IMPORTANCE OF ALTERNATIVE REACTOR VESSEL FLOW PATHS DURING LOSS-OF-COOLANT ACCIDENT SCENARIOS WITH DEBRIS-INDUCED CORE BLOCKAGE

A Thesis

by

TIMOTHY MICHAEL CROOK

Submitted to the Office of Graduate and Professional Studies of
Texas A&M University
in partial fulfillment of the requirements for the degree of

MASTER OF SCIENCE

Chair of Committee,	Yassin Hassan
Committee Members,	Rodolfo Vaghetto
	William Marlow
	Maria King
Head of Department,	Andreas Polycarpou

May 2017

Major Subject: Mechanical Engineering

ABSTRACT

During the initial stage of a Loss of Coolant Accident (LOCA), known as the blowdown phase, the high-temperature and pressure break flow can impinge on thermal insulation and generate a substantial amount of debris in containment. This debris can accumulate in the sump compartment and become a major safety concern by potentially impacting the capabilities of the Emergency Core Cooling System (ECCS). Debris can accumulate in the sump and could cause ECCS pump head loss and/or pass through the filtering systems (debris bed and sump strainer) into the reactor primary system during the long-term cooling phase. This scenario and its possible downstream effects are of primary concern under the US Nuclear Regulatory Commission (NRC) Generic Safety Issue 191 (GSI-191).

If the debris was to bypass the filtering and accumulate at the core inlet, core flow could theoretically decrease, affecting the core coolability (decay heat removal). If the debris accumulation at the lower core plate was high enough, it could potentially block the flow through the base of the core, the primary coolant flow path. In an even more severe scenario, debris could block all flow from the bottom of the core by blocking both the core inlet and core baffle/barrel bypass.

Under such conditions, core coolability is dependent on coolant reaching the core through alternative flow paths. One of these key flow paths is the core bypass (baffle/barrel). Additionally, the effectiveness of bypass flow in reaching and cooling the core is heavily impacted by certain plant specific features. One such plant specific feature, which the presence or lack of can have a major impact on core coolability, are the pressure

relief (LOCA) holes. When the core inlet and bypass are unavailable, coolant must reach the core from the top by passing through the upper head region or steam generators.

The primary response during each phase of a double-end guillotine (DEG) Loss-of-Coolant Accident (LOCA) was analyzed for the model of a typical 4-loop Pressurized Water Reactor (PWR) using RELAP5-3D. The effectiveness of core cooling under the hypothesized cold leg break with full core inlet blockage was analyzed for models with and without pressure relief holes. Core cooling was also evaluated under a hypothesized hot leg DEG break with full core inlet and core bypass blockages, with particular emphasis on comparing simplified upper head geometry to a more realistic and more conservative model.

The results of the cold leg break with inlet blockage simulation showed that the presence of alternative flow paths from the bypass into the core may significantly increase core coolability and prevent cladding temperatures from reaching safety limits, while the lack of LOCA holes may lead to a conservative over-prediction of the cladding temperature. The simulation results also showed the impact of LOCA holes on total liquid level maintained in the core and driving flow through the bypass. The hot leg full inlet/bypass blockage showed that the upper head path was able to provide sufficient coolant, even under conservative models. These simulation results help inform safety impacts of some severe accidents under GSI-191 and serve as a reminder of the importance of modeling plant-specific features when performing best-estimate safety calculations.

ACKNOWLEDGMENTS

I first want to thank Dr. Yassin Hassan for being an amazing advisor and mentor. I'm not sure I'll ever manage to do as many things simultaneously as you do. Running our department, seriously caring about students, making time to go to nearly every student organization meeting, opening up your home for thanksgiving, the list goes on. Yet you always had time to talk one-on-one. Thank you.

I want to thank Dr. Rodolfo Vaghetto for being a great boss and mentor for this project. Your oversight and continuity of knowledge through all of the RELAP work before and after me was incredibly valuable.

I owe many thanks to Alessandro for being an awesome office mate for three-plus years. I especially appreciated your espresso machine and great conversations over a cup of espresso. It got me through many an afternoon.

Thanks to Ernie Kee for his contributions to the project, but more importantly, for being an inspiration. You got me interested in PRA and helped me become passionate about this project through understanding the whole scope of the GSI-191 problem.

Thank you to my other committee members, Dr. Maria King and Dr. William Marlow for your participation in this process.

Last, I want to thank Dr. Leslie Dewan for making finishing this thesis a condition for employment. Sadly, it was much needed motivation to actually finish writing it.

CONTRIBUTORS AND FUNDING SOURCES

Contributors

This work was supervised by a thesis committee consisting of Professors Yassin Hassan, Rodolfo Vaghetto, and William Marlow of the Department of Nuclear Engineering and Professor Maria King of the Department of Mechanical Engineering.

All work for the thesis was completed by Timothy Crook, in collaboration with Mr. Alessandro Vanni, Dr. Rodolfo Vaghetto, Dr. Yassin Hassan, and Mr. Ernie Kee of the Texas A&M University Department of Nuclear Engineering, and Mr. Cliff Davis of the Idaho National Laboratory.

Funding Sources

Graduate study was supported by a Graduate Assistant Research position under Dr. Yassin Hassan from Texas A&M University.

NOMENCLATURE

BWR	Boiling Water Reactor
CL	Cold Leg
CLL	Collapsed Liquid Level
CRDM	Control Rod Drive Mechanism
DEG	Double-Ended Guillotine
ECCS	Emergency Core Cooling System
GSI	Generic Safety Issue
HHSI	High Head (High Pressure) Safety Injection
HL	Hot Leg
LHSI	Low Head (Low Pressure) Safety Injection
LOCA	Loss-of-Coolant Accident
LOCA Holes	"LOCA" Pressure Relief Holes
NPP	Nuclear Power Plant
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
PCT	Peak Cladding Temperature
P-Grid	Protection-Grid
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Cracking Corrosion
RHR	Residual Heat Removal

RPV	Reactor Pressure Vessel
RWST	Refueling Water Storage Tank
SG	Steam Generator
SI	Safety Injection
SSO	Sump Switchover
TDJ	(RELAP5) Time Dependent Junction
TDV	(RELAP5) Time Dependent Volume

TABLE OF CONTENTS

LIST OF FIGURES

FIGURE Page

LIST OF TABLES

TABLE Page

1. INTRODUCTION

In the United States, civilian applications of nuclear technology fall under the oversight of the Nuclear Regulatory Commission (NRC), an independent agency created "to ensure the safe use of radioactive materials for beneficial civilian purposes while protecting people and the environment." [1] As part of its regulatory duties, the NRC often identifies potential issues that could affect the safety of nuclear power plants. Often, these fall under the purview of the Generic Issues Program [2] if they "could affect multiple entities under NRC jurisdiction." Typically, NRC attention to these potential issues are motivated by incidents that occurred at a single power plant and could potentially affect many other plants with similar components.

The potential for one such Generic Safety Issue (GSI), related to potential loss of sump pumping power, or Net Positive Suction Head (NPSH), due to possible debris accumulation on the sump pump strainer after a Loss-of-Coolant Accident (LOCA) was theorized by the NRC in 1985. In 1992, an incident occurred at the Barsebäck Boiling Water Reactor (BWR) Nuclear Power Plant (NPP) in Sweden. A LOCA occurred at a safety release valve, causing a hot steam jet to impinge and dislodge insulation, which was transported to the condensation pool in the lower part of containment. Once recirculation occurred and the pumps began to draw water from the pool, a pressure drop alarm was triggered by cavitation which had occurred in the containment spray system pumps. It was determined that the problem was due to the clogging at the strainer due to its small surface area. A year later, a US BWR, Perry NPP had similar issues with strainers being blocked by particulate matter and later fiberglass insulation. These events prompted the NRC to issue a bulletin to BWR licensees mandating they address this potential problem. [3]

The primary solution to this issue was rather simple for BWRs. Enlarging the strainer surface area and removing or replacing fibrous insulation not rated to withstand LOCA conditions addressed the concerns for the vast majority of BWR plants. Despite not occurring at any Pressurizer Water Reactors (PWRs), the NRC determined they should also address this potential event. PWRs did not have such a simple fix due to a number of factors, including differences in design of BWR suppression pools compared to PWR sump pools, so the NRC opened GSI-191 to address this scenario. [3]

A risk-informed approach was determined to be the best method of addressing GSI-191 in PWRs. A risk-informed approach utilizes insights from probabilistic risk assessment (PRA) in conjunction with other engineering insights to determine the actual risk to core damage. [4] For example, the location and size of break can have a widely varied effect on the amount of debris generated during a LOCA. Additionally, some debris may bypass the strainer and could therefore impact the primary system. Texas A&M has performed numerous debris distribution and penetration studies, as well as computational thermal hydraulic analyses in support of these possible scenarios. [5] Following this train of logic, if sufficient debris were to penetrate the sump strainer and enter the primary system, it could theoretically accumulate at small flow area flow paths, and possibly affect core coolability. These are generically referred to as “downstream effects.”

One such major region of concern for a downstream flow blockage effect is the core inlet. This is theoretically possible because of the filter system at the base of fuel assemblies, consisting of a debris filter and a Protective-Grid (P-Grid). [6] During normal operation, greater than 90% of coolant flow follows the flow path through the base of fuel assemblies, as illustrated in Figure 1-1. When a break occurs, this is still the predominant flow path for coolant to reach the core, so if debris were to pass through the sump strainers, it would likely be transported to the core inlet.

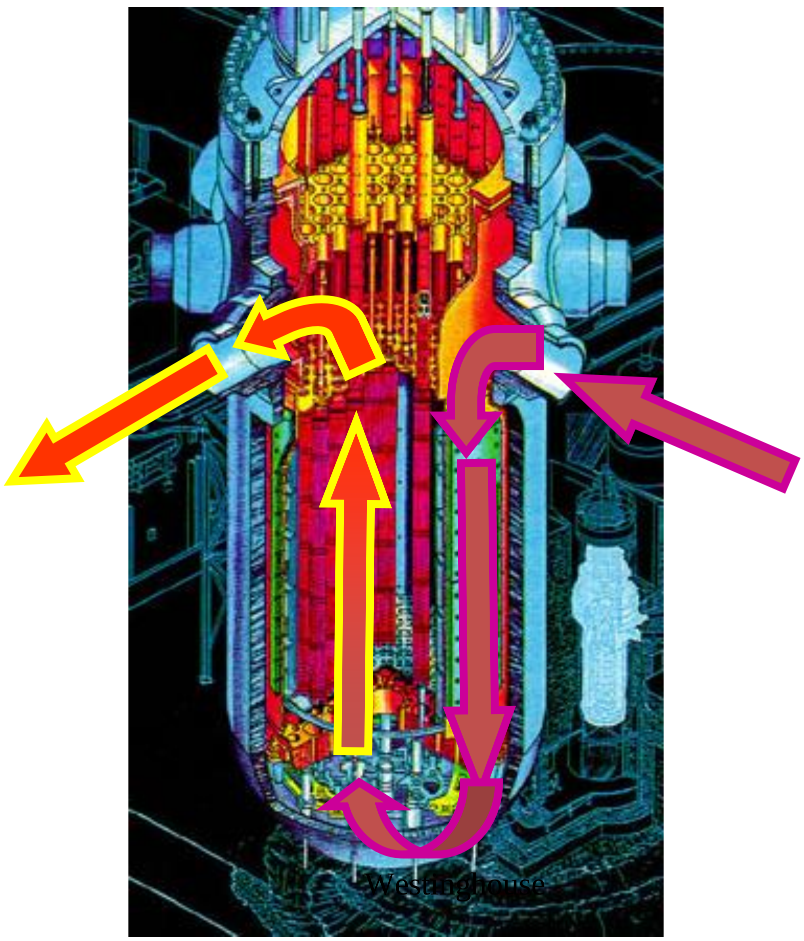


Figure 1-1. Nominal Westinghouse PWR Core Coolant Flow Path [7]

If debris were to sufficiently accumulate in this region, it could block coolant flow to the core, and alternative flow paths, such as the core bypass and upper head sprays could become essential to maintaining core coolability. Additionally, if the bypass and core were both simultaneously blocked due to excessive debris accumulation, the available alternative flow path for coolant is more severely impacted. Additionally, a number of plant specific features, such as pressure relief (“LOCA”) holes (between the bypass and core) may play an important role in ensuring core coolability.

This study intends to examine the possible safety impacts of debris blockage during a large break loss-of-coolant accident. As a possible scenario relevant to GSI-191, this analysis supports closure as the general procedure can be applied to multiple units. Using RELAP5 to model the primary system and each stage of the accident, the consequences of both a CL DEG with core inlet blockage and HL DEG with full core and bypass blockage can be evaluated for a generic Westinghouse four-loop PWR. Because the analysis focuses on the impact of specific alternative flow paths for each break, the plant specific design could have a great deal of impact on the final results. For example, the presence or lack of LOCA holes, or the design of the upper support plate and guide tubes may vary and cause much different results for otherwise identical breaks. Therefore, it is more useful to look at the more severe bounding cases as done in this study. The model can then be refined to match an exact plant unit and a specific analysis. It is expected that this methodology, while somewhat conservative in this instance, provide future guidance and outline the importance of considering plant-specific features in best-estimate simulations.

2. OVERVIEW OF COOLANT FLOW PATHS

If debris is transported to the primary system, it is possible for it to accumulate in the reactor vessel and cause core blockage. In order to determine the degree of blockage, the nominal flow paths through the primary vessel must be considered. Figure 2-1 displays these flow paths extensively. Table 2-1 explains the labelling of each flow path and key potential blockage location as well as showing the corresponding fraction of flow. The flow from the downcomer (A) is further subdivided (D-H) as noted (they sum to the flow of A). These values are approximate for a Westinghouse 4-Loop PWR, but may vary (especially with respect to designed upper head flow). [8] This is due to subsequent modifications to upper head flow in many plants to compensate for Primary Water Stress Cracking Corrosion (PWSCC) in the Control Rod Drive Mechanism (CRDM), increasing the upper head flow from the cold leg by approximately two percent directly diverted from the downcomer (originally destined for direct fuel contact heat removal). [9] The model simulated in this thesis will assume these modifications. Table 2-2 describes the structural components of the Reactor Pressure Vessel (RPV) with flow through them that are relevant to this study. Further functional and structural description available in [8] and [9]. Individual flow path's impact on core coolability is discussed in Section 5.

