ANALYSIS REPORT

Post-LOCA PWR Core Inlet Blockage Assessment

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SAFETY MARGINS BRANCH

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EXECUTIVE SUMMARY

The objective of this study is to assess the impact that Pressurized Water Reactor (PWR) core inlet blockage conditions would have on the ability of the Emergency Core Cooling System (ECCS) to provide long term core cooling during the recirculation phase after the Refueling Water Storage Tank (RWST) inventory is depleted. The TRACE analysis models the entire primary and secondary system of a typical four loop Westinghouse standard 412 PWR. The TRACE analysis was performed for a double ended cold leg break (DECLB) for the periods before and after the start of recirculation. Core Inlet blockage conditions were assumed to occur at the start of recirculation when cooling water is started to be pumped from the reactor water sump. Predictions for a range of core inlet blockages at the highest decay heat value at the start of cooling water recirculation are specifically assessed because of the possible presence of debris in the containment sump water. An unblocked core inlet and three inlet core blockage conditions of 75%, 87.5% and 94.8% were analyzed. The analysis results using the TRACE computer code show that the PWR core can be sufficiently cooled in the recirculation phase with inlet blockage conditions up to 94.8%. Crossflow through and around the rod bundles can exist in core areas where blockages do not exist. Crossflow in the reactor core downstream of the blocked inlet area provides sufficient flow and cooling to adequately maintain acceptable clad temperatures for all cases analyzed. As expected, the 94.8% blocked case predicts the maximum clad temperature of about 280 °F (137.8 °C) which is approximately 4 °F (2.3 °C) above the unblocked case. The analyses for the cases with the smaller blockages predict temperature rises less than for the 94.8% blocked case.

Detailed three-dimensional CFD analyses of the PWR reactor core with three assumed inlet blockages were performed using the FLUENT computer code in order to verify the acceptability of the TRACE predictions which employed a less detailed core model. The FLUENT analysis uses a single-phase steady-state solution approach and models only the core region; whereas the TRACE provides a single or two-phase transient solution of the entire PWR primary system including the reactor core. TRACE calculated results provide the FLUENT boundary conditions specified at the core inlet and outlet. The FLUENT analyses do not include two-phase effects such as increased pressure drop, and do not include phase change effects such as those involving heat of vaporization considerations. These calculational assumptions limit the applicability of the results calculated at the upper parts of the core where two-phase conditions exist.

The FLUENT predictions produce the same basic trends in core flow as observed from the TRACE analyses. The CFD analysis predicts that sufficient crossflow and cooling is provided to the core areas immediately downstream of an inlet blockage. Because of the two-phase calculational limitations, FLUENT predicts unrealistically high core temperatures because heat of vaporization is not considered. Since the TRACE calculations predict two-phase conditions within the core and FLUENT assumes one-phase core flow, the FLUENT results are only used to verify the acceptability of the flow redistribution downstream of the assumed inlet blockage. In this regard, FLUENT and TRACE predict sufficient flow redistribution and cooling downstream of a core inlet blockage at recirculation.

TRACE Calculated Peak Clad Temperatures After Recirculation

Analyzed Case	Calculated Peak Clad Temperature
Unblocked Inlet	276 °F (135.5 °C)
75% Inlet Blockage	278 °F (136.5 °C)
87.5% Inlet Blockage	277 °F (136.1 °C)
~95% Inlet Blockage	280 °F (137.8 °C)

1.0 INTRODUCTION

The objective of this study is to assess the impact that Pressurized Water Reactor (PWR) core inlet blockage conditions would have on the ability of the Emergency Core Cooling System (ECCS) to provide long term core cooling during the recirculation phase after the Refueling Water Storage Tank (RWST) inventory is depleted. The inlet blockage could be caused by debris which passes the sump screens after the start of sump water recirculation. Since the ability of the core to capture debris is not currently known, this study will provide TRACE code predictions for a range of assumed core inlet blockages. The predictions of the TRACE code will be compared with similar computational fluid dynamics (CFD) analyses of a PWR core performed using the FLUENT code.

1.1 Thermal-Hydraulic Computer Code Methods

The TRACE computer code is a modern consolidation of various computer codes which have been used to analyze fluid-thermal transients in pressurized water reactor and boiling water reactor (BWR) systems. The TRACE computer code is designed to analyze both PWR and BWR systems. The TRACE thermal-hydraulic correlations and calculational methods have and are continuing to be updated to include the latest technological information and advances.

FLUENT is a modern computational fluid dynamics (CFD) code which solves the Reynolds-Averaged Navier-Stokes equations, including heat transfer, for a three-dimensional domain. FLUENT has been used for a diverse set of applications in many industries including power generation. FLUENT's porous media formulation has been employed for the analysis described in this report.

2.0 PWR CORE INLET ANALYSIS USING TRACE

This analysis employs a TRACE input model developed for analyzing a typical four loop Westinghouse standard 412 PWR with 17 x 17 fuel assemblies. The TRACE model includes both primary and secondary system components including Model F steam generators, a pressurizer, hot and cold leg piping, and a reactor vessel. Table 1 summarizes the assumptions used for the reported analyses. A full double ended cold leg break was assumed in order to maximize the mass lost from the primary system. The safety injection and containment flows were maximized in order to model an early recirculation time with larger core decay heat. With the assumed Refueling Water Storage Tank (RWST) volume, the recirculation time is calculated to occur about 1200 seconds following the loss-of-coolant accident (LOCA).

