Using NASA's GFSSP Code for Steady State and Transient Modeling of Gas Cooled Reactor Passive Safety Systems

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INTRODUCTION

In the design and development of passive cooling systems for the gas cooled fast reactor (GFR), it was found that industry standard system codes (RELAP, MARS-GCR) were cumbersome and not intended to be used for nuclear system design. These codes do not allow for common design rationale to be used through flexibility in selection of correlations or methods that are utilized. These restrictions, due to the licensing basis of such codes, prevent a designer from making preliminary decisions and conclusions on system design.

Initially, a FORTRAN code system was developed (LOCA-COLA) for the analysis of passive safety systems in gas cooled fast reactors. The LOCA-COLA package was unique in its ability to implement mixed convection heat transfer and friction factor correlations, two items that are imperative in GFR safety analysis. However, the author quickly saw the limitations of this option due to the complexities in the network solutions required by system codes. Therefore the author eventually found NASA's Generalized Fluid System Simulation Program (GFSSP) code package. GFSSP is a general-purpose computer program for analyzing steady state and time-dependent flowrate, pressure, temperature and concentrations in a complex flow networks. The program is capable of modeling phase changes, compressibility, mixture thermodynamics, and external body forces such as gravity and centrifugal.

This paper covers some initial investigations into the use of NASA's GFSSP code package for modeling and simulating passive emergency cooling systems for GFRs. The code is benchmarked against the previous results obtained from LOCA-COLA and RELAP. Finally, some future uses of GFSSP are explored.

DESCRIPTION OF THE ACTUAL WORK

Initial Work with RELAP/ATHENA and MARS-GCR

During the design and analysis of passive cooling safety systems for the GFR, it was found that existing thermalhydraulic code packages such as RELAP/ATHENA or MARS-GCR are cumbersome for performing these types of preliminary analysis. Although these codes are robust and heavily verified and validated as part of the nuclear industry's safety analysis arsenal, they were never meant to be used for design analysis. A major issue with these codes is the lack of flexibility in adding user defined correlations and the difficulty in quickly testing trade-off scenarios or determining impacts on design decisions.

Development and Previous Work with LOCA-COLA

As part of the GFR research effort, a group at MIT developed the LOCA-COLA code, in FORTRAN, as a way to perform quick design calculations on passive and active gas cooled reactor emergency systems [1,2,3]. The code was developed to perform a simple single closed loop system analysis using a one-dimensional lumped parameter approach and solving for core and cooler performance using a finite-differencing scheme. It also utilized existing NIST routines for calling the thermophysical properties of fluids.

The ability to incorporate user defined heat transfer and friction factor coefficients and the implementation of a mixed convection map were later added. Finally the ability to perform multiple parallel channel solutions in the core was added.

This code was benchmarked against RELAP/ATHENA and MARS-GCR in order to insure base functionality in comparison to the highly developed system codes [1]. The comparison study was performed on a simplified GFR passive/active safety system design as shown in Figure 1. The results will be described in the later section when they are compared with the current work.