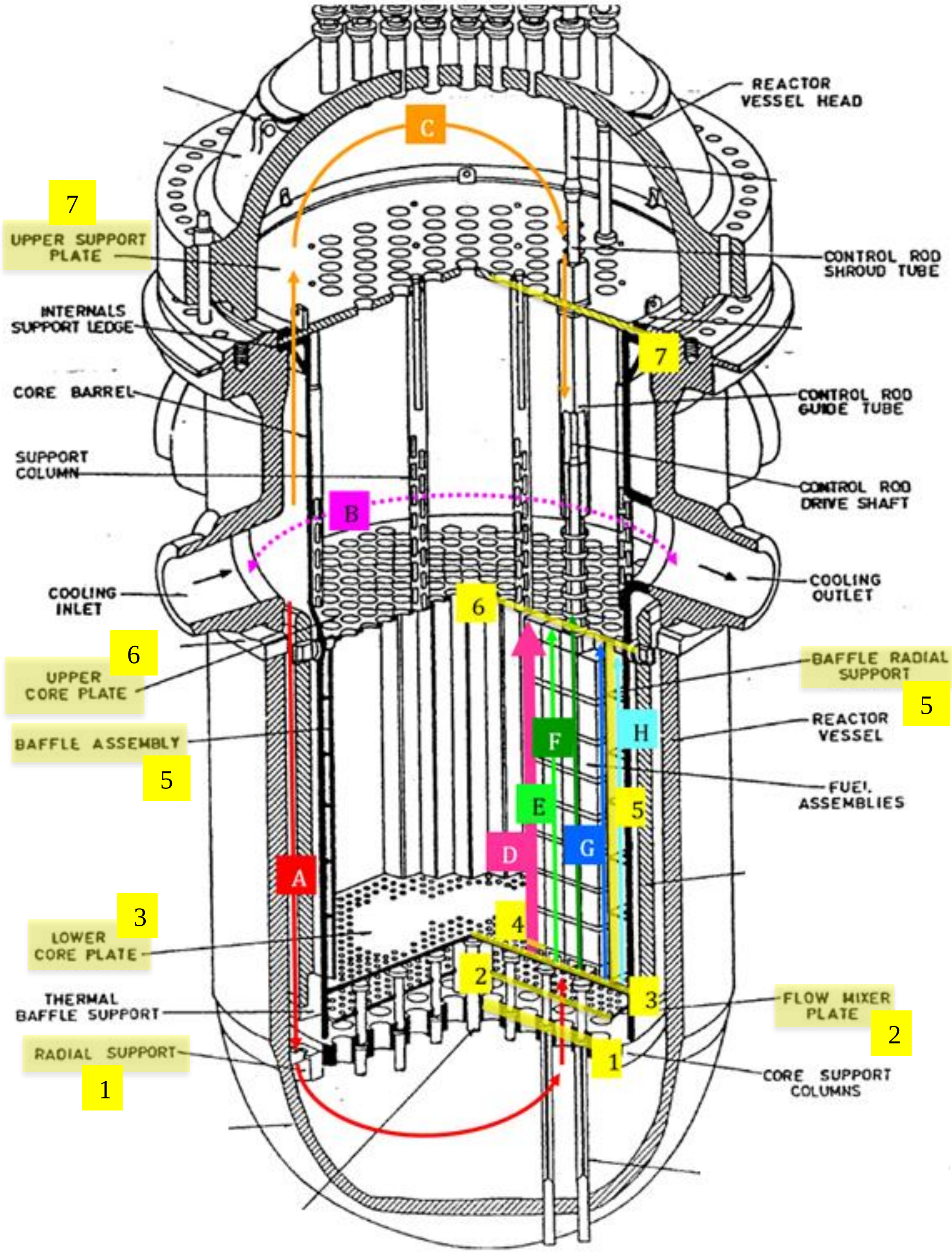


Figure 2-1. Vessel Flow Paths and Key Structural Locations [8]

Table 2-1. Vessel Flow Paths

Path (subpath)[1]	Approx. Percent of Total Flow (%)[2]	Description
A	96.5% to 98.5%	Downcomer to lower plenum to core region flow
B	<1%	Hot leg/cold keg designed leakage (nozzle bypass flow)
C	1-2.5%	Upper head spray and instrumentation cooling
D (A)	91.5% to 93.5%	Heat removal by direct contact with fuel
E (A)	2%	Empty guide tube and instrumentation rods (without thimble plugs in many updated designs)
F (A)	2%	Control rod and in-core instrumentation cooling (via guide tubes)
G (A)	0.5%	Gap between outer fuel assemblies and baffle
H (A)	0.5%	Baffle-barrel flow “core bypass”

NOTES:

1 – The parenthetical letter indicates which primary flow path the sub components are part of. IE, the flow paths D-H all originate from the downcomer flow and as such sum to equal that percentage of total flow.

2 – These values are approximate, and while given as a range, vary from plant to plant, therefore some plants may exceed or may not meet these given values.

Table 2-2. Key Structural Locations

(#) Structure	Flow Path	Description
(1) Radial Support (Bottom Support Plate)	All A (Flow passes through down from downcomer into lower plenum and then back up from lower plenum region)	Primarily structural. Large penetrations for flow and instrumentation thimbles.
(2) Flow Mixer Plate (Intermediate Diffusor Plate)	All A	Perforated plate between Lower Core Plate and Bottom Support Plate Creates uniform coolant flow to fuel assemblies. Not used in all Westinghouse units.
(3) Lower Core Plate	A to D/E/F/G/H	Structurally supports and positions fuel assemblies. Distribute coolant flow to core, preferentially towards center of core (hotter assembly locations).
(4) Fuel Assembly Base	A to D	Includes bottom nozzle, debris filter, and P-Grid.
(5) Baffle/Bypass Assembly	G to H (through LOCA holes)	Core bypass region. If available, LOCA holes are located along the vertical baffle plates. Secured to core barrel by horizontal former plates.
(6) Upper Core Plate	D/E/F/G/H to Upper Plenum	Aligns upper and lower core support structure, fuel assemblies, and control rods. Typically, coolant flows up from the core, and radially out through the outlet nozzles (hot legs).
(7) Upper Support Plate	C, through spray nozzles (Cold Leg to Upper Head) and through control rod guide tubes (Upper Head to Upper Plenum)	Primarily for attachment for other upper internals (support columns and control rod guide tubes) through designed perforations. Serves as dividing boundary between upper plenum and reactor vessel head. Structural design varies in Westinghouse RPVs, but purpose and flow path is generally the same. May have additional perforations between head and plenum, but these are not included in this study for conservatism.

3. POSTULATED BLOCKAGES AND IMPACT ON FLOW PATHS

Once the nominal coolant flow paths are understood, plausible accident scenarios under GSI-191 can be postulated and their consequences can be estimated through simulations.

3.1 Postulated Core Inlet Blockage Scenario

During a LOCA a substantial amount of debris may be generated in containment during the blowdown phase. This stage of the accident could generate debris because the hot water jet from the break could spray onto piping insulation, breaking it down into fibrous debris. This debris can then be transported to the containment floor by break flow or containment cooling spray. This is demonstrated in Figure 3-1 and Figure 3-2. [10]

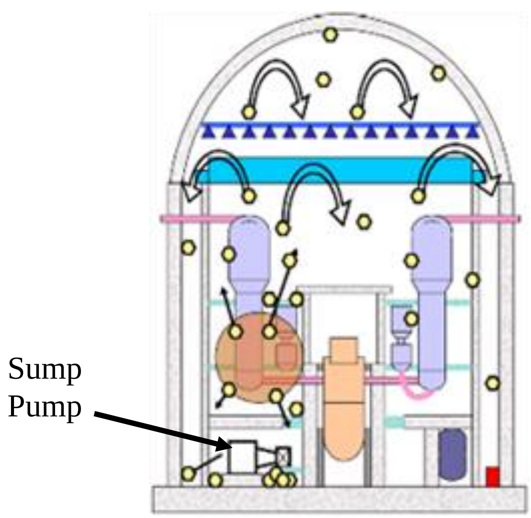


Figure 3-1. Depiction of Debris Generation Around Break [10]

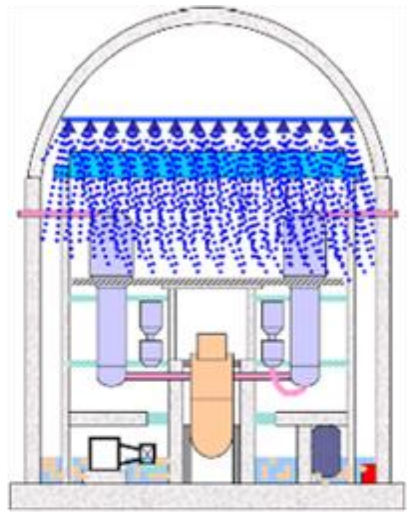


Figure 3-2. Depiction of Containment Cooling Spray [10]

Initially, the Emergency Core Cooling System (ECCS) draws water from the Refueling Water Storage Tank (RWST). All this time, injected water eventually flows out of the break and travels by gravity to the lower containment. When the RWST reaches a very low level, the sump switchover (SSO) is engaged and water is drawn for recirculation from the sump pump in the lower containment. These pumps have large strainers designed to prevent debris penetration into the primary system, such as those pictured in Figure 3-3. Debris beds tend to form on the strainers and prevent further penetration into the primary system, however, fine debris may penetrate the strainers, especially immediately after SSO. The initial postulated accident was a loss of NPSH on these strainers due to significant debris bed, therefore many plants increased the surface area of their strainers. This further could increase the risk that fine debris may reach the primary system.

Figure 3-3. Example Sump Strainer [12]

In the case that debris does enter the primary system, the most likely transport path is through the downcomer. The debris is then most likely to accumulate at the base of fuel assemblies due to the design of the debris filter and P-Grid. If the debris were to sufficiently accumulate at the core inlet region, the core flow could theoretically decrease, affecting the core coolability. Figure 3-4 shows the complexity of the filter system at the bottom of each assembly and why it is the most likely place for debris to accumulate in the lower core.

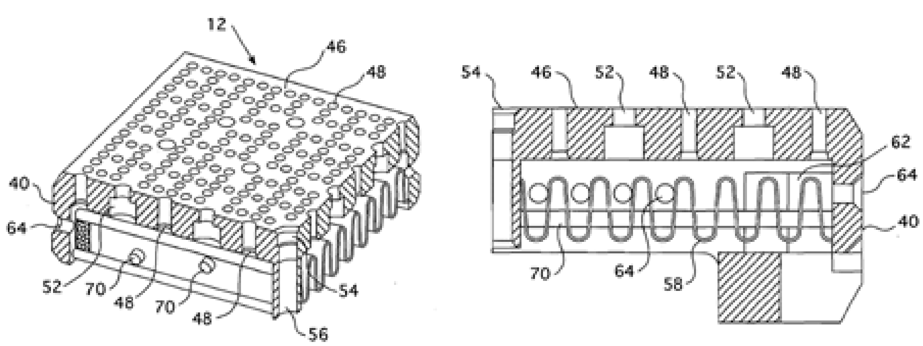
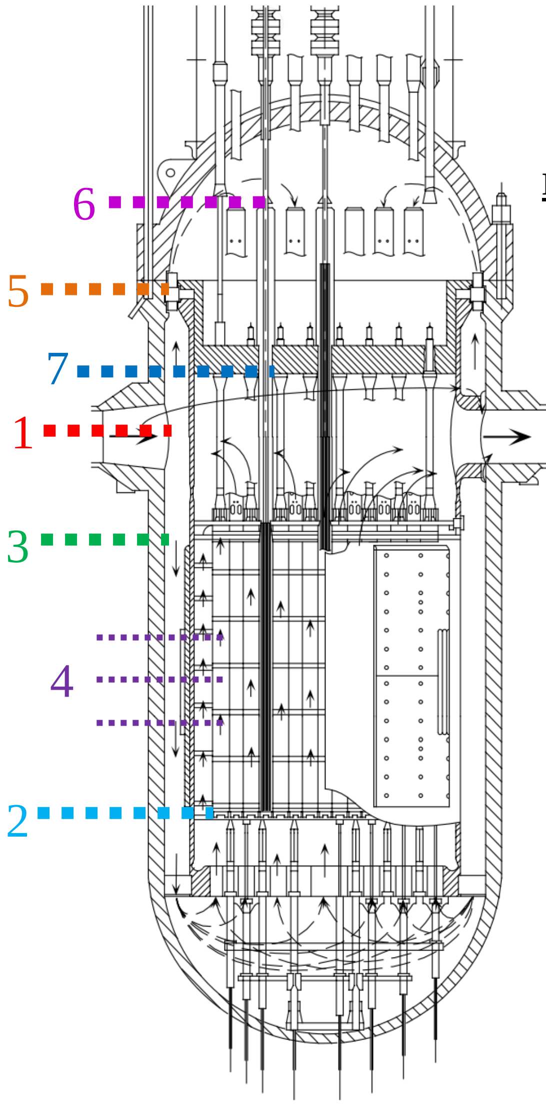


Figure 3-4. Patent Drawing of Assembly P-Grid and Debris Filter [13]

3.2 Baffle/Barrel Bypass Flow and Pressure Relief "LOCA" Holes

Under such core inlet blockage conditions, the removal of decay heat would only be possible by coolant flow reaching the core through alternative flow paths, such as the barrel/baffle core bypass region (simply referred to as "bypass" in this study). While the baffle could potentially be blocked by debris as well, as considered later, it has a larger and less restrictive flow area compared to the fuel assembly bases, therefore, considering it to be free may be reasonable given CL DEG debris generation and penetration estimates. There are certain plant specific features that can play a major role in core cooling from this bypass flow. One of these of key interest is the pressure relief holes, also known as "LOCA holes." In a typical PWR reactor vessel, LOCA holes exist in the baffle plates, enabling cross-flow between the core bypass and the core. These holes are arranged in rings symmetrically at discrete axial heights.