Table 1: Analysis Assumptions

Break Size	Double Ended Cold Leg Break (DECLB)
Recirculation Start Time	1200 seconds
Safety Injection	Two Train High Pressure Safety Injection (HPSI) Two Train Low Pressure Safety Injection (LPSI)
Containment Spray	Two Trains
Containment Spray Flowrate	3100 gpm per train
Core Blockage	At Core Inlet (Bottom of Elevation Level 5)
Baffle-Core Cross Flow	No Cross Flow Between Core and Baffle Region
Fuel Assembly	17 x 17
Initial Operating Power	3.459×10^9 W

The reactor vessel is modeled using the TRACE three-dimensional Vessel component with twenty-six axial levels, four radial rings and eight 45° segments. The core is modeled using fourteen axial levels. Figures 1 and 2 show the nodalization of the reactor vessel. No bypass flow is assumed to be present across the core baffle. The core flow resistances are typical of the Watts Bar core design.

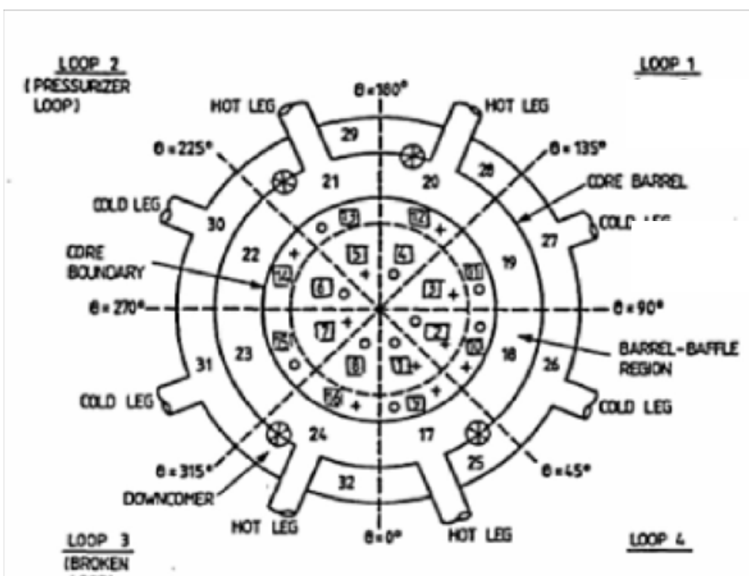

Figure 1: Reactor Vessel Model - Cross Section View

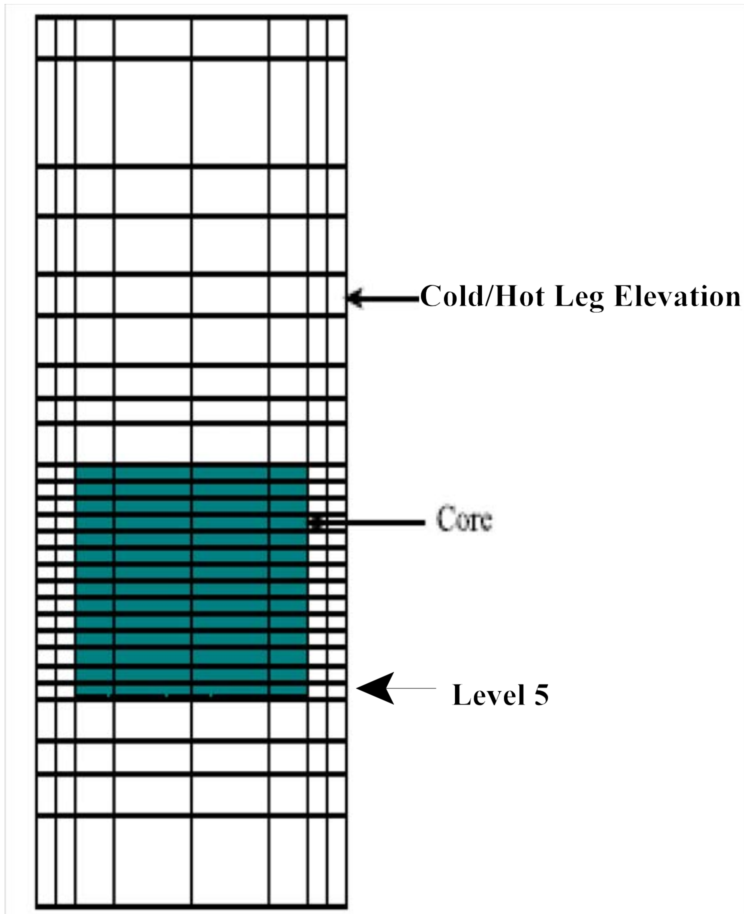

Figure 2: Reactor Vessel Model - Axial View

TRACE analyses were performed for an unblocked core inlet and for core inlet blockages of 75%, 87.5% and 94.8% at the bottom of vessel elevation 5. The core inlet blockages were assumed to instantaneously occur at 1200 second which is the start of water recirculation. Figures 3, 4 and 5 illustrate the three assumed blockage geometries. Figure 6 shows the total and individual loop safety injection water flows before recirculation. The remaining figures show the TRACE calculated results for the core unblocked and three blocked conditions. These figures indicate that water flow reaches all parts of the core and sufficient core cooling is provided for all analyzed blocked and unblocked cases. Figure 7 plots the maximum core rod temperature for unblocked and blocked core cases. The TRACE calculated peak clad temperatures (PCT) are listed in Table 2. As expected the maximum clad temperature of about 280 °F (137.8 °C) is calculated to exist for the 94.8% blocked case; this predicted temperature is approximately 4 °F (2.3 °C) above the unblocked case. The analyses for the cases with the smaller blockages predict temperature rises less than for the 94.8% blocked case. The core collapsed level for the unblocked and blocked cases are shown on Figure 8. There is no significant difference in calculated collapsed level for the cases analyzed. Figure 9 indicates that calculated void fractions immediately above the blocked area are larger with increases in blocked area. However, Figure 10 indicates that no significant differences in void fraction are calculated to exist at the core outlet for all cases analyzed.

