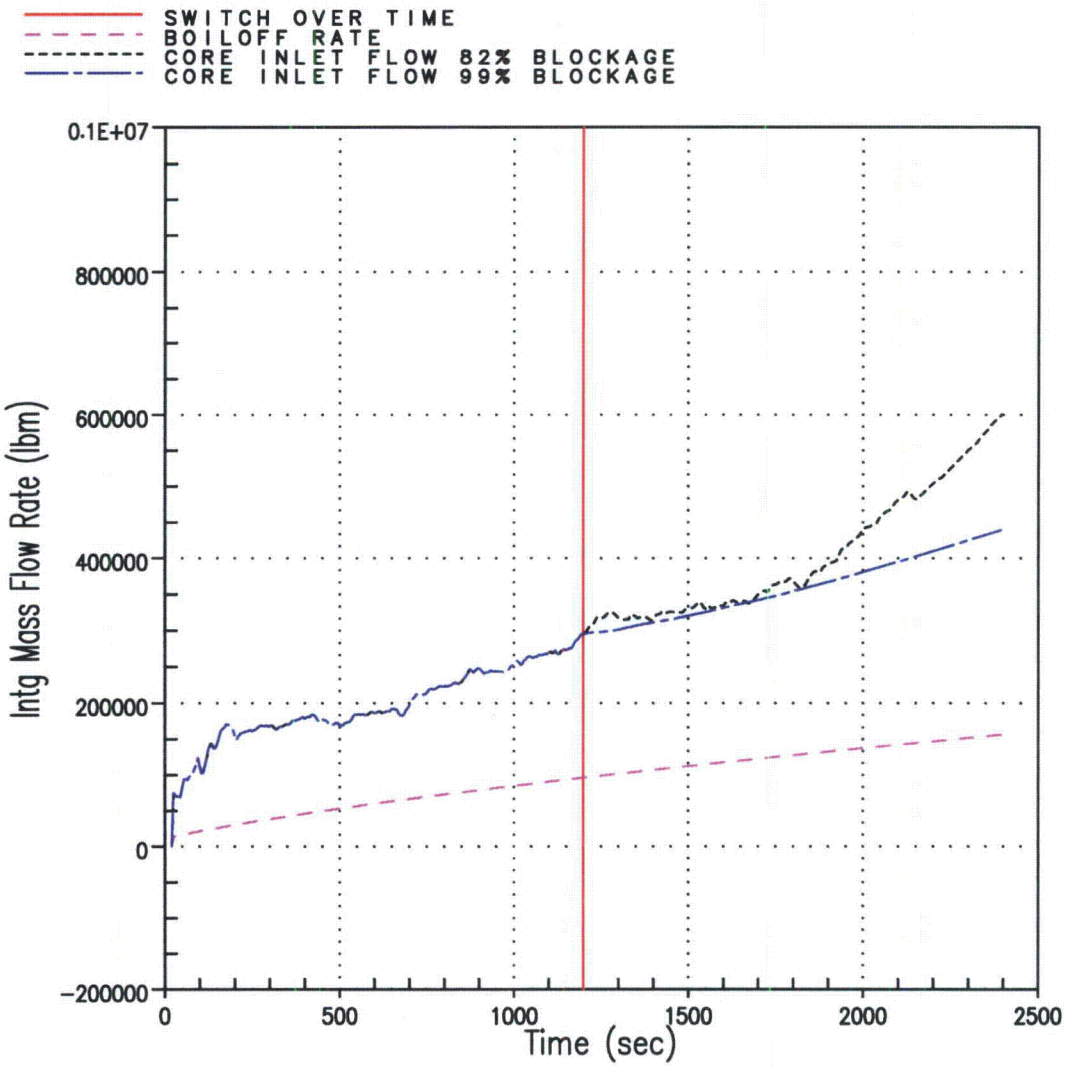


Figure B-13 Integrated Core Flow vs. Core Boil-off for Case 1 and Case 2

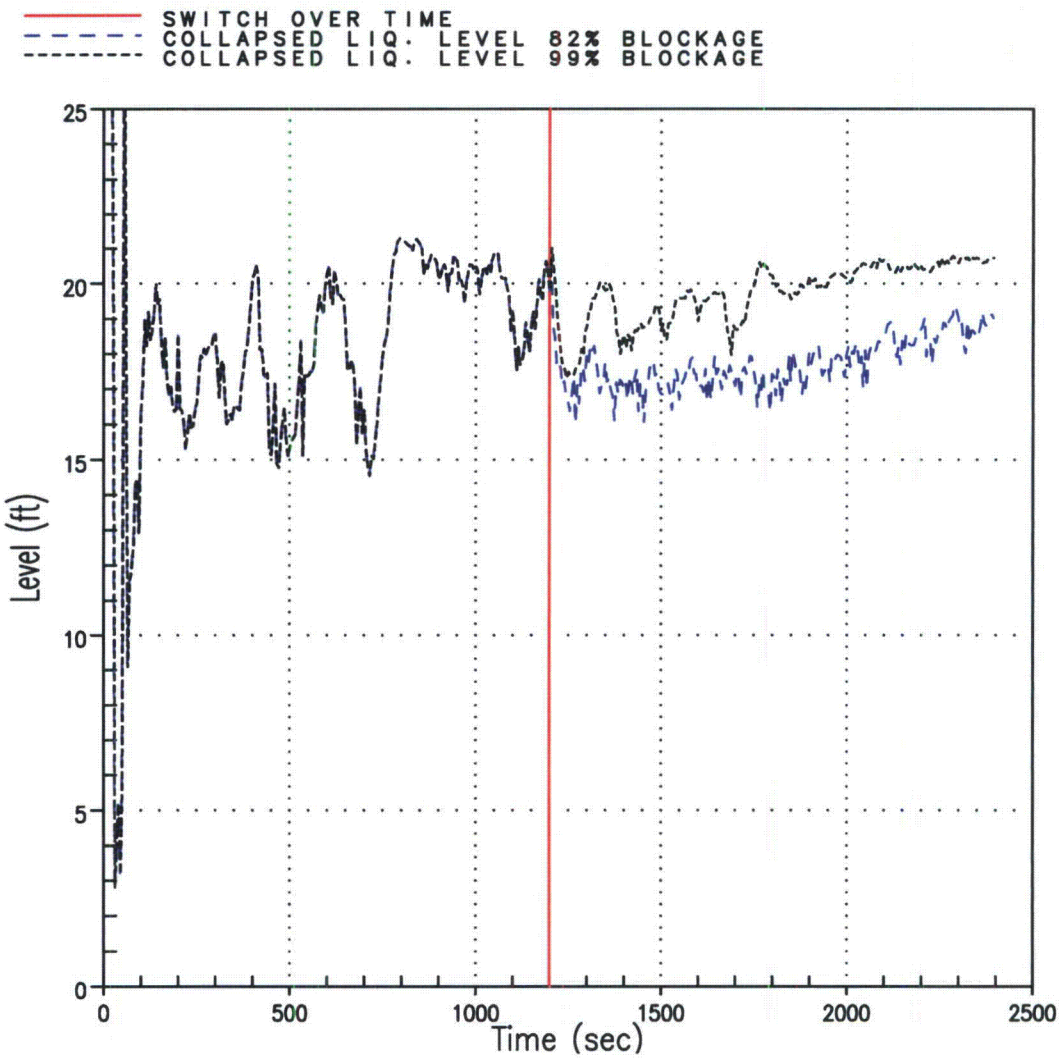


Figure B-14 Broken Loop DC Channel Collapsed Liquid Level for Case 1 and Case 2

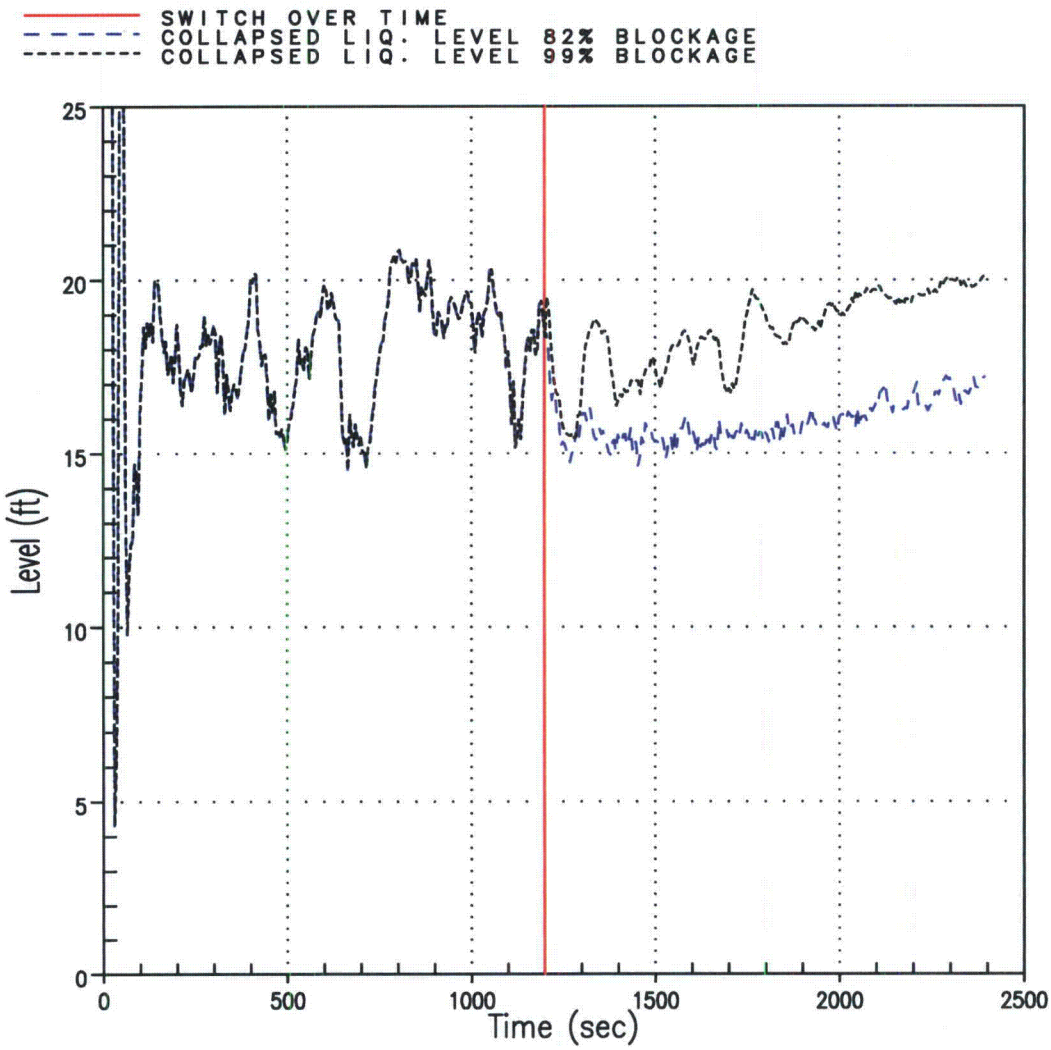


Figure B-15 Intact Loop DC Channel Collapsed Liquid Level for Case 1 and Case 2

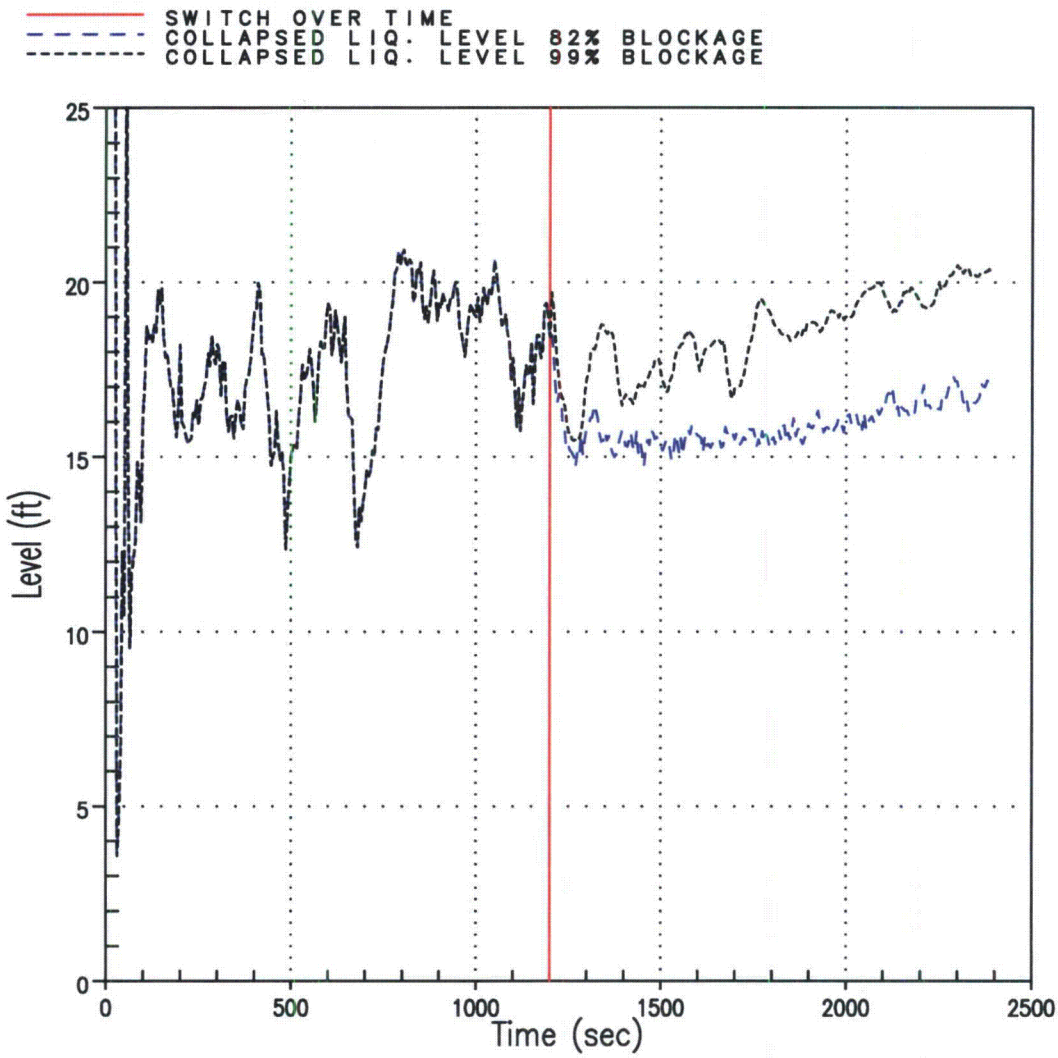


Figure B-16 Pressurizer Loop DC Channel Collapsed Liquid Level for Case 1 and Case 2

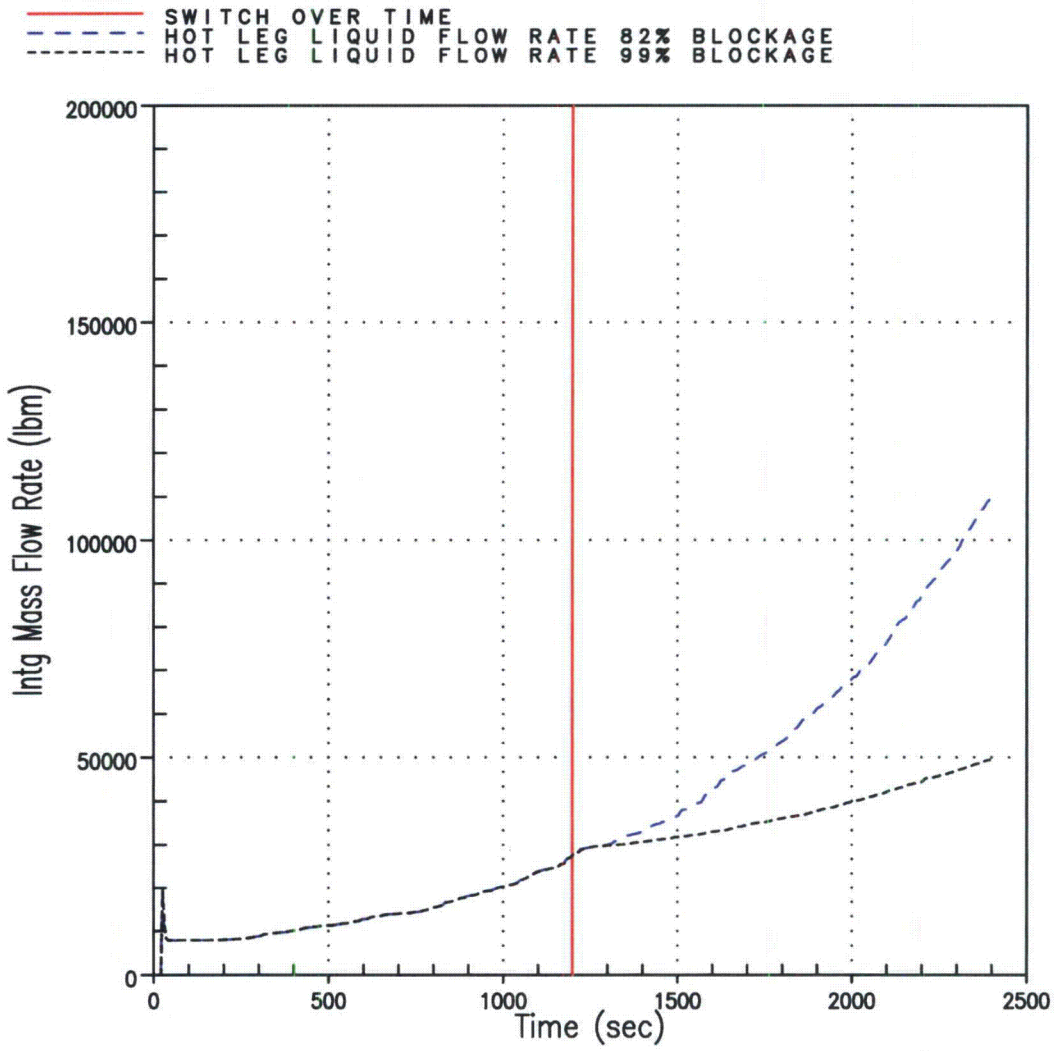


Figure B-17 Pressurizer Loop Hot Leg Integrated Liquid Flow for Case 1 and Case 2

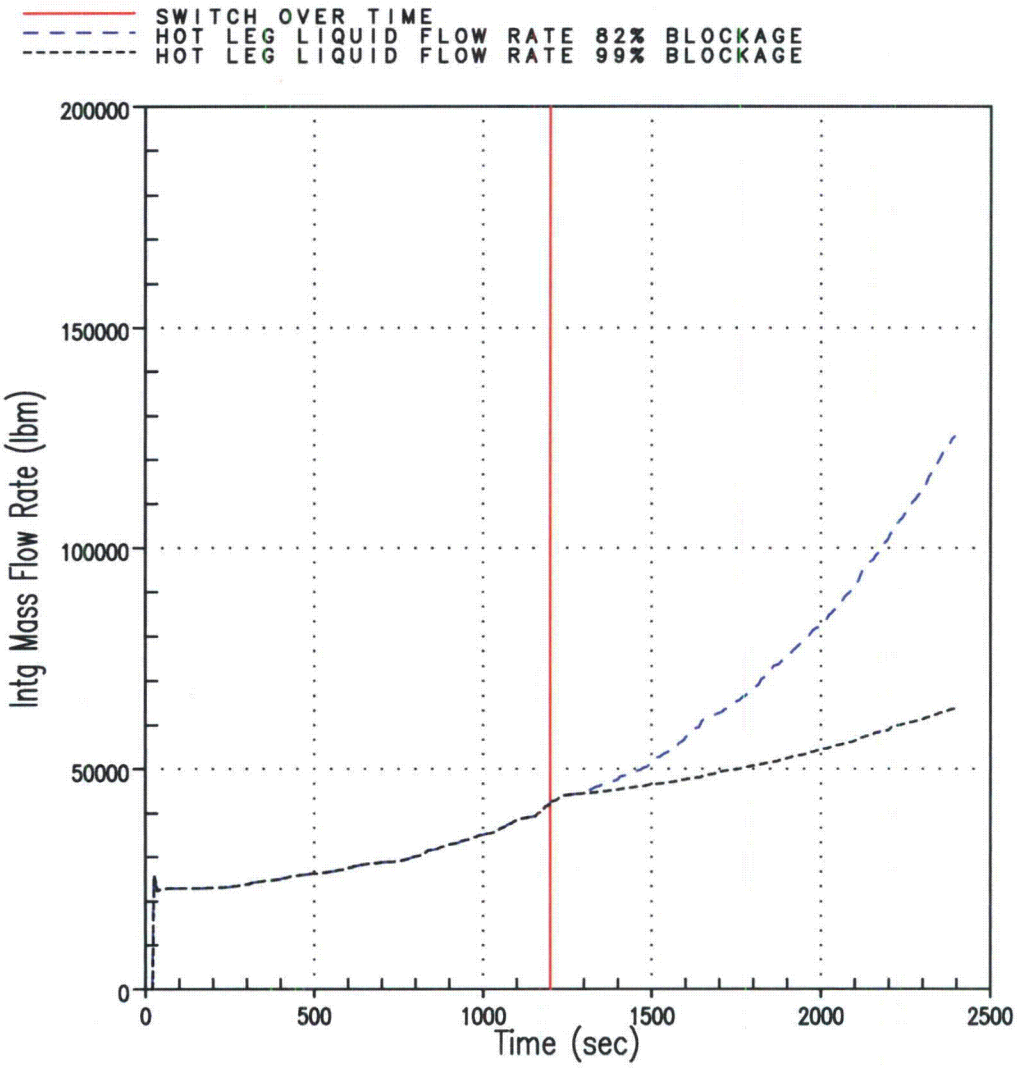


Figure B-18 Intact Loop Hot Leg Integrated Liquid Flow for Case 1 and Case 2

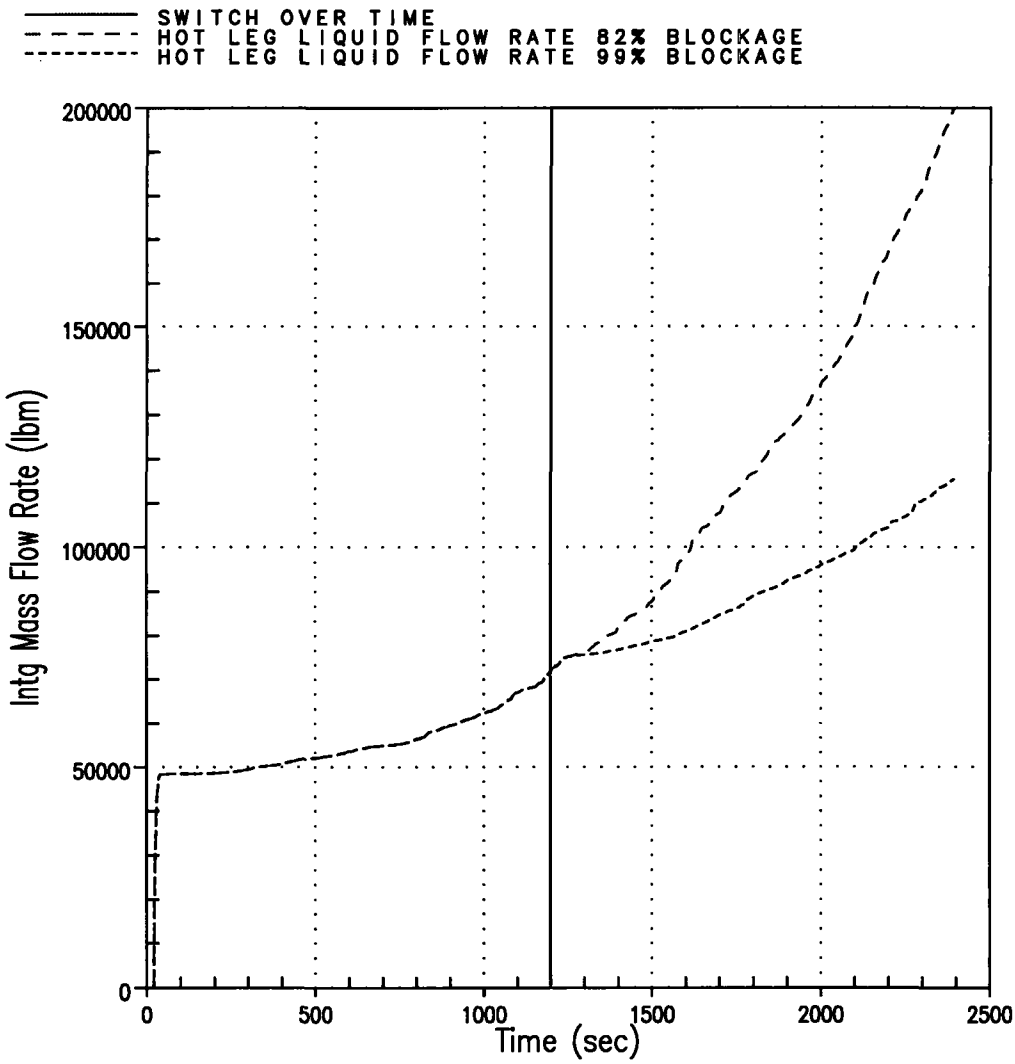


Figure B-19 Broken Loop Hot Leg Integrated Liquid Flow for Case 1 and Case 2

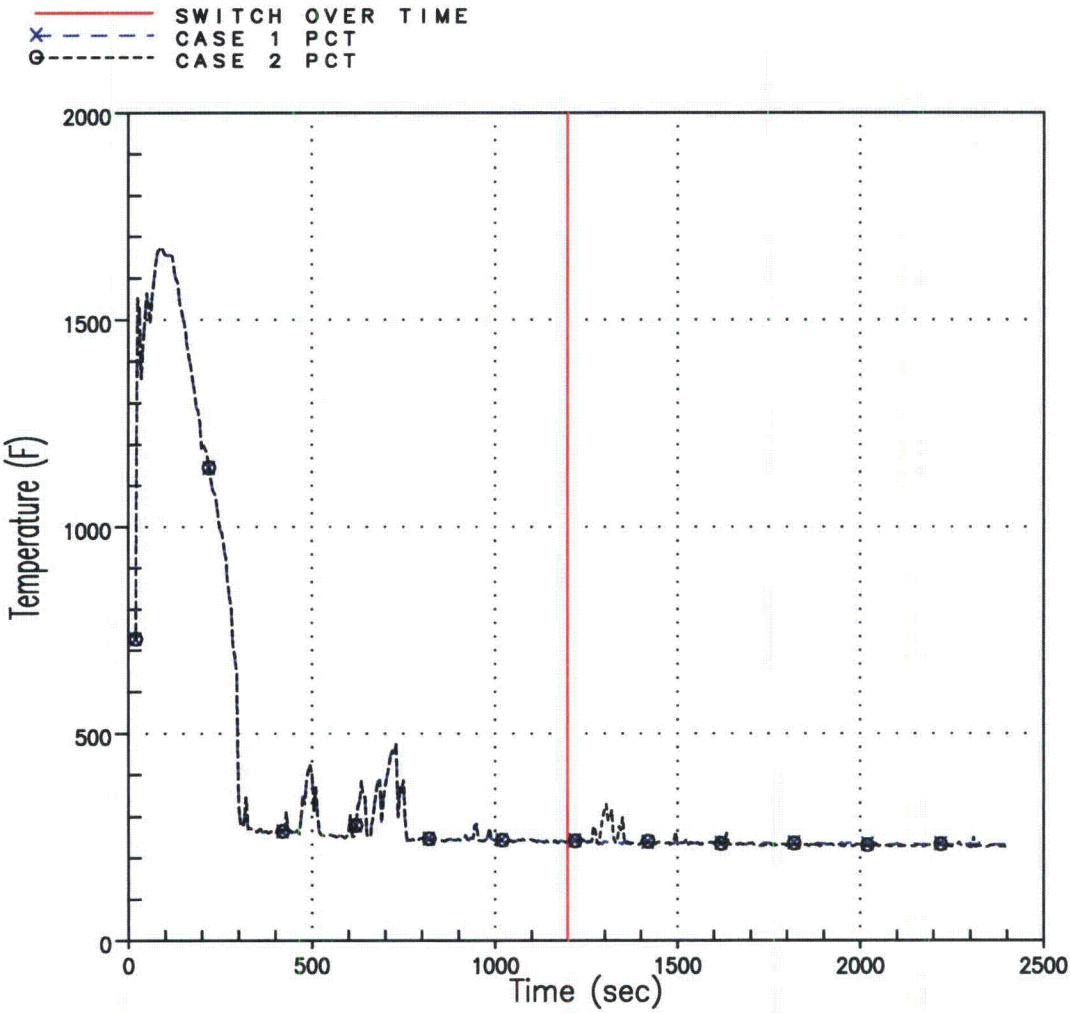


Figure B-20 Case 1 and Case 2 Hot Rod PCT

B.4 SUMMARY

The effects of 82% and 99.4% blockage of the core inlet flow area were examined using WC/T. A comparison between the calculated rates and the flow rate needed to match core boil-off showed ample flow in the core to replace boil-off after core blockage occurred. Also, the PCT plot of the hot rod shows the PCT occurs in traditional LOCA analysis space and after roughly 300 seconds the core is quenched and no significant heatup occurs thereafter. Because no late heat up occurs, the maximum local and core-wide oxidation calculations for traditional analyses are still considered applicable. It is concluded that sufficient liquid can enter the core to remove core decay heat once the plant has switched to sump recirculation with up to 99.4% core blockage.