During core inlet blockage in a design without LOCA holes, water must flow down the downcomer, up the bypass, then back into the core from the top. In a cold leg break, this process is primarily hydrostatically driven. Key heights are marked in Figure 3-5, where 1 is the cold leg inlet center, 2 is the core inlet/bypass entrance, 3 is the top of the core, and 4 are representative heights for possible LOCA holes. In a plant with LOCA holes, water does not need to reach the top of the core height before reaching the core from the bypass as it may pass through LOCA holes at lower heights.



Height	Location
1	cold leg inlet center
2	core inlet/bypass entrance
3	top of core
4	LOCA holes
5	upper head sprays
6	top of the guide tubes
7	base of the upper support plate

Figure 3-5. Westinghouse PWR Vessel with Key Heights Noted [8]

3.3 Full Core and Core Bypass Blockage

In consideration of a more conservative blockage scenario, debris might block both the core inlet and the bypass. This is considered plausible when the baffle entrance or lower core plates are plugged. In this case, all flow through the downcomer is inconsequential and coolant can only reach the core through alternative flow paths such as the hot leg/cold leg leakage or upper head sprays.

3.4 Hot Leg/Cold Leg Leakage Flow

During normal operation, a small amount of coolant passes directly between vessel inlet and outlet nozzles via the nozzle bypass flow. This coolant leakage passes through the "small clearances between the core barrel outlet nozzles and the reactor vessel outlet nozzles." Due to the design of this leakage, credit is not taken for it in these scenarios. This is both plausible and conservative. It is plausible because the large frictional losses make this flow path least preferential, and small thermal swelling of the vessel could result in the clearance being reduced such that the pathway is closed. Additionally, it is more conservative to exclude this flow path from analysis because it provides near direct coolant access to the top of the core through the upper plenum.

3.5 Upper Head Flow

In order for flow to access the core from the upper head, it must first pass through the upper head sprays (height noted as 5 in Figure 3-5), flow through the top of the guide tubes (6), down into the upper plenum where it can flow into the core. Another key height for this alternative flow path region is the base of the Upper Support Plate (7). If the plate is perforated, coolant can bypass the guide tubes and directly access the upper plenum. If this is not the case, the coolant must fill up the head region to the height of the guide tubes before becoming available for core cooling. The conservative modelling methodology assumes no perforations exist for flow to pass directly through the upper support plate.

Another feature of note that varies from plant to plant is the specific design of the control rod guide tubes. The base of the tubes actually ends at the bottom of the upper core plate. In short, this could provide direct coolant access to the core once the upper head has sufficiently filled. However, some guide tubes are perforated in the upper plenum region, and thus flow could exit the tubes into the upper plenum after passing below the upper support plate. The conservative modelling methodology assumes perforations do exist for guide tubes below the upper support plate.

4. RELAP5 MODEL DESCRIPTION

RELAP5-3D [14] was used to simulate the primary system. The primary system input model was based on a typical 4-loop Westinghouse PWR. RELAP5-3D is a best-estimate system code which has been used extensively for modelling primary systems in steady-state as well as responses of the primary systems during transients, including LOCA scenarios in LWRs. The code, developed by Idaho National Labs, has undergone a thorough validation process for LOCA scenarios and includes physical models for prediction of phenomena through all the phases of a LOCA, including blowdown, refill, reflood, and long-term cooling. Beyond its thorough validation and being regarded well by the NRC and industry, RELAP5 was chosen because of its ability to model the entire primary system in sufficient detail, as well as model each scenario event, such as the break or blockage, in a single tool.

The containment boundary data was obtained from relevant power plant simulator data of a typical large, dry containment simulation using MELCOR, imposing the boundary conditions at the break such as break flow and enthalpy supplied by the RELAP5-3D model. MELCOR is a US NRC code which has been extensively used to perform calculation of the containment response during different types of accidents in LWRs, including LOCA. Containment data was imposed as a boundary condition for containment pressure and sump temperature in the RELAP5-3D model. When using MELCOR with RELAP, this allowed the two models to pass data in a dynamic manner resulting in more realistic simulations. Additional information about the MELCOR model is available in [15] and [11] and is not within the purview of this study.

4.1 Primary System Model

The primary system, Figure 4-1, was modeled with one-dimensional components for these simulations. All four coolant loops are independently modeled (as opposed to lumped modelling for non-break or non-pressurizer loops). The non-pressurizer loops are 2, 3, and 4, with the pressurizer on loop 1. All of the loops have steam generators that include both the primary and secondary side with heat exchange structures. The reactor core and bypass are modelled as two vertical pipes. The core includes three heat structures, simulating the average channel, hot channel (without the hottest rod), and the hottest rod.

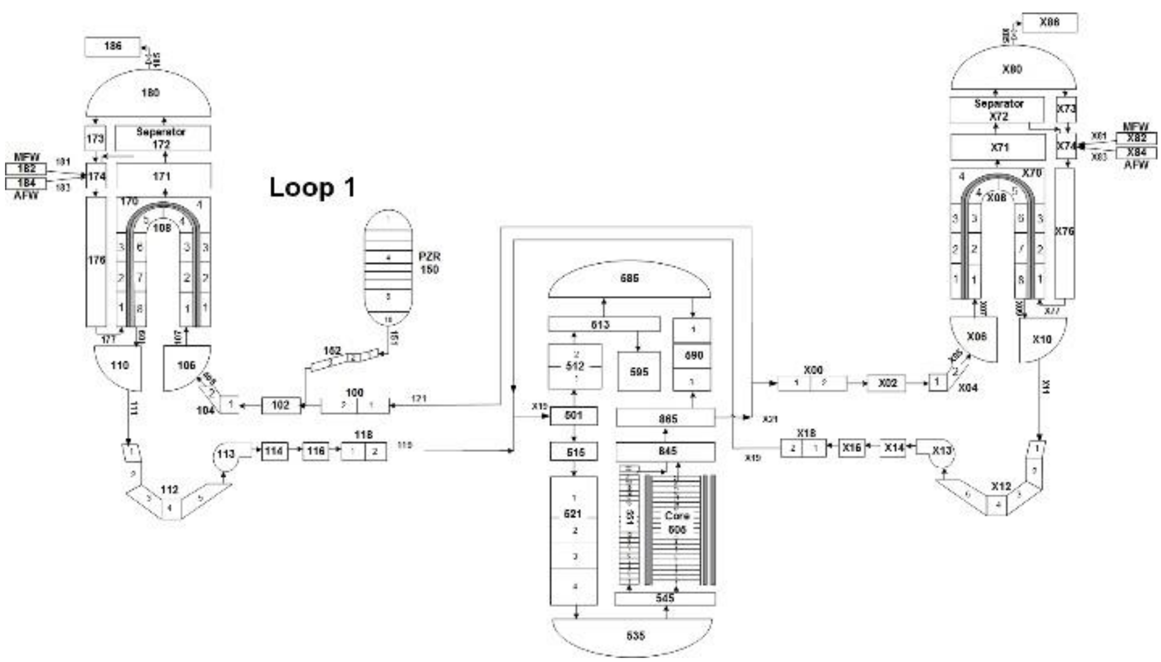


Figure 4-1. RELAP5 Primary System Nodalization

All of the pumps included in the input contain realistic pump characteristic tables (velocity of the liquid vs pressure of the primary system) and account for pressure losses through the suction and discharge lines. Control variables are included in the RELAP5-3D model to simulate the main control logic of the primary system as extensively as necessary during a LOCA scenario, including the reactor control rod scram, charging flow initiation, sump switchover, ECCS pumps actuation, and secondary side feedwater trips to name a few. This model was validated at steady-state against typical plant data. More information as well as detailed nodalization diagrams are available in [16] and [15].

4.2 ECCS Nodalization

The nodalization diagram adopted for the ECCS, including charging pumps, SI pumps, RHR pumps, and RHR heat exchangers, is shown in Figure 4-2. Trip control functions were defined to switch between the phases of the injection (Safety injection, cold leg (CL) recirculation, and hot leg (HL) recirculation).

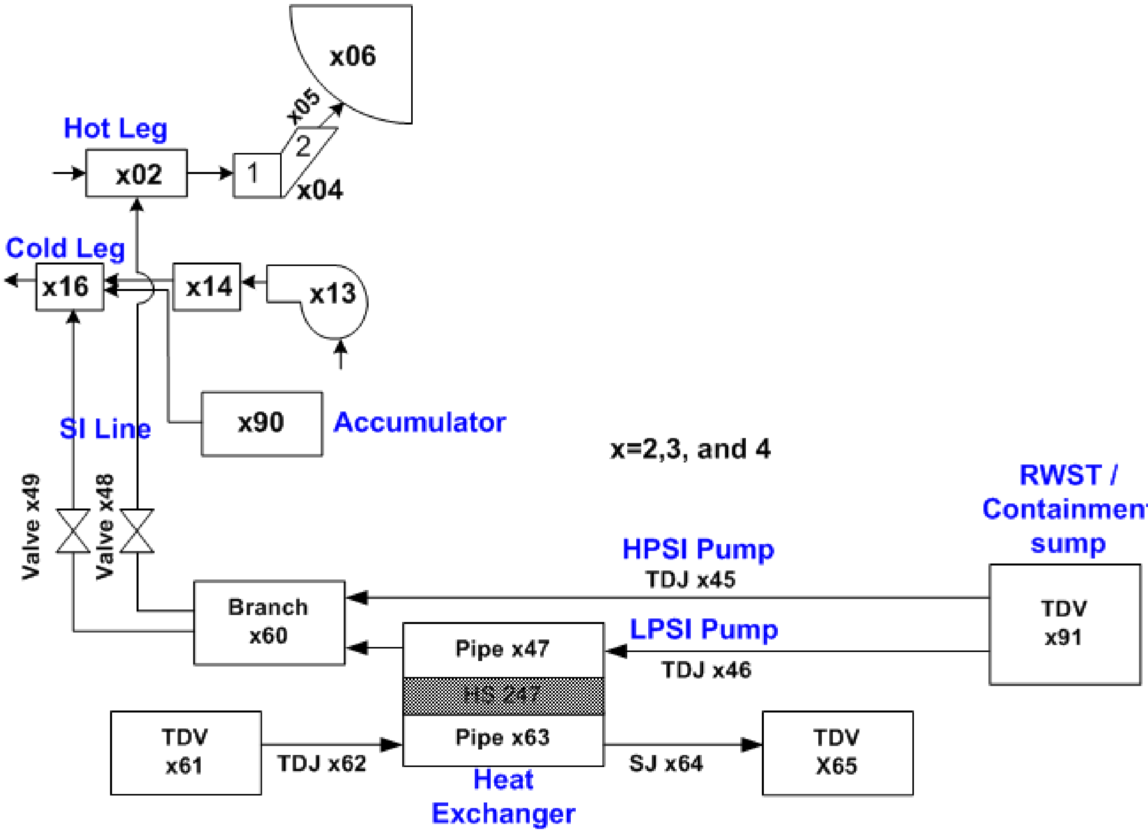


Figure 4-2. RELAP5-3D Safety Injection Nodalization Diagram
(X = loop number = 2, 3, or 4)

The RELAP5-3D ECCS includes the following features for each train:

- One Accumulator simulated using the accumulator component (Components x90). A control variable was defined in order to isolate the accumulator to account for special LOCA manual operation procedures.
- One high pressure safety injection (HHSI) injecting directly to the primary system. The pump was modeled with a time-dependent junction (X45) where

a table of the liquid velocity to be injected as a function of the pressure of the primary system injection location was defined.

- One low pressure safety injection (LHSI, TDJ x46) injecting into the primary system through the residual heat removal (RHR) exchanger. This pump was modeled with the same approach applied to the HHSI.
- One RHR exchanger, connected downstream from the LHSI pump simulated with one pipe component (X47) for the tube side of the heat exchanger, one pipe component (X63) for the shell side of the heat exchanger and one heat structure, connected to both pipes to simulate the heat exchanger tube walls. Component Cooling Water (CCW) thermal-hydraulic conditions were imposed through a time-dependent volume (X61) and mass flow rate was imposed through a time-dependent junction (X62).

The three phases of injection, namely safety injection, cold leg recirculation, and hot leg recirculation, are included in the model. During the safety injection phase (from the ECCS actuation to the sump switchover time), the ECCS draws water from the RWST (volume x91) and discharges into three of the cold legs (volumes 216, 316, and 416). These volumes were chosen for the break location because it is most conservative to inject one train directly into the break volume in loop 3. This allows for up to an entire train of injection to be lost by simply passing out of the break, which would be possible in the worst-case scenario in which the break occurs in the pipe section connected to ECCS injection in a real plant. The next stages of recirculation draw water from pumps located in the sump compartment rather than the RWST. Control logic was defined to switch between the cold leg recirculation phase, when the ECCS flow is injected into the primary system through the cold leg (via trip valve x49), and the hot leg recirculation phase, when two of the three trains (loops 2 & 3) are manually switched to the hot leg injection (via trip valve x48).

The suction volume of the ECCS pumps (RWST during the safety injection phase, and containment sump during the recirculation cooling phase), is modeled with a time-dependent volume (TDV x91) where the specific boundary conditions are supplied by RELAP tables during safety injection and by MELCOR or simulator data during CL and HL recirculation. The switch from the RWST to sump conditions (recirculation phase) is controlled by a logic control function. As mentioned before, the HHSI and LHSI were simulated with time-dependent junctions. To accurately simulate the pumped volumetric flow, the velocity in the junction is used as the reference parameter in order to account for the density of the injected water (coming from the RWST or sump) as it changes during the transient.

4.3 Break Model Nodalization

The DEG break was simulated using a series of three valve components, one connecting the two pipe sections (300-02 and 302), and one connecting each of the same pipes to break volumes (13 and 14), as seen in Figure 4-3. The hot leg ECCS injection for loop 3 enters pipe component 300-02. For a cold leg break, the pipes would be 316 and 318-1, with cold leg injection into 316. The choice of a break location adjacent to the ECCS injection volume is an additional conservatism as coolant is injected in the closest location to the break possible. This, along with the correlations used at the break are modeled as suggested by the NRC and Idaho National Laboratory for licensing. [17]

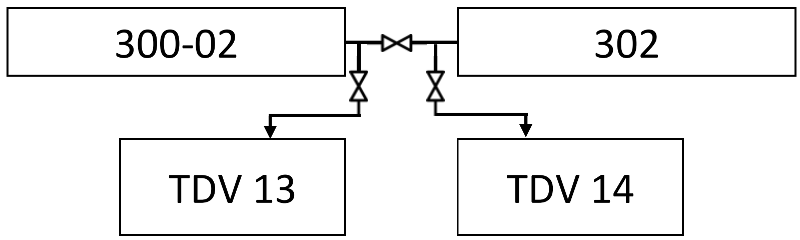


Figure 4-3. Hot Leg DEG Break Nodalization

The break transient occurs after 300s of steady-state simulation. The trip valves simulating the break open instantaneously at that time, while the valve connecting the hot/cold leg pipes instantaneously closes.