Table 2: TRACE Calculated Peak Clad Temperatures After Recirculation

Analyzed Case	Peak Clad Temperature
Unblocked Inlet	276 °F (135.5 °C)
75% Inlet Blockage	278 °F (136.5 °C)
87.5% Inlet Blockage	277 °F (136.1 °C)
94.8% Inlet Blockage	280 °F (137.8 °C)

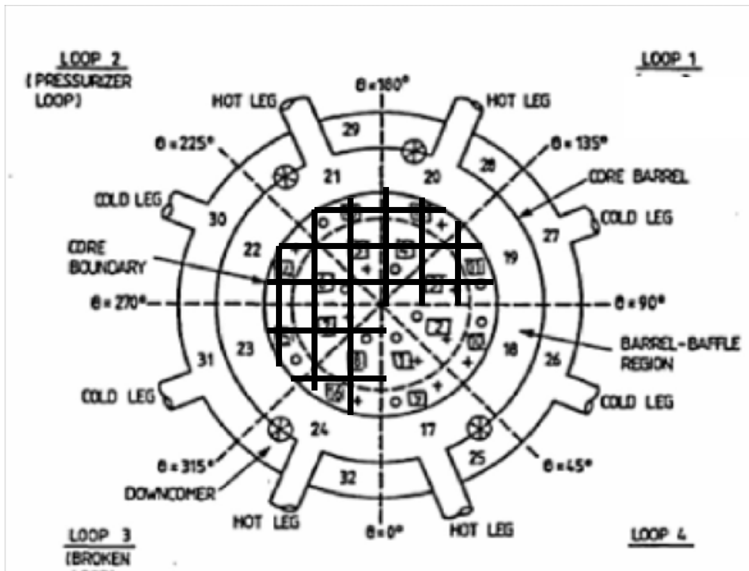

Figure 3: 75% Core Inlet Blockage

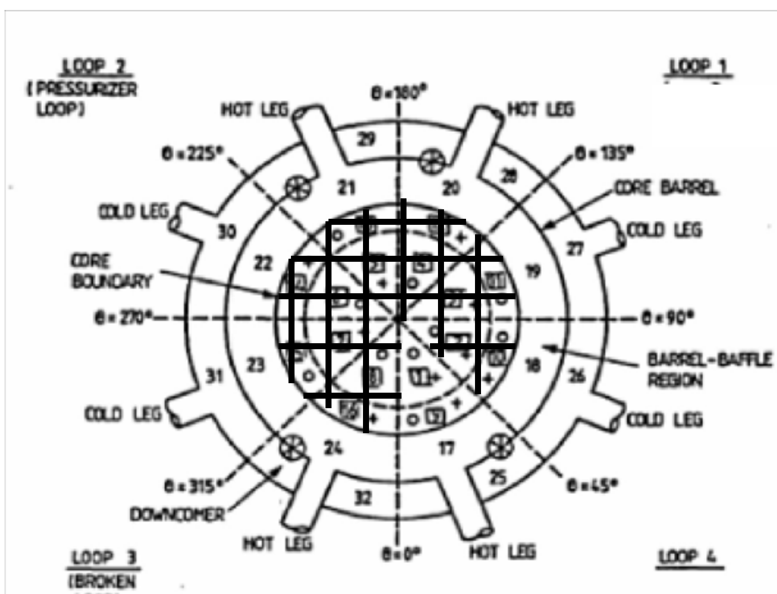

Figure 4: 87.5% Core Inlet Blockage

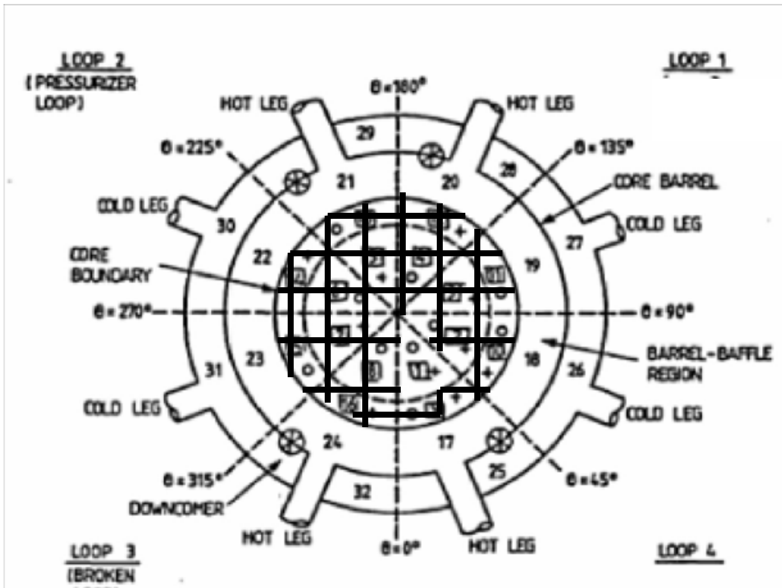

Figure 5: 94.8% Core Inlet Blockage

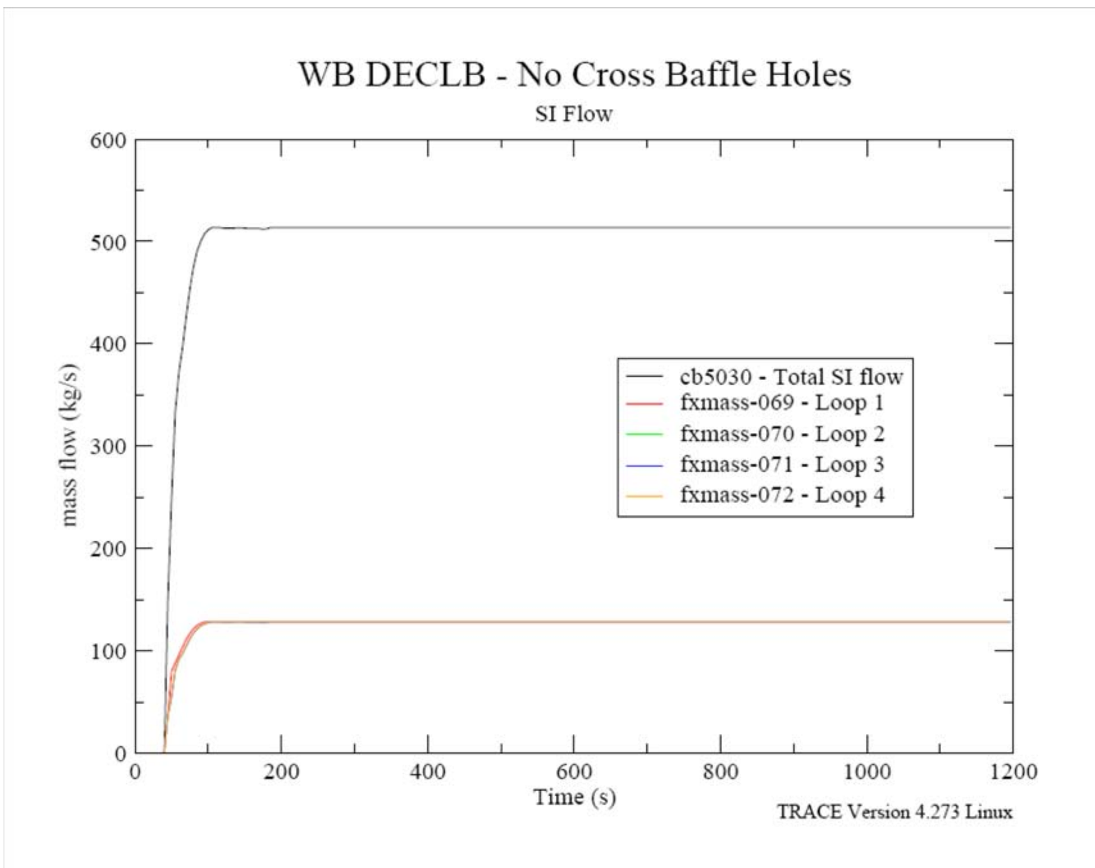