B.5 ADDITIONAL WC/T CALCULATIONS

Several additional WC/T analyses were peformed in support of the effort documented by this report. These WC/T runs were performed at the request of the 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. The documentation for these additional calculations include 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.

B.5.1 Method Discussion & Input

Two WC/T runs made in support of WCAP-16793-NP are described in Section B-3. These analyses demonstrated that up to 99.4% of the core inlet could be blocked and still maintain sufficient flow to reach the core to remove core decay heat. In order to assess the blockage level that would reduce core flow below that necessary to match coolant boil-off, modifications 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 section is Case 2, or the more restricted flow area case, from Section B-3. 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 LTCC. 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.

3. For the first approach, the flow area of the hot channel, Channel 13 (see Figure B-7), 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, 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. These cases are discussed in Section B.3.2.

For the second approach, the loss coefficients of the hot channel, Channel 13 (see Figure B-7), were increased in increments until the flow rate into the core was less than the core boil-off rate. These cases are discussed in Section B.3.3.

B.5.2 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.

B.5.3 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.

B.5.4 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 plots for this case are shown in Figures B-21 through B-27. Figures B-21 and B-22 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 B-23 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 B-24. There are no significant PCT excursions after the core is blocked. Figure B-25 displays the collapsed liquid level of the average assembly core channel (Channel 11 of Figure B-7). 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 B-26 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 B-27. 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 plots for this case are shown in Figures B-28 through B-34. Figures B-28 and B-29 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 cannot 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 B-30). In addition, Figure B-31 shows that the PCT increases until the end of the transient once the core liquid level, shown in Figure B-32, is reduced to a level that the core becomes unquenched. Continuing with the trend discussed above, the void fraction at the core exit (Figure B-33) shows that only vapor is present. The core pressure drop is displayed in Figure B-34. The figure displays a pressure drop of roughly 4 psi at the core inlet as a result of the blockage at the core inlet.

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 LTCC.

B.5.5 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 plots for this case are shown in Figures B-35 through B-41. Figures B-35 and B-36 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 B-36 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 B-37 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 B-38. There are no significant PCT excursions after the core inlet loss coefficient is increased. Figure B-39 displays the collapsed liquid level of the average assembly core channel (Channel 11 of Figure B-7). The figure shows that the collapsed liquid level drops slightly at the 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 B-40 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 B-41. 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 plots for this case are shown in Figures B-42 through B-48. Figures B-42 and B-43 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 B-43 shows a similar behavior as was shown in Figure B-36. 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 B-44 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 B-45. There are no significant PCT excursions after the core inlet loss coefficient is increased. Figure B-46 displays the collapsed liquid level of the average assembly core channel (Channel 11 of Figure B-7). 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 B-47 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 B-48. The figure displays an increased pressure drop of roughly 2 psi as blockage at the core inlet is increased. As the conditions in the 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 plots for this case are shown in Figures B-49 through B-55. Figures B-49 and B-50 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 B-51). In addition, it is displayed in Figure B-52 that the PCT increases until the end of the transient once the core liquid level, shown in Figure B-53, is reduced to a level that the core becomes unquenched. Continuing with the trend discussed above, the void fraction at the core exit (Figure B-54) shows that only vapor is present. The core pressure drop is displayed in Figure B-55. 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 C_D = 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 LTCC.

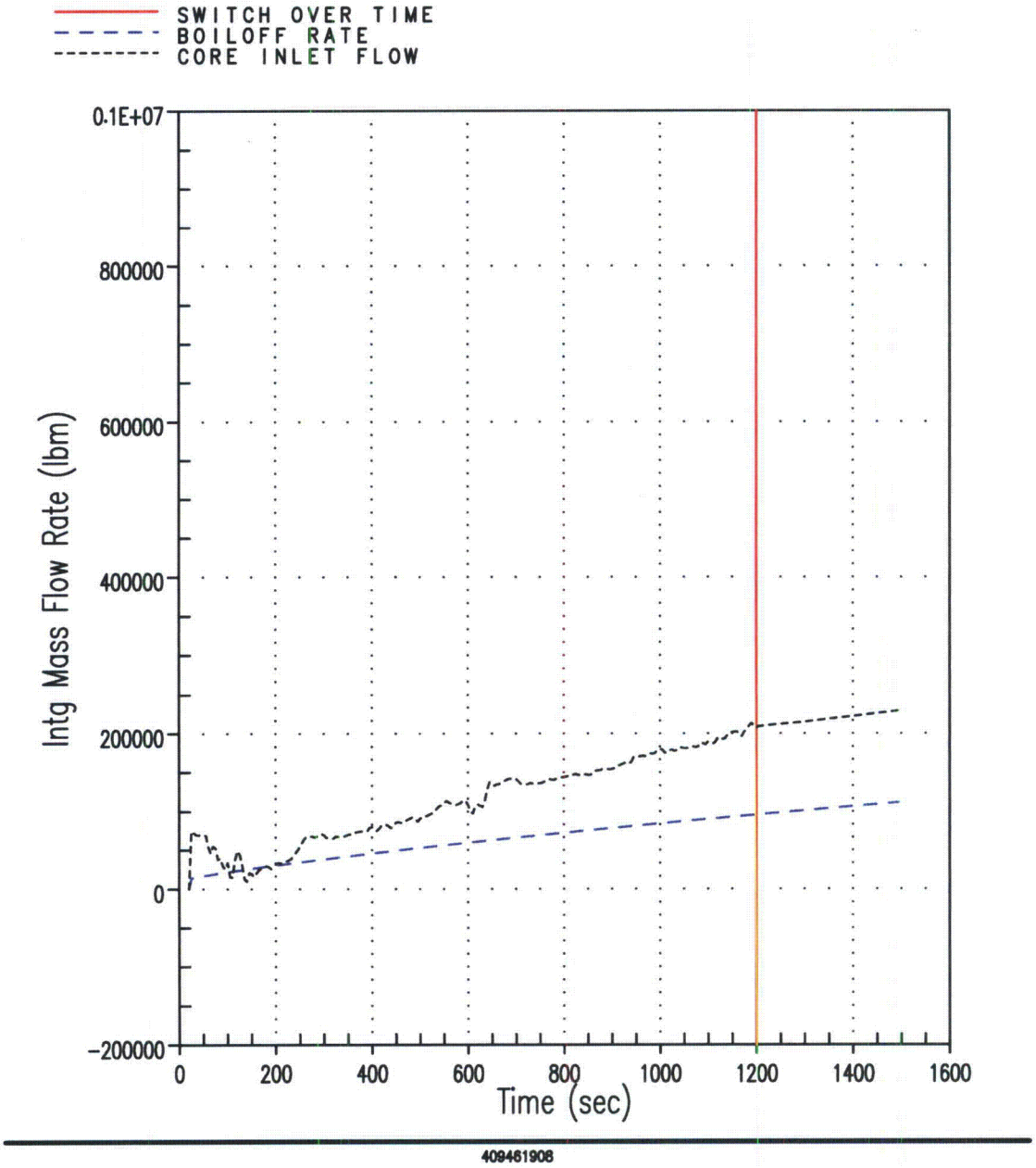


Figure B-21 Integrated Core Flow vs. Core Boil-off for Channel 13 Flow Reduction 50%

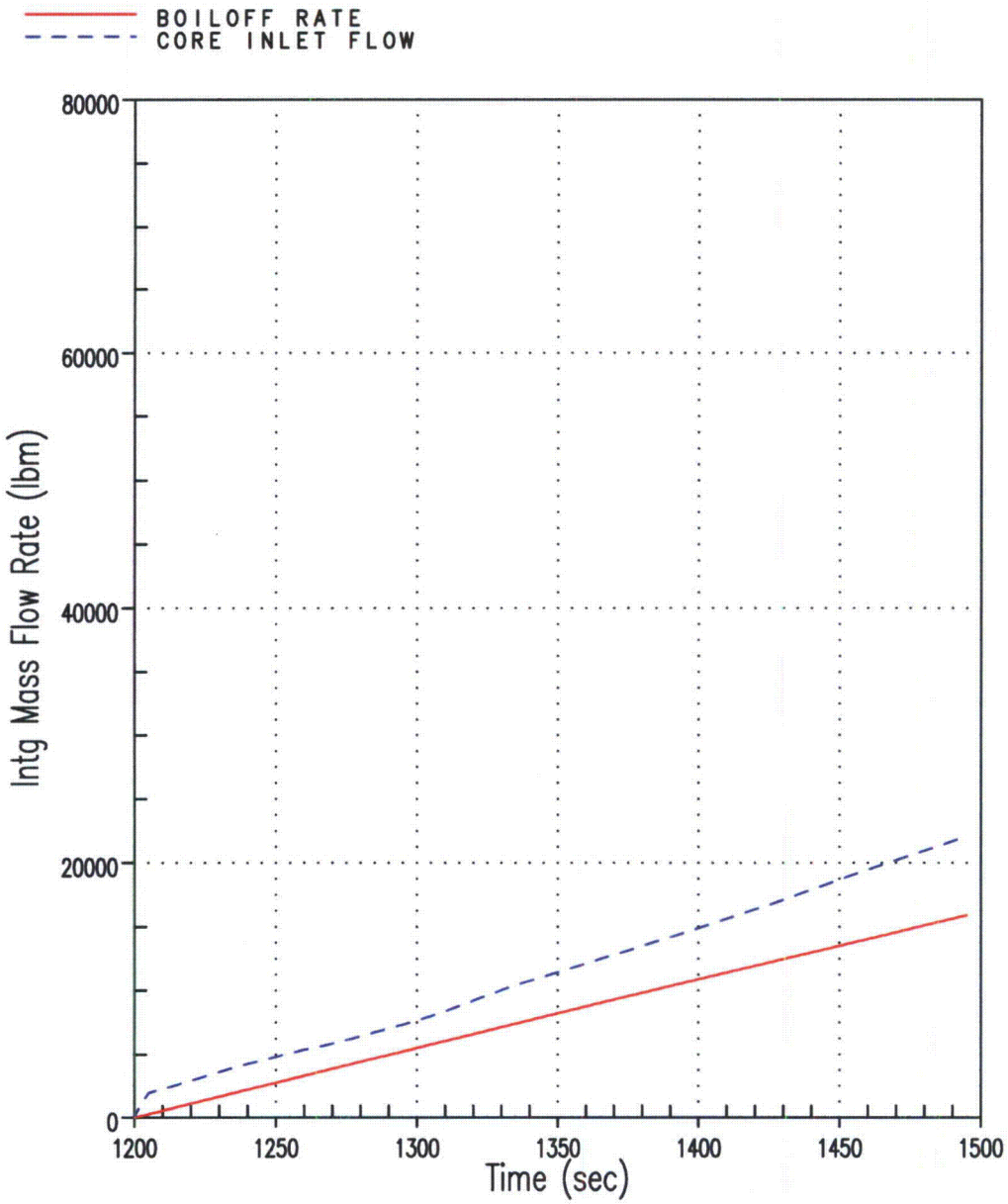


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Figure B-22 Integrated Core Flow vs. Core Boil-off for Channel 13 Flow Reduction 50% Case (Shifted Scale)

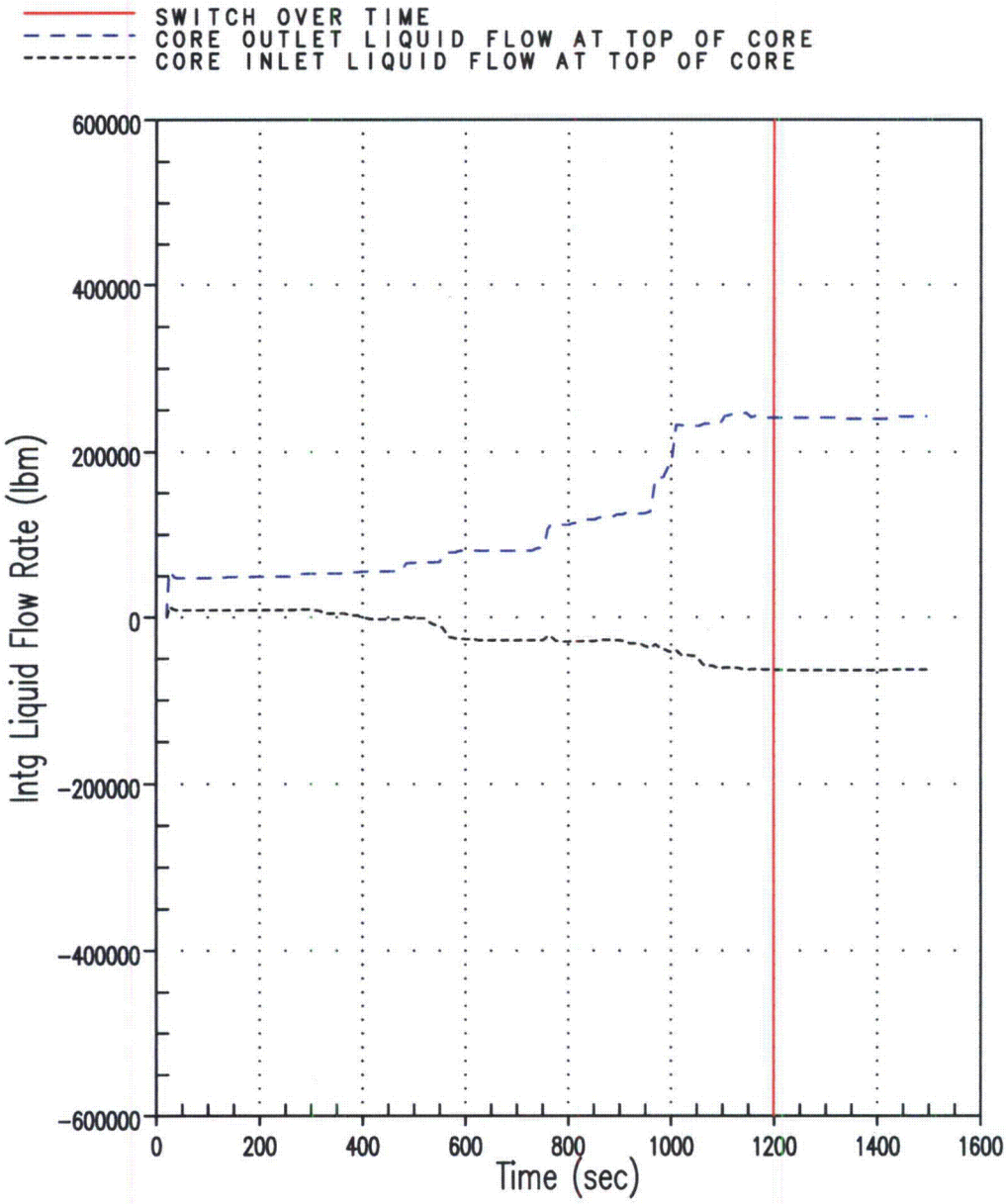

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Figure B-23 Total Integrated Liquid Flow at the Top of the Core for Channel 13 Flow Reduction 50% Case (Positive/Outlet flow represents HA, GT, AVG channels; Negative/Inlet flow represent LP channel)

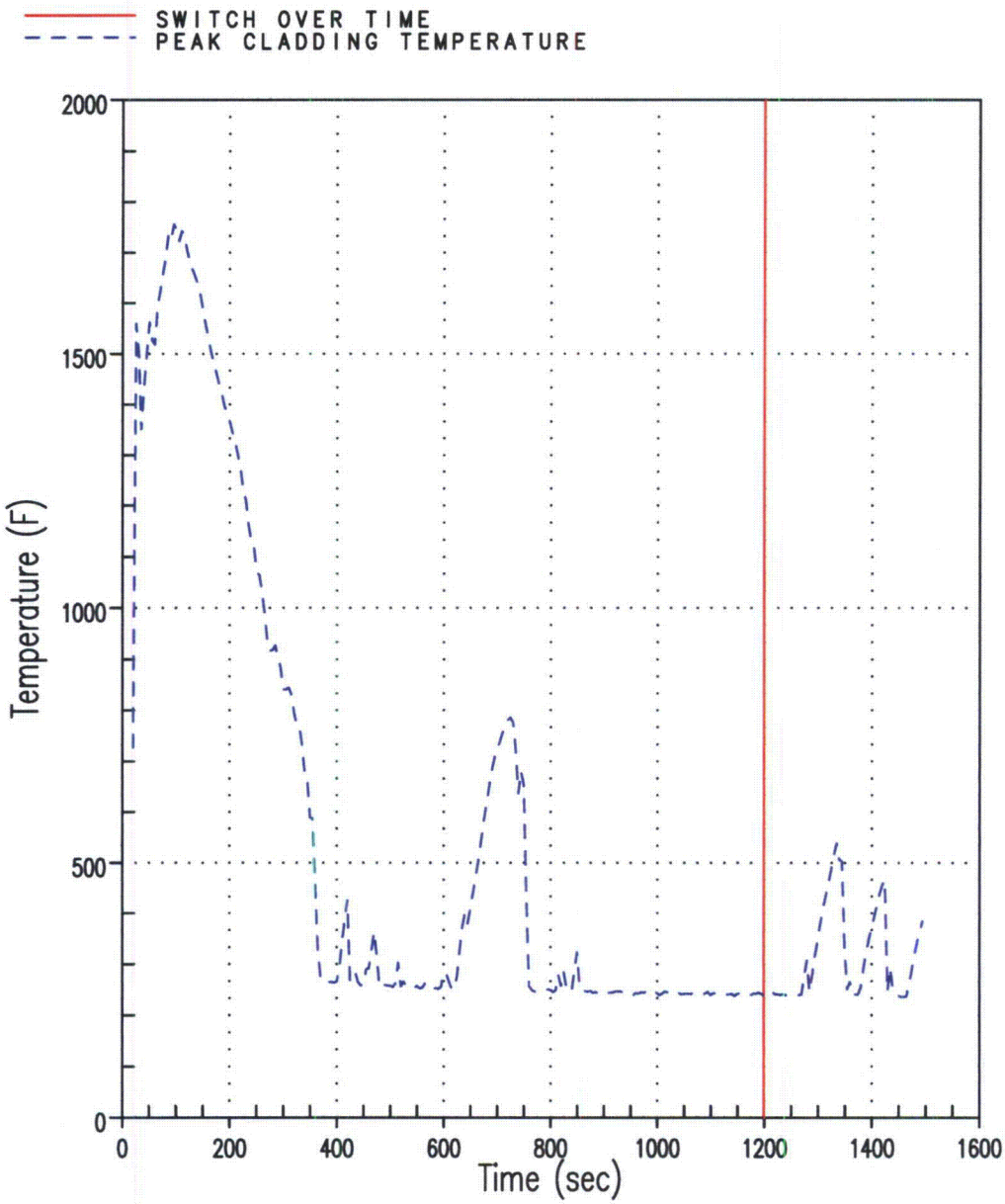

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Figure B-24 Hot Rod PCT for Channel 13 Flow Reduction 50% Case

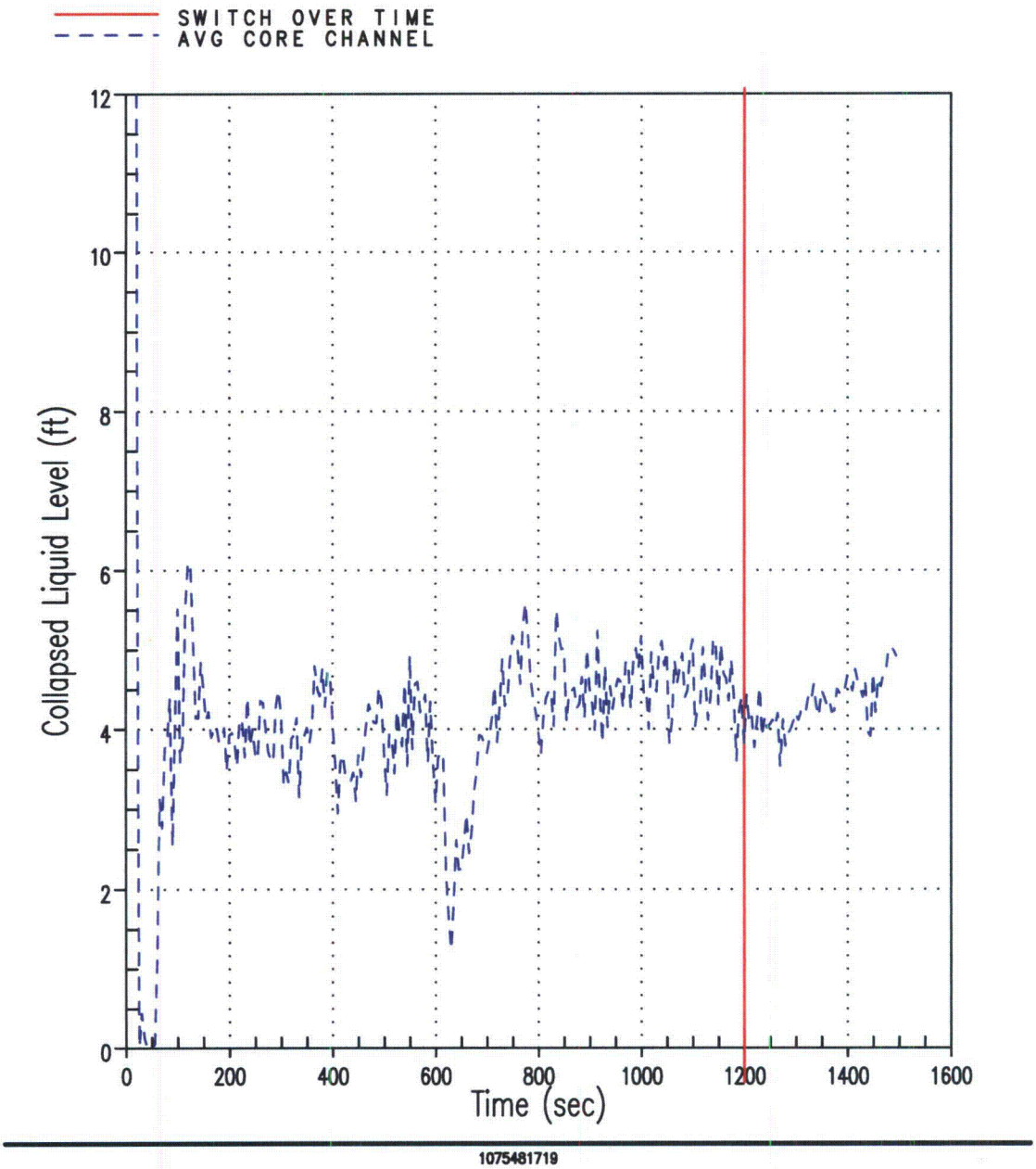

Figure B-25 Average Core Channel CLL for Channel 13 Flow Reduction 50% Case

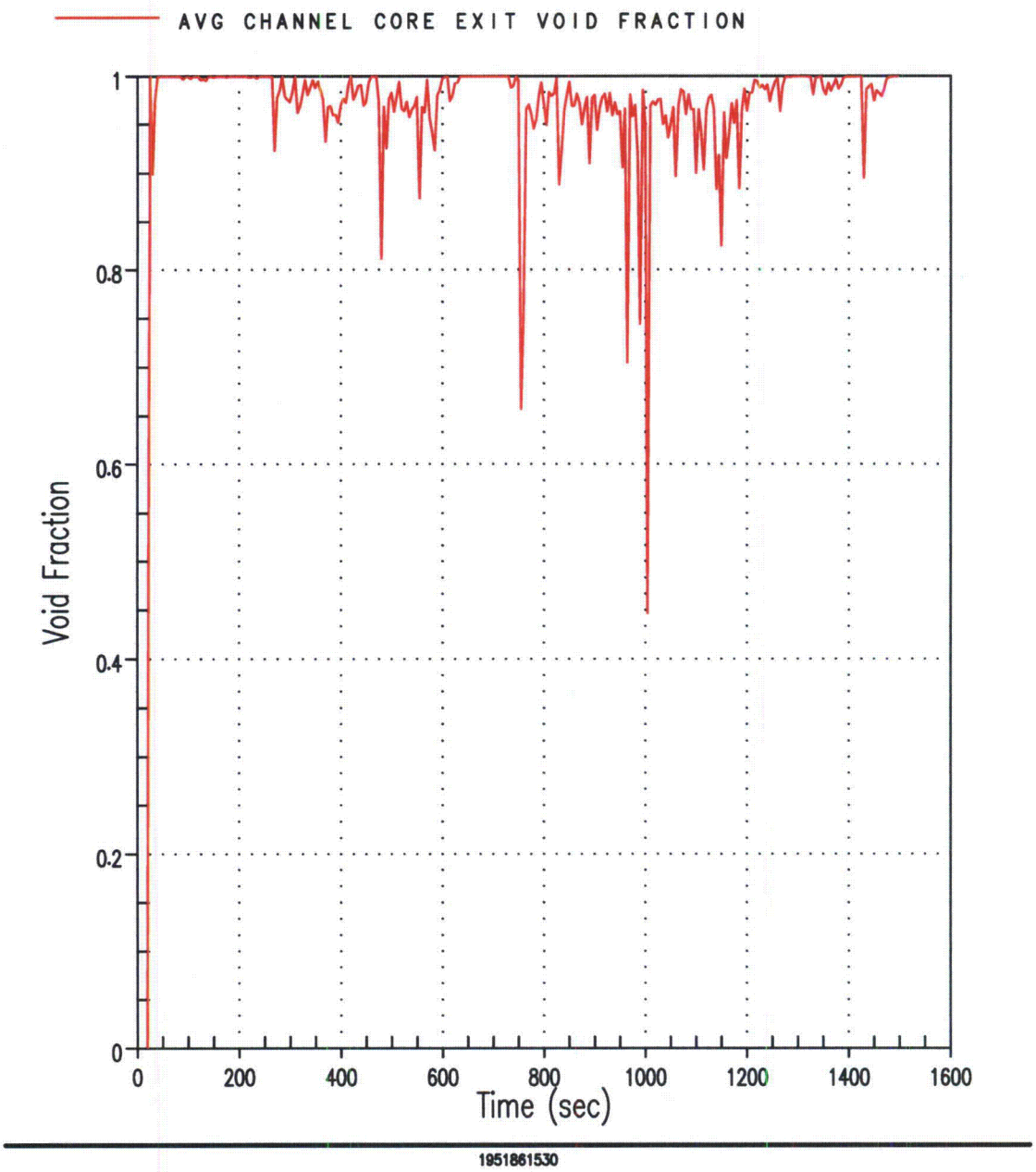

Figure B-26 Void Fraction at the Exit of the Average Core Channel for Channel 13 Flow Reduction 50% Case