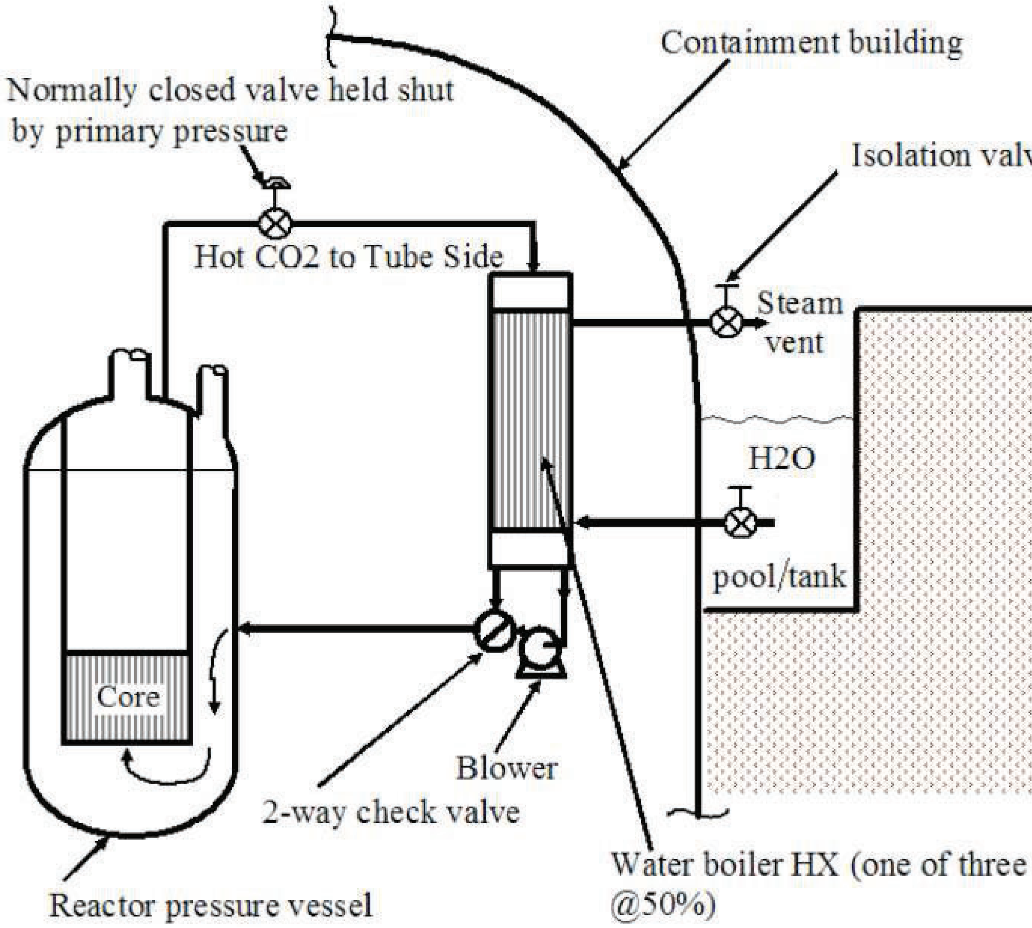

Fig. 1. Diagram of simplified GFR passive/active safety system [1].

Although successful in its implementation of mixed convection correlations for gases, LOCA-COLA left much to be desired in terms of usability and robustness. LOCACOLA helped to design several interesting passive and passive/active emergency core cooling systems for GFRs. It also was used to find some interesting phenomenon that can occur during the emergency operations. Nevertheless, LOCA-COLA was also quite limited due to its ability to only produce steady state results for a single loop system. For this reason, the search continued for a better way to design and analyze reactor systems.

Other Code Attempts

Attempts were made to implement new calculation methods for designing passive emergency systems. One path led to MATLAB's Simulink and Simscape Package. This package is promising and is making headway into the defense and IT industries due to the ability to have strict verification and validation of the produced user defined systems. One successful attempt at using the Simulink and Simscape and Simulink platforms to do nuclear engineering analysis is that of the University of Pittsburg's UPANTHER [4]. The U-PANTHER work is an excellent display of how this software can be used to develop simplified nuclear reactor simulators. However, MATLAB's Simulink and Simscape package in its current implementation is limited for use as a design platform. This is due to the need to custom code many new blocks to handle thermal-hydraulic calculations considered routine for any functioning nuclear engineering design code. Also many of the MATLAB functions are locked and do not allow for inspection if one desires to understand what is happening inside the black box.

The disappointment created by the MATLAB attempt pushed the research direction to find what is the basic functionality desired in thermal-hydraulic system design software. There is a desire to have an ability to use 1-D empirical formulations that are fundamentally understandable by an engineer and an ability to add user defined correlations. There is a desire to have nodal or network analysis methods to solve for multiple interconnected flow loops. There is a need to have finite difference or finite volume solution methods that allow for dense and sparse nodelization depending on the system and its components. Good engineering design software should have a simple but useful graphical user interface to diagram the system and allow drag-and-drop placement of components. There is a requirement to have integral fluid properties and the ability to create user-defined properties. There is a desire to have a preexisting catalog of common thermal-hydraulic components with the flexibility to create user-defined components. There is a need to perform both steady state and transient analysis. Finally there is the need to have solid mathematical basis in the solution engine. All of these were culminated in NASA's GFSSP.