The break empties into two time-dependent volumes (TDV). The time-dependent volumes simulate the break discharge volume and contain realistic containment pressure profiles in response to a DEG accident obtained from MELCOR simulations.

4.4 Blockage Simulations

The blockage was performed according to two methodologies depending on the simulation. To simulate the core inlet (only) blockage, the forward k-loss coefficient on the core inlet junction (J545-01) was artificially increased to a sufficiently high number (1.0×10^{6}) to prevent flow into the core at the time of sump switchover. The k-loss coefficient of the bypass inlet junction was left unchanged to represent a free (unblocked) bypass. In order to perform this method of blockage, and to compare a core blockage against a simulation without blockage, the simulation was run without blockage in entirety, and a restart file was created and restarted at the time of sump switchover with the blockage changes to relevant junctions. Thus, data from simulations of this nature will be the exact same until SSO/Blockage time.

For the full core and core bypass simulation, a different approach was taken as some coolant could still flow into the core through the lower core plate in initial simulations, despite high k-loss value. This was likely because the upper head sprays also had significant frictional losses. Therefore, in order to determine the explicit impact of upper head cooling, the core blockage was modeled by removing the junctions connecting the lower core plate to the bypass (J545-02) and core pipes (J545-01). This method of blockage simulation ensures conservatism in calculations as water can only reach the core from the top. Blockage for this scenario occurs 360 seconds after sump switchover. This is based on the time estimated for debris to accumulate at the base of the core. A zoomed nodalization of the relevant regions is produced in Figure 4-4.

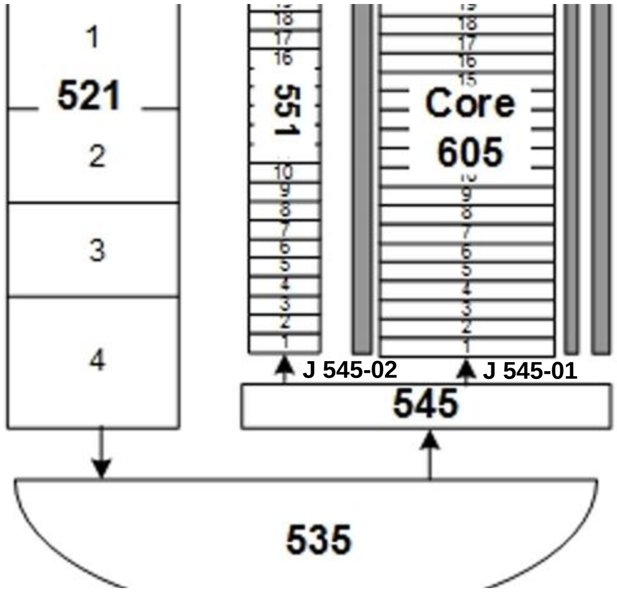


Figure 4-4. Blockage Nodalization

4.5 LOCA Holes Model Modification

The models with LOCA holes are a modification of the standard primary model without LOCA holes. All dimensions, components, and simulation conditions were maintained and the LOCA holes have been included in the model. In the model, these holes are classified by their level's height (1 through 3), located at various spacing increments from the bottom of the baffle plate. Three different input models with LOCA holes were produced, with one, two, or three levels of LOCA holes.

The RELAP5-3D model simulates the core bypass and the core as two separate vertical pipes. Each pipe is nodalized into 21 equally sized volumes. For this simulation, the LOCA holes are modeled as cross-junctions between the two pipes, occurring at the midplane of nodes 6, 9, and 12, as shown in Figure 4-5. Because the core model used in this simulation is the one-dimensional pipe component, the LOCA holes at each level were lumped and each cross-junction represents all of the LOCA holes that occur in a single axial level. The result of this "lumping" approach is three LOCA holes with the same hydraulic diameter as a single hole and the total flow area equal to the sum of all of the holes on each level. The height and number of levels, as well as the size and number of individual holes each level may vary for individual plants, and this particular arrangement was assumed valid for this study.

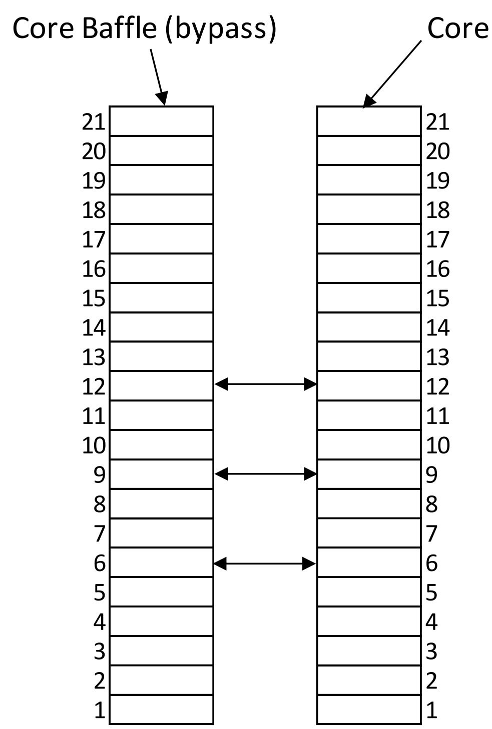


Figure 4-5. RELAP5-3D Nodalization Showing Placement of Three Equivalent LOCA Holes

In order to estimate the K-loss coefficient of each level of equivalent LOCA holes, a number of correlations have been considered. The most basic geometry considered was for flow through an orifice with sharp edges, where the LOCA hole is modeled as an orifice, suggesting the selection of the correlation [18] shown in Equation 6.1 valid for sufficiently high Reynolds numbers for such flow geometries:

$$k = \frac{\Delta p}{\rho w_0^2/2} = \left(1 + 0.707\sqrt{1 - \frac{F_0}{F_1}} - \frac{F_0}{F_1}\right)^2 \qquad \text{(Eq. 4.1)}$$

In Equation 6.1, F_0 and F_1 are the flow area of hole and the injection ambient. The k-loss coefficients (forward and reverse) implemented in the RELAP5-3D model were estimated conservatively assuming the flow area of the core and baffle regions (F_1) to be infinitely larger compared to that of individual LOCA holes (F_0). This implies that the ratio $F_0 / F_1 = 0$ in Equation 6.1. With this assumption the resulting k-loss coefficient was found to be:

$$k = \frac{\Delta p}{\rho w_0^2/2} = 2.914 \qquad \text{(Eq. 4.2)}$$

4.6 Upper Head Nodalization

In earlier simulations, the nodalization of the upper head followed a simplified version of typical upper head nodalization. [14] This was not expected to impact simulations where coolant could still reach the core through the baffle/bypass, but it was necessary to ensure the nodalization accurately reflected the true upper head geometry when the only path for coolant to the core was through this region. In order to improve the realism of the model, the upper head region was re-nodalized from the simplified nodalization to reflect a more detailed representation of each key region. Additionally, the conservative assumption was made that the water must reach the top of the guide tubes to travel through them (i.e. no perforations in the top tube section or upper support plate as mentioned previously). The simplified nodalization is shown below in Figure 4-6, with the updated nodalization in Figure 4-7. In the figures, the blue regions were unchanged between nodalizations and represent the core inlet (501), downcomer entry (501), upper plenum (865), and upper core plate (845).

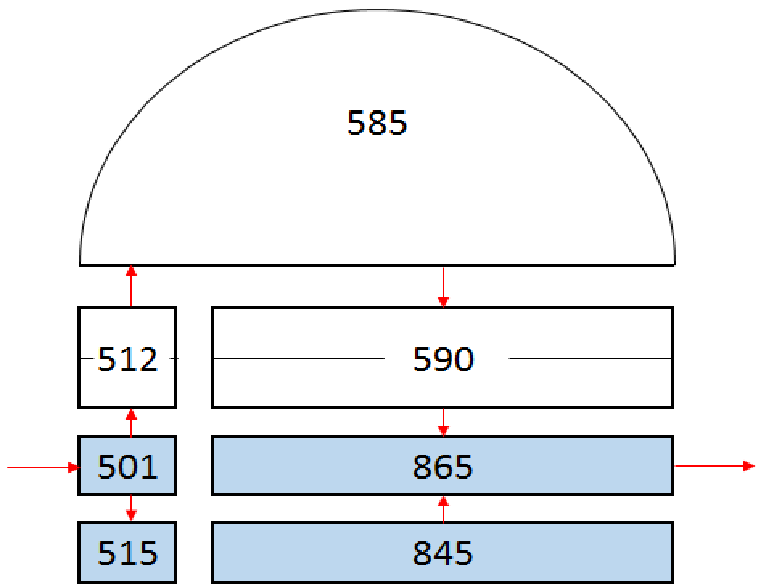


Figure 4-6. Simplified Upper Head Nodalization

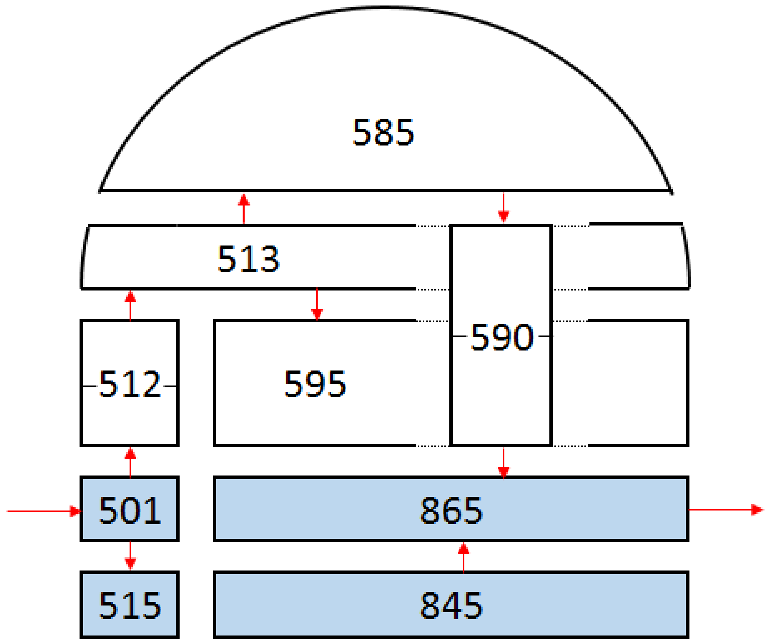


Figure 4-7. Detailed Upper Head Nodalization

The simplified model allowed for water to reach the vessel exit through the guide tube section in an easier manner as the exit of the upper sprays (512) is at the same height as the top of the tube section (590). The detailed nodalization conservatively models these at discreet heights, requiring water to first fill a “dead end” volume (595), and a lower section of the dome (513) before entering the guide tubes (590). Additionally, passive heat structures were added extensively to the detailed model representing the vessel and all internal piping and structures (some existed previously, but only for the outer RPV).

In the example of a Westinghouse plant found in the RELAP5 user manual (volume V) [14] the upper head is nodalized as seen in Figure 4-8. This example looks like a sort of hybrid between the simplified and detailed models. The key flow paths in the example model are:

1) Guide tubes (129) connect the upper dome (126) to the bottom of the upper core plate (118)

2) The upper plenum (122) also connects the upper head/dome (126) to the vessel exit (120)

3) HL/CL leakage connects (102) with (120)

The first flow path is reflected in the detailed nodalization, but the tubes open into the vessel exit (upper plenum) rather than the bottom of the upper core plate. The detailed model is more realistic (with given assumptions) than the example given because of the large number of perforations in the guide tubes in the vessel exit section as well as the larger flow area of the perforations relative to the smaller flow area in the tubes themselves. This means that once coolant passes below the upper support plate, it can be considered as unrestricted, resulting in a more conservative and realistic model than one which directs all guide tube flow directly to the top of the core. Additionally, control rods in the inserted configuration (as it would be during a shutdown) at the base of the guide tubes may restrict flow in that region, further compounding uncertainty.

The second flow path was reflected in the simplified nodalization as pipe 590. To ensure conservatism in calculations, as well as assess the impact on cooling from the guide tubes specifically, the new nodalization only has flow from the upper dome to the vessel exit through the guide tubes. This is realistic because some reactor designs may not have additional flow paths through the upper support plate.

The third flow path, HL/CL leakage has significant pressure loss terms and a very small flow area as it is a designed leak, as previously mentioned. Due to uncertainty in performance or availability of this flow path during an accident scenario with possible thermal swelling, it is not modeled. It is more conservative to remove it as it provides a flow path at the same elevation as the upper plenum.

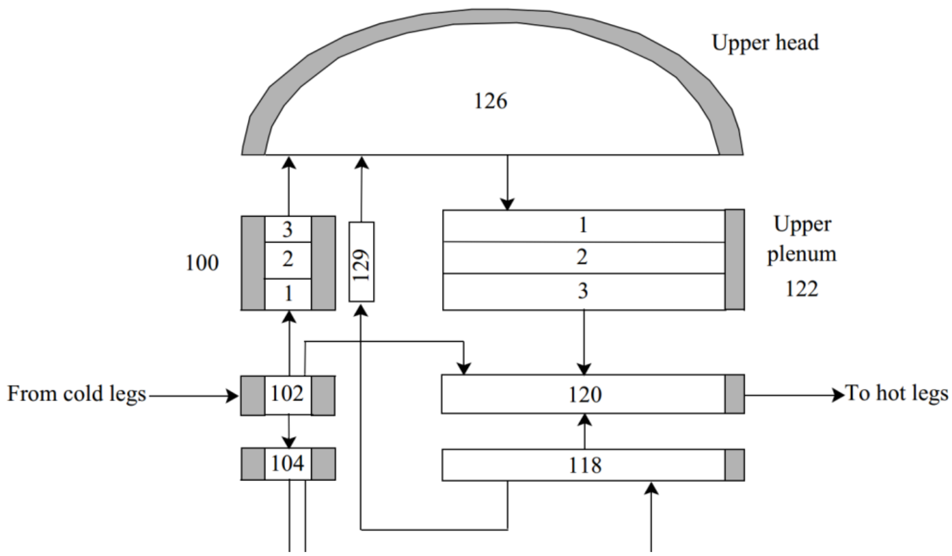


Figure 4-8. RELAP Manual Upper Head Nodalization for Westinghouse PWR [13]

4.7 General Simulation Procedure

All transient simulations have been preceded by a steady-state simulation, to best replicate the initial conditions of the nuclear power plant before the break. The transients begin with 300 seconds of null-transient, to insure stabilization of the initial conditions of the transient simulations.