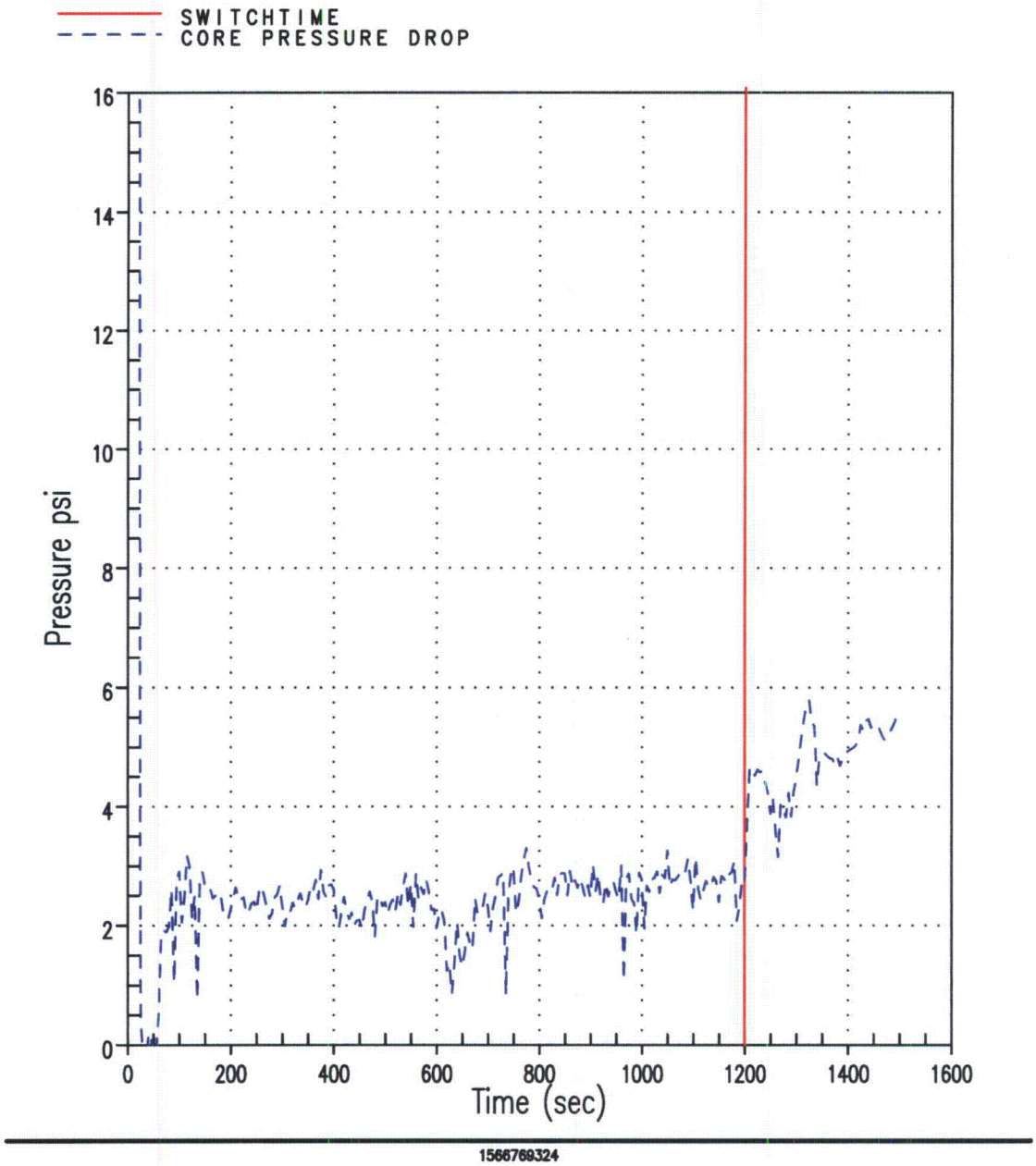

Figure B-27 Core Pressure Drop for Channel 13 Flow Reduction 50% Case

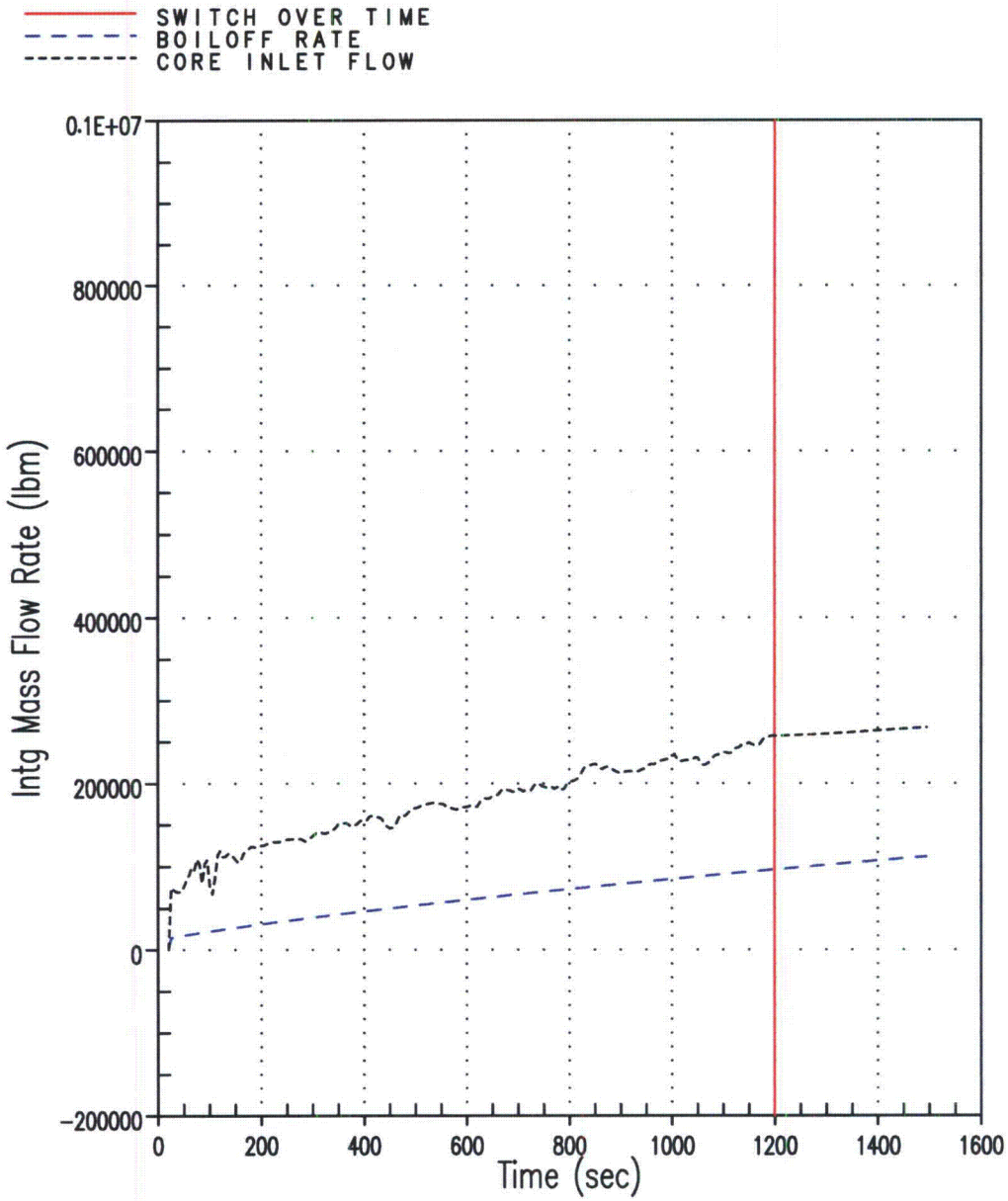

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Figure B-28 Integrated Core Flow vs. Core Boil-off for Channel 13 Flow Reduction 80% Case

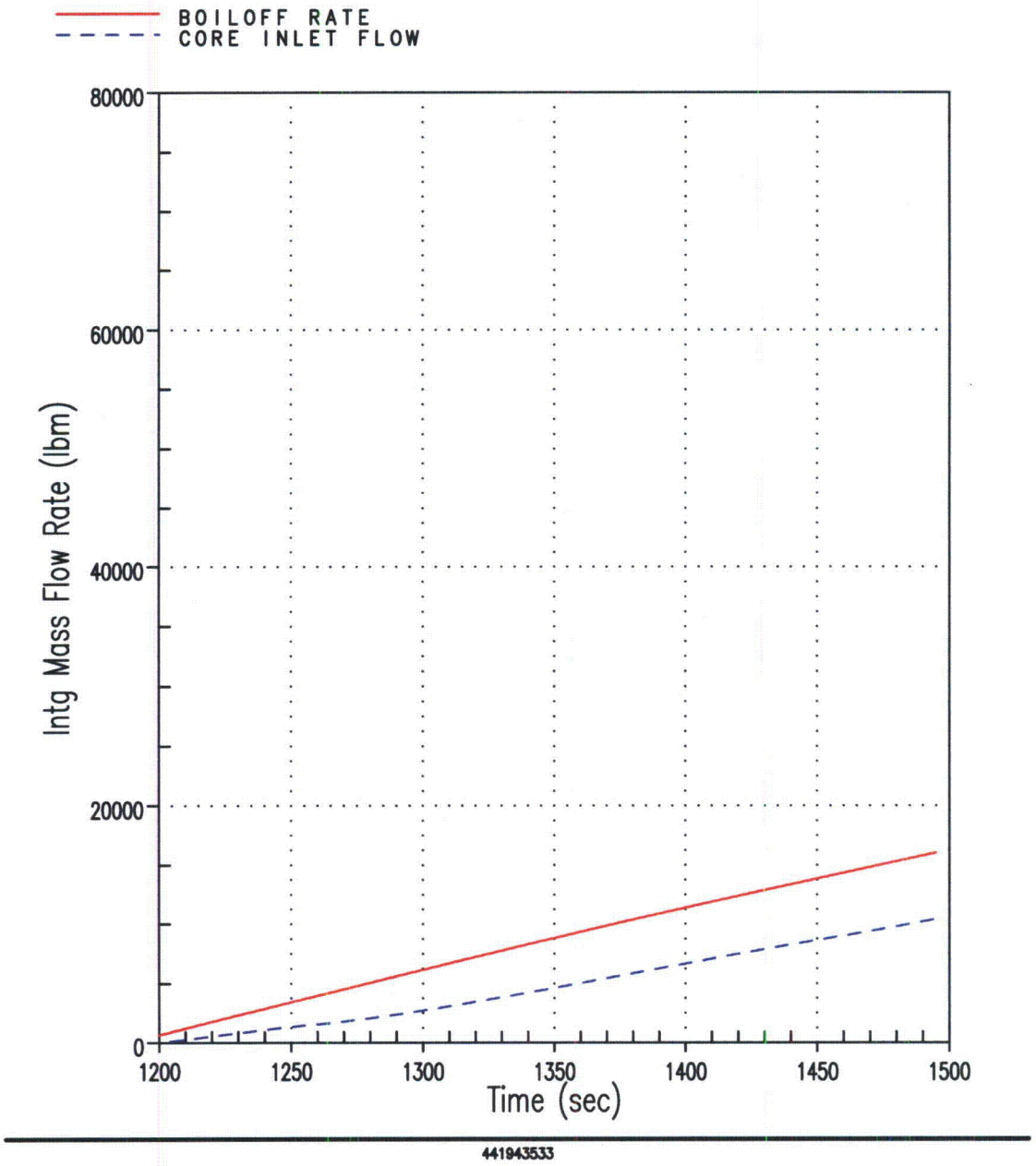

Figure B-29 Integrated Core Flow vs. Core Boil-off for Channel 13 Flow Reduction 80% Case (Shifted Scale)

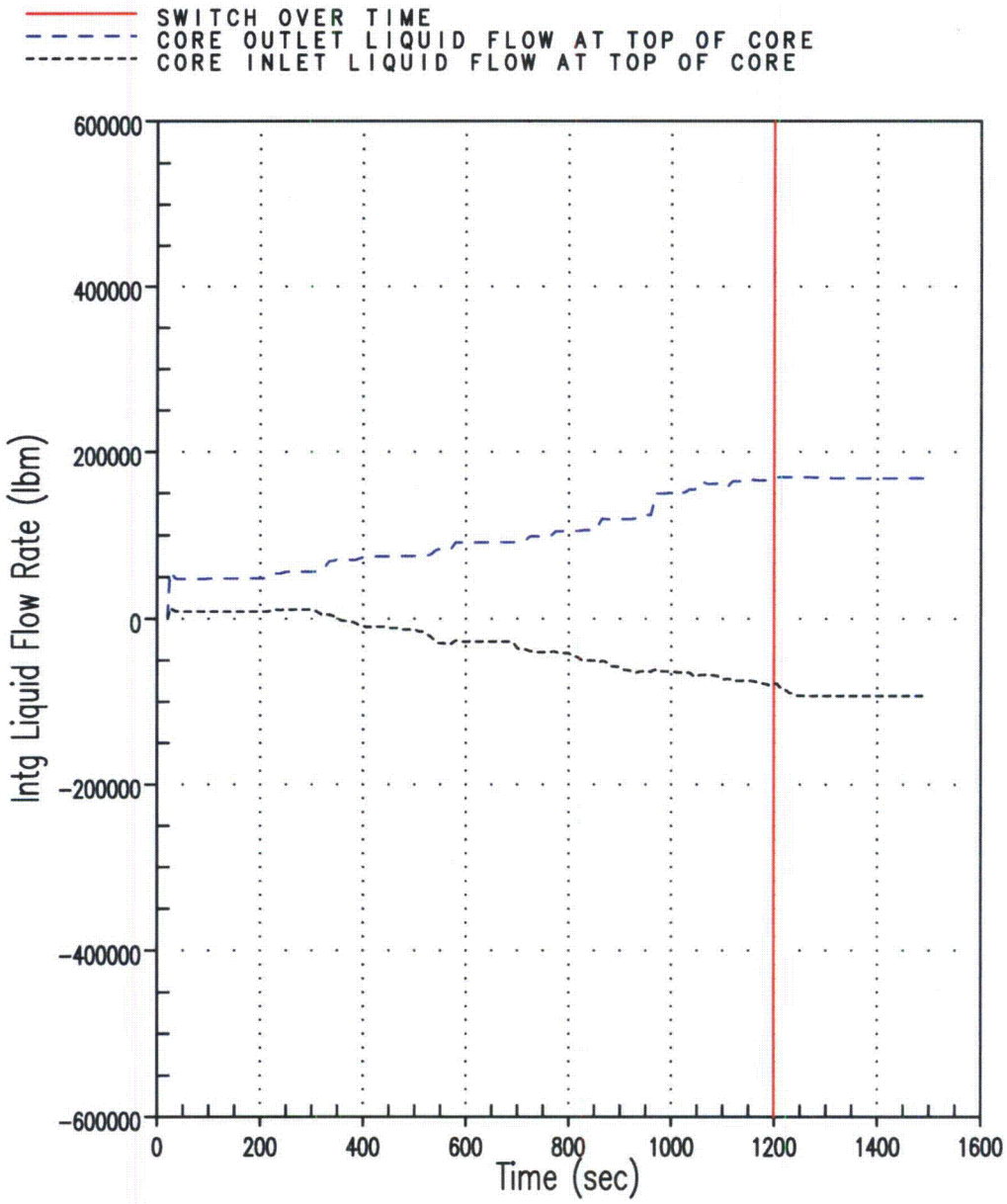

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Figure B-30 Total Integrated Liquid Flow at the Top of the Core for Channel 13 Flow Reduction 80% Case (Positive/Outlet flow represents HA, GT, AVG channels; Negative/Inlet flow represent LP channel)

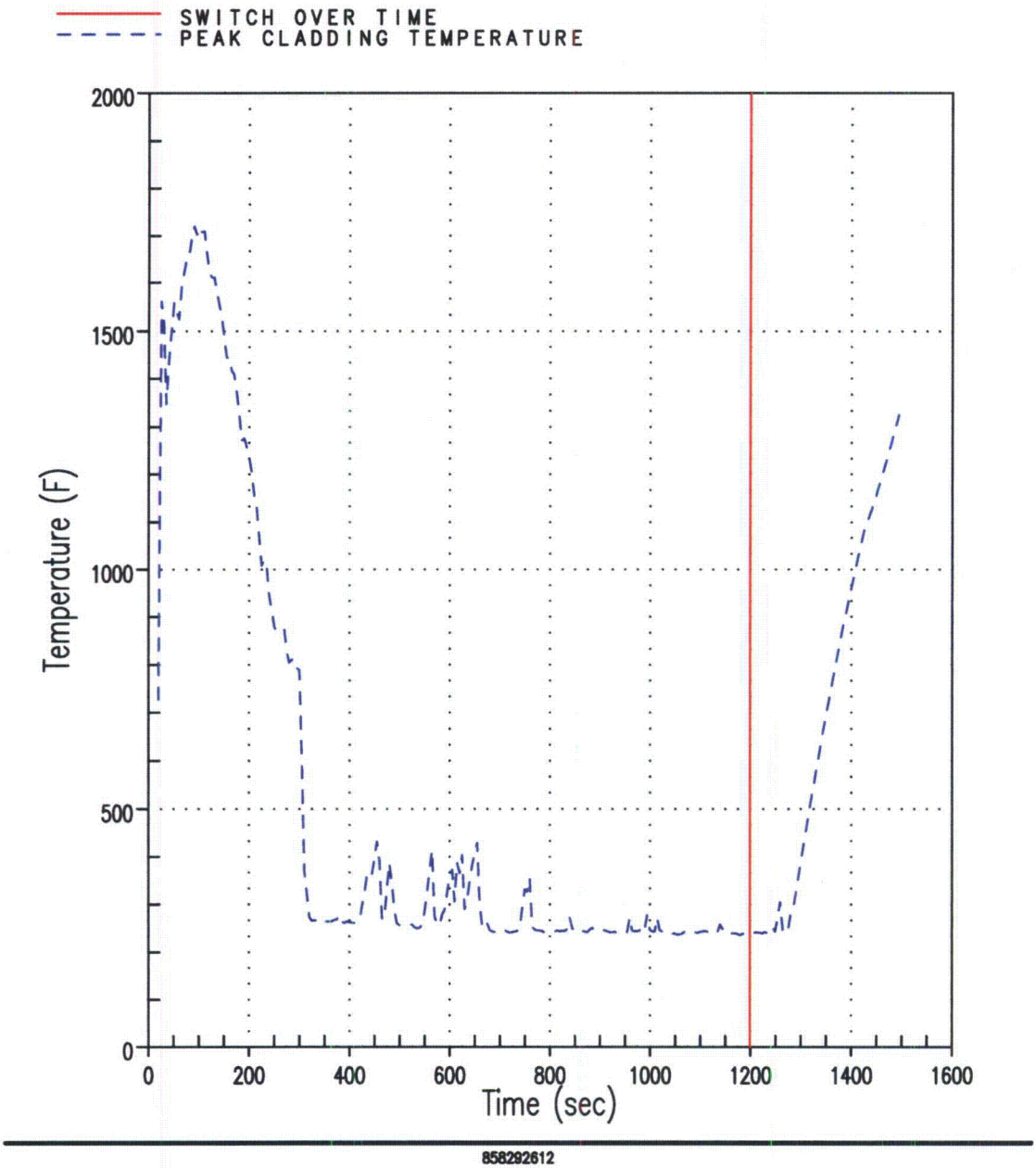


Figure B-31 Hot Rod PCT for Channel 13 Flow Reduction 80% Case

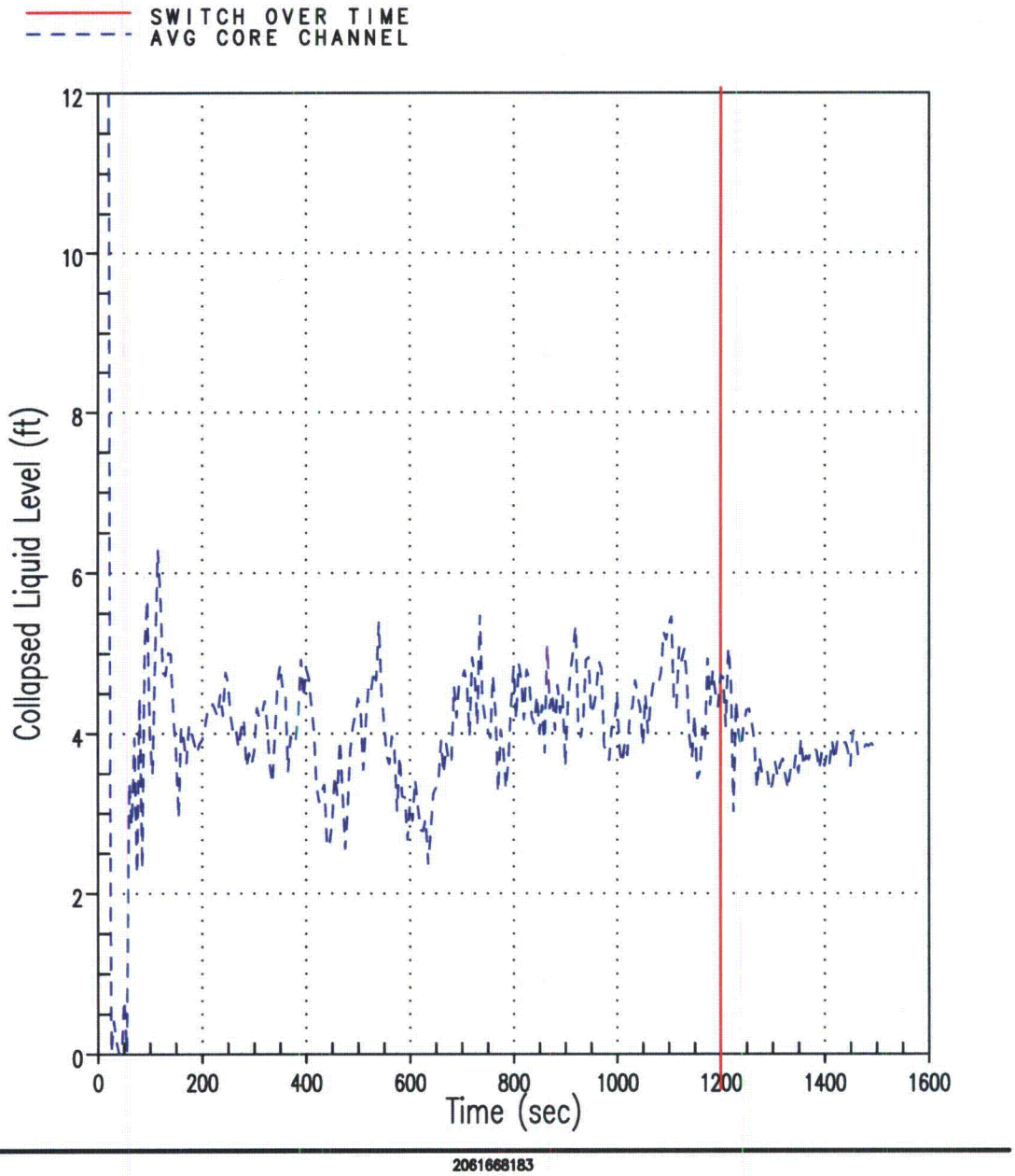


Figure B-32 Average Core Channel CLL for Channel 13 Flow Reduction 80% Case

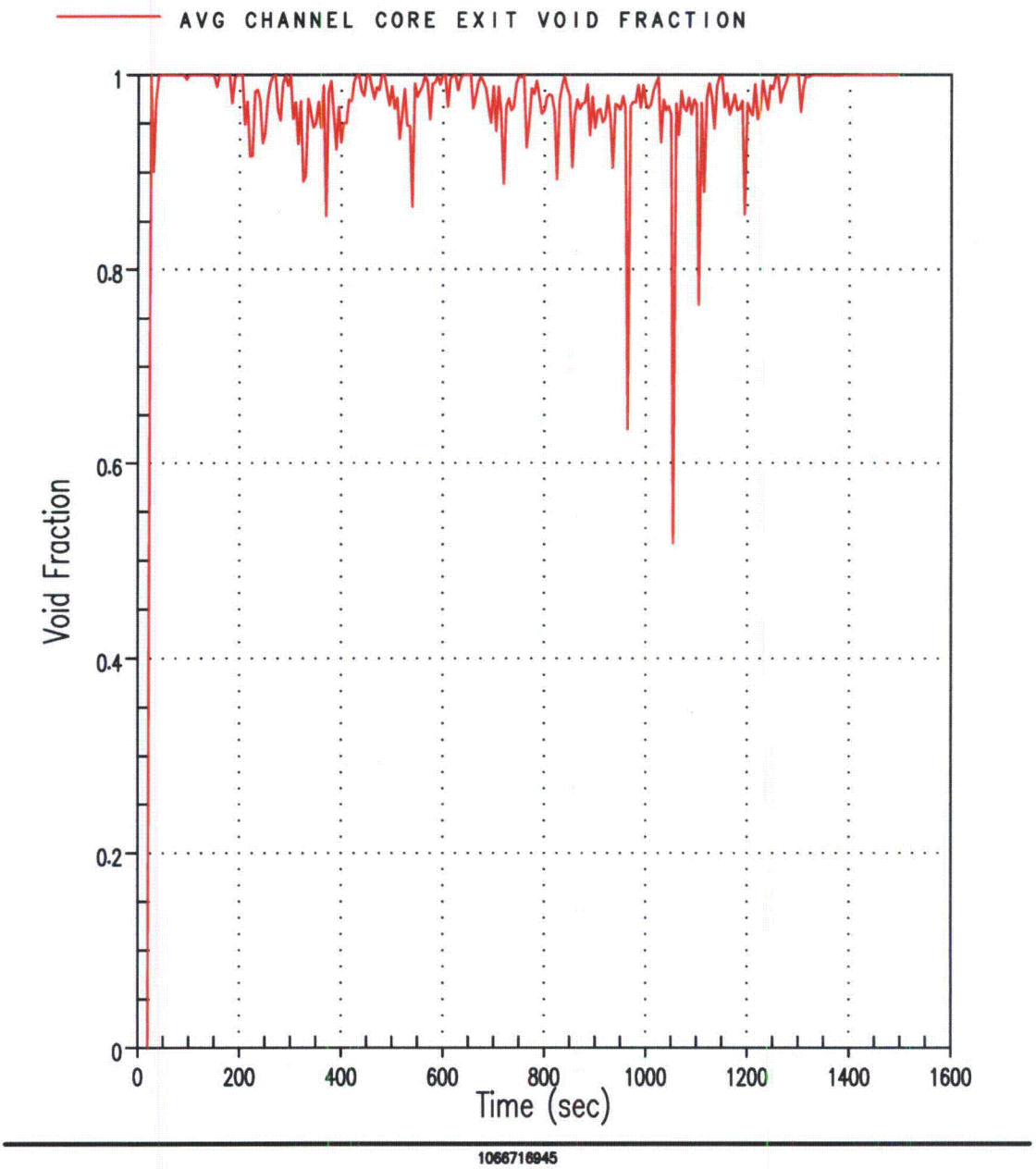


Figure B-33 Void Fraction at the Exit of the Average Core Channel for Channel 13 Flow Reduction 80% Case