NASA's GFSSP

In the 1990's, engineers at NASA at Marshall Space Flight Center (MSFC) found a very similar problem to that encountered here. The engineering team was working on new turbopump designs and found the specific purpose codes created for analyzing these systems often required extensive changes to the original coding to be utilized for analyzing new designs. This prompted the creation of GFSSP in 1994 with the goal to create a user friendly and modular program that could be used by an undergraduate engineer to rapidly develop thermal-hydraulic system models [5].

On further inspection, NASA's GFSSP is freely available to researchers and the educational version is freely available for students. The mathematics behind the code is robust and its methods have been verified and validated using some of NASA's extensive experimental data. It also contains nearly all the features a nuclear thermal-hydraulic designer would desire. For this reason, GFSSP was selected as a new method to perform nuclear thermal-hydraulic design and analysis. This project is starting with using GFSSP to perform analysis on GFR passive emergency systems to determine its utility and performance for nuclear engineering thermal-hydraulics.

GFSSP for Steady State Analysis of GFR Passive Emergency Systems

The GFR passive safety system shown in Fig. 1 was selected as an excellent trial case for using GFSSP. This system has already been analyzed in the previous work using RELAP/ATHENA, MARS-GCR, and LOCA-COLA[1]. The generalized passive (natural convection) emergency cooling loop schematic that was analyzed is shown in Figure 2. Table I and II provide the geometry of the nodal elements.

Table I. Geometry of the GFR Simplified Passive (Natural Convection) Emergency Cooling Loop [1]

Node Number	**# of parallel channels**	**D_e (m)**	**A(m^2)**
1	1	1.12	1.0
2	11557	0.011	9.50332E-05
3	11557	0.011	9.50332E-05
4	11557	0.011	9.50332E-05
5	1	2.0	3.14159
6	1	2.5	4.90874
7	1	2.0	3.14159
8	11557	0.011	9.50332E-05
9	1	2.5	4.90874
10	1	2.0	3.14159

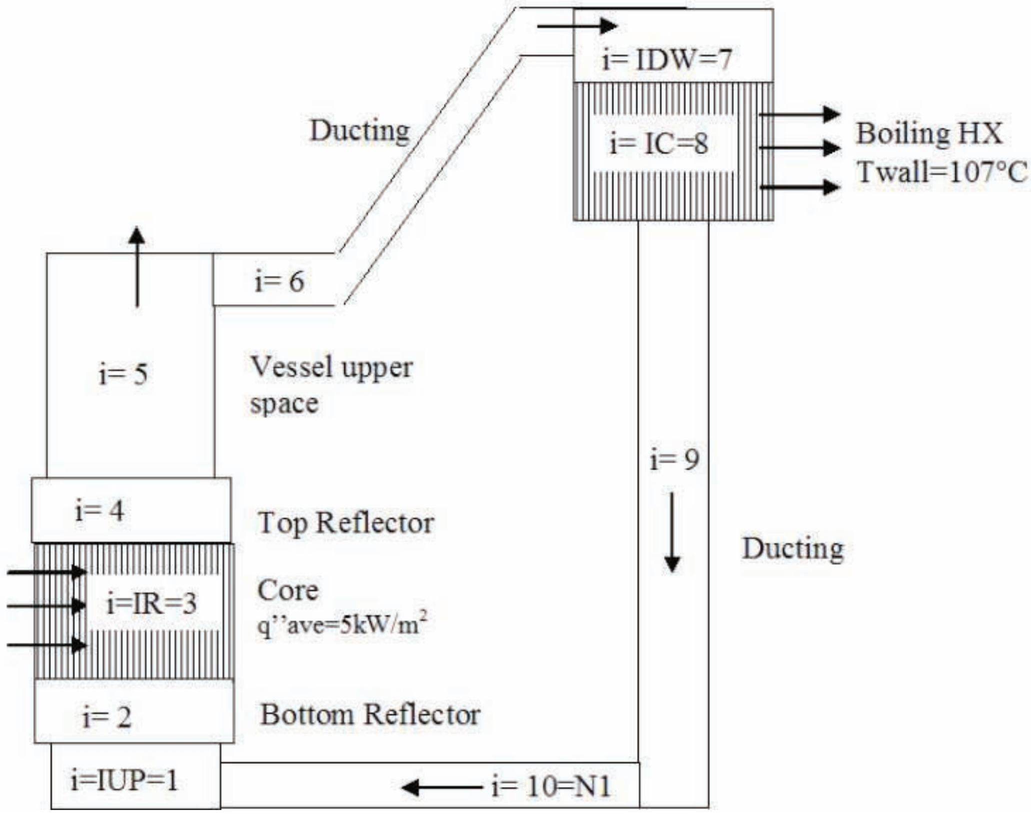

Fig. 2. Nodelization schematic of the simplified GFR passive (natural convection) emergency cooling loop [1].