Figure 6: Safety Injection Flow Before Recirculation Start

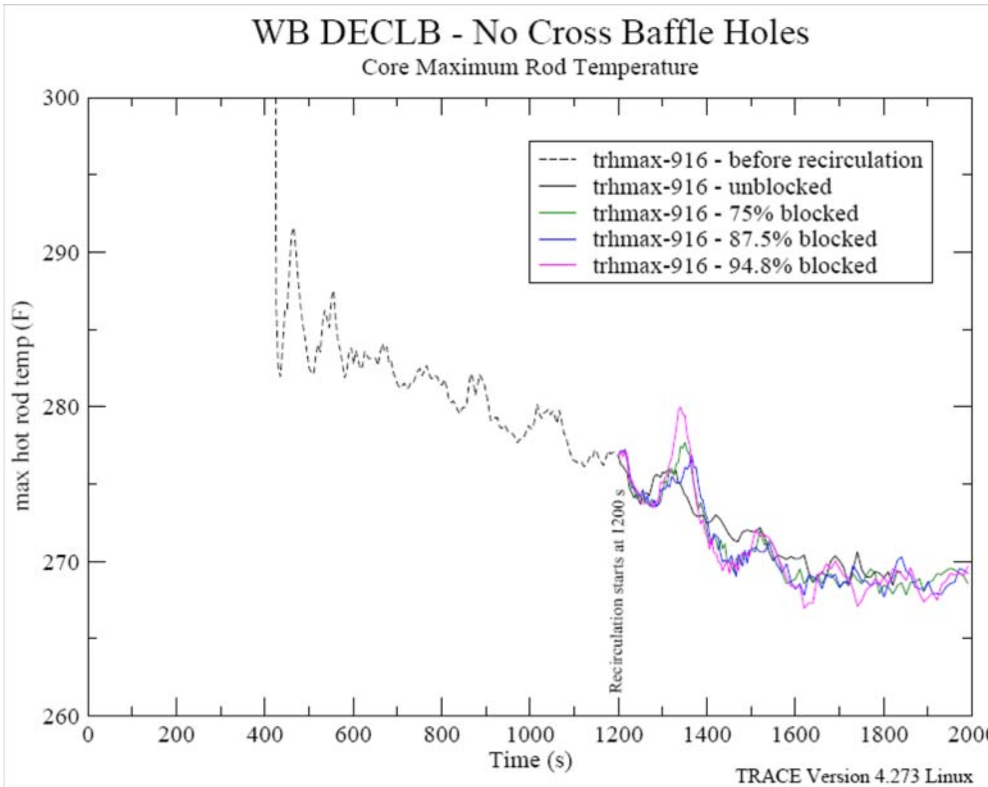

Figure 7: Core Rod Maximum Temperature

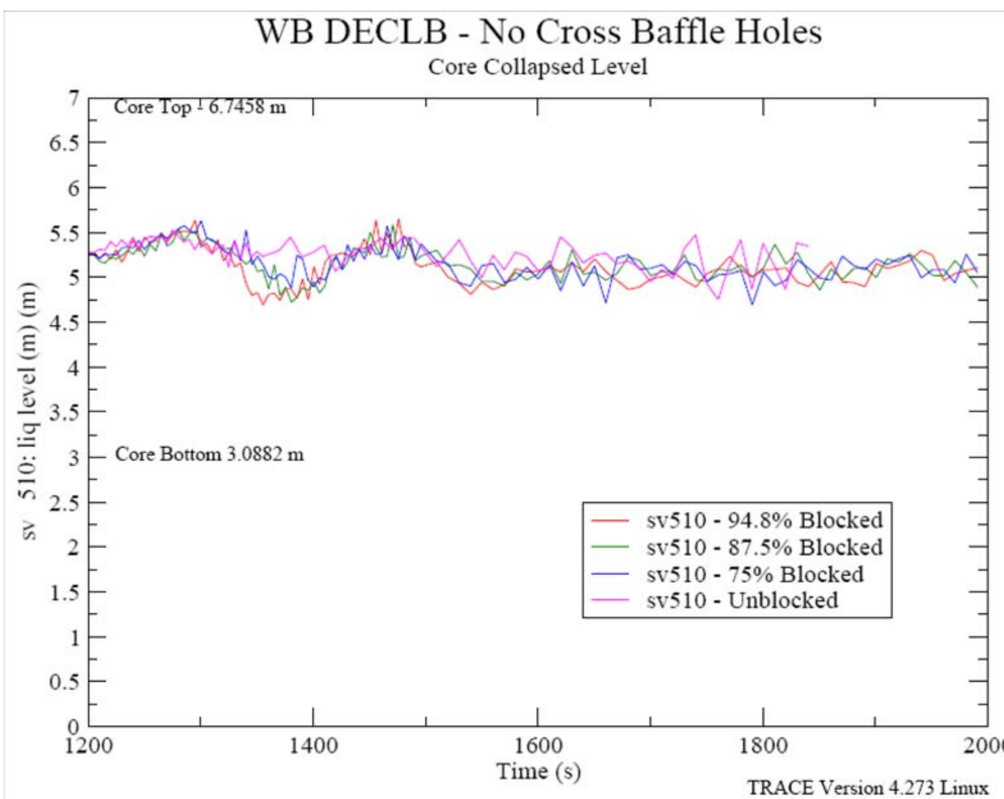

Figure 8: Core Collapsed Level

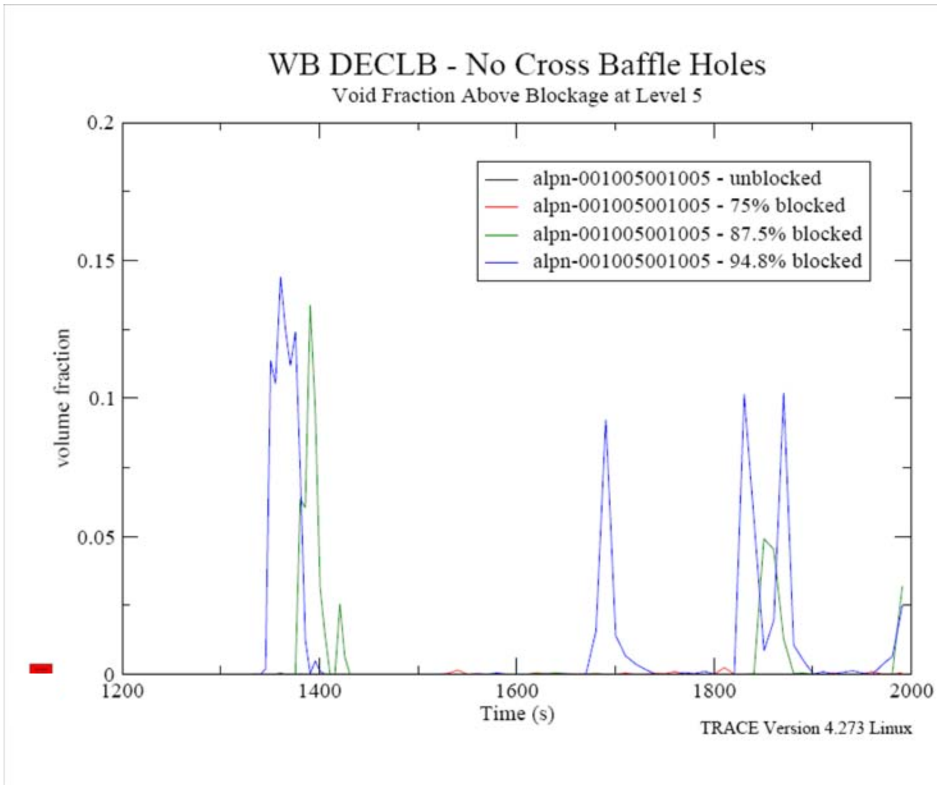

Figure 9: Core Entrance Void Fraction Above Blockage

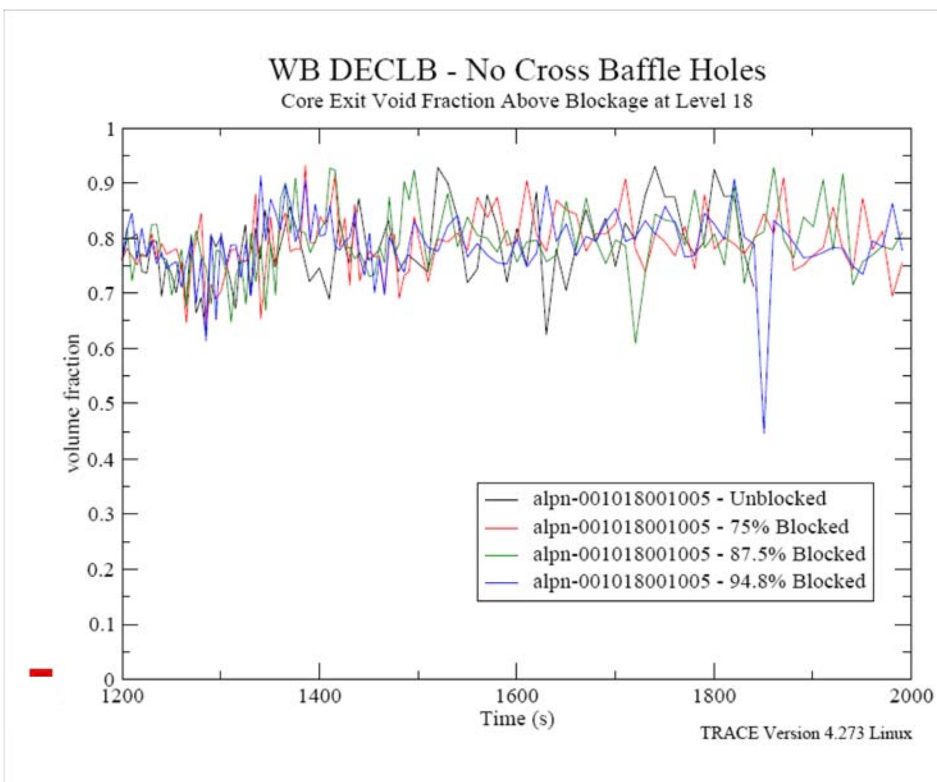

Figure 10: Core Exit Void Fraction Above Blockage

3.0 CFD PWR CORE INLET ANALYSIS

The objective of the FLUENT analysis is to use a detailed core model with better spatial resolution than the TRACE model to predict flow conditions in a core whose inlet is blocked with debris. The FLUENT flow predictions can be compared with the TRACE results to verify the acceptability of the less detailed TRACE core model. The FLUENT analysis models only the reactor core using the "porous CFD" mode. The FLUENT analysis calculates a steady-state snapshot of time when the core inlet blockage is instantaneously assumed to occur at 1200 seconds, the start of recirculation. The core inlet and outlet conditions predicted from the TRACE analysis have been used as boundary conditions for the FLUENT core flow analysis.

The FLUENT analysis uses radial symmetry to model one-half of the reactor core. The FLUENT core model uses 96.5 separate assemblies or channels. The porous media approach is used to model the flow losses of the fuel rods, spacer grids, mixing vanes and nozzles. Twelve axial fuel regions are modeled. Axial power variations are modeled by specifying different power in each of the twelve axial regions. Figure 11 shows the core FLUENT model. Core flow distributions were obtained for axial blockages of 75%, 87.5% and 95% at the bottom nozzle (Figure 12).