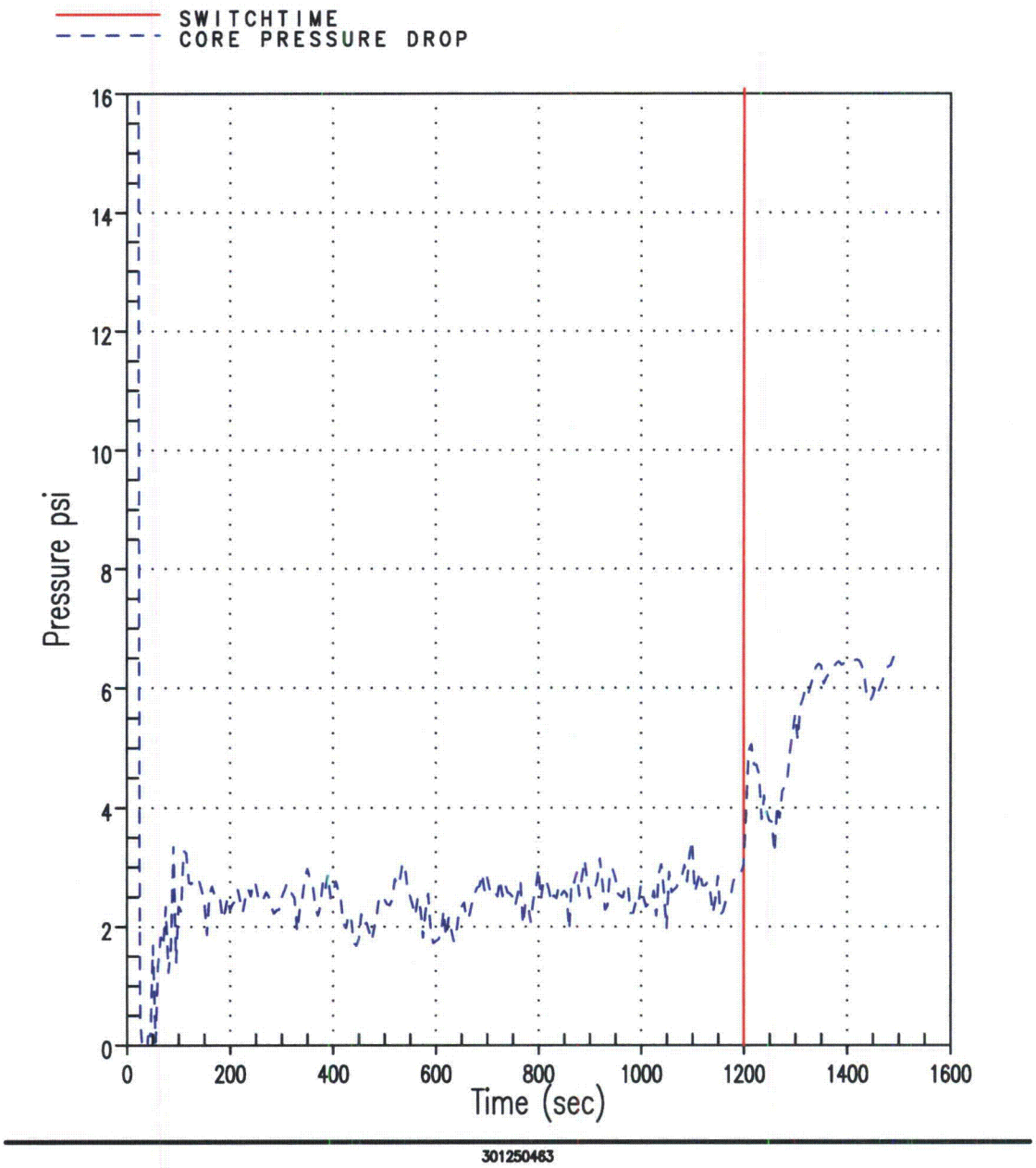
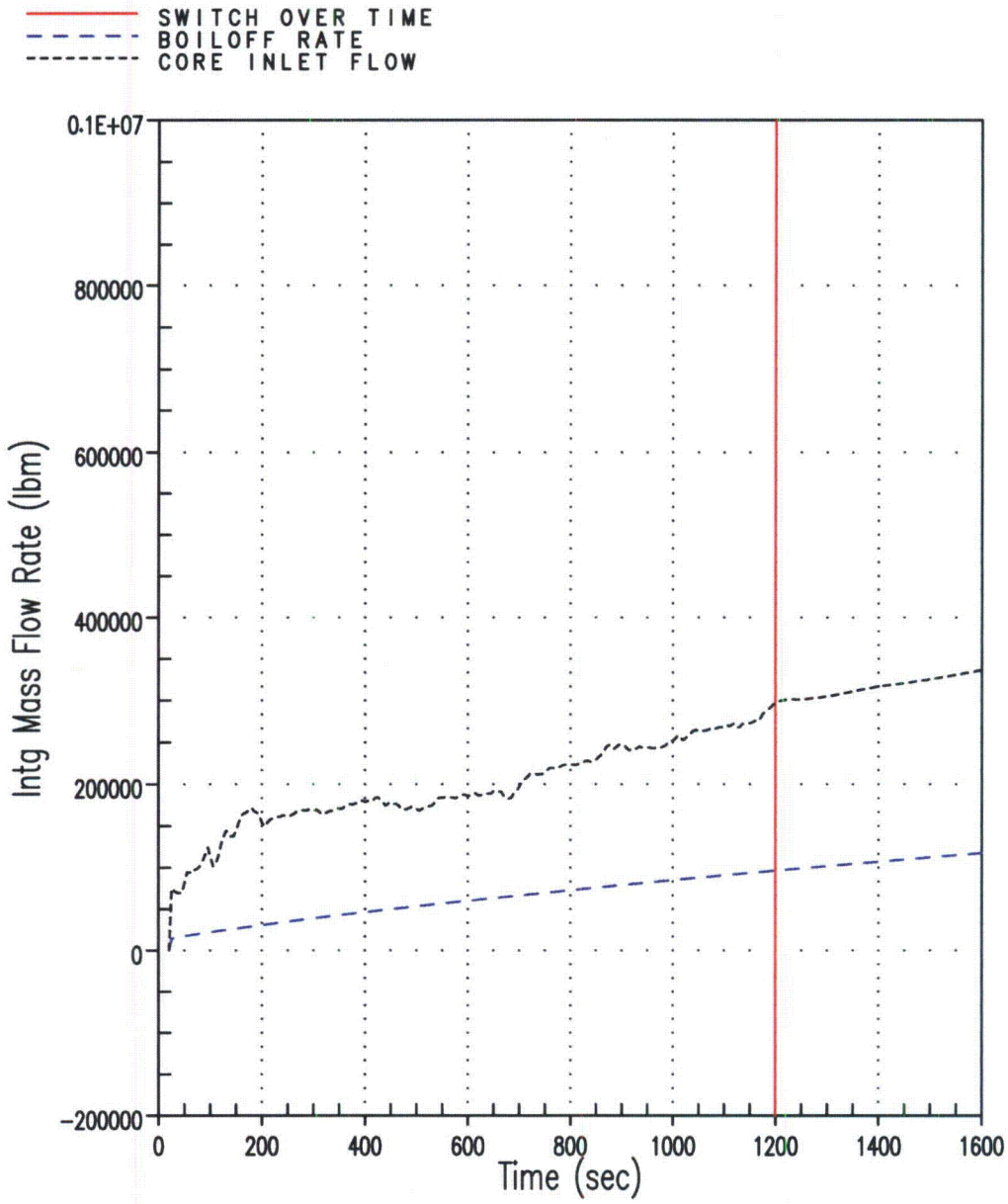


Figure B-34 Core Pressure Drop for Channel 13 Flow Reduction 80% Case



221025277

Figure B-35 Integrated Core Flow vs. Core Boil-off for Uniform C_D = 50,000 Case

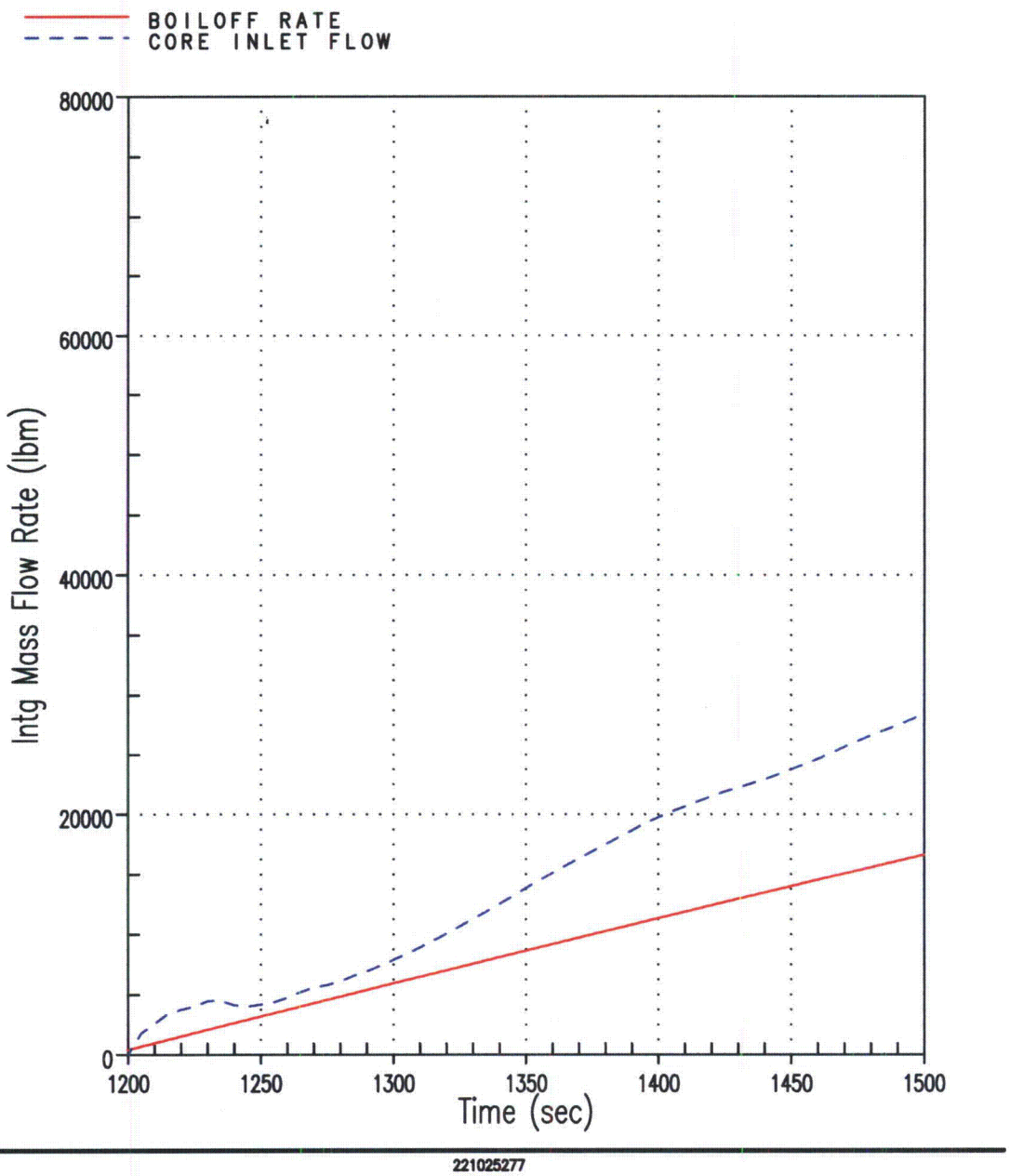


Figure B-36 Integrated Core Flow vs. Core Boil-off for Uniform C_D = 50,000 Case (Shifted Scale)

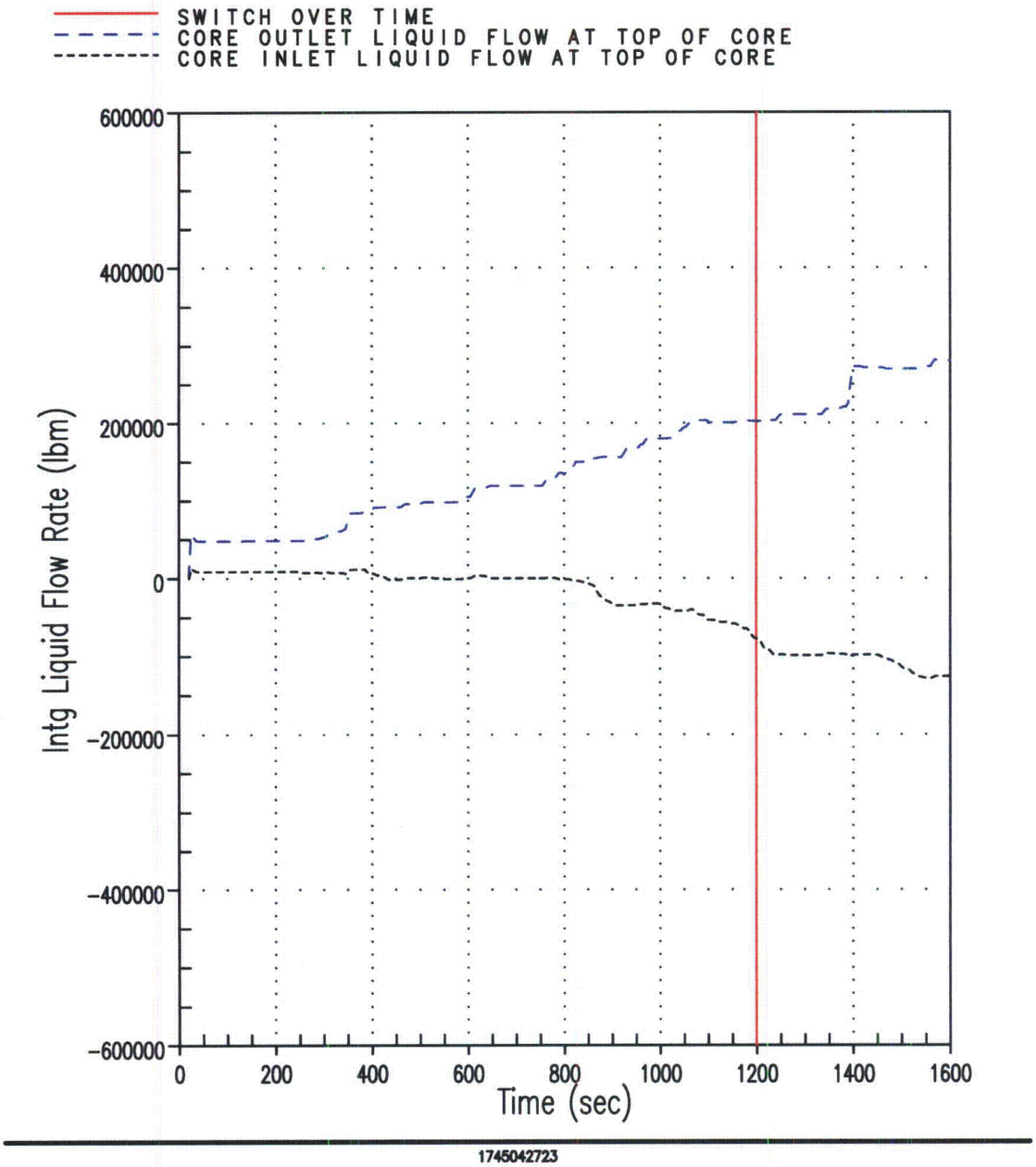


Figure B-37 Total Integrated Liquid Flow at the Top of the Core for Uniform C_D = 50,000 Case (Positive/Outlet flow represents HA, GT, AVG channels; Negative/Inlet flow represent LP channel)

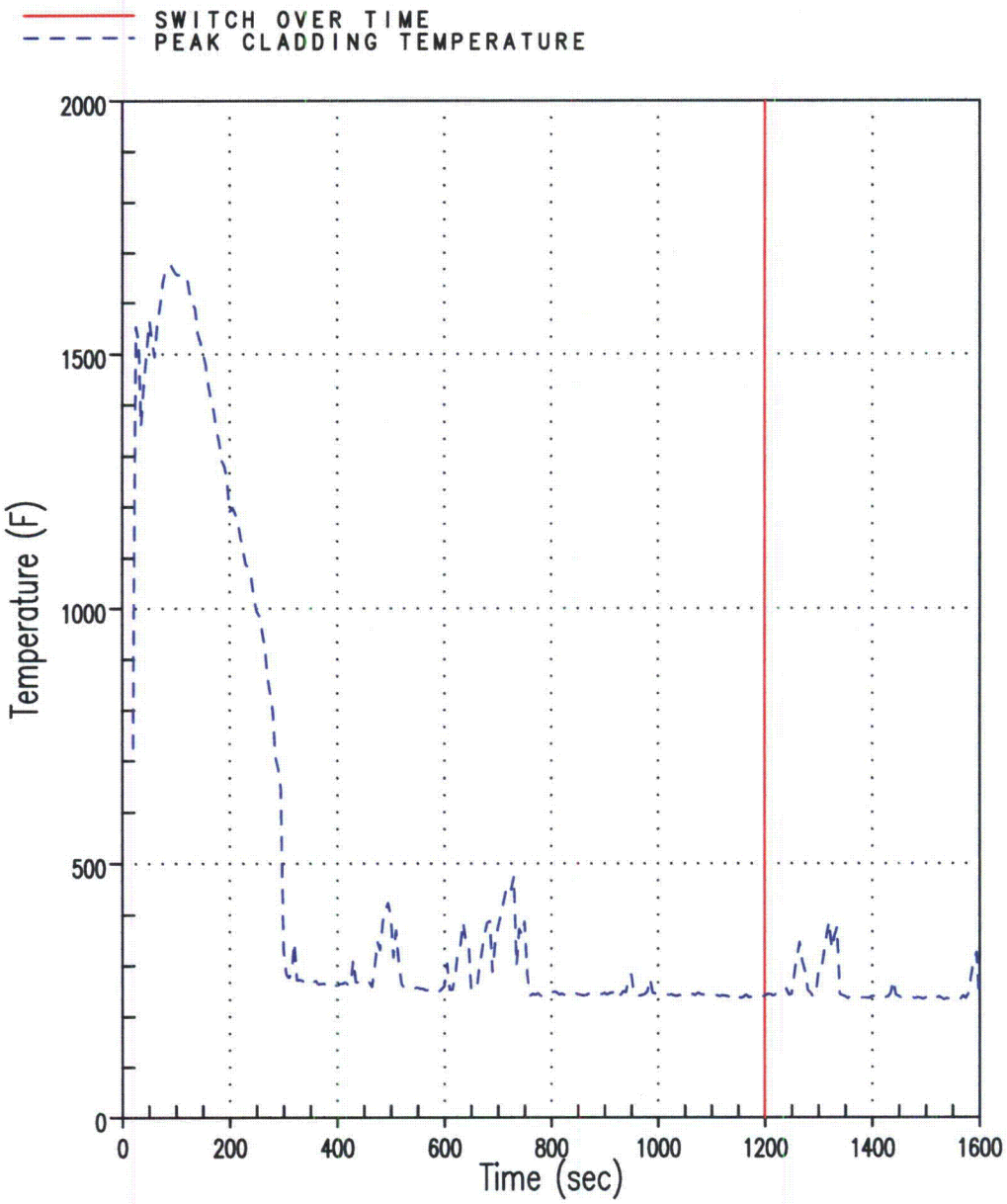


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Figure B-38 Hot Rod PCT for Uniform C_D = 50,000 Case

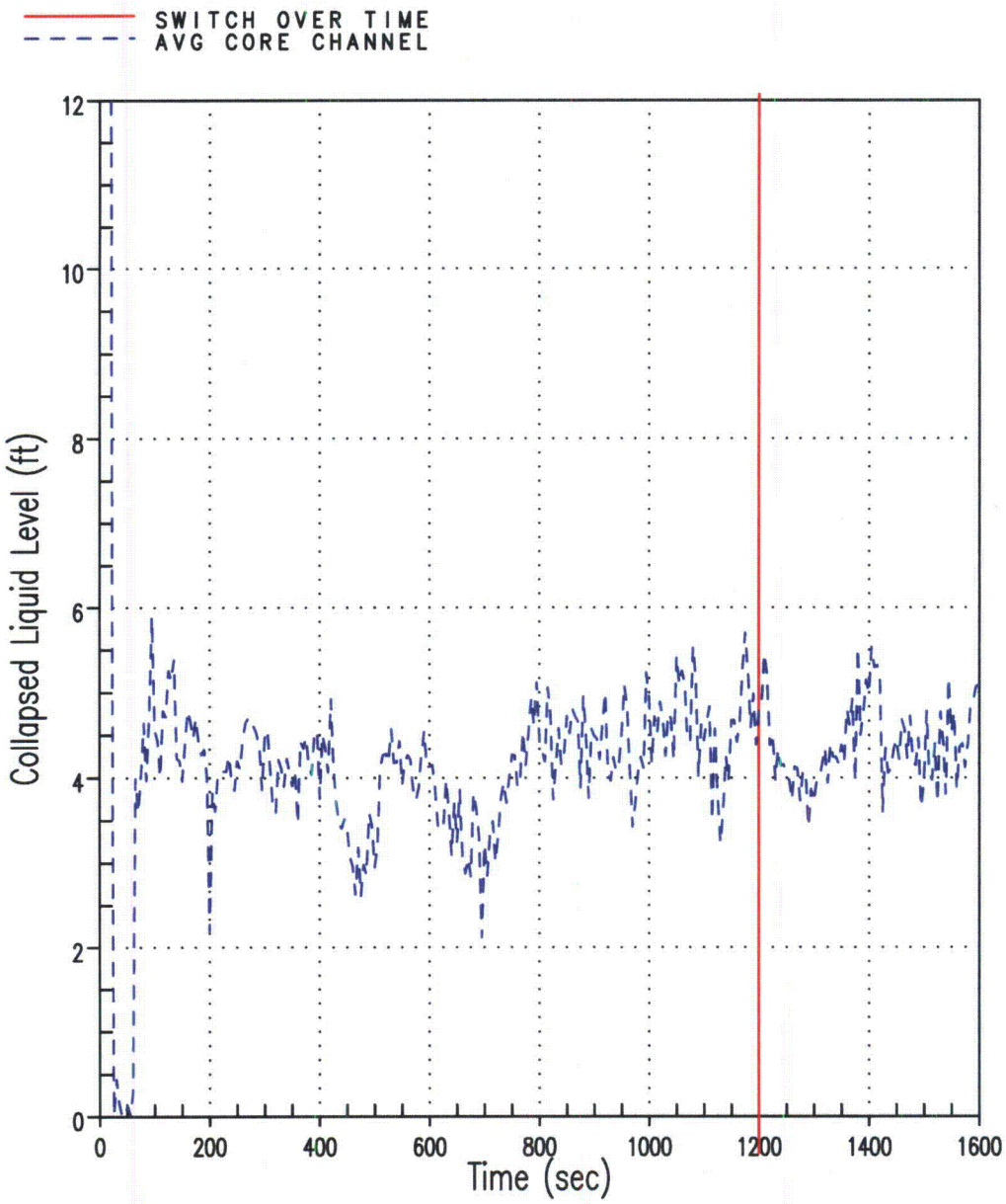

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Figure B-39 Average Core Channel Collapsed Liquid Level for Uniform C_D = 50,000 Case