Table II. Geometry of the GFR Simplified Passive (Natural Convection) Emergency Cooling Loop [1]

#	Height (m)	Length (m)	Roughness (mm)	Form loss
1	1.0	1.0	0.46	0.5
2	0.5	0.5	0.1	0.8
3	2.5	2.5	0.1	0.1
4	0.5	0.5	0.1	1.0
5	5.5	5.5	0.46	2.0
6	4.0	12.0	0.46	1.0
7	1.0	1.0	0.46	0.5
8	1.5	1.5	0.46	1.0
9	11.5	12.0	0.46	2.5
10	0.0	1.5	0.46	1.0

For the simulation, CO_2 was selected as the coolant. The total decay power generated in the core is assumed to be 5MWth, which is equivalent to a decay power level of about 0.8% of full power for the GFR. The system was assumed to have 5000kPa of pressure from the containment after blowdown. A model was created in GFSSP as shown in Figure 3. Table III shows a comparison of the key parameters that are calculated by each code.

Table III. Comparison of Code Performance [1].

Parameter	LOCACOLA	MARS-GCR	RELAP/ATHENA	GFSSP
Mass flow rate (kg/s)	8.74	8.83	8.90	8.89
Core inlet temp. (K)	440	441	430	440
Core outlet temp. (K)	951	949	935	943
Maximum cladding temp. (K)	1067	1048	1027	1158
Heat transfer coefficient - core (W/m^2- K)	28-41	38-40	39-43	20

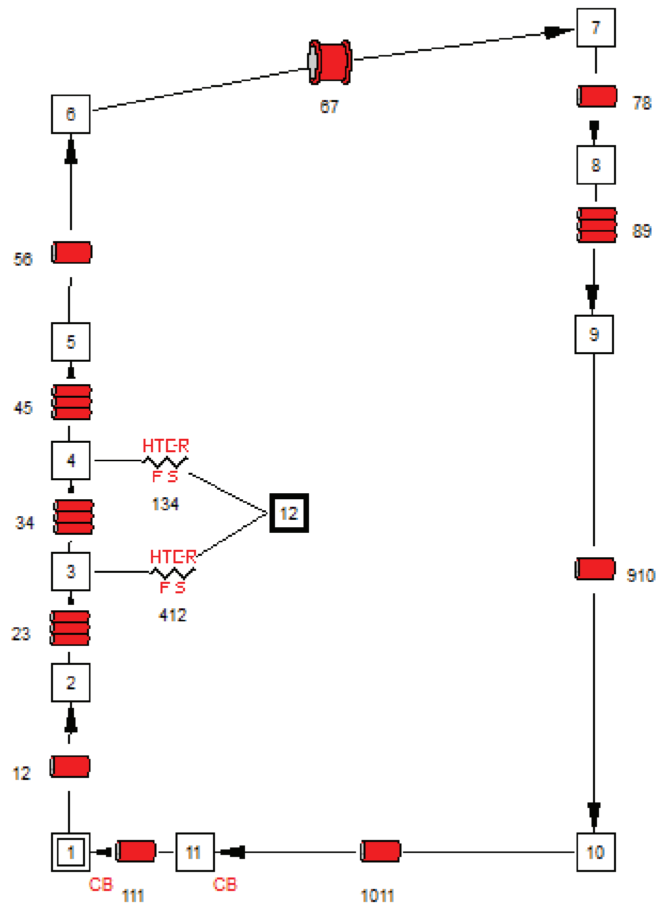

Fig. 3. Schematic of GFSSP Nodalization

In Table III it can be seen that the performance of GFSSP is quite close to the other selected codes. The largest deviation is with regards to calculated cladding temperature due to the difference in heat transfer coefficient, an error of 8%. The GFSSP code uses a Dittus-Boelter type correlation for heat transfer coefficient, whereas the other codes use Gnielinksi's correlation. Future work will use GFSSP's user defined functionality to add Gnielinksi's correlation for heat transfer coefficient in addition to the ability to modify heat transfer correlation when mixed convection occurs locally.