At 300 seconds, the DEG break is initiated according to the procedure previously described. The sump switchover trip time varies depending on break size and ECCS actuation, but is very similar for DEG breaks in either leg. The blockage trip is initiated either at the time of SSO or is delayed by a few minutes according to time it would take to for debris to sufficiently accumulate. Regardless of specific timing, the blockage is full and instantaneous once triggered. The simulation proceeds through long term cooling flow recirculation according to the ECCS procedure described in Section 4.2, with HLSO occurring at a fixed time, approximately five and a half hours after the break, regardless of break size. After the HLSO, the simulation will continue to run in this actuation until reaching quasi-steady state. This is summarized in Table 4-1.

Table 4-1. Simulation Procedure Summary

Simulation Time (seconds) *[Not to Scale]*

Time (s) *not to scale*	-300	0	~1800	~2150	~20000	∞
Trip	n/a	Break	SSO/ Blockage (no delay)	Blockage (w/ delay)	HLSO	n/a

<table>
<tr><td>Simulation Phase</td><td>Steady-State</td><td>Safety Injection</td><td colspan="3">Long Term Cooling</td></tr>
<tr><td>ECCS Draw</td><td>n/a</td><td>RWST</td><td colspan="3">Sump Pump (Recirculation)</td></tr>
<tr><td>ECCS Injection</td><td>n/a</td><td colspan="3">3 Trains Cold Leg</td><td>2x Hot Leg, 1x Cold Leg</td></tr>
</table>

5. SIMULATION RESULTS

When evaluating simulation results, the most important safety parameter for any simulation is core coolability. There are a number of factors that play into core coolability, but the key indicators are peak cladding temperature (PCT) and coolant inventory.

Peak cladding temperatures are measured in RELAP heat structures, and are pulled directly from output data. The safety limit of PCT of interest to this study is 1478K due to cladding oxidation limits. It should be noted that regulatory limits may be lower, especially during long term cooling.

Coolant inventory is most easily examined through the Core Collapsed Liquid Level (CLL). CLL is a method of looking at overall void fraction in a region, as coolant is often dispersed and not entirely liquid when the core depressurizes in an accident scenario. Core CLL is calculated from the RELAP liquid fraction (volumetric liquid fraction) multiplied by the height of each node. The nodes across the core are then summed and a representative height of liquid vs vapor for the entire core is available as a parameter of interest. CLL can be expressed mathematically according to the following formula:

$$\mathrm{CLL} = \sum_{node=1}^{21} (\mathrm{height_{node}} \cdot \alpha_{\mathrm{node}}) \qquad \text{(Eq. 5.1)}$$

Core collapsed liquid level is calculated from the liquid fraction in each of the core's 21 nodes multiplied by the height of each node. Additionally, flow patterns in the primary system are important for explaining system behavior and coolant availability, and integral mass flow shows these flow patterns as a trend, making it especially useful. Other parameters may be of interest to specific scenarios and will be discussed as relevant.

5.1 Cold Leg DEG Break with Core Inlet Blockage

The most severe scenario to consider under the premise of this study would be a double-ended guillotine break in the cold leg. Obviously the DEG break is the largest size possible, which allows for rapid blowdown and uninhibited exit of coolant through the break during safety injection. The cold leg break is more severe than the hot leg break as coolant is not forced to go through the RPV before exiting the break. This may also mean debris is not forced through the core, but a sufficient amount of debris reaching the lower core to induce inlet blockage is assumed for the purposes of this study. It qualitatively results in an almost entirely hydrostatically driven process of coolant supply which is confirmed quantitatively.

To determine impact of LOCA holes as an alternative flow path, a series of input models were prepared, one without LOCA holes, and three with LOCA holes, including one level, two levels, and three levels of LOCA holes. The LOCA holes start at the bottom as depicted by Figure 4-5, such that the single LOCA hole model is the lowest hole and the two LOCA hole model has the lower two holes. The standard simulation procedure was carried out for each input model, summarized in Table 5-1 with a color scheme to assist in identifying data in this section. For simulations without core blockage, the key core coolability parameters are essentially indistinguishable, therefore only the model without LOCA holes is displayed (in green).

Table 5-1. Transient Simulations Matrix for CL DEG with Core Inlet Blockage

	Simulation Phase		
	Steady-State	Safety Injection	Long-Term Cooling
Without LOCA Holes	Null Transient	DEG CL Transient	Core Blockage
			No Core Blockage
With 1 LOCA Hole			Core Blockage
			No Core Blockage
With 2 LOCA Holes			Core Blockage
			No Core Blockage
With 3 LOCA Holes			Core Blockage
			No Core Blockage

In the first simulation performed, the two CL DEG LOCA transients were performed on the model without LOCA holes, one with core blockage at the sump switchover and one without. For the blockage simulation, the core inlet was instantaneously blocked as described in the previous section. After blockage occurs the maximum peak cladding temperature rapidly exceeded the safety limit. Figure 5-1 shows this comparison of two runs of the model without LOCA holes near the sump switchover time. Later simulation times are not of direct relevance to results as the simulation is considered over if the safety margins are exceeded. The HLSO would also allow water to simply flow directly onto the top of the core, making this alternative flow path irrelevant.

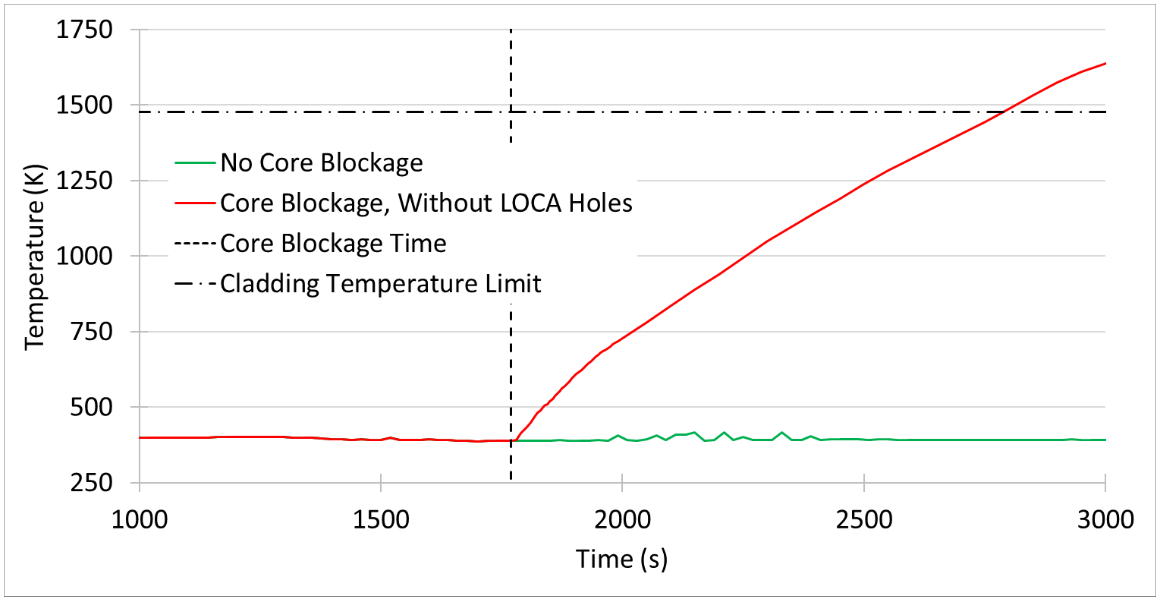


Figure 5-1. Maximum Peak Cladding Temperature (Base Case and Core Blockage Scenario without LOCA Holes)

As mentioned in Section 3.2, In a cold leg DEG core blockage transient scenario the supply of coolant to the core is primarily determined by the hydrostatic head between the cold leg and the top of the core. Before blockage, the hydrostatic head must overcome the frictional losses through the flow path between the ECCS injection location (cold leg) and the top of the core, including the inlet nozzle, downcomer, and lower core plates. Due to the lower pressure drop between the injection point and the break, the injected water is preferentially directed toward the break (either directly out from loop 3, or into the annular vessel entrance region then out the broken leg from loops 2 and 4), while a relatively small fraction goes through the downcomer. Before the break this is usually enough to replace the water evaporating in the core, however after the blockage, the flow area is reduced and the frictional losses are higher as flow must pass through the bypass (including the horizontal former plates) to reach the top of the core.

Figure 5-2 gives a visual representation of this phenomena. This may explain the lack of core coolability and the subsequent increase of the cladding temperature shown in Figure 5-1 for the core blockage scenario.

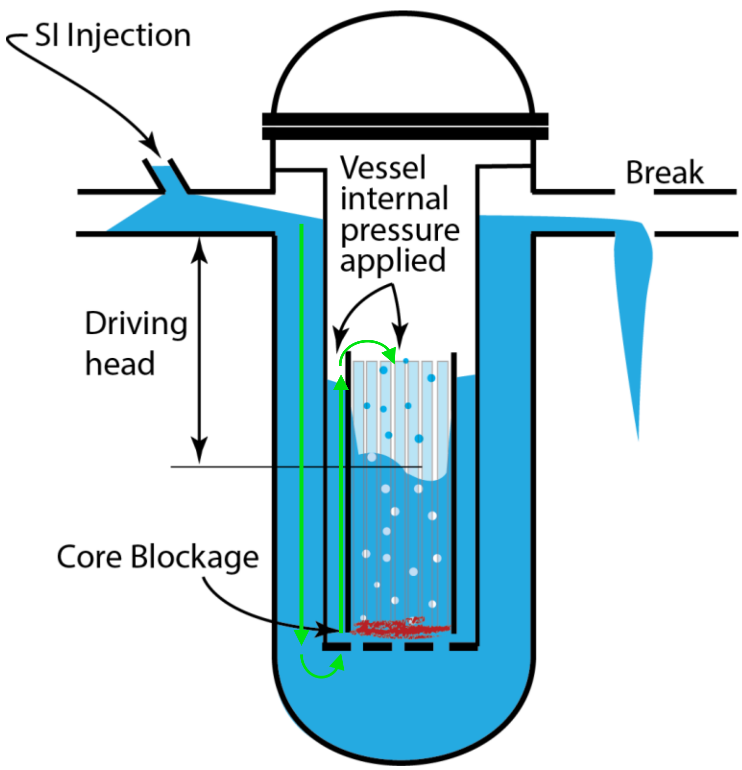


Figure 5-2. Flow Path Scheme during Core Blockage (without LOCA holes)

Due to height of the cold leg, a significant amount water is not expected to pass through the upper head in a cold leg DEG break. Because the bypass is therefore the only realistic route during a CL break for cooling water to take in the event of a core inlet blockage, as explained in section 3.2, it would be expected that the LOCA holes would provide additional flow paths to the core and positively affect the core coolability under core blockage conditions. Therefore, adding this feature to the model and investigating its effects is very important to GSI-191 research as well as model realism.

The addition of LOCA holes is expected to increase the supply of coolant to the core through holes located at lower elevations as the effective hydrostatic head is higher than the top of the bypass. This is visually represented in Figure 5-3.

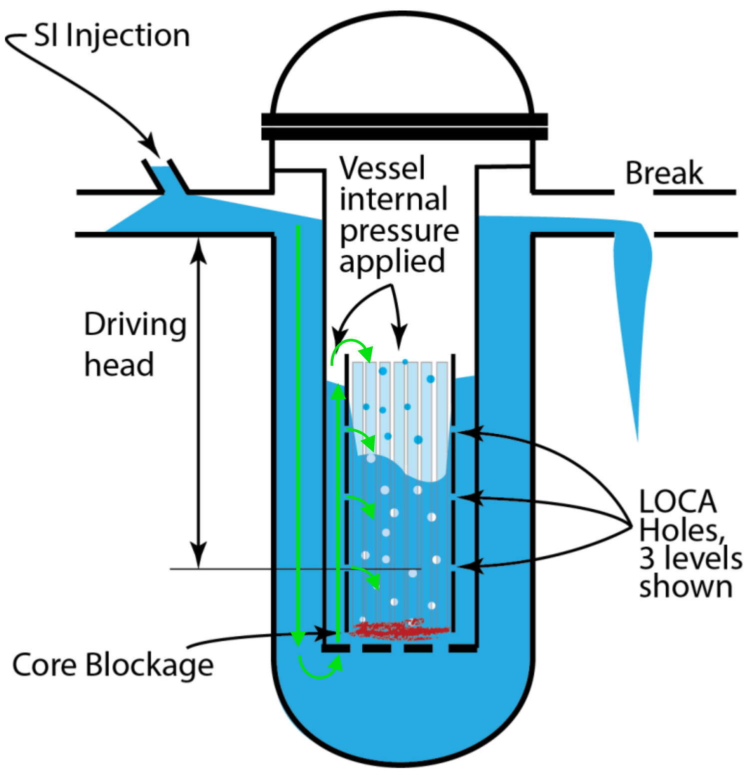


Figure 5-3. Flow Path Scheme during Core Blockage (with LOCA holes)

The comparison of peak cladding temperature for the core blockage simulations with and without the LOCA holes is shown in Figure 5-4. The effect of the addition of LOCA holes in the transient simulation showed an improvement of the core coolability, maintaining the peak cladding temperature below the safety limit.