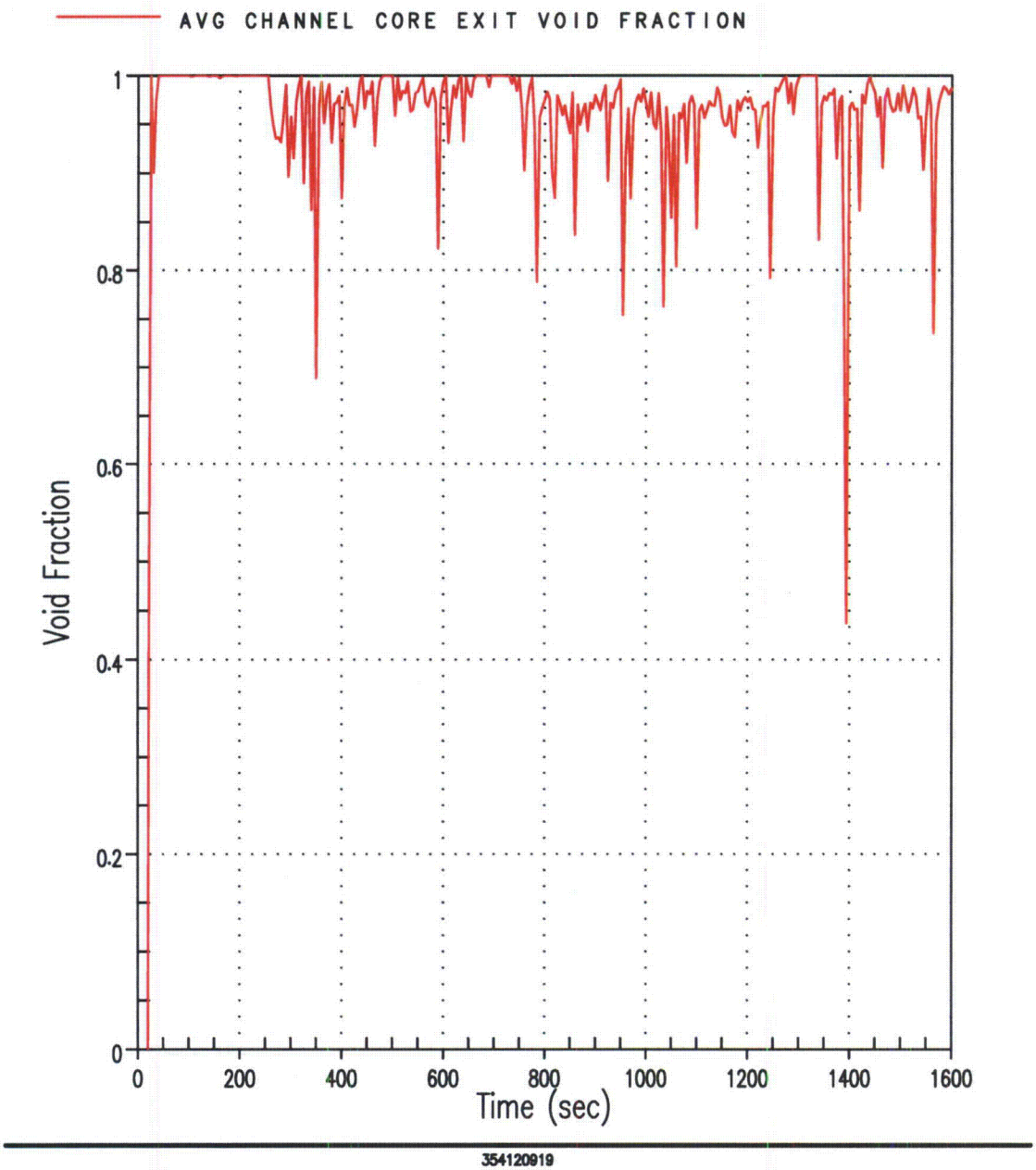

Figure B-40 Void Fraction at the Exit of the Average Core Channel for Uniform C_D = 50,000 Case

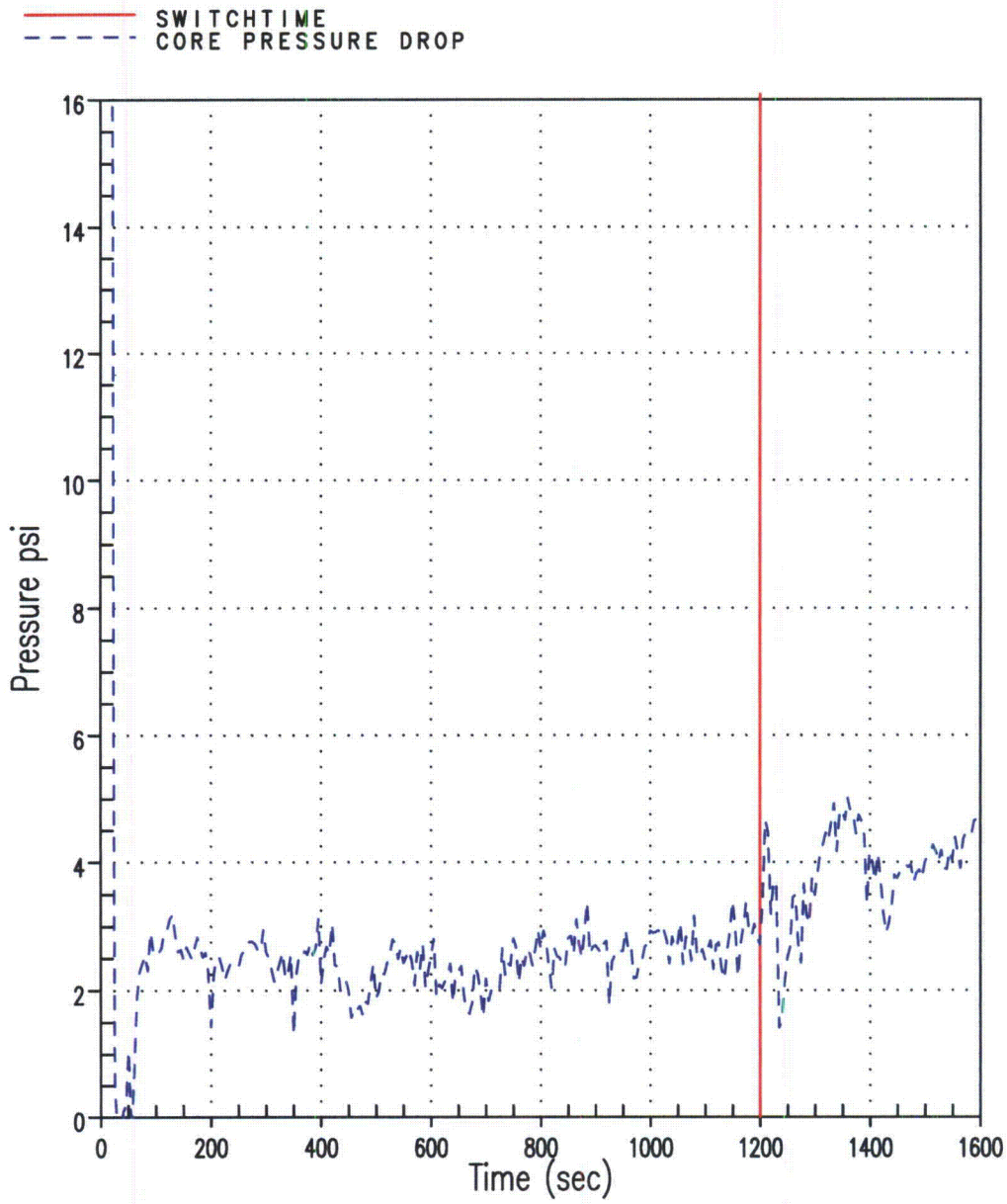

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Figure B-41 Core Pressure Drop for Uniform C_D = 50,000 Case

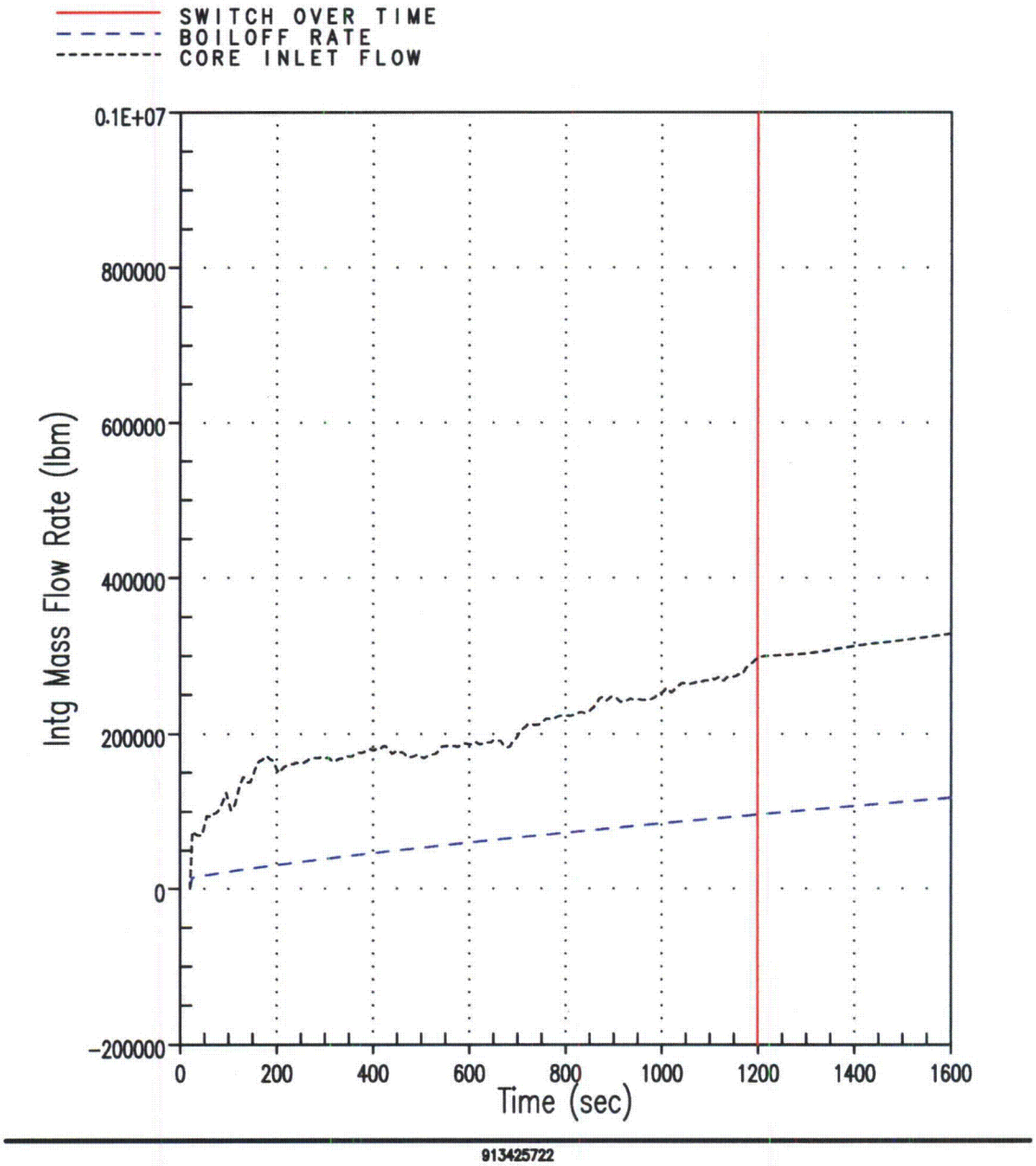
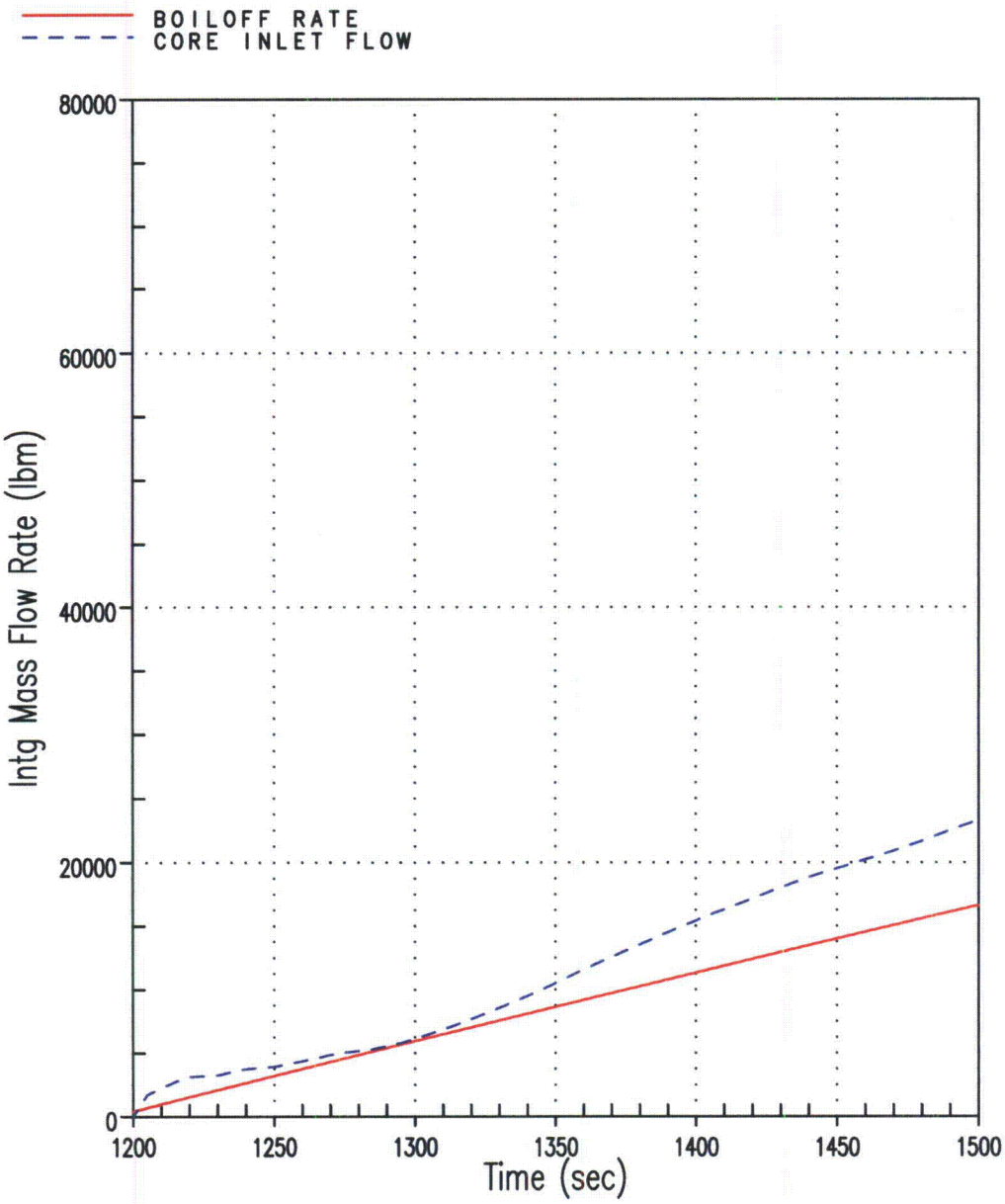


Figure B-42 Integrated Core Flow vs. Core Boil-off for Uniform C_D = 100,000 Case



913425722

Figure B-43 Integrated Core Flow vs. Boil-off for Uniform C_D = 100,000 Case (Shifted Scale)

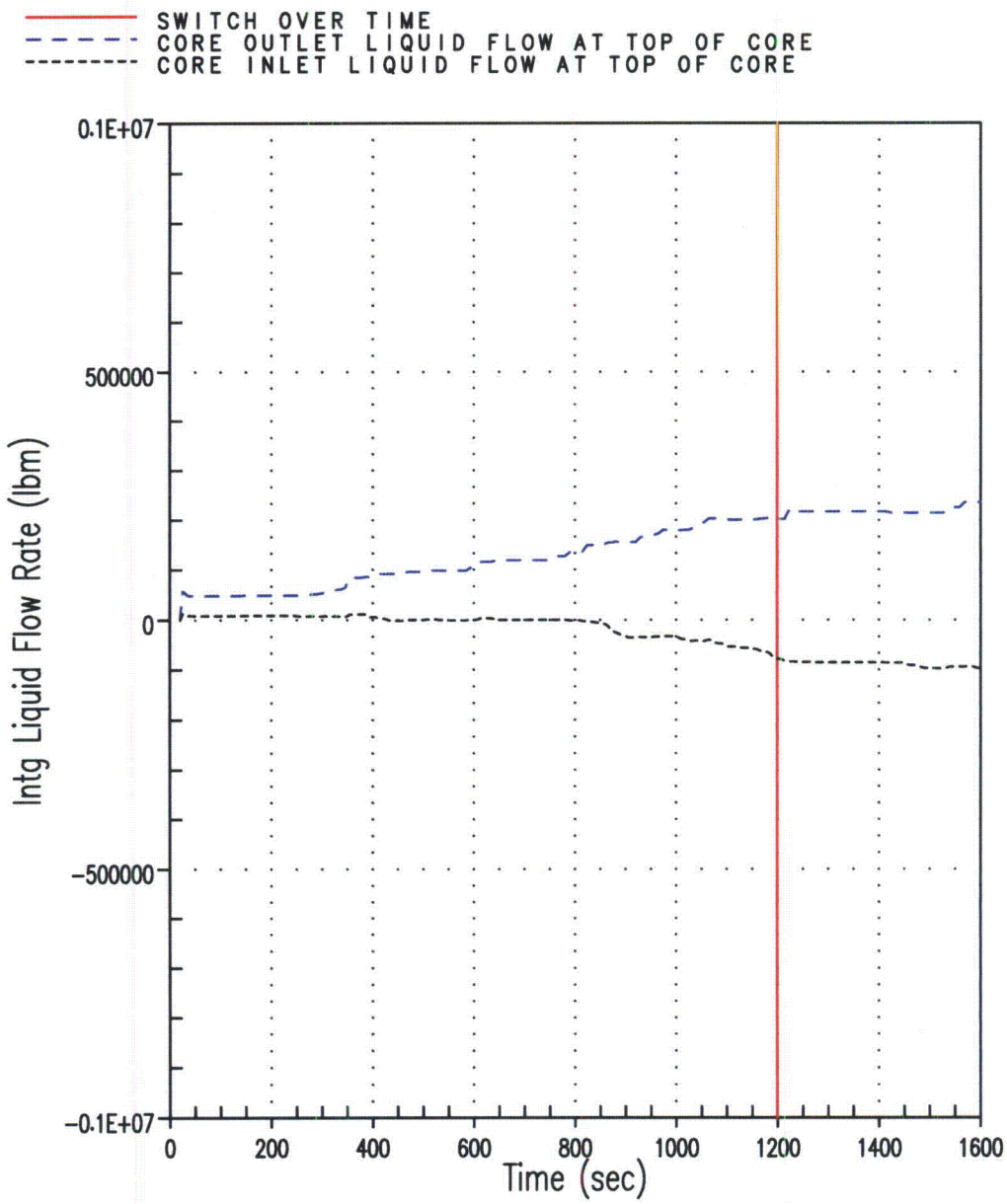


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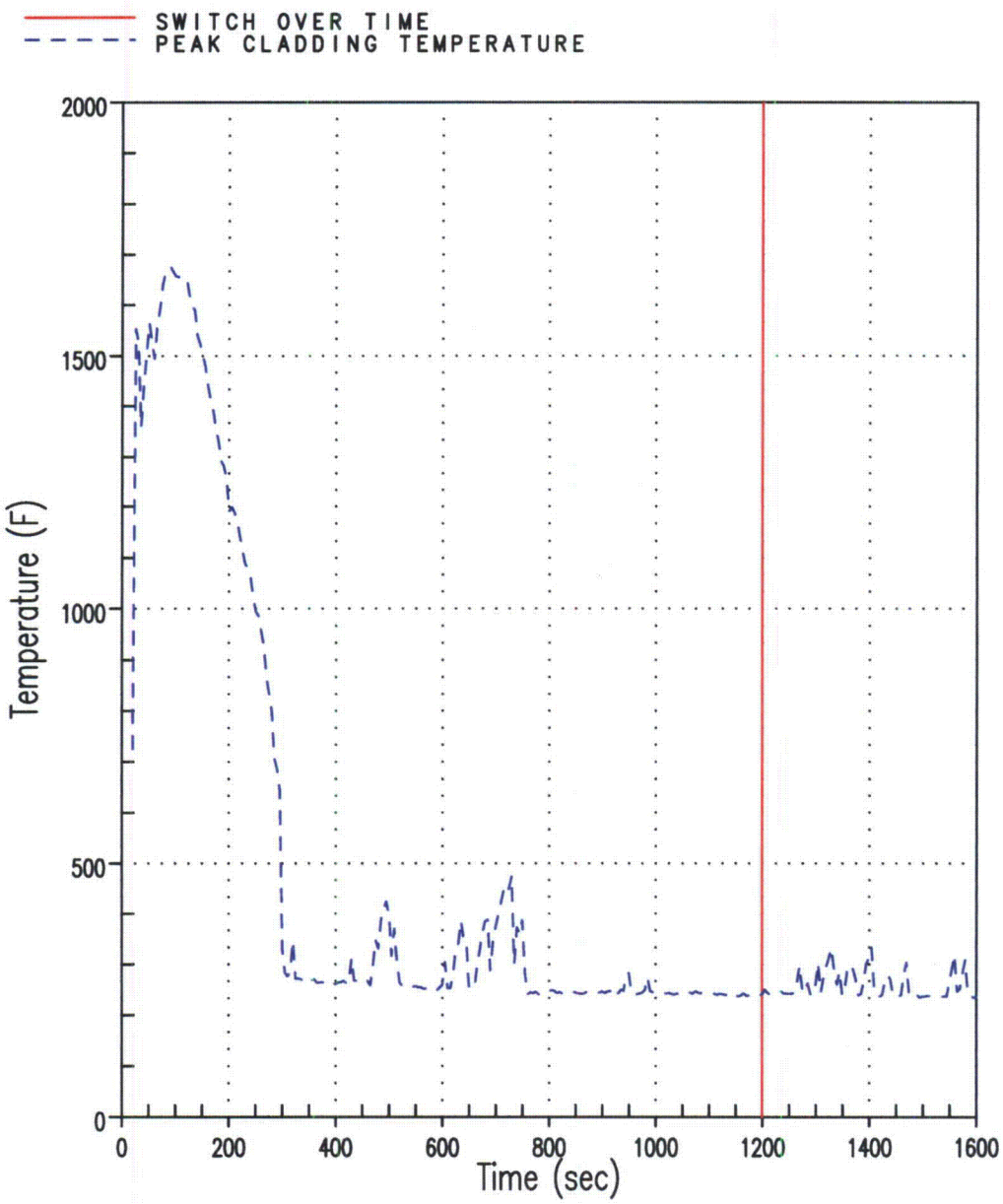
Figure B-44 Total Integrated Liquid Flow at the Top of the Core for Uniform C_D = 100,000 Case (Positive/Outlet flow represents HA, GT, AVG channels; Negative/Inlet flow represent LP channel)



2013607248

Figure B-45 Hot Rod PCT for Uniform C_D = 100,000 Case

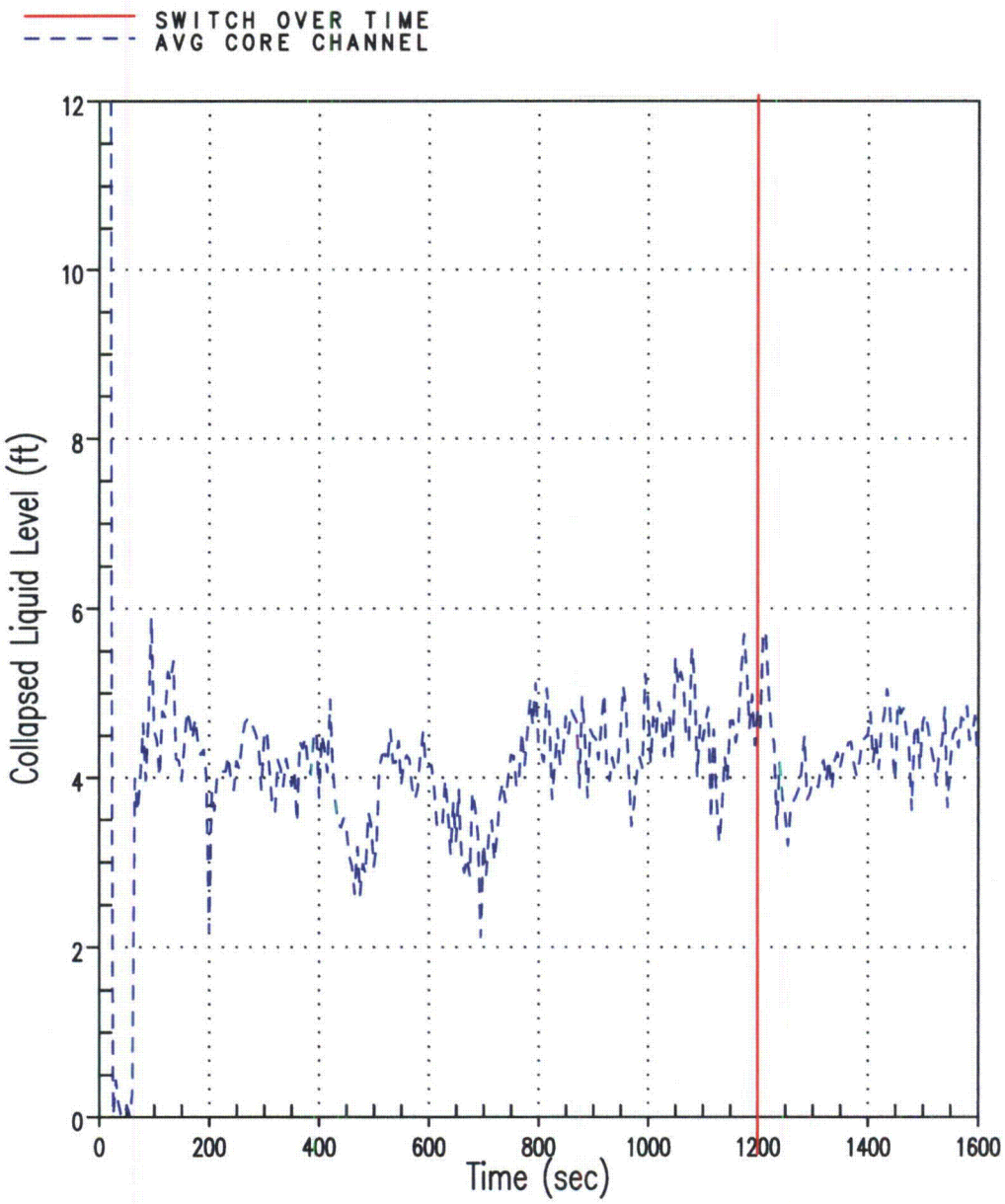


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Figure B-46 Average Core Channel Collapsed Liquid Level for Uniform C_D = 100,000 Case