GFSSP for Transient Analysis of GFR During Emergency Cooling Operations

Previous work on the GFR safety system identified the very important phenomenon of hot channel flow starvation that can occur during natural convection at low pressures. The work of Hejzlar, et.al. [6] describes how the hot channel can potentially starve it's flow by the increase of viscosity of gas with temperature in this pressure and temperature during laminar natural convection. This phenomenon was discovered with the LOCA-COLA code after the ability to calculate hot channels was implemented. Initially the phenomenon was believed to be a numerical instability in the code due to the sensitivity of the output to minor changes in the input parameters. However, further investigation found other works [7,8,9] describing the laminar-instability problem. This issue was further defined as a key thermal fluid phenomenon in prismatic gas-cooled reactors [10,11] due to its potential to cause catastrophic failure.

Steady State Calculation of Parallel Channels

Although LOCA-COLA was sufficient to identify the laminar-instability problem, it is a steady-state solver and was unable to determine the time scale of the onset of this instability. This is an ideal candidate to test GFSSP for transient modeling performance. The first step of the simulation is to recreate the steady state conditions and check them against those of LOCA-

COLA. A simplified two-channel model was created as shown in Figure 4. The model contains two of the 11557 GFR channels. The model was split into 4 axial nodes to obtain some of the axial peaking profile from the previous LOCA-COLA analysis. The model utilizes GFSSP's functionality of a fixed inlet mass flow rate at the header which was set to 8.74 kg/s. The inlet conditions were also fixed at 5MPa and 167°C to match the LOCA-COLA simulation. In order to simulate the transient behavior, the fluid node volumes, the thermal node masses, and heat transfer areas were calculated and input. The solid nodes representing the matrix fuel material properties were approximated as those for graphite for the purposes thermal conductivity and heat capacity. The GFSSP capability to calculate buoyancy induced flow behavior is also included. Finally the heat rate was calculated and input for each node to simulate the heat generation with an axial peaking factor.