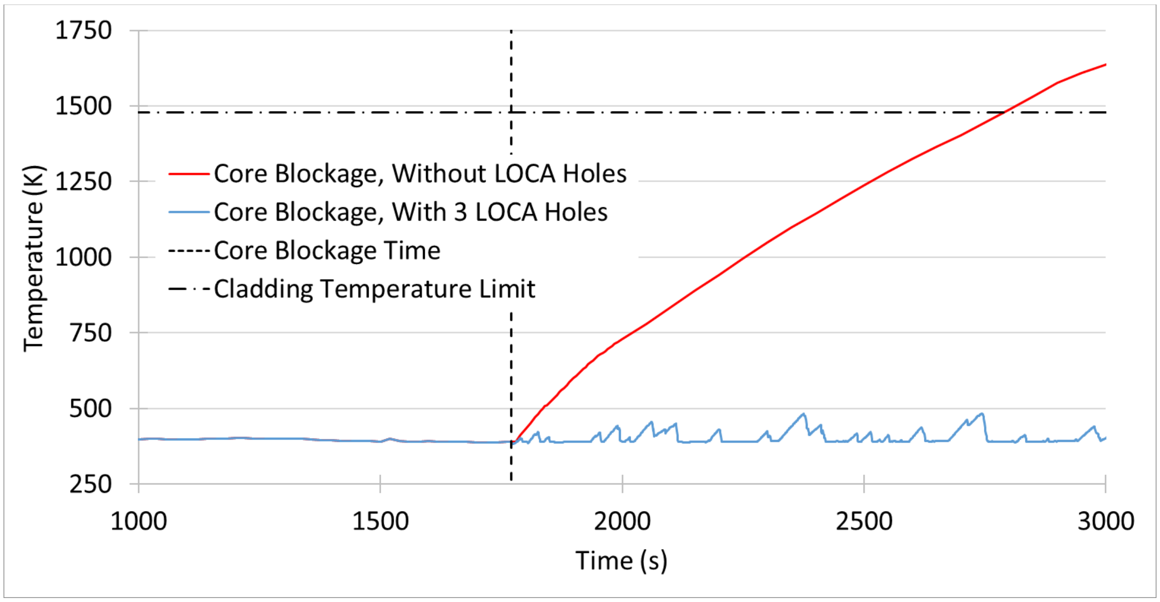


Figure 5-4. Maximum Peak Cladding Temperature (Core Blockage with and without LOCA Holes)

The impact of the addition of LOCA holes is so substantial, that even a single row of holes at the lowest elevation is predicted by the simulation to be sufficient to prevent cladding from exceeding failure limits. The comparison of all three input models with LOCA holes is shown in Figure 5-5 (with a zoomed temperature scale to better show differences). The comparison of the single LOCA hole model is compared to the results without blockage (with the same temperature scale) in Figure 5-6 and shows that while there are minor cladding temperature spikes, the presence of a single level of LOCA holes significantly reduces the impact of core blockage.

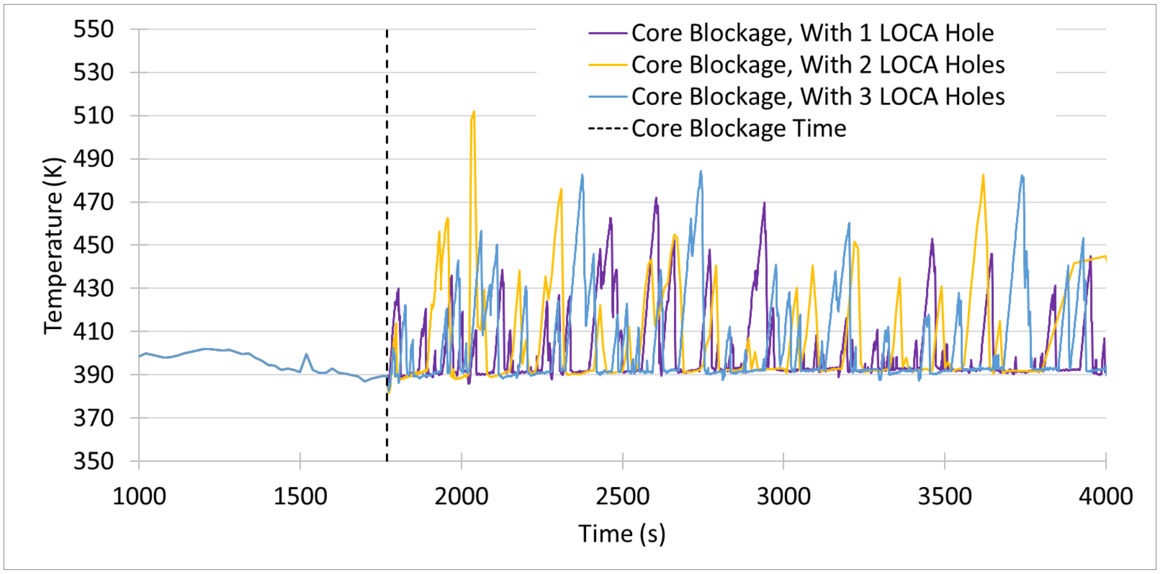


Figure 5-5. Maximum Peak Cladding Temperature (Core Blockage for All Three Models with LOCA Holes)

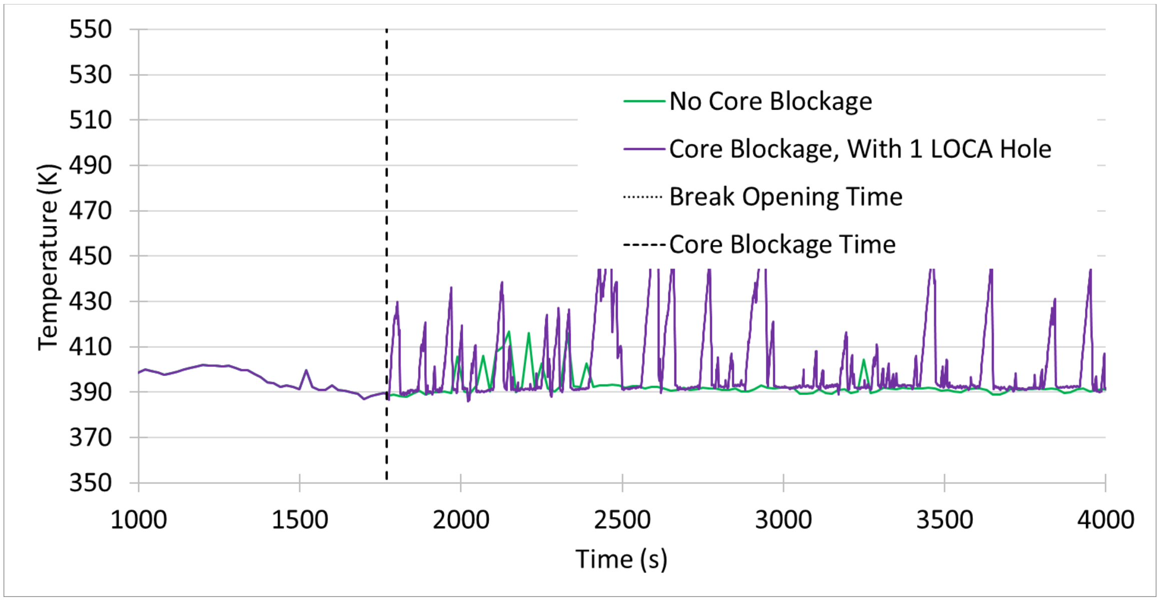


Figure 5-6. Maximum Peak Cladding Temperature (Core Blockage for 1 Level of LOCA Holes Compared to No Core Blockage)

The core collapsed liquid level is shown in Figure 5-7. The effect of the LOCA holes is shown as a higher core liquid level (blue) than the simulation without LOCA holes (red). The simulation with LOCA holes clearly has a lower total liquid volume than a simulation without blockage (green), but the amount of liquid is sufficient to keep maintain an adequate heat removal rate. Figure 5-8 shows the same data, but for the model with just one LOCA hole. The similarities in liquid level between the one and three LOCA hole models and the impact of a single level of LOCA holes on PCT indicates that the lowest height LOCA hole may be the most impactful on core coolability, regardless of total number of levels of LOCA holes in the baffle.

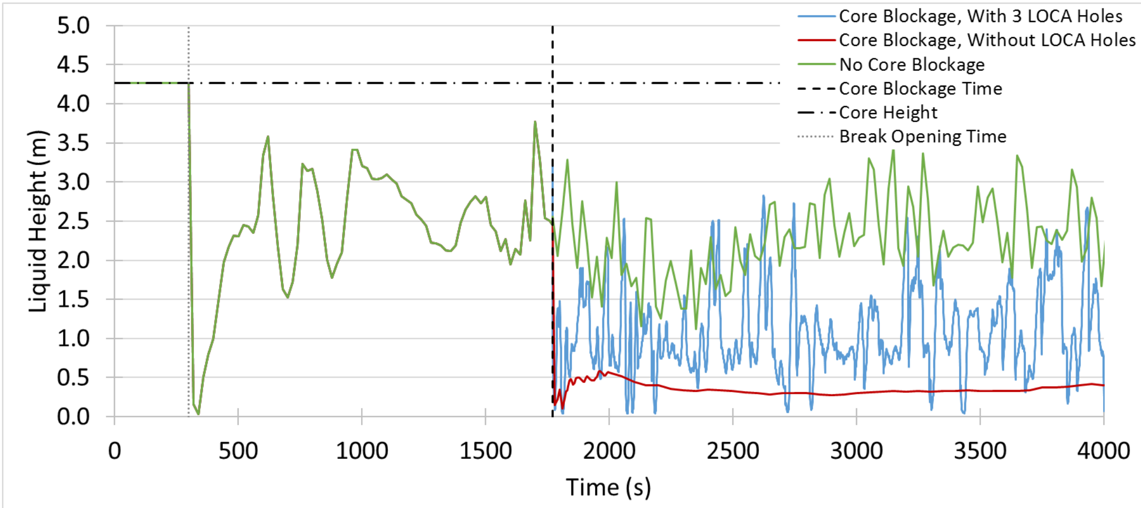


Figure 5-7. Core Collapsed Liquid Level (Unblocked Core and Core Blockage Scenario with 3 LOCA Hole Levels and without LOCA Holes)

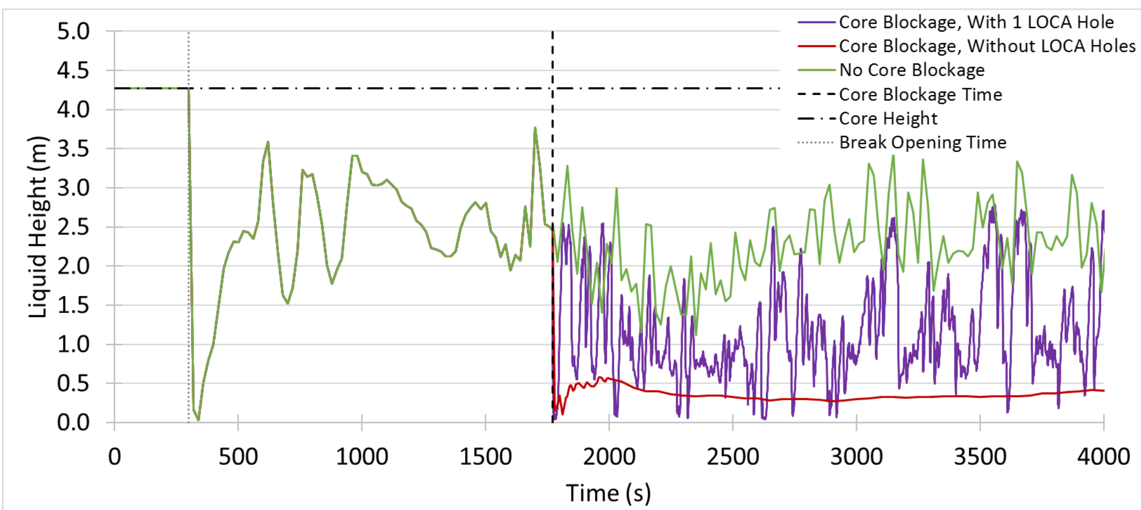


Figure 5-8. Core Collapsed Liquid Level (Unblocked Core and Core Blockage Scenario with 1 LOCA Hole Level and without LOCA Holes)

Bypass integral flow mass, shown in Figure 5-9, represents the total mass of the flow through bypass region over time, with data artificially zeroed at each trip (0s and

1770s). Examining core bypass flow is the most effective way to see if the coolant has the opportunity to reach the core, as the bypass is the dominant flow path for coolant to reach the blocked core. From 0-300 seconds, it is clear a great deal of coolant is forced through the bypass during normal, steady operation. This gives some scale for subsequent flow. After the break, there is little relative flow in through the bypass. After the core blockage time, the results showed a net increase in the mass flowing through the bypass inlet as an effect of the presence of the LOCA holes, whilst in the model without the LOCA holes, the net flow was positive but predicted to be substantially lower.

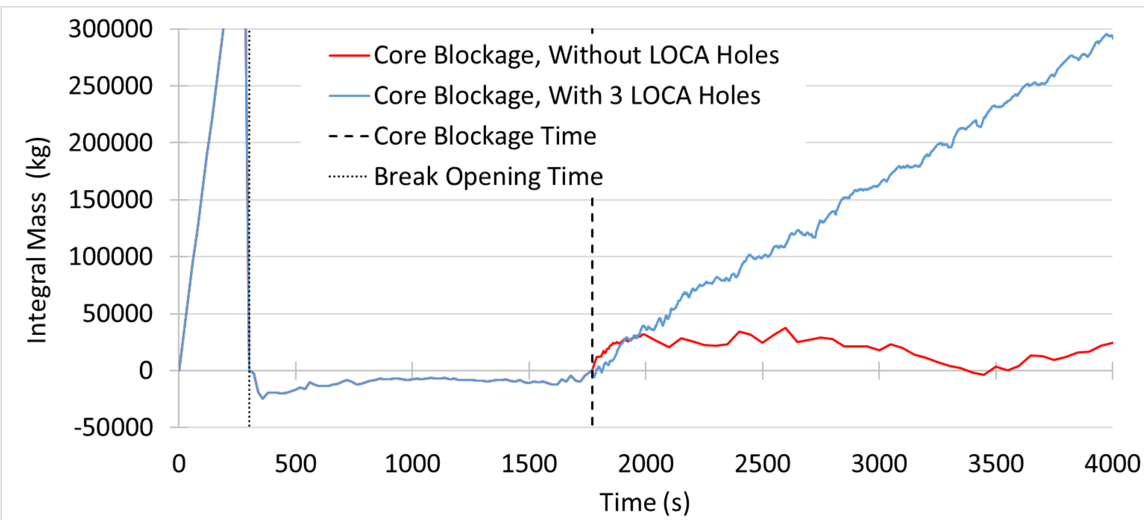


Figure 5-9. Bypass Integral Flow Mass (Core Blockage Scenario with and without LOCA Holes)

In summary, cladding temperature was predicted to increase in the core blockage scenario if LOCA holes are not accounted for, due to a substantial decrease of the water supplied to the core. The effect of LOCA holes in improving core coolability by increasing coolant supply to the core through the bypass was predicted by the simulations. The simulation also predicted that even a single level of LOCA holes at the lowest height was sufficient to cool the core for extended periods of time.