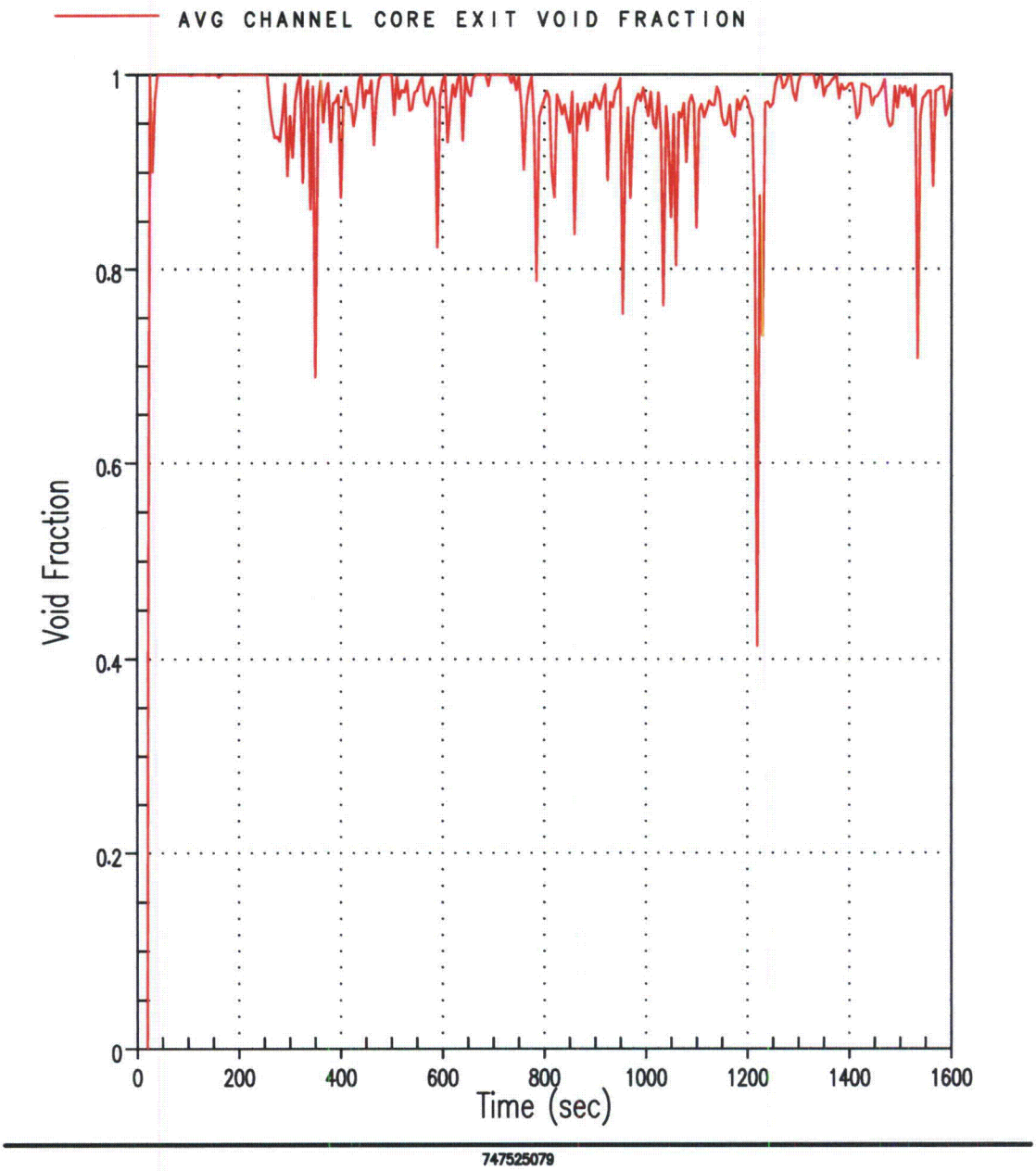
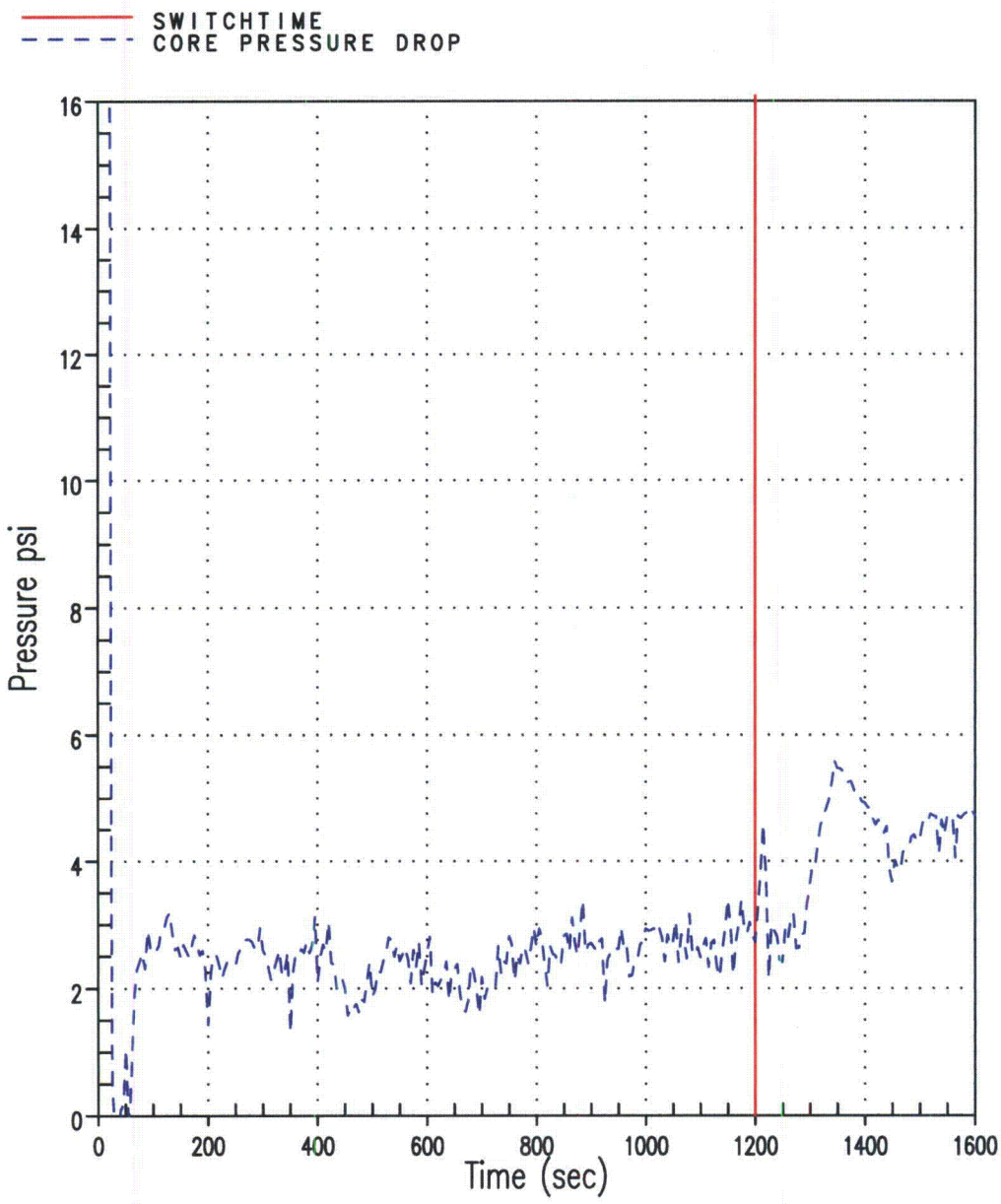


Figure B-47 Void Fraction at Exit of Average Core Channel for Uniform C_D = 100,000 Case



1552950110

Figure B-48 Core Pressure Drop for Uniform C_D = 100,000 Case

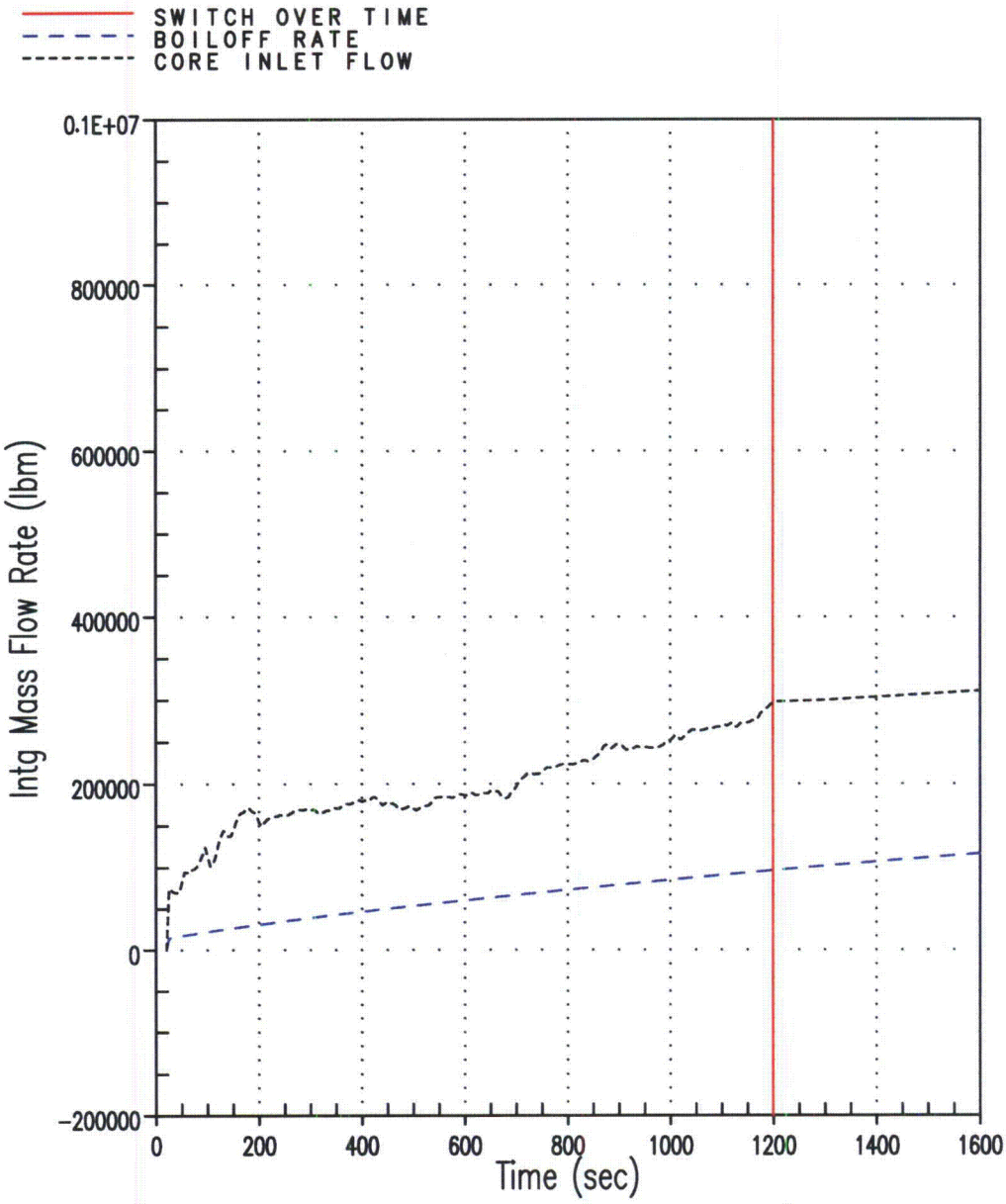


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Figure B-49 Integrated Core Flow vs. Boil-off for Uniform C_D = 1,000,000 Case

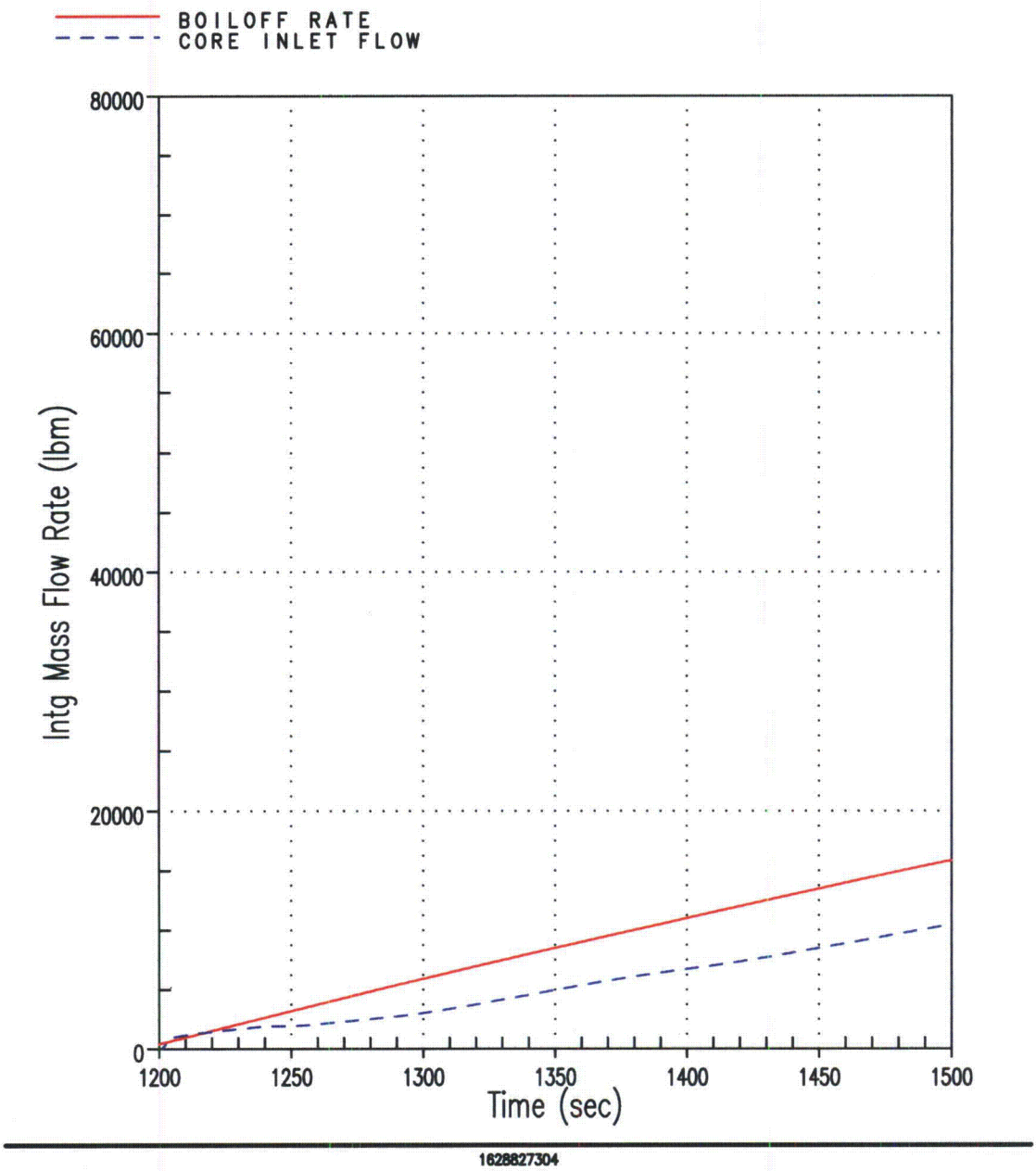
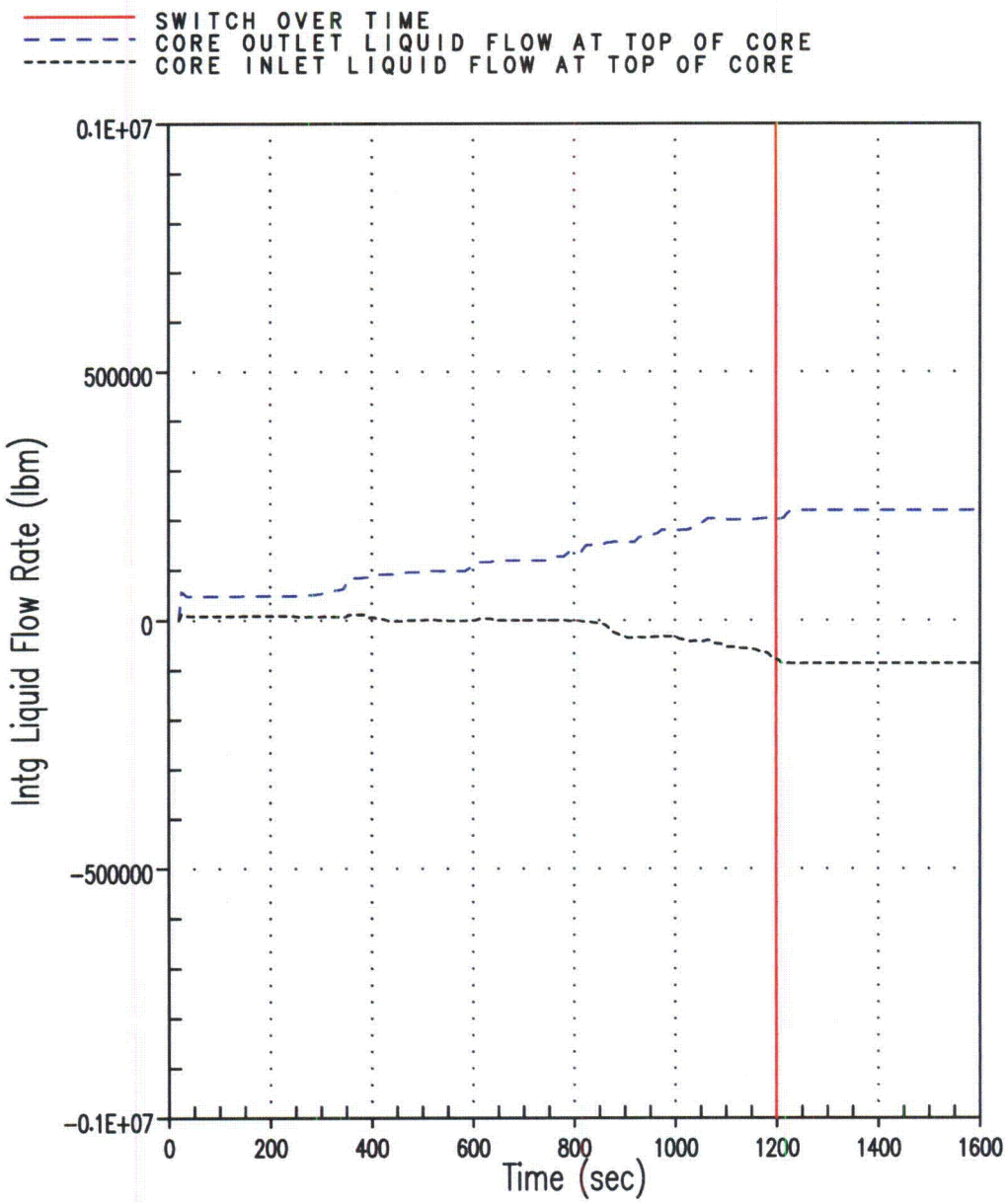
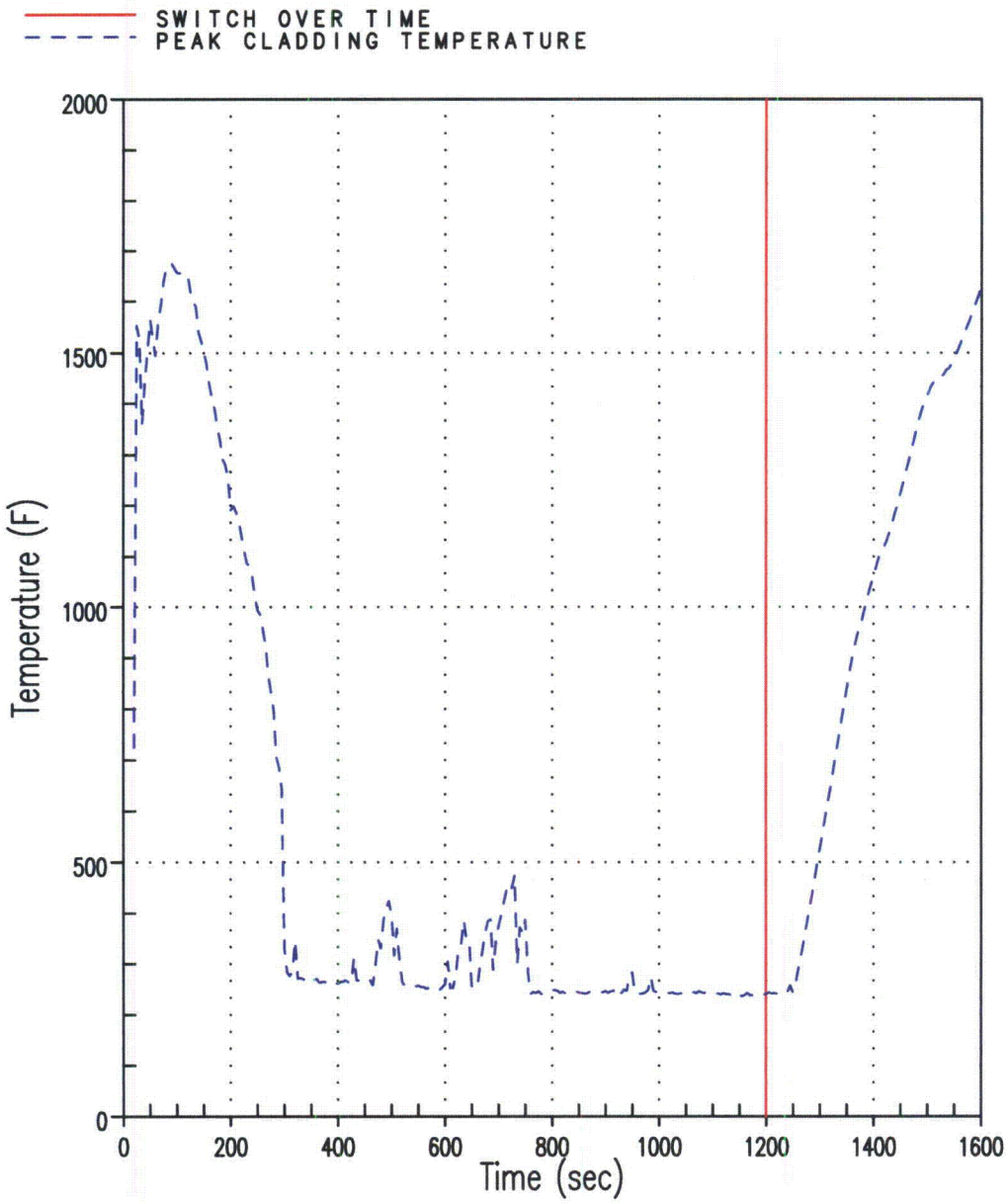


Figure B-50 Integrated Core Flow vs. Boil-off for Uniform C_D = 1,000,000 Case (Shifted Scale)



973451765

Figure B-51 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)



566586943

Figure B-52 Hot Rod PCT for Uniform C_D = 1,000,000 Case

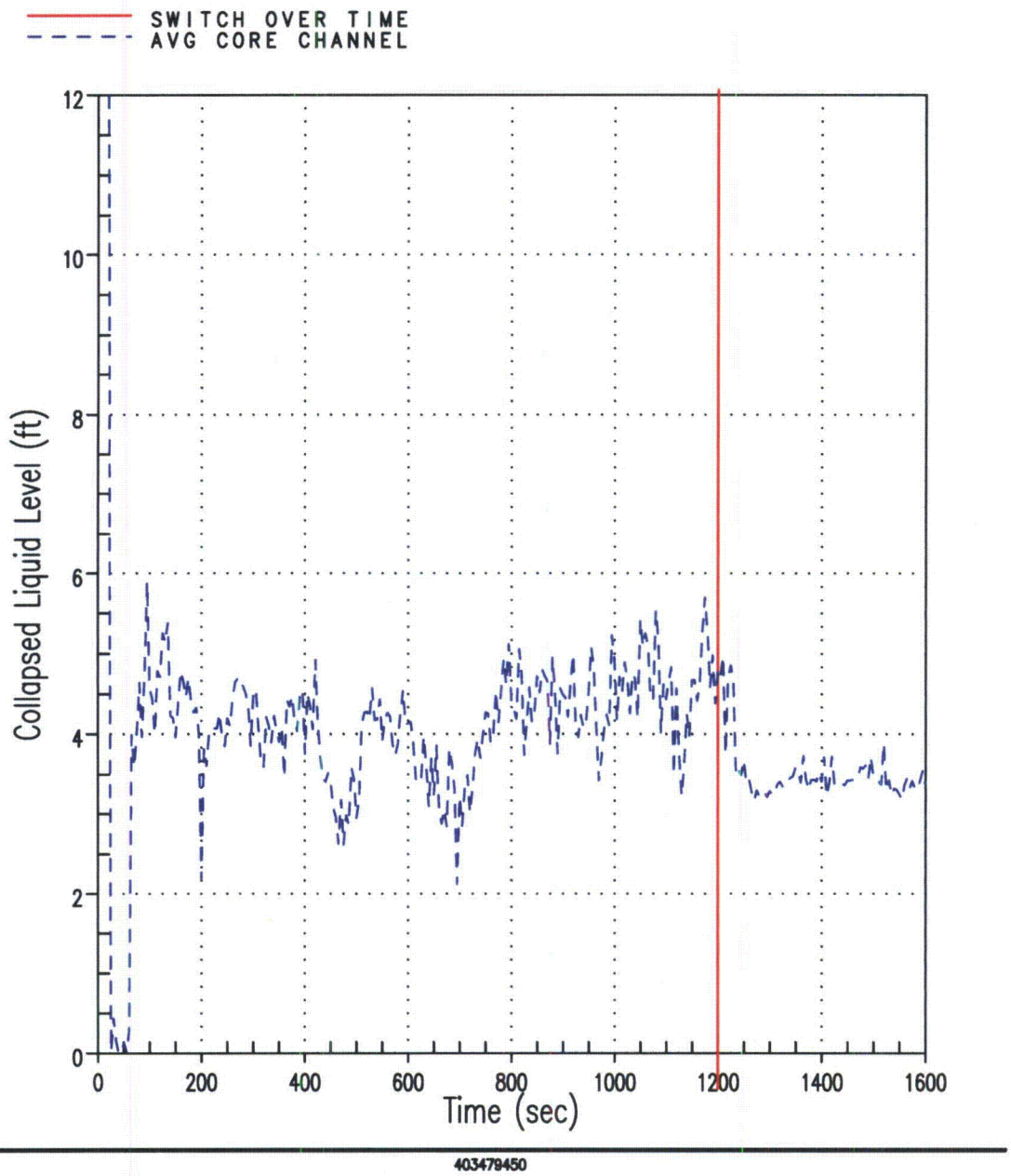


Figure B-53 Average Core Channel Collapsed Liquid Level for Uniform C_D = 1,000,000 Case