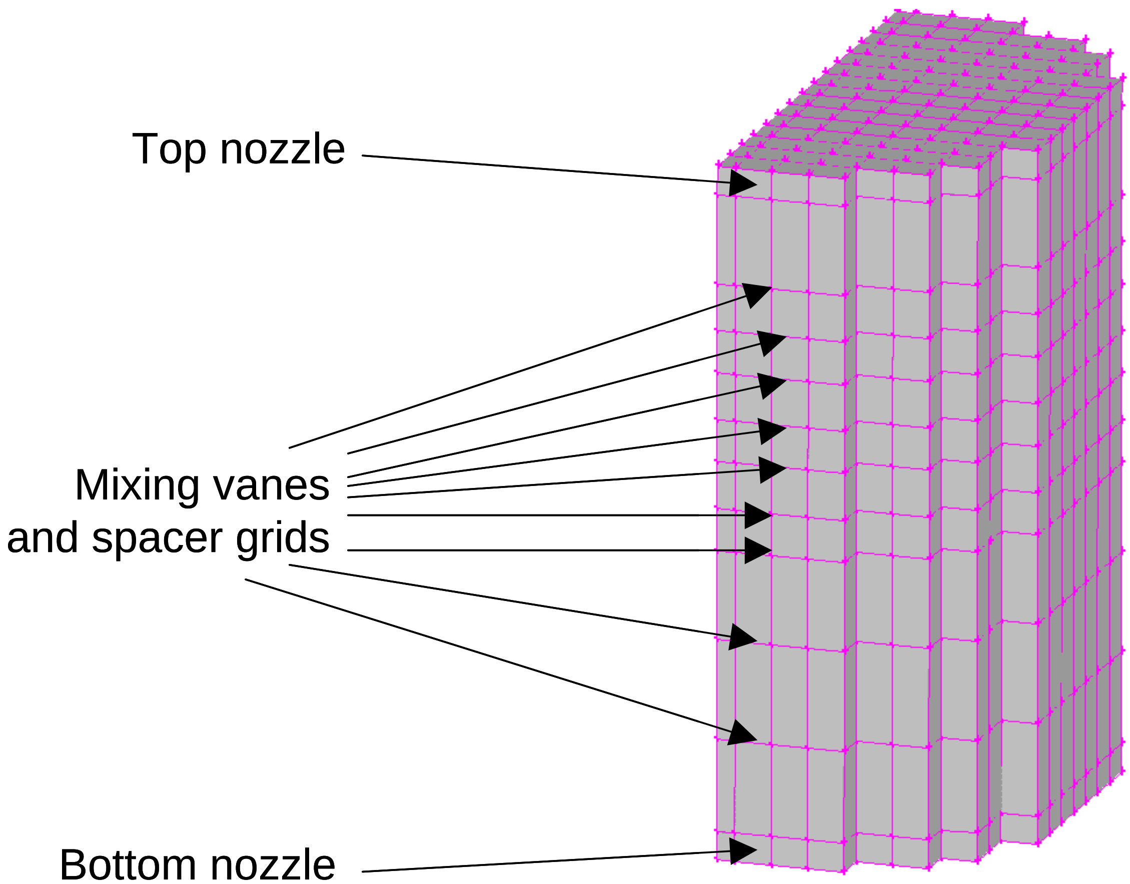

Figure 11: FLUENT Core Model

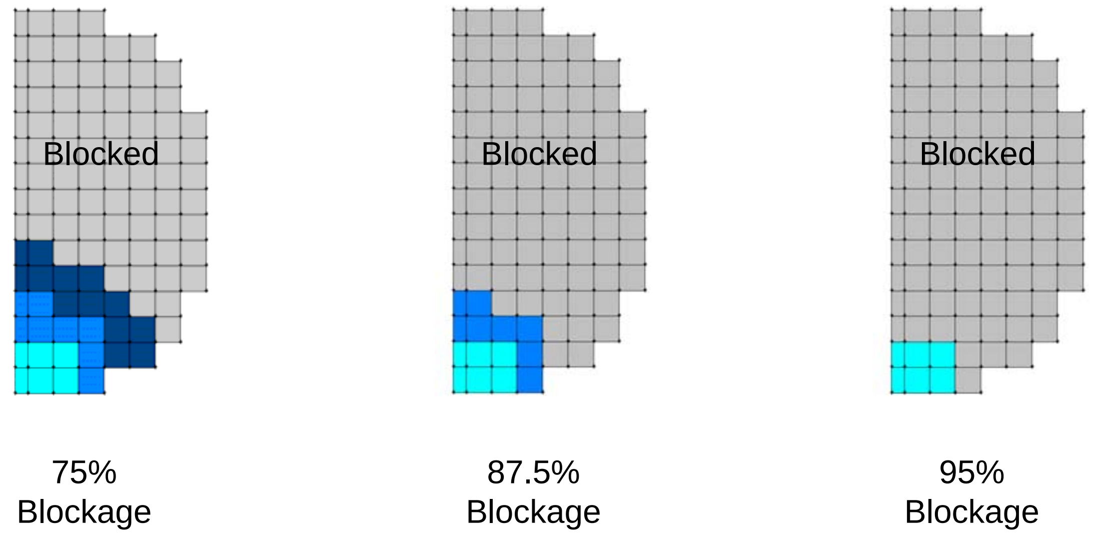

Figure 12: Core Blockage Cases Analyzed Using FLUENT

The characteristic mesh size is 3.5 cm in the three coordinate directions. All cells have an aspect ratio of 1. The model uses 410,000 hexahedral cells; each assembly is composed of 6 by 6 by 118 cells. The FLUENT transient solver is used to obtain quasi-steady state results after core inlet blockage is assumed. The model assumes the presence of one-phase flow using the standard k-ε turbulence model. The FLUENT analyses do not model two-phase flow and do not account for latent heat of vaporization or any other phase change effects. Temperature dependent fluid properties are used; however, solid properties are assumed constant.

The axial and radial power shapes were taken from the TRACE model. The core power is specified using a source term with ten axial levels with a peaking factor of 1.31. The total power is assumed to be 72.5 Mw which is 2% of full power. The radial peaking factors are 1.136 for ring 1, 0.901 for ring 2 and 1.932 for the hot assemblies.

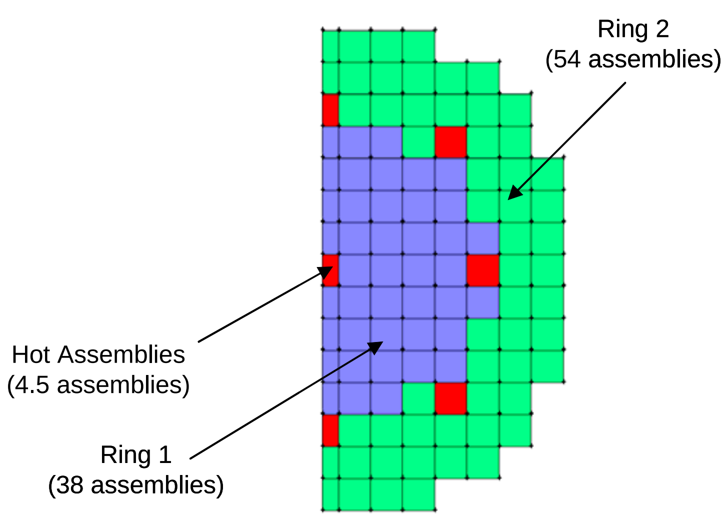

Figure 13: Core Radial Power Shape

The following lists the assumptions, limitations and observations associated with the FLUENT analyses.