While there are not yet other studies or simulations with which to compare the validity of the simulation results with the addition of LOCA holes, the base model has been thoroughly vetted and the LOCA holes were added in a valid manner, therefore, the results should be considered valid. This work shows the importance of modeling plant specific features when performing best-estimate calculations to support the GSI-191 research and shows the degree of impact a single design feature may have on core coolability.

5.2 Hot Leg DEG Break with Full Core/Bypass Blockage

While a cold leg DEG was determined to be the most severe scenario, it was not chosen for the full core and bypass blockage due to a number of factors. First and foremost, the cladding reaches a failure limit in a cold leg DEG with full blockage regardless of nodalization as coolant is initially injected into the cold legs and therefore preferentially flows out of the break. Second, due to the ECCS injecting into the cold leg before HLSO, a hot leg break may force more debris down the downcomer and through the core than a cold leg break which could result in more significant blockage. While obviously conservative in general, this makes the full core and core bypass scenario more likely to occur with a hot leg break.

During a hot leg break with full core blockage, the injected coolant has one of two (considered) paths to exit the primary through the break:

1) By passing through the upper head sprays, through the guide tubes, and exiting the upper plenum via the broken loop's HL nozzle
2) By filling a Steam Generator (SG) and effectively reversing the normal coolant flow path. This can happen according to two subpaths:
    a. If the broken loop is refilled in this manner, coolant can directly exit over the U-tubes an through the break;
    b. Otherwise, if it fills a non-broken SG, it must pass over the U-tubes and through the upper plenum still, and exit the upper plenum via the broken loop's HL nozzle.

This is due to safety injection actuation before the HLSO, where all three SI trains are injecting into the cold leg. Pushed by the SI pump head, which operates at an inverse to primary pressure, that is, it injects a larger volume of coolant in response to lower primary pressures, there is sufficient force in this configuration for either scenario to occur. However, because the elevation of the steam generator is much higher than the upper head sprays, coolant is at least initially directed through the sprays after core/bypass blockage.

As noted in Section 4.6 the initial model of the upper head was rather simplistic, but was considered sufficient as the upper head flows were not expected to have a significant impact on core coolability in a cold leg break or hot leg break with free bypass. However, examining a hot leg break under the scenario of full core/bypass blockage requires accurate modeling as it becomes a very important flow path. The simplistic model showed that the core survived, but it allowed flow to immediately pass into the upper plenum from the sprays due to not accounting for key upper head elevations. Therefore, core survivability under the more realistic, more conservative model was of interest and a comparison with the simplistic model was necessary.

The first figure of merit when examining the differences between the models was Peak Cladding Temperature (PCT). The results from the new design showed that the peak cladding temperatures were not significantly different from the reference, and the core was kept well below safety limits during the full HL DEG break and blockage transient. There is a "stepped" temperature increase for both models, thanks to the delay of 360s after the sump switchover before the blockage occurs, resulting in the first jump from warmer sump recirculation water, and 360 seconds later, another increase due to the blockage itself. The cladding temperature These results are seen in Figure 5-10.

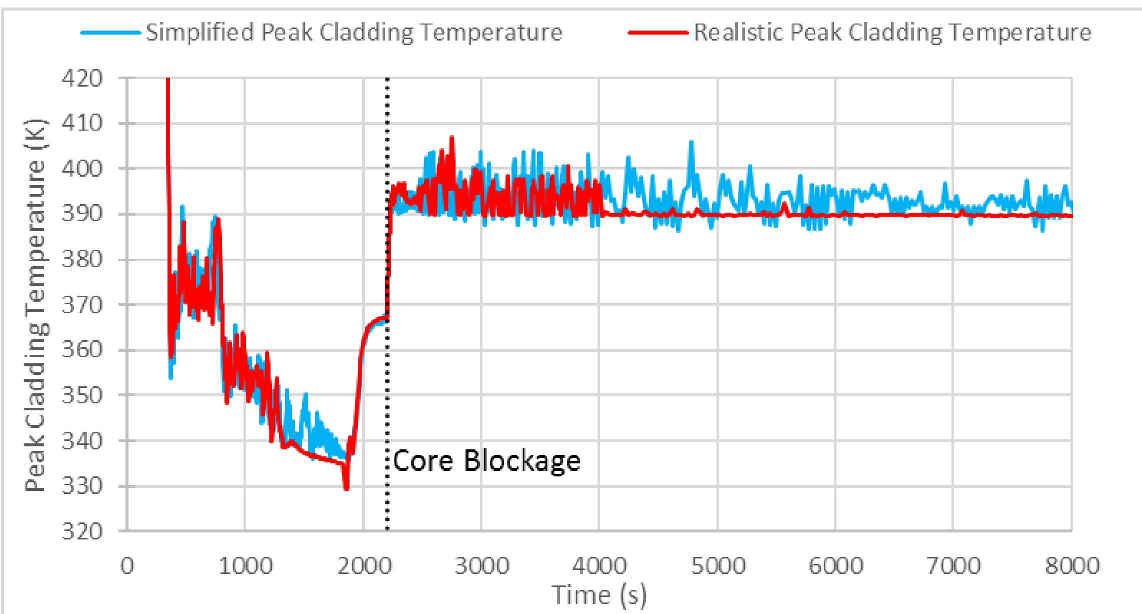


Figure 5-10. Maximum Peak Cladding Temperatures (HL DEG)

As Figure 5-11 shows, the average Core CLL is very similar for both models, with less fluctuation from the newer model. The reduction in fluctuation is primarily due to the more constricted and realistic geometry, which does not permit rapid movement of large amounts of fluid from the upper head to the upper plenum (and thus the core).

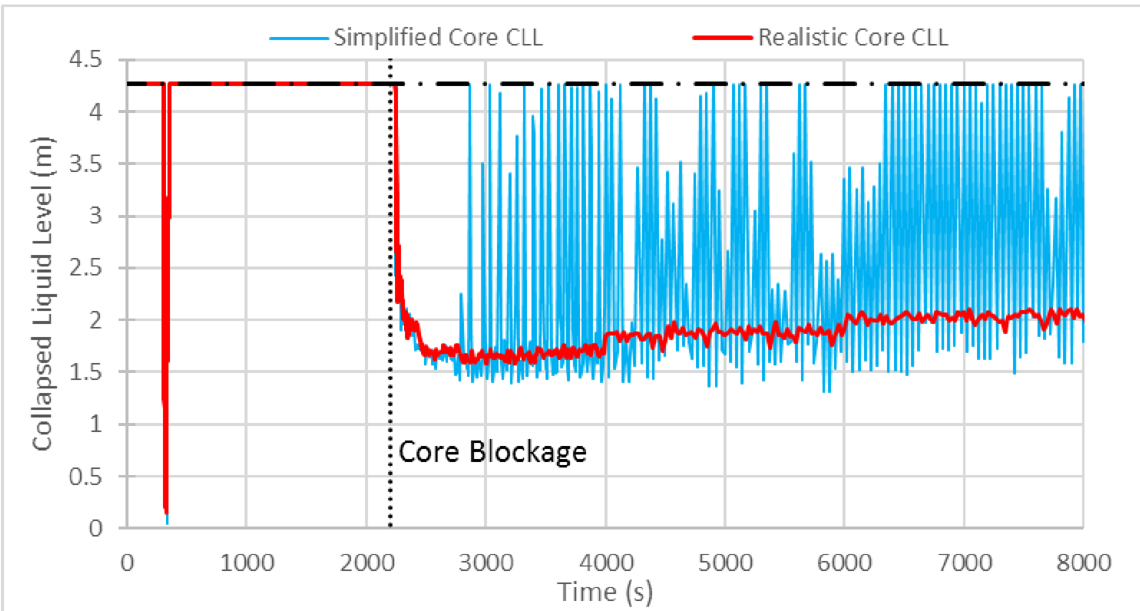


Figure 5-11. Core Collapsed Liquid Levels (HL DEG)

The integral flow plot in Figure 5-12 is "zeroed" artificially before the break and again at the core blockage to aid in distinguishing the flow patterns at each stage of the transient. Positive slopes on the plot indicate flow in the marked direction, while negative slopes indicate the opposite.

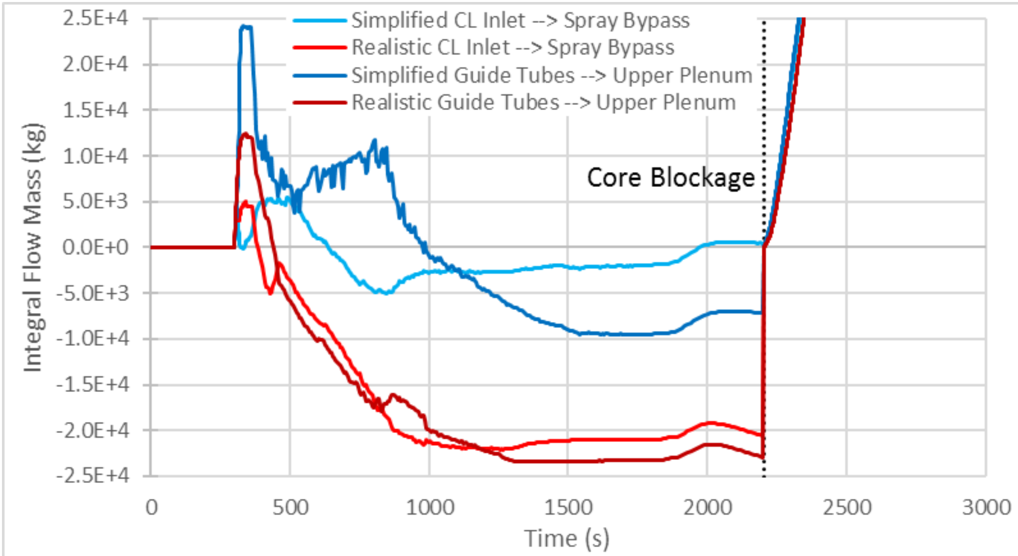


Figure 5-12. Upper Head Integral Flow Behaviors (HL DEG)

The results make sense qualitatively as the steep, positive slopes immediately after the break (300-320s) in the Guide Tubes to Upper Plenum/Vessel Exit direction indicate that both upper head volumes are emptying, mostly through that preferential and larger flow area (rather than reverse flow through the spray nozzles) with the simplified model emptying more because of the lack of the "dead volume" below spray nozzle height. When the two slopes diverge (positive guide tube to plenum, negative CL inlet to spray flows), the upper head area is emptying (and conversely filling when they converge), which can be confirmed by examining the liquid fraction of the total upper head volumes (volumes above 501/865 in Figures Figure 4-6 and Figure 4-7). The reference case tends to exhibit greater degrees of emptying as it has a larger flow area across the upper support plate. After the core blockage, a strong positive slope is expected and seen, as this becomes the only path for water to exit the vessel.

The liquid fraction plotted against integral flow in Figure 5-13 confirms the expectations of the initial loss of coolant in the upper head immediately after the break. Additionally, the expected head emptying when integral flows diverge is confirmed. Some clear examples of this behavior are marked in Figure 5-14, Figure 5-15, and Figure 5-16, for the initial emptying, simplified upper head, and realistic upper head, respectively.