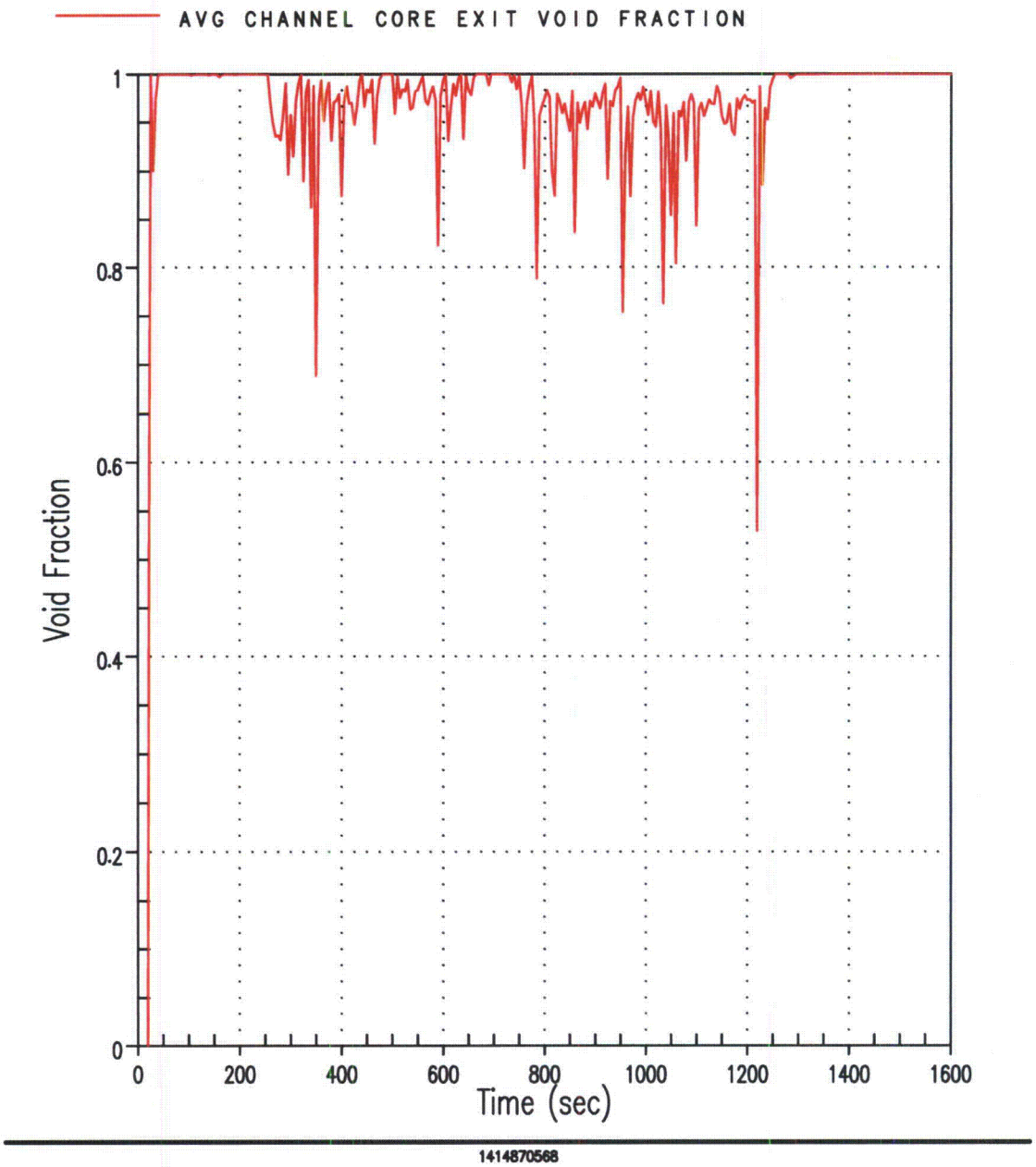


Figure B-54 Void Fraction at Exit of Average Core Channel for Uniform C_D = 1,000,000 Case

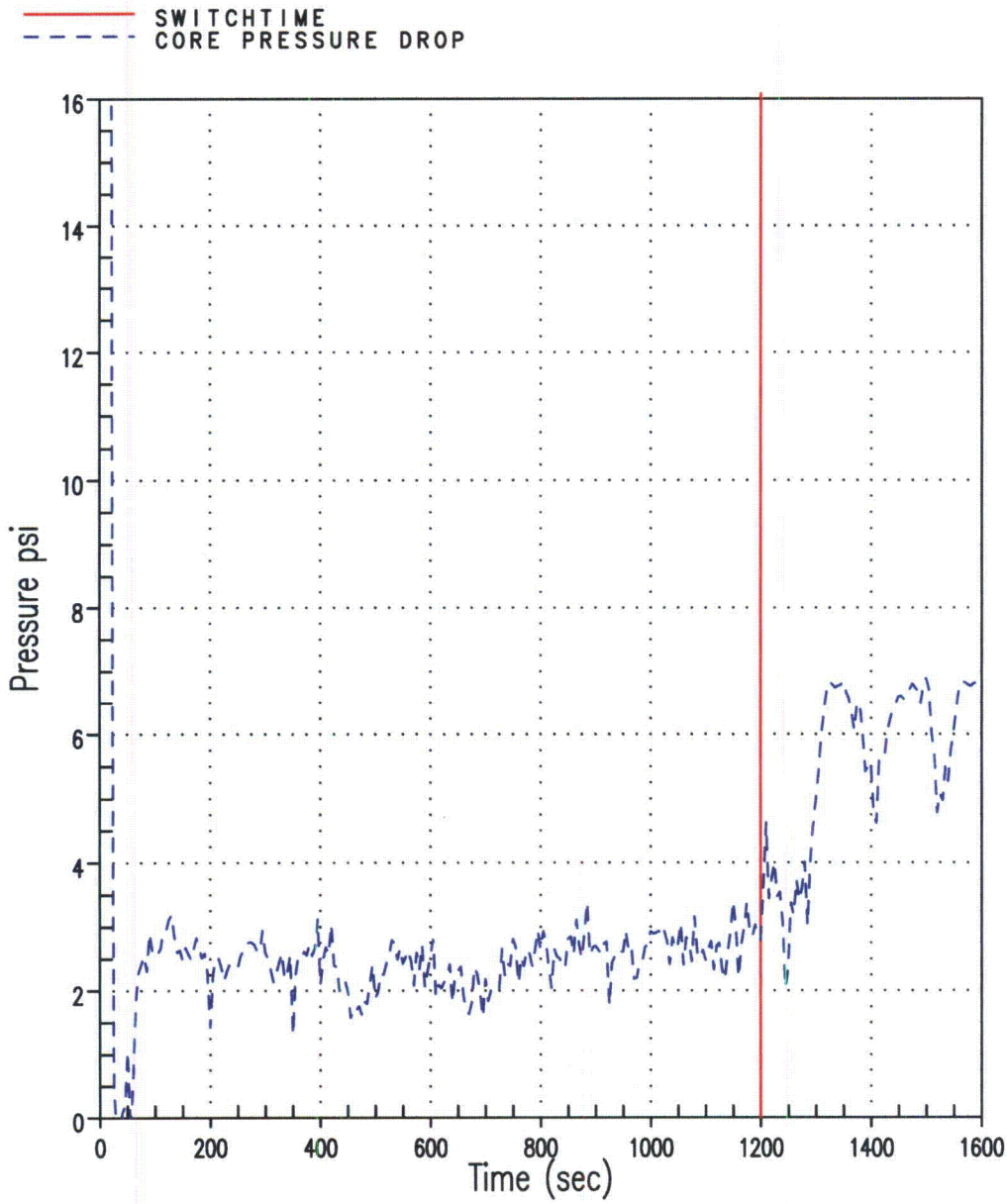


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Figure B-55 Core Pressure Drop for Uniform C_D = 1,000,000 Case

B.6 EFFECT OF THICK METAL STORED ENERGY ON COOLANT TEMPERATURE

The WC/T calculations were also used to evaluate the potential for ECCS coolant to absorb thermal energy from the thick metal reactor vessel components during recirculation of coolant from the containment building sump. This was done to assess the potential effect of changes on the recirculating coolant temperature on solubility limits of post-accident chemical products.

The thermal energy stored in the thick RV shell and the RV B/B is small, as demonstrated in the following discussion, 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 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 CL 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 CL 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 HL 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 ECCS is realigned to draw suction from the reactor containment building sump from the 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 WC/T calculations reported in Section B-3, 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 B-56, Comparison of Reactor Vessel Metal Temperature at Bottom of Fuel; Outside Diameter versus Inside Diameter, and Figure B-57, 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 WC/T calculations for a postulated CL break. From Figures B-56 and B-57, 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 B-58, 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 B-59, 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 B-60, 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 B-61 and B-62, 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 CL 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 CL or HL 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 HL break. The thickness of a RV 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 RV thick metal during recirculation is due to the convective resistance between the RV surface and the fluid.

The stainless steel cladding on the inside of the RV 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 WC/T calculations for a postulated CL 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.

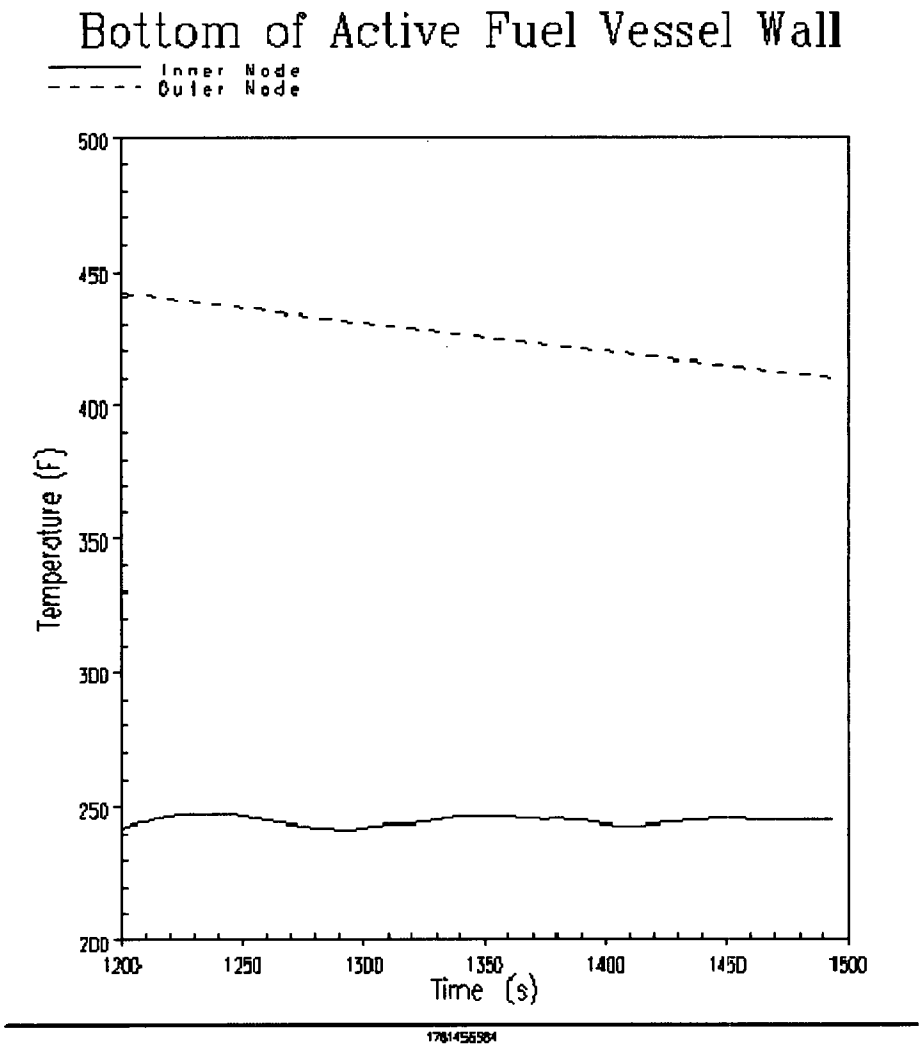

Figure B-56 Comparison of Reactor Vessel Metal Temperature at Bottom of Fuel; Outside Diameter versus Inside Diameter

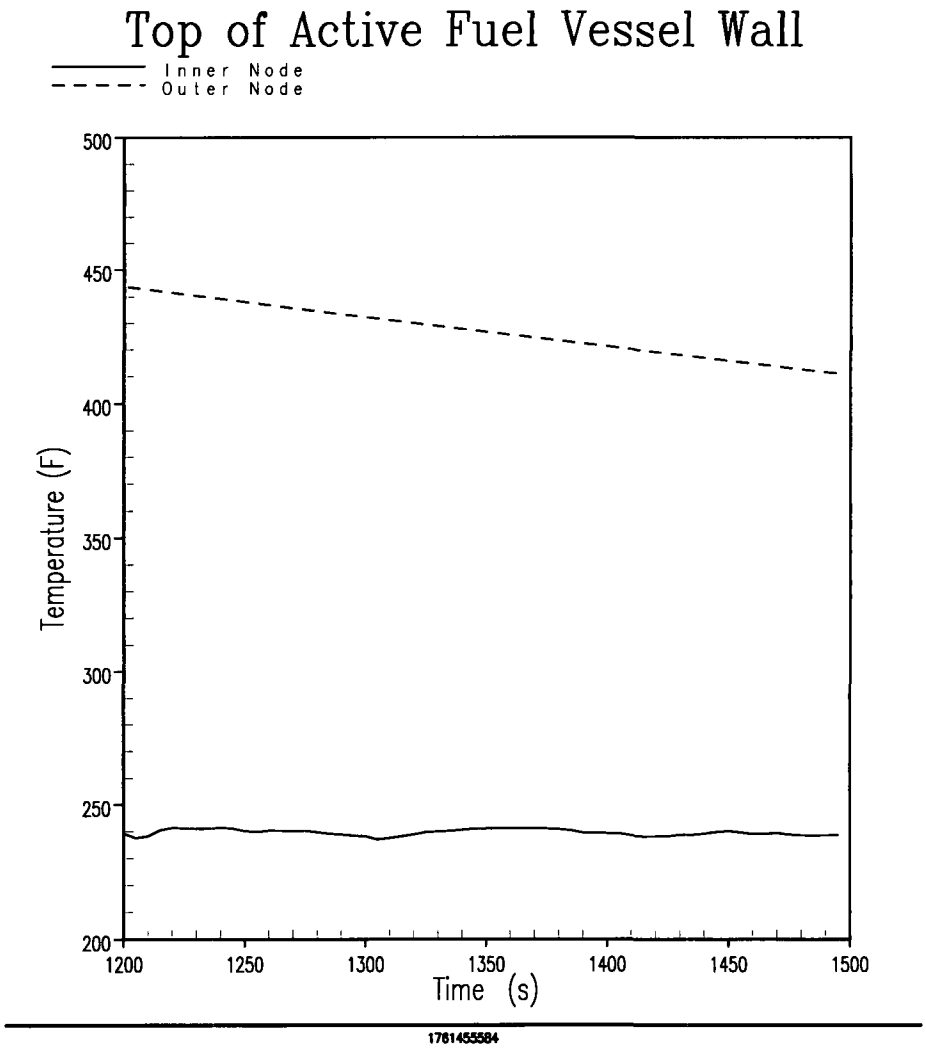

Figure B-57 Comparison of Reactor Vessel Metal Temperature at Top of Fuel; Outside Diameter versus Inside Diameter

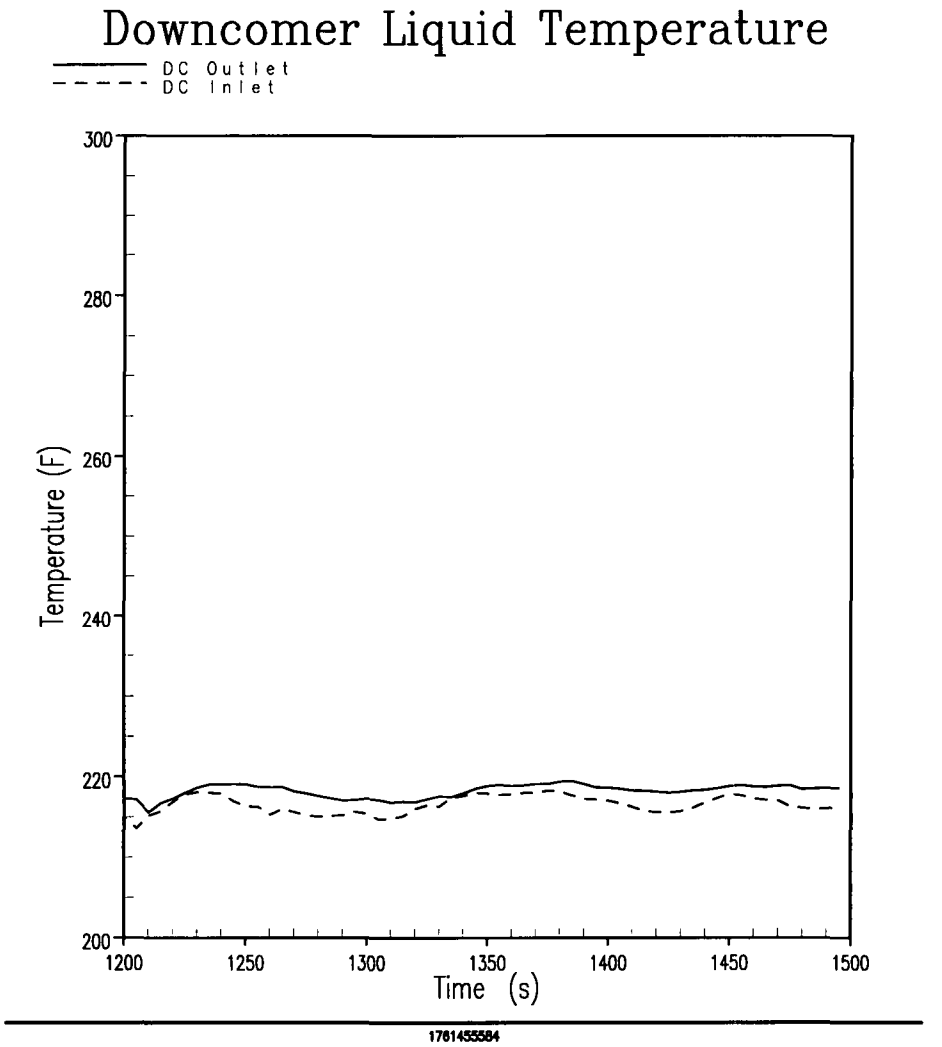

Figure B-58 Comparison of Fluid Temperature at Top and Bottom of Downcomer

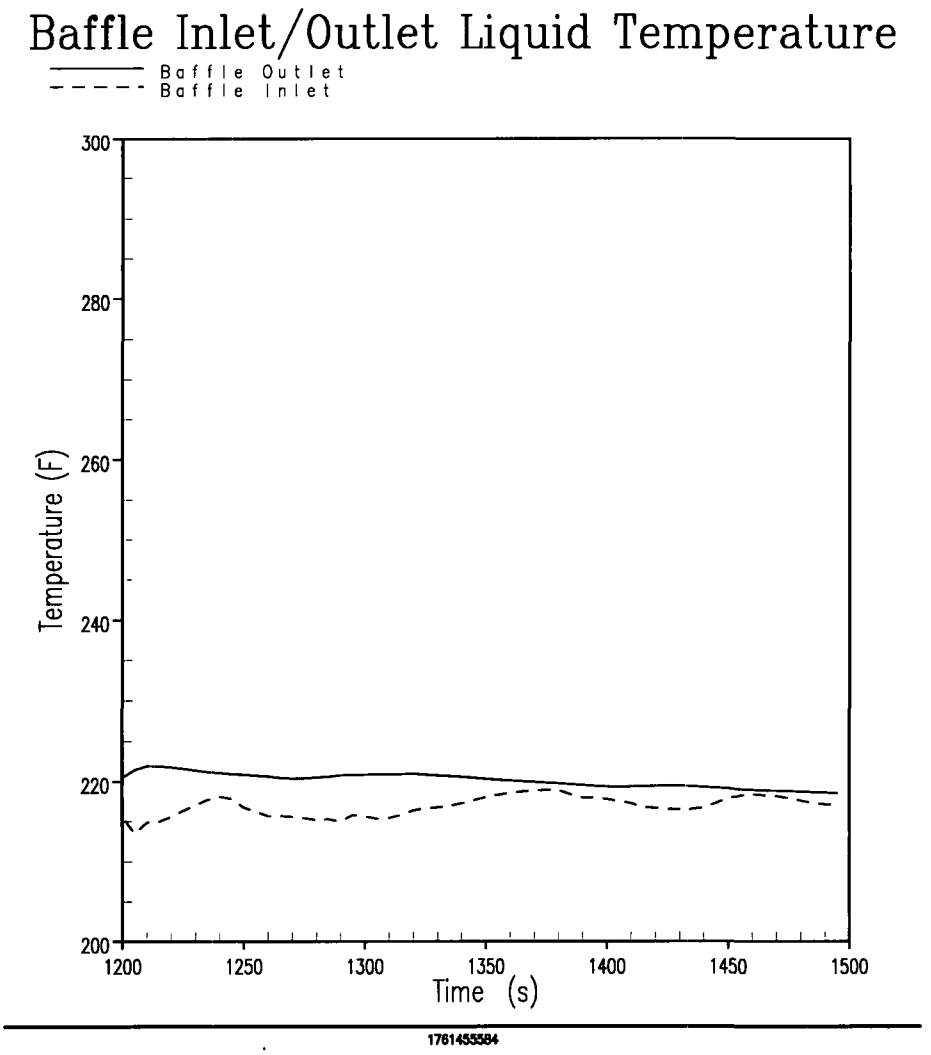

Figure B-59 Comparison of Fluid Temperature at Top and Bottom of Baffle

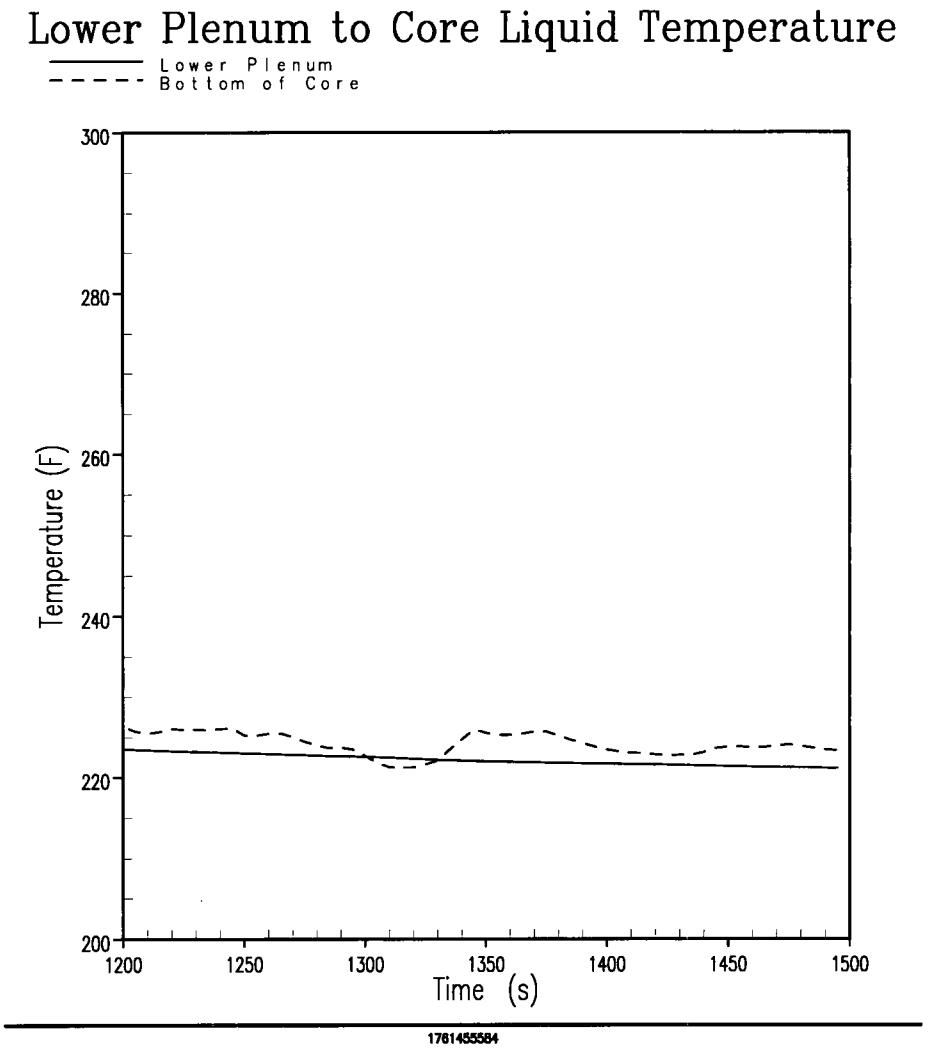

Figure B-60 Comparison of Fluid Temperature in Lower Plenum to Core Inlet

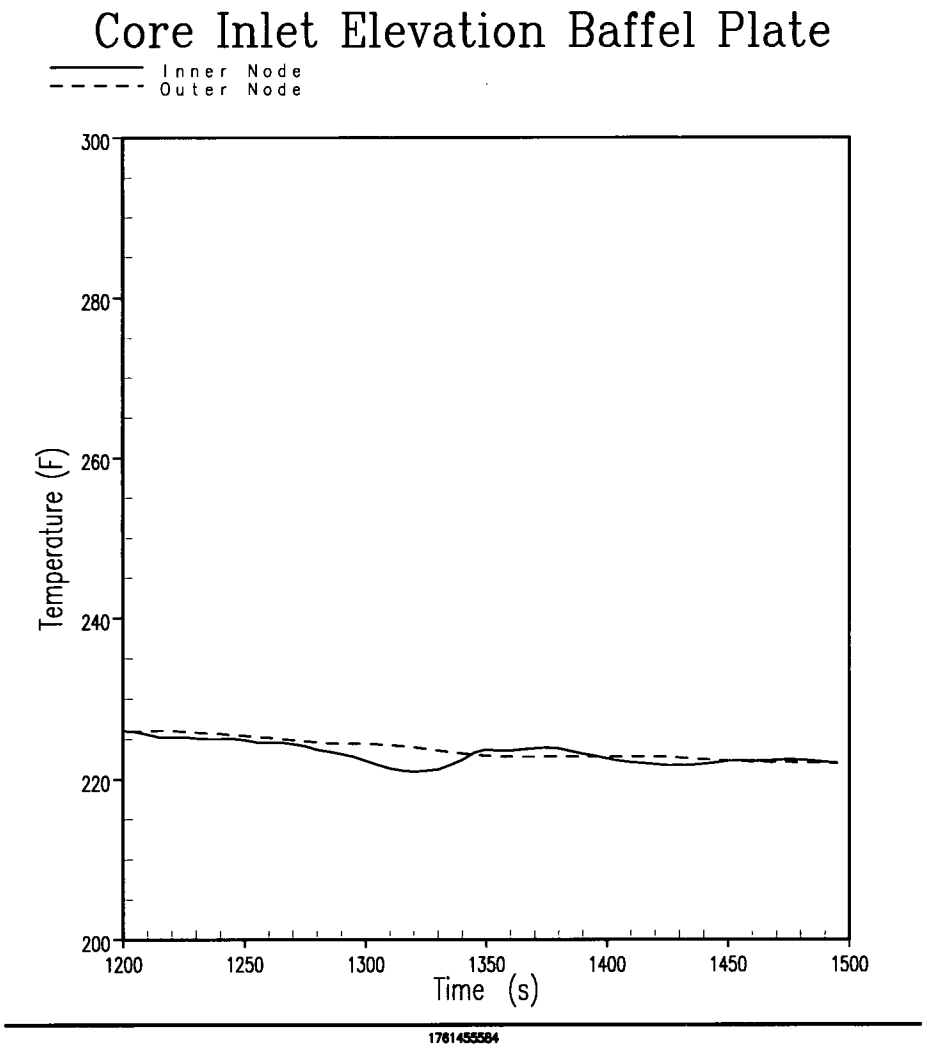

Figure B-61 Comparison of Fluid Temperature Between Core Inlet and Inside Baffle

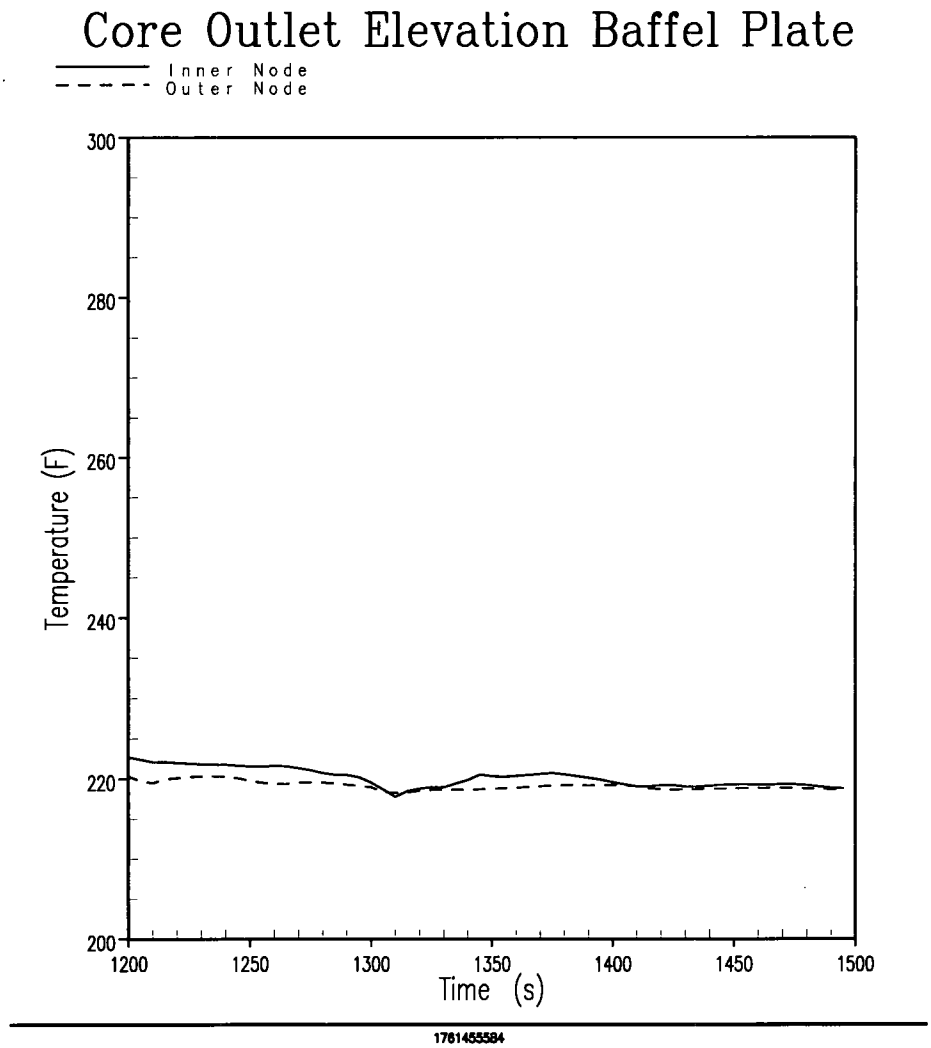

Figure B-62 Comparison of Fluid Temperature at Core Outlet and Top Baffle

B.7 REFERENCES

B-1. Thermophysical Properties of Fluid Systems, http://webbook.nist.gov/chemistry/fluid/, April 11, 2007.