- In order to maintain reasonable computational times, it is not practical to model the entire primary system. Consequently, the FLUENT analysis uses radial symmetry to only model one-half of the reactor core.
- The FLUENT analysis solves for a quasi-steady flow distribution. This assumption is reasonable because of the length of the time-scales, tens of minutes, for the period of interest.
- The boundary conditions for the core FLUENT model are obtained from the TRACE analyses. The FLUENT model assumes a velocity boundary condition at the core inlet with the water velocity corresponding to the TRACE mass flow rate prediction at a temperature of 373 °K (100 °C, 212 °F), and a pressure boundary conditions at the core outlet at a temperature of 500 °K (227 °C, 440 °F). The TRACE results are oscillatory in time, but steady-state approximations are employed, consistent with the quasi-steady-state assumption. Table 3 lists the inlet velocity boundary conditions used in the FLUENT analyses.

Table 3: Core Inlet Velocity Boundary Condition from TRACE Analyses

Analyzed Case	Core Flowrate	Inlet Boundary Velocity
75% Inlet Blockage	200 kg/s	0.094 m/s
87.5% Inlet Blockage	150 kg/s	0.141 m/s
95% Inlet Blockage	100 kg/s	0.226 m/s

- No two-phase effects, such as increased pressure drop or phase change considerations such as accounting for the latent heat of vaporization, are modeled. These considerations become important if the void fraction at the top of the core becomes high. It is important to note that the TRACE code predicts about 80% void fraction at the core exit for all analyzed cases.
- The FLUENT model utilizes anisotropic porous media flow resistance and isotropic porosity. This approach primarily impacts the turbulence modeling since individual structures are not characterized. Sensitivity studies showed no overall effect from using a different turbulence model or assuming laminar flow probably due to the porous media modeling representation.
- Viscous losses are not modeled and FLUENT's superficial velocity formulation was employed. A later sensitivity study demonstrated that this assumption did not impact calculated results.
- The simulations did suffer from high turbulent viscosity ratios near the exit, and back-flow in the 95% blockage case. These disturbances are thought to be a result of the zero pressure gradient at the exit, and would benefit from modeling of the upper plenum and hot leg attachments.
- No grid-sensitivity study was performed. Due to the geometric simplicity of the porous media representation, cell aspect ratios and equi-angle skews are near ideal. The 3.5 cm cell size is reasonable, and not expected to have an impact on the results.

The FLUENT results indicate that radial flow spreads very quickly in the core downstream of the blockage location. This flow spreading is illustrated in Figure 14 which shows the flow field for the 75% core inlet blockage case.

The FLUENT study concludes that the areas of the core behind the blocked assemblies are effectively cooled for all the cases analyzed. The results indicate that increased blockage up to 95% results in higher exit temperatures, but do not result in larger radial temperature variations.

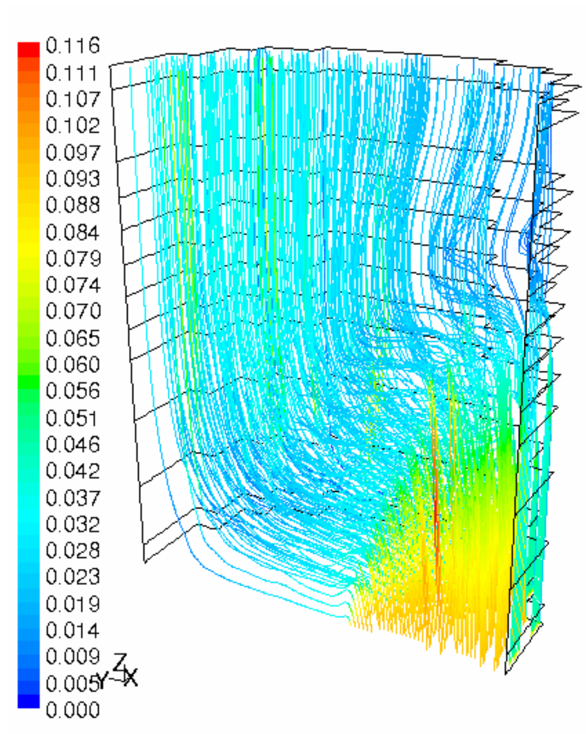

Figure 14: Flow Field for 75% Core Inlet Blockage

4.0 CONCLUSIONS

The objective of this study is to assess the impact that PWR core inlet blockage conditions would have on the ability of the ECCS to provide long term core cooling during the recirculation phase after the RWST inventory is depleted. The TRACE analysis models the entire primary and secondary system of a typical four loop Westinghouse standard 412 PWR following a LOCA. The TRACE analysis was performed for a double ended cold leg break (DECLB) for the periods before and after the start of recirculation. Core Inlet blockage conditions were assumed to occur at the start of recirculation when cooling water is started to be pumped from the reactor water sump. Predictions for a range of core inlet blockages at the highest decay heat value at the start of cooling water recirculation are specifically assessed because of the possible presence of debris in the containment sump water. An unblocked core inlet and three inlet core blockage conditions of 75%, 87.5% and 94.8% were analyzed. The analysis results using the TRACE computer code show that the PWR core can be sufficiently cooled in the recirculation phase with inlet blockage conditions up to 94.8%. Crossflow through and around the rod bundles can exist in core areas where blockages do not exist. Crossflow downstream of the blocked inlet area provides sufficient flow and cooling to adequately maintain acceptable clad temperatures for all cases analyzed. As expected, the 94.8% blocked case predicts the maximum clad temperature of about 280 °F (137.8 °C) which is approximately 4 °F (2.3 °C) above the unblocked case. The analyses for the cases with the smaller blockages predict temperature rises less than for the 94.8% blocked case.

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5.0 REFERENCES

(1) "**TRAC-M/FORTRAN 90 Theory Manual**," U. S. Nuclear Regulatory Commission, NUREG/CR-6724, July 2001.

(2) **FLUENT 6.2 Users Guide**, Fluent Inc., 2006.