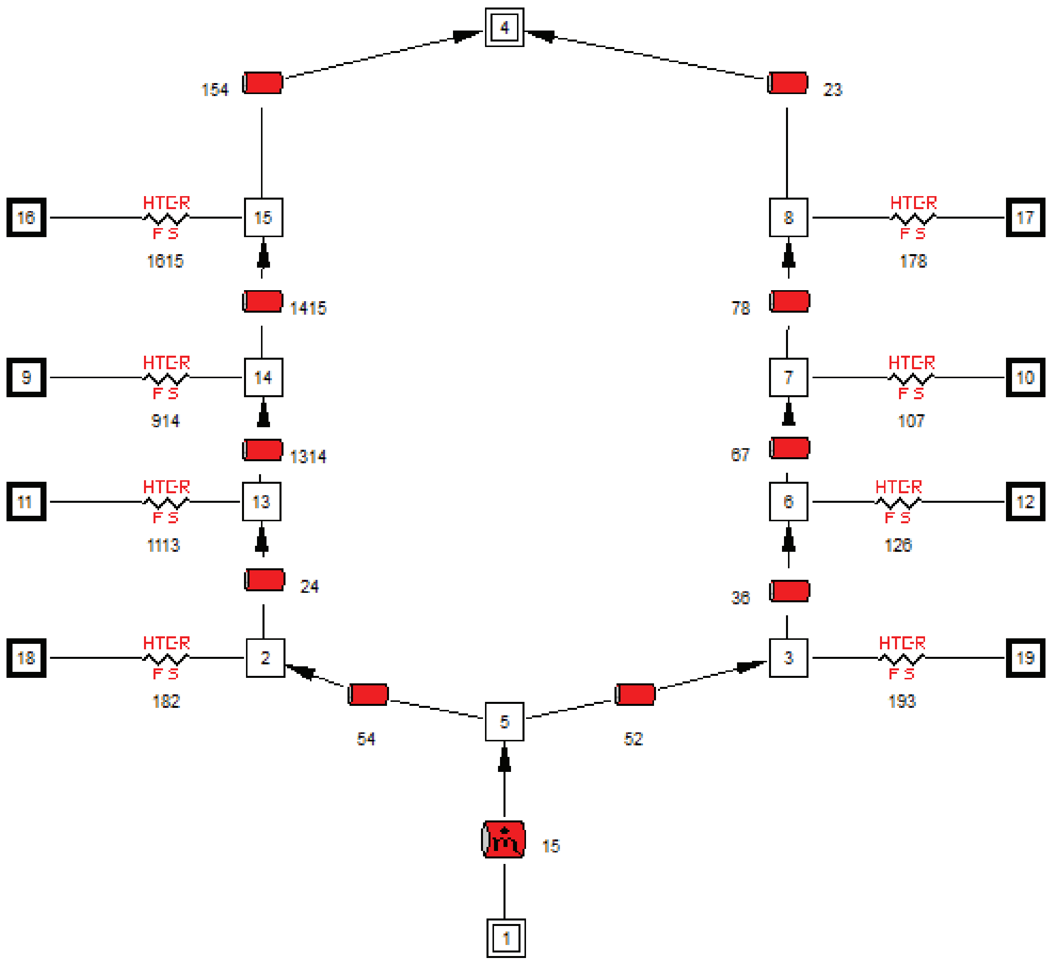

Fig. 4. Multichannel Flow Instability Model in GFSSP

Initially the model was run as a steady state simulation without the hot channel (radial peaking factor) as a check of performance with LOCA-COLA. The flows in the channels were found to be equivalent and the resulting bulk and wall temperatures were in good agreement with those results found by LOCA-COLA as shown in Figure 5.

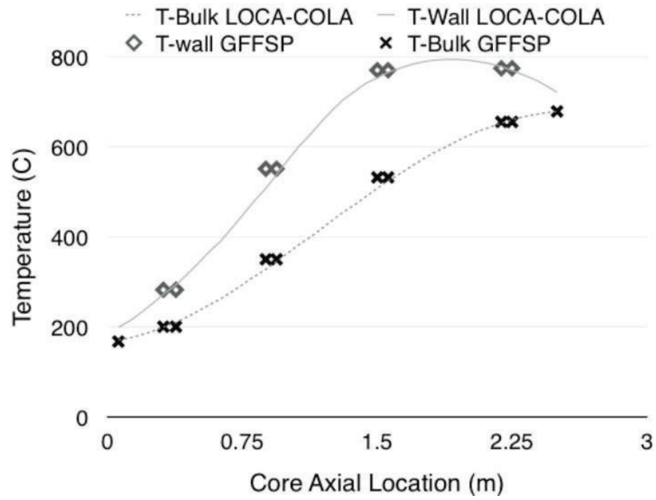

Fig. 5. Axial Temperature Profile Comparison of LOCA-COLA and GFSSP

One important note is that the solutions created using GFSSP utilize the Dittus-Boelter type correlation for the calculation of heat transfer coefficients. The steady state analysis values of heat transfer coefficient are in the transitional region between laminar and turbulent flow. LOCA-COLA has the additional capability to calculate buoyancy induced increases or decreases in heat transfer coefficient. GFSSP does not have this capability inherently available. However, in this simulation Reynolds numbers of around 4000 being close to laminar at the heat fluxes investigated do not have a significant increase due to the buoyancy effects typical in gas channel heated upward flow. This could explain the slightly higher wall temperatures calculated by GFSSP when compared to LOCA-COLA.

Transient Analysis of the Hot Channel Instability

The steady state solution values were then utilized as the starting point for the transient simulation. First the transient simulation was performed without the hot channel to insure the numerical stability of the initial steady state solution. The results remained stable and equivalent to the steady state solution. The hot channel factor was included as a 1.25 radial peaking factor and adjustments were made to the heat rates in the hot channel solid nodes. The simulation was run for 1000 seconds with 1-second time steps.