APPENDIX C
FUEL CLAD HEAT-UP UNDERNEATH GRIDS

C.1 INTRODUCTION

In an extreme case, it has been postulated that the volume between the fuel rod and spacer grid could completely fill with debris. An evaluation was performed to determine the cladding surface temperature of a fuel rod within a fuel grid when the rod is plated with debris in a post-LOCA recirculation environment. A parametric study was performed to show the effects on the maximum temperature of the fuel rod within the spacer grid caused by varying debris thickness and the thermal conductivity of the debris. A detailed discussion of this calculation, including a discussion of assumptions and boundary conditions, is presented here.

C.2 ANSYS DISCUSSION

The ANSYS mechanical software was used for the cladding heat up behind fuel grids. This software is in common use internationally to solve a wide range of mechanical engineering problems. The use of the thermal analysis capability for this model is in accordance and consistent with standard industry practices for ANSYS mechanical software and other similar engineering problem solving software.

The ANSYS mechanical software offers a comprehensive product solution for structural linear/nonlinear and dynamics analysis. The product offers a complete set of elements behavior, material models, and equation solvers for a wide range of engineering problems. In addition, ANSYS offers thermal analysis and coupled-physics capabilities involving acoustic, piezoelectric, thermal-structural, and thermal-electric analysis. For the cladding heat up calculations, only the thermal solution capabilities of the ANSYS mechanical software were used.

C.3 METHOD

A model of a single fuel rod was created in Solidworks® and imported into ANSYS. The model was cut down to a "1 quarter pie piece" in order to reduce the size of the model to be analyzed, while maintaining symmetry. In order to conservatively model the convection, the clad was divided into 20 zones. To preserve accuracy of this model, no convection was assumed to occur at the planes of symmetry. This was done to ensure that the convection was modeled only on the outer surface of the fuel rod assembly, and not on the surface of the cladding under the grids and the debris. A similar technique of dividing one large portion of the modeled fuel rod into multiple smaller segments was used to simulate the layers of debris in the runs using a thin debris layer (10 mils and under) in order to allow ANSYS to more easily generate a mesh. Once in ANSYS, all bodies in the model were meshed using a refined mesh size of 0.05, in order to create a finer mesh while still allowing the program to complete the analysis in a reasonable amount of time.

After the mesh was generated, a constant heat flux was set on the entire inner surface of the cladding, and convection heat transfer, with a constant convection coefficient set on the entire outer surface of the rod assembly. The mission time as defined in WCAP-16406-P-A (Reference C-1) is 30 days. In order to allow ANSYS to accurately model the simulation, the minimum time step was set to 10 seconds, but the

maximum time step was set to 400 hours. This allowed the simulation to speed up after steady state conditions had been reached, but still ensured that the final conditions had been achieved, without having to perform multiple runs for each scenario. Each model, with the exception of the clean rod, was run four times to collect data for debris thermal conductivities of 0.3, 0.5, and 0.9 $\left(\frac{\text{BTU}}{\text{hr} * \text{ft} * {}^{\circ}\text{F}}\right)$. A new model was also generated for each debris thickness, from 5 to 50 mils.

C.4 MODEL INPUTS

The values in the Tables C-1 through C-4 are taken from the WC/T model described in Appendix B, thus providing for consistency between this calculation and the core inlet blockage calculation. Table C-1 summarizes the physical dimensions of the fuel rod model. Table C-2 lists the location of the standard and mixing vane grids modeled in this calculation. Table C-3 identifies the location of spacer and mixing vane grids along the length of the fuel rod.

Table C-1 General Fuel Rod Dimensions

Dimension	Value
Outer Cladding Diameter (in)	0.36
Cladding Thickness (in)	0.0225
Rod Length (in)	144
Grid Thickness (in)	0.018
Large Grid Length (in)	2.25
Small Grid Length (in)	0.475

Table C-2 Grid Locations

Grid Type	Elevation from Base (in)
Large	24.57
Large	45.07
Large	65.67
Small	76.77
Large	86.17
Small	97.37
Large	106.77
Small	117.87
Large	127.27

Table C-3 Fuel Rod Zone Description

Zone	Begin (in)	End (in)	Description	Debris
1	0.000	24.570	Clad – 1	No
2	24.570	26.820	Large Grid – 1	No
3	26.820	45.070	Clad – 2	No
4	45.070	47.320	Large Grid - 2	No
5	47.320	65.670	Clad – 3	No
6	65.670	67.920	Large Grid – 3	No
7	67.920	76.770	Clad – 4	No
8	76.770	77.245	Small Grid – 1	No
9	77.245	86.170	Clad – 5	No
10	86.170	88.420	Large Grid – 4	No
11	88.420	96.000	Clad – 6a *	No
12	96.000	97.370	Clad – 6b *	Yes
13	97.370	97.845	Small Grid – 2	Yes
14	97.845	106.770	Clad – 7	Yes
15	106.770	109.020	Large Grid – 5	Yes
16	109.020	117.870	Clad – 8	Yes
17	117.870	118.345	Small Grid – 3	Yes
18	118.345	127.270	Clad – 9	Yes
19	127.270	129.520	Large Grid – 6	Yes
20	129.520	144.000	Clad – 10	Yes

* The clad modeled in Section 6 was segregated into two parts, Zone 11 and Zone 12. This was done to provide for the simulation of oxide, crud and/or chemical product deposition over the fuel rod elevation extending from 96.000 in. to 144.000 in. Therefore, Zone 12 is modeled with a layer of material (oxide, crud, and/or chemical product deposition), while Zone 11 is modeled as a clean-surface fuel rod.

Table C-4 lists the average values of the thermal hydraulic boundary conditions used for the single fuel rod heat-up calculations. These values were taken from the WC/T output for the LOCA simulation at 1200 seconds after the initiation of the transient from Appendix B; the time of switchover from injection from the RWST to recirculation from the containment sump. The values listed in the table are averages of the values taken at multiple points along the surface of the fuel rod surface. Table C-5 lists the thermal properties of the cladding material modeled in the WC/T analysis and were used for the fuel rod heat-up calculation described in this appendix. Table C-6 lists the range of values for the two input parameters that were varied for the calculations described in this appendix.

Table C-4 Heat Transfer Inputs	
Input	**Value**
Liquid Heat Transfer Coefficient $\left(\frac{BTU}{hr * ft * °F}\right)$	638.32
Vapor Heat Transfer Coefficient $\left(\frac{BTU}{hr * ft * °F}\right)$	17.30
Ambient Liquid Temperature (°F)	194.02
Ambient Vapor Temperature (°F)	224.95
Heat Flux, Outer Cladding Surface $\left(\frac{BTU}{hr * ft^2}\right)$	6508.93

Table C-5 Cladding Thermal Properties		
Temp (°F)	**k** $\left(\frac{BTU}{hr * ft * °F}\right)$	**Cp** $\left(\frac{BTU}{lb_M * °F}\right)$
200	7.984	0.07044
250	8.129	0.07183
300	8.274	0.07276
350	8.419	0.07356
400	8.564	0.07436
450	8.709	0.07516
500	8.854	0.07596
550	8.999	0.07676
600	9.144	0.07756
650	9.289	0.07836
700	9.434	0.07914
750	9.595	0.07979
800	9.860	0.08044
850	10.13	0.08109
900	10.39	0.08174
950	10.65	0.08239
1000	10.92	0.08303

Table C-6 Variable Ranges		
Property	**Lower Value**	**Upper Value**
Debris Thermal Conductivity $\left(\frac{BTU}{hr*ft*°F}\right)$	0.1	0.9
Debris Thickness (mils)	0	50

The accepted EPRI value for crud thermal conductivity according to EPRI Project document, "Boron-induced Offset Anomaly (BOA) Risk Assessment Tool" is $0.5\left(\frac{BTU}{hr*ft*°F}\right)$. In order to perform a sensitivity study, values of 0.1, 0.3 and 0.9 $\left(\frac{BTU}{hr*ft*°F}\right)$ were also analyzed. The crud thickness is modeled up to a maximum value of 50 mils as this bounds what the operating plants would be expected to experience.

C.5 ASSUMPTIONS

- The top third of the fuel rod (4 ft) was assumed to be covered in a layer of debris, the bottom two thirds (8 ft) was assumed to remain clean throughout the event.
- All debris that reaches the fuel rod was evenly distributed over the affected area, and was modeled with uniform thermal properties.
- The heat generated by the fuel pellets was modeled as a constant heat flux exerted on the entire inner surface of the cladding (but was based on an outer heat flux value). This assumption was made to simplify the model by removing the fuel pellets and the gap, while maintaining thermal accuracy.
- The cladding material type did not factor into this evaluation, since the value was ultimately based on the heat flux at the cladding surface.
- No convection occurred under the grids in the fuel rod assembly. This assumption was made to maintain conservatism, as the actual value will be less than the value on the surface of the assembly, but the exact value is unknown.
- Grids were assumed to have the same thermal properties as cladding.
- Debris was assumed to have the same thermal properties as crud. The accepted EPRI Value for the crud thermal conductivity was $0.5\left(\frac{BTU}{hr*ft*°F}\right)$.

C.6 RESULTS

The calculated maximum clad temperatures are summarized in Table C-7 and are shown graphically in Figure C-1.

Table C-7 Maximum Clad Temperatures (T_{MAX})

	Debris Thermal Conductivity $\left(\frac{BTU}{hr * ft * °F}\right)$			
	0.1	**0.3**	**0.5**	**0.9**
Debris Thickness (mils)	T_{MAX}	T_{MAX}	T_{MAX}	T_{MAX}
0	260°F	260°F	260°F	260°F
5	–	269°F	266°F	264°F
10	305°F	277°F	271°F	266°F
15	–	284°F	275°F	269°F
20	–	291°F	280°F	271°F
25	–	297°F	284°F	274°F
30	386°F	303°F	288°F	276°F
35	–	310°F	291°F	278°F
40	–	316°F	295°F	281°F
45	–	322°F	299°F	283°F
50	474°F	327°F	302°F	285°F

The calculated maximum clad temperatures all occur under a grid on the upper section of the fuel rod assembly. Assuming the minimum thermal conductivity of the debris collected in the grid and assuming a debris thickness of 50 mils, a maximum cladding temperature behind a grid of 474°F is calculated. This calculated temperature is well below the 800°F LTCC acceptance basis identified in Appendix A. Thus, the clad surface temperature acceptance basis of 800°F identified in Appendix A is satisfied.

The temperatures calculated with this model are conservative. The calculation assumed no flow through the grid. Thus, some coolant flow is expected to pass through the grid, cooling the clad surface. Not accounting for flow through the debris captured between the grid and the clad provides for the calculation of a conservatively high clad surface temperature.

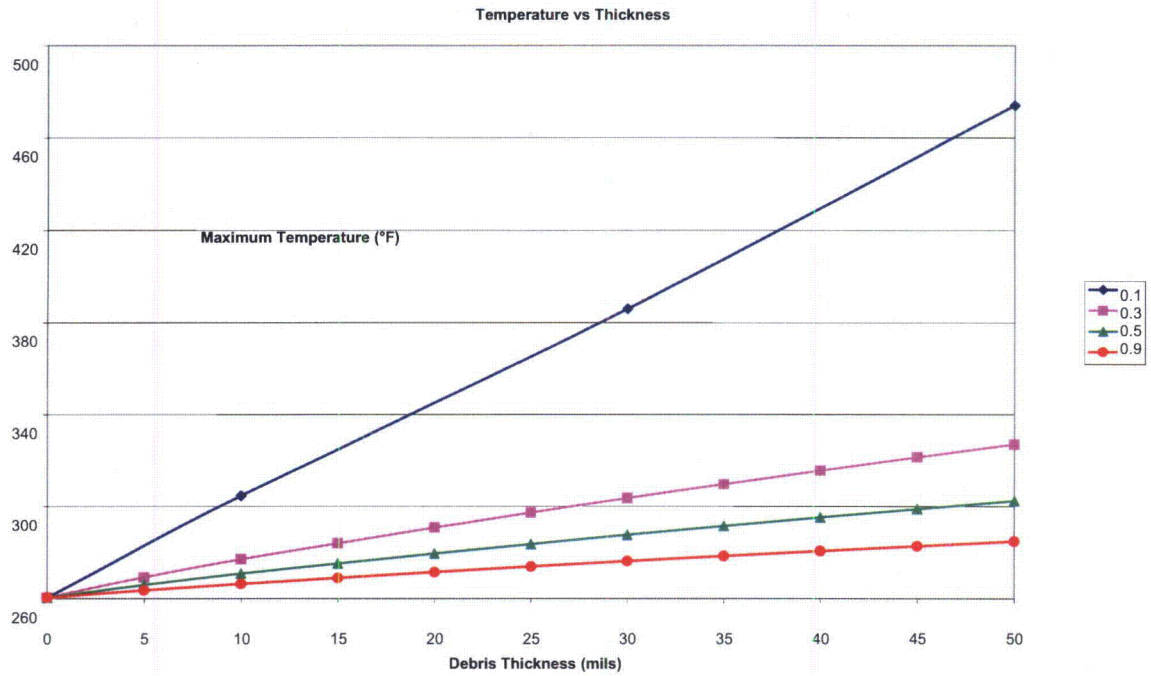

Figure C-1 Temperature vs. Deposition Thickness and Thermal Conductivity

Comparing these results to those of Appendix D, the corresponding Appendix D temperatures are approximately 15°F to 86°F hotter. The reason for this is that the Appendix D analysis used a bulk fluid temperature that was greater than 25°F hotter, and a heat flux that was 25 percent larger, and considered oxide and crud layers, each 100 microns (4 mils) thick, in addition to a 50 mil layer of precipitate. These additional layers also contributed several °F to the temperature increase predicted by the calculations described in Appendix D.

C.7 REFERENCES

C-1 WCAP-16406-P-A, Revision 1, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," March 2008.

APPENDIX D
FUEL CLAD HEAT-UP BETWEEN GRIDS

D.1 INTRODUCTION

In an extreme case, it has been postulated that debris could adhere to the fuel rods and impede decay heat removal. A parametric study was performed to show the effects on the maximum temperature of the fuel rod due to deposited debris by varying debris thickness and the thermal conductivity of the debris. A detailed discussion of this calculation, including a discussion of assumptions and boundary conditions, is presented here.

D.2 DESCRIPTION

This appendix provides an engineering analysis of the heat transfer behavior associated with fuel cladding that has a coating or layering of oxide, crud, and debris precipitate. Boundary conditions are decay heat of the fuel and two phase thermal hydraulic conditions associated with the core in a post LOCA environment.

The acceptance criterion used was the 800°F value identified in Appendix A. The temperature at the oxide/clad interface was compared to the acceptance criterion; this location represented the OD surface of the cladding. As noted in Appendix A, this temperature was chosen as autoclave testing has demonstrated that it is a value below which excessive cladding oxidation and hydrogen embrittlement has been demonstrated to not to occur.

D.3 METHODOLOGY

This analysis considered the cladding as being surrounded by concentric layers of oxide, crud, and chemical precipitate, with no gaps between them. The source of heat was decay heat in a post-LOCA environment, and the section of rod analyzed was assumed to be fully exposed to a two-phase liquid/vapor environment in the core. This analysis used the generic resistance form of the heat transfer equation, for a radial coordinate system.

Figure D-1 provides a graphical depiction of the analysis model.

D.4 INPUTS/ASSUMPTIONS

- This analysis only considered heat conduction in the radial direction. No axial heat conduction was assumed to occur.
- This analysis did not assume the presence of any grid components.
- The fuel rod power value was assumed to be a constant value of 0.226 kW/ft. This is a reasonable value for the peak power level at 20 minutes post-LOCA.

- The fuel cladding outside diameter was assumed to be 0.360 in. (a typical value for Westinghouse fuel).

- The cladding material type does not factor into this evaluation, since the calculation uses the heat flux at the cladding/oxide interface.

- The total convective heat transfer coefficient was assumed to be a constant value of 650 BTU/hr-ft^2-°F. This is a reasonable value at 20 minutes post-LOCA.

- The assumed conductive heat transfer coefficients were:

  – Oxide – Constant value of 1.27 BTU/hr-ft-°F (based on accepted PWR industry experience).

  – Crud – Constant value of 0.3 BTU/hr-ft-°F (based on accepted EPRI Value).

  – Debris Precipitate – Analysis cases considered are 0.1, 0.3, 0.5, and 0.9 BTU/hr-ft-°F.

  The values of thermal conductivity represent bounding values for a silicate precipitate (0.1 BTU/hr-ft-°F), a bounding value for a calcium-based precipitate (0.3 BTU/hr-ft-°F), a value for crud reported by EPRI (0.5 BTU/hr-ft-°F), and enhanced heat transfer through a highly conductive and porous medium (0.9 BTU/hr-ft-°F).

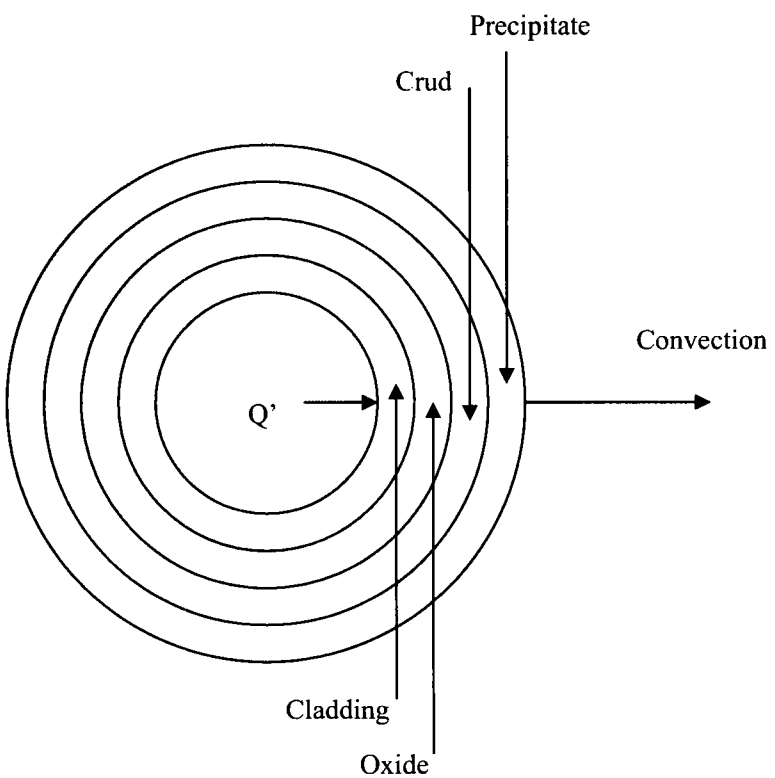

Figure D-1 Heat Transfer Model (not to scale)

- The thickness of each material was assumed to remain uniform around the circumference of the fuel rod:

  - Oxide – Constant value of 100 microns (0.004 in) (based on upper bound of PWR industry experience).

  - Crud – Constant value of 100 microns (0.004 in) (based on upper bound of PWR industry experience).

  - Debris Precipitate – Analysis cases varied from 0 to 50 mils (0.05 in) in 10 mil increments.

- No contact resistance was assumed to exist between material layers. The term 'contact resistance' refers to the resistance to the transmission of heat across the boundary of two adjacent solids. This resistance to heat flow is due to gases or vacant spaces between the two solids.

  The development of the oxide layer and the deposition of the crud layer on the oxide, both which occur at power operations, are gradual and occur over time. The oxide provides nucleation sites for the deposition of the crud and the crud adheres to the outer oxide layer. The thermal conductivity of both the clad oxide layer and the crud already account for the morphology of their formation, including gases or vacant spaces. Since the crud adheres to the outer clad oxide layer by attaching itself to surface irregularities in the oxide layer, additional surface resistance was evaluated and found not to be appropriate during long-term core cooling.

  Similarly, the deposition of the debris layer on the crud surface is also gradual and occurs over time. The deposition on and adhesion to the surface of the crud layer is evaluated to be similar to that of the crud onto the clad oxide layer. Considering that a conservatively small thermal conductivity value for the debris deposition of 0.1 Btu/(hr-ft-°F) is used for the parametric study, the use of a contact resistance is evaluated to be both inappropriate and overly conservative.

- The bulk fluid temperature (T_∞) was assumed to be 250°F. This is a reasonable and expected value for the post-LOCA fluid temperature within the core.

D.5 RESULTS (TABLE/FIGURE)

Table D-1 lists the clad/oxide interface temperatures for each of the four values of precipitate thermal conductivity analyzed.

Table D-1 Clad/Oxide Interface Temperature vs. Chemical Precipitate Thickness

Chemical Precipitate Thickness (mils)	$k_{precipitate}$ BTU/hr-ft-°F			
	0.1	0.3	0.5	0.9
0	273°F	273°F	273°F	273°F
10	336°F	293°F	285°F	279°F
20	396°F	313°F	296°F	286°F
30	453°F	331°F	308°F	291°F
40	508°F	350°F	318°F	297°F
50	560°F	367°F	328°F	302°F

Figure D-2 plots the clad/oxide interface temperature as a function of chemical precipitate thickness for four values of precipitate thermal conductivity.

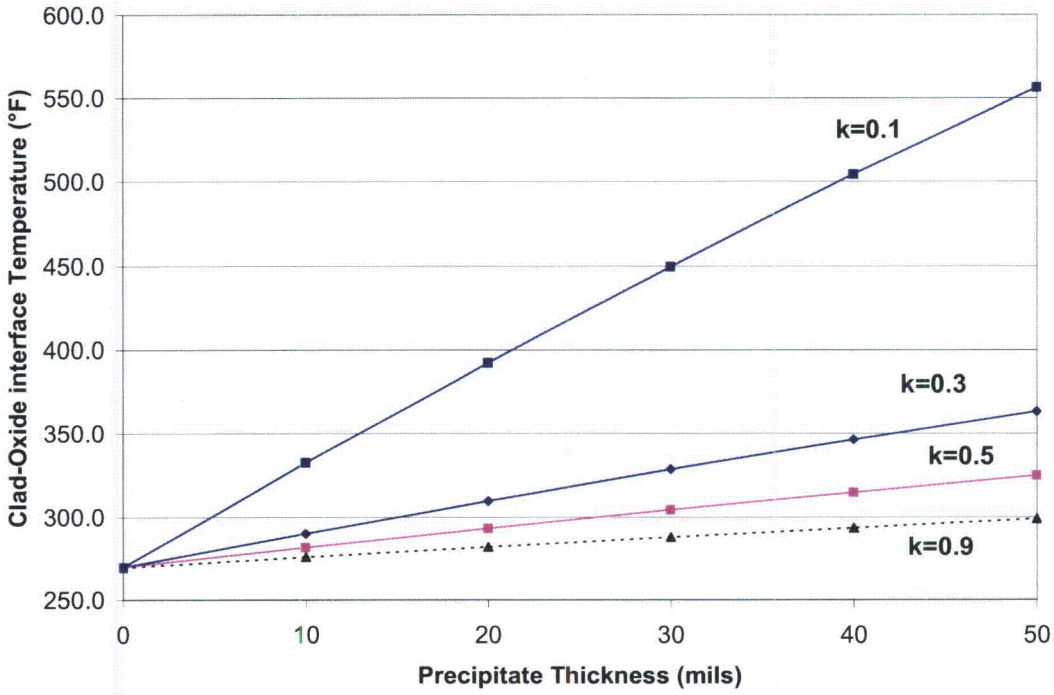

Figure D-2 Clad-Oxide Interface Temperature vs. Chemical Precipitate Thickness