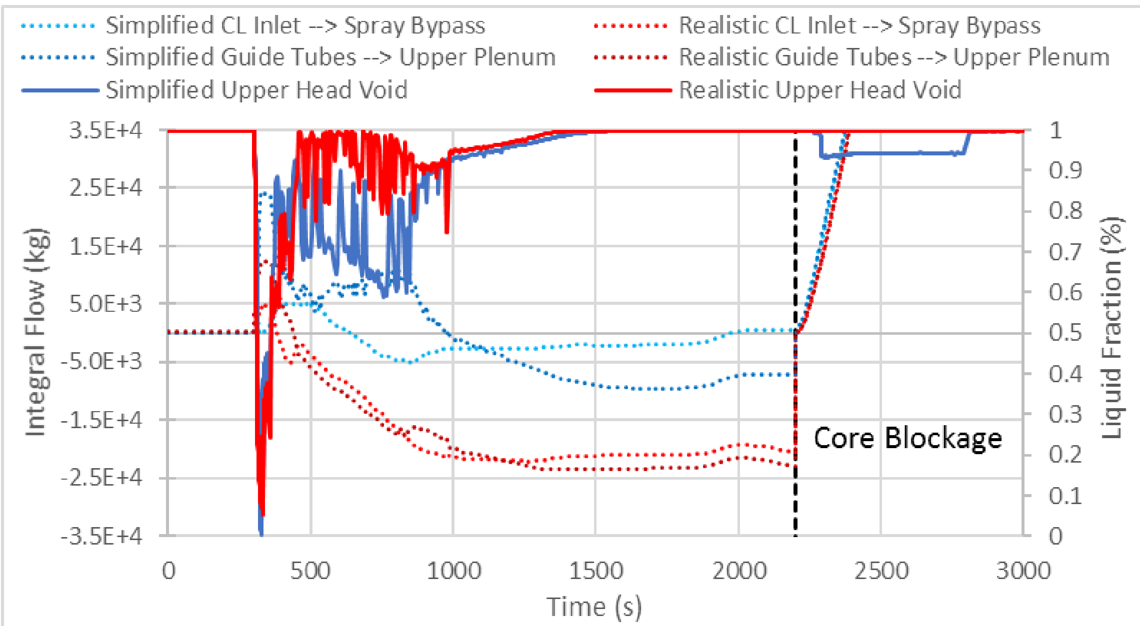


Figure 5-13. Upper Head Integral Flow Mass vs. Liquid Fraction (HL DEG)

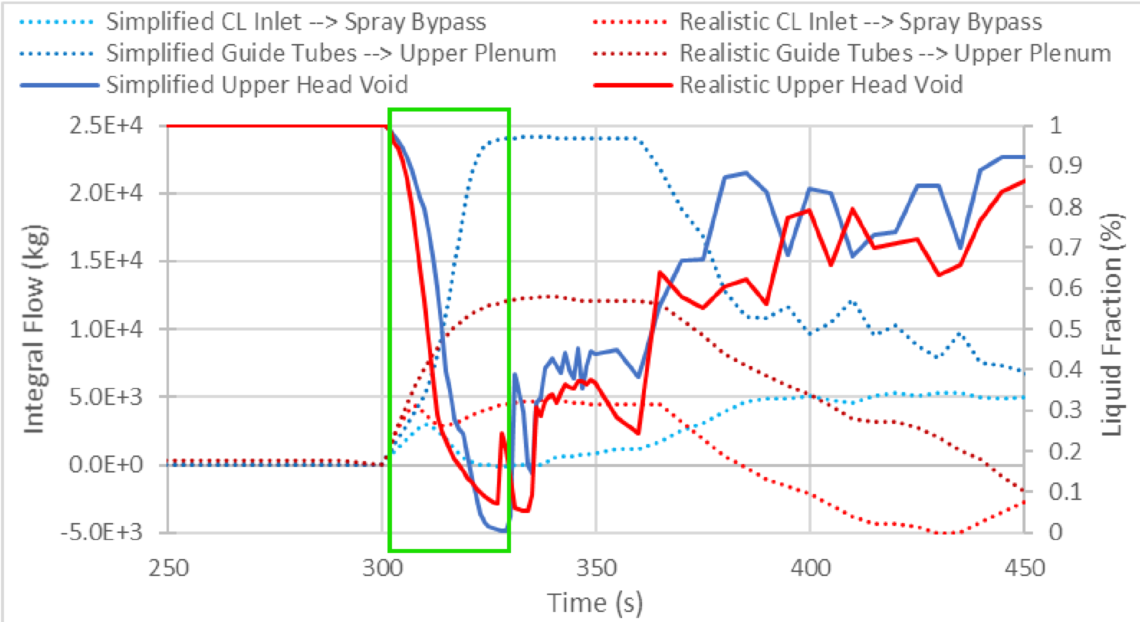


Figure 5-14. Flow and Void Behavior During Initial Break and Subsequent Rapid Upper Head Emptying

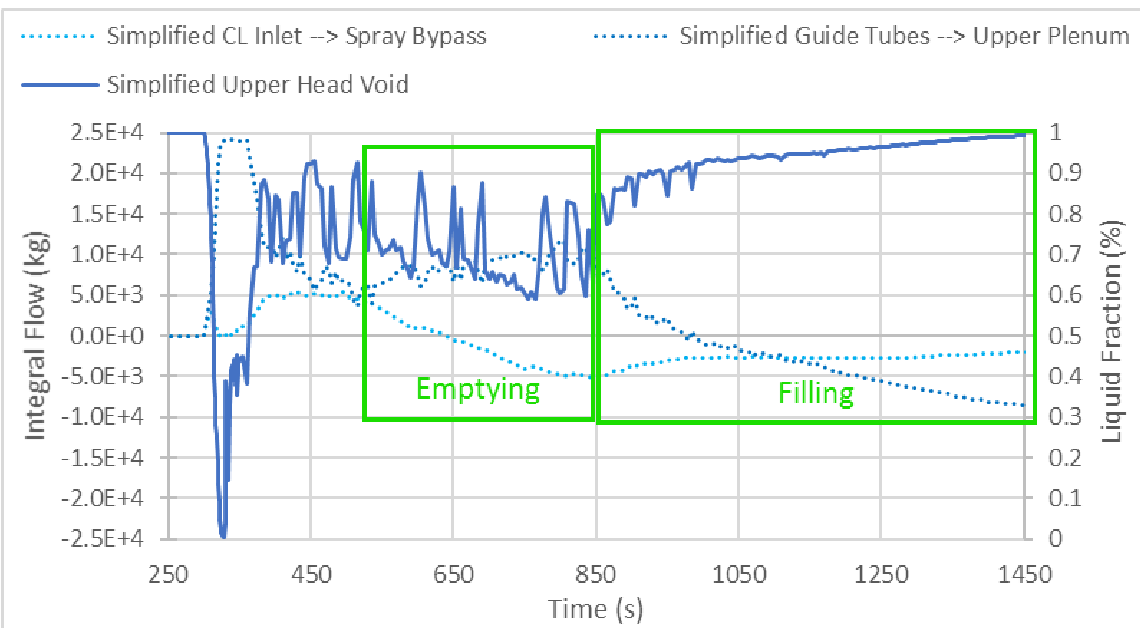


Figure 5-15. Flow and Void Behavior for Simplified Upper Head

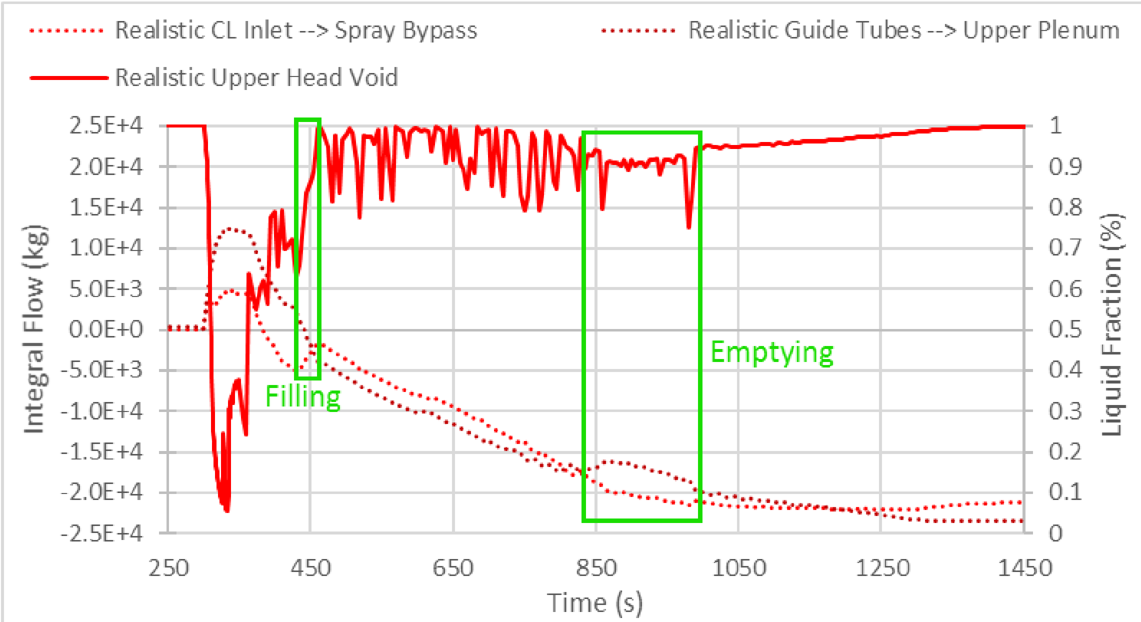


Figure 5-16. Flow and Void Behavior for Realistic Upper Head

In considering the other coolant flow path, via steam generator refill, there are a number of factors that may influence if a steam generator refills, which steam generator refills if this occurs, and how quickly after the blockage it will refill if such behavior occurs. Due to the large number of factors that play a role in this behavior, and the difficulty in conducting and validating a separate effects test specifically for SG Refill, this study does not consider it in detail. The data obtained in this study indicates that SG refills may have no significant effect on the core coolability in a HL DEG core/bypass blockage, which is somewhat reassuring as it essentially shows safety margins could potentially be maintained with the loss of a portion of ECCS injection, which was already somewhat confirmed in the CL break as the break location allowed an entire train of injection to pass directly out of the break.

Overall the more realistically nodalized model seems to better represents expected behavior by taking into account geometrical effects on flow paths and restrictions due to upper head internal components. The results showed that the more realistic geometry had little effect on overall core coolability, with the peak cladding temperature remaining well below safety limits in both cases. This is reassuring in the case of an accident where this flow path could become vitally relevant when it may not have been previously considered.

6. CONCLUSIONS

Under some of the most severe possible scenarios for debris penetration, core coolability is still possible due to the numerous flow paths for coolant to reach the core. This study demonstrated the importance of ensuring models accurately reflect a plant's specific features such as LOCA holes, a single feature which may prove to be the difference in surviving a cold leg DEG under hypothetical core blockage. The changes made to emphasize realism and ensure conservatism in the upper head showed that under some of the most conservative modelling assumptions, core coolability could be maintained during a hot leg DEG break with full core and bypass blockage.

This work represents significant contribution towards the risk-informed closure of GSI-191. While the probabilistic analyses may show this sort of large break accident and further conditions for this scenario to occur to the degree postulated as unlikely to occur, they represent bounding conditions and add deterministic data in support of problem analysis. The ability to iterate over varying predicted break sizes and blockage configurations in RELAP shows a cooperation between probabilistic and deterministic calculation methods.

This thesis also represents the framework for thermal hydraulic safety analysis established by Texas A&M University to address this regulatory burden for typical Westinghouse four-loop PWRs. RELAP is a well-documented and NRC approved code. The plant specific features, such as ECCS configuration, presence of LOCA holes, structural upper head design can be easily modified and accident components such as break size, location, and blockage can be varied according to needs. If this general methodology is approved by the NRC, it can be applied to many similar plants, where, if survivability of severe accidents and blockages is ensured, may help close the GSI-191 issue for them as well.

REFERENCES

[1] United States Nuclear Regulatory Commission, "About NRC," 5 April 2016. [Online]. Available: http://www.nrc.gov/about-nrc.html. [Accessed 1 December 2016].

[2] United States Nuclear Regulatory Commission, "Generic Issue Program," 6 August 2016. [Online]. Available: http://www.nrc.gov/about-nrc/regulatory/gen-issues.html. [Accessed 1 December 2016].

[3] E. A. Lahti and J. Zhang, "Generic safety issue number 191 (GSI-191) - status and research activities.," *Nuclear Saety and Simulation*, vol. 5, no. 2, pp. 92-98, 2014.

[4] Z. Mohaghegh, E. Kee, S. Reihani, R. Kazemi, D. Johnson and et al, "Risk-Informed Resolution of Generic Safety Issue 191," in *ANS PSA 2013 International Topical Meeting on Probabilistic Safety Assessment and Analysis*, Columbia, SC, USA, 2013.

[5] Texas A&M University, Department of Nuclear Engineering, "Experimental Activities for GSI-191 at Texas A&M," 2016. [Online]. Available: http://nuclear-research.tamu.edu/media/45678/tamu_gsi_191_poster.pdf. [Accessed 1 December 2016].

[6] J. Yan, K. Yuan, E. Tatli, D. Huegel and Z. Karoutas, "CFD Predictions of Pressure Drop for the Inlet Region of a PWR Fuel Assembly," Westinghouse Electric Company, Columbia, SC, 2010.

[7] Westinghouse Electric Corporation Water Reactor Divisions, The Westinghouse Pressurized Water Reactor Nuclear Power Plant, Pittsburgh: Westinghouse, 1984.

[8] Westinghouse Technology, "0519-R304P - Westinghouse Technology Manual 3.1 - Reactor Vessel and Internals," U.S Nuclear Regulatory Commission, 2011.

[9] K. H. Luk, "Pressurized-Water Reactor Internals Aging Degredation Study - Phase I," Oak Ridge National Laboratory, Oak Ridge, TN, 1993.

[10] Texas A&M University Department of Nuclear Engineering, "Thermal Hydraulic Research Laboratory - Projects - GSI-191," 2016. [Online]. Available: http://nuclear-research.tamu.edu/thermal-hydraulic-research-laboratory/projects/gsi-191/. [Accessed 1 December 2016].

[11] Performance Contracting Inc., "Sump Strainer Screens," Engineered Systems Group (ESG), [Online]. Available: http://www.pciesg.com/. [Accessed 1 December 2016].

[12] Y. Aleshin, "Nuclear Fuel Assembly Debris Filter Bottom Nozzle". Columbia, SC, US Patent 20110164719, 7 July 2011.

[13] Idaho National Laboratory, "RELAP5-3D User’s Manual, INEEL-EXT-98-00834," Idaho National Laboratory, Idaho Falls, ID, US, 2005.

[14] R. Vaghetto, B. A. Beeny, Y. A. Hassan and K. Vierow, "Analysis of Long-Term Cooling of a LOCA by Coupling RELAP5-3D and MELCOR," in *ANS Annual Meeting*, Chicago, IL, US, 2012.

[15] R. Vaghetto, B. A. Beeny, Y. A. Hassan and K. Vierow, "Dry Containment Response during a Loss of Coolant Accident using RELAP5-3D and MELCOR," in *ANS Annual Meeting*, Atlanta, GA, US, 2013.

[16] T. M. Crook, R. Vaghetto, A. Vanni and Y. A. Hassan, "Emergency Core Cooling System Sensitivity Analysis for a Four-Loop Pressurized Water Reactor with Three Independent Injection Trains," in *ANS Winter Meeting*, Anaheim, CA, US, 2014.

[17] R. R. Schultz and C. B. Davis, "Recommended Models & Correlations and Code Assessment Matrix for Creating a 10CFR50.46 Licensing-Version of RELAP5-3D: Pressurized Water Reactors," Idaho National Laboratory, Idaho Falls, ID, US, 2003.

[18] I. E. Idelchik, Handbook of Hydraulic Resistance. 2nd Ed., Washington, D.C.: NRC (Reprinted), 1986.

[19] R. Vaghetto, A. Franklin and Y. A. Hassan, "Sensitivity Study of Hypothetical Debris-Generated Core Blockage Scenarios," in *ANS Winter Meeting*, Washington, D.C., US, 2013.