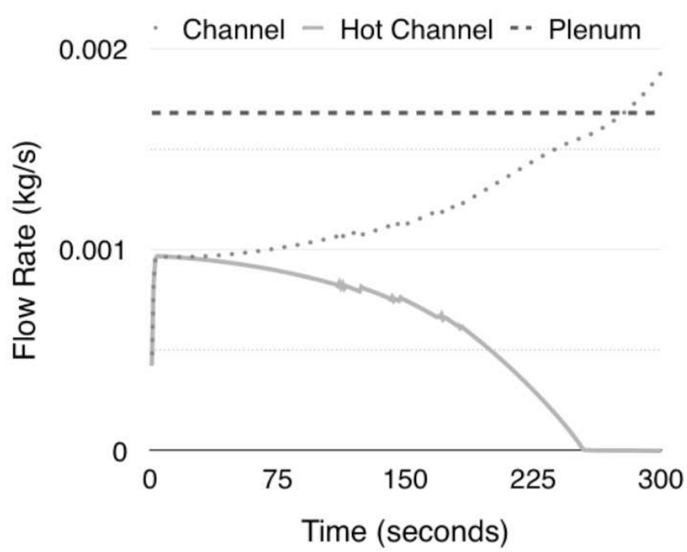

Fig. 6. Transient Normal Channel, Hot Channel, and Plenum Flow Rates

The hot channel flow rate begins to quickly and significantly diverge from the normal channel flow rate within the first 20 to 30 seconds as seen in Figure 6. At 272 seconds the hot channel flow rate is effectively zero and completely starved. Looking at the transient wall temperature at the axial peak nodes shown in Figure 7, it can be seen that the starved hot channel temperature increases and the normal channel decreases. This is due to the starvation of the hot channel and the increased flow into the normal channel.

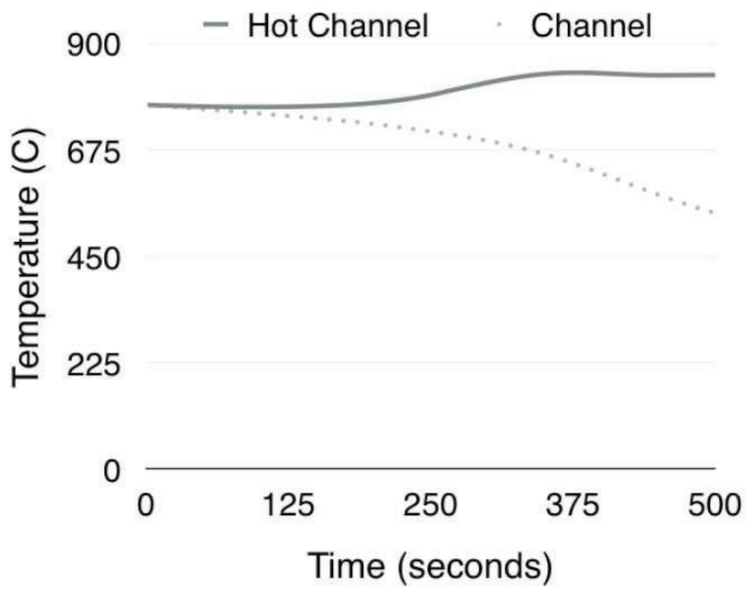

Fig. 7. Transient Wall Temperature of the Normal and Hot Channel at the Axial Peaking Location

RESULTS

In conclusion this paper demonstrated the abilities of GFSSP to model both steady state system and transient system phenomenon that occur in GFR safety system design and analysis. The simplicity and versatility of GFSSP allows for quick and intuitive modeling of systems important to the thermal-hydraulic designer and analyst. Results of the preliminary calculations performed showed the timescale of transient onset of hot channel flow starvation in a GFR during emergency cooling. This result proves the importance of the laminar-instability problem for gas cooled reactor designs. Further investigations are warranted to create a more detailed theory of this timescale. GFSSP is a significant tool for the exploration and design of gas cooled reactor systems. The studies above showed its capability to perform steady state and transient analysis on these systems. The verification and validation of NASA systems and designs gives GFSSP credibility for use outside on nuclear reactor systems. It is believed that through further validation, GFSSP can be an excellent tool for nuclear engineers.

Future work will explore GFSSP's ability to incorporate user defined subroutines in order to incorporate buoyancy induced heat transfer and flow correlations similar to those identified in the previous work of LOCA-COLA and also in more recent works of Lee et. al., Kim, et. al., and Wibisono, et. al. If these capabilities are incorporated, the ability to model transient behavior of a passive natural convection safety system performance from full power though SCRAM and LOCA blowdown will be explored.

NOMENCLATURE

A = flow area (m_2)
D_e = equivalent flow diameter (m)
GFR = constant

ENDNOTES

The authors would like to acknowledge the hard work of the NASA Marshall Space Flight Center. The freely available GFSSP software is an excellent usage of federal tax money for a truly valuable piece of intellectual property.

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