TRACG ANALYSIS OF BOILING WATER REACTOR CONTROL ROD DROP ACCIDENT TO OPTIMIZE ANALYSIS METHODOLOGY

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ABSTRACT

During transient events, it is important to confirm that the plant maintains compliance to regulatory standards that provide safety criteria to prevent the release of fission products. The design basis Reactivity Insertion Accident (RIA) considered for Boiling Water Reactors (BWRs) is the Control Rod Drop Accident (CRDA). During this postulated scenario, a control blade becomes decoupled from the control rod drive mechanism as the drive is withdrawn and remains lodged in place. At a later time, the blade falls out to the position of the drive causing a reactivity insertion into the core. This leads to a power excursion and terminates due to the Doppler Effect or upon completion of the SCRAM. Many BWR plants adopt a Banked Position Withdrawal Sequence (BPWS) methodology that was developed to minimize the control rod worth and mitigate the consequences of the CRDA event. The BPWS methodology places restrictions on control rod movement to ensure that no CRDA could exceed the applicable event limits by reducing the incremental control rod reactivity worth to acceptable values. In this paper, the best estimate model code TRACG, the GE Hitachi (GEH) Nuclear Energy proprietary version of the Transient Reactor Analysis Code, TRAC, is used to model CRDA event to ensure that the plant is compliant with the new more restrictive regulatory standards. TRACG is a three-dimensional, two-fluid representation of two-phase flow for the transient reactor thermal-hydraulics, which is coupled to the Global Nuclear Fuel (GNF) core simulator PANAC to provide the three-dimensional transient neutron diffusion for the calculation of the transient power response as a function of the transient thermal-hydraulic conditions. The analysis presented in this paper is aimed to model the CRDA event using the best-estimate TRACG system code. These results will be compared to the three-dimensional transient neutron diffusion adiabatic representation of CRDA available in PANAC. Results of the analyses include the realistic best estimate calculation of the enthalpy rise resulting from the event in a generic BWR/6 and demonstration of the large margin available when compared to the US Nuclear Regulatory Commission enthalpy criteria limits published in Standard Review Plan 4.2 Fuel System Design.

KEYWORDS
Control Rod Drop Accident, enthalpy rise, positive reactivity, 3D neutron kinetics, TRACG

1. INTRODUCTION

The design basis Reactivity Insertion Accident (RIA) event considered for Boiling Water Reactor (BWR) is the Control Rod Drop Accident (CRDA). During this postulated scenario a control blade becomes decoupled from its drive mechanism, the drive is withdrawn but the blade remains lodged in place. At a

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later time, the blade falls out to the position of the drive causing a rapid reactivity insertion into the core. This leads to a power excursion. The event terminates due to the Doppler Effect or upon completion of the SCRAM. This event presents a thermo-mechanical challenge to the fuel; as a result conservative regulatory standards have been put in place. It is important to understand the plant's response to the accident to be sure the plant is compliant with the regulatory standards.

In order to mitigate these effects, the Licensing Topical Report, NEDO-21231, describes the Banked Position Withdrawal Sequence (BPWS) methodology [1]. It deals exclusively with the description, performance, and design requirements of BPWS which virtually eliminates CRDA as an accident of any concern, by maintaining incremental rod worths to relatively low values. Currently, BPWS is used for all US BWR plants. For applicable plants, BPWS successfully mitigates the impact of a CRDA to meet the specific fuel enthalpy criteria.

The US Nuclear Regulatory Commission has enthalpy criteria and guidance established for reactivity-initiated events under NUREG-0800 in Standard Review Plan 4.2 Fuel System Design. Previously, the 280 cal/g design limit and the 170 cal/g fuel cladding failure threshold were used for CRDA. Appendix B Revision 3 of the Standard Review Plan provides an updated enthalpy criteria based on the correlation between fission gas release and the maximum fuel enthalpy increase. For zero hydrogen content the fuel enthalpy rise limit is 150 cal/g, from 75 ppm to 150 ppm the limit decreases to 60 cal/g and a final decrease at a less steep slope until 300 ppm ending at a fuel enthalpy rise of 50 cal/g [2, 3].

In this paper, the best estimate model code TRACG [4], the GEH Nuclear Energy proprietary version of the Transient Reactor Analysis Code TRAC, is used to model CRDA event to ensure that the plant's response meets the new regulatory criteria. The TRACG analysis will aid in demonstrating the margin available in comparison to the CRDA regulatory criteria for the plant. The limiting control blade is identified using PANAC [5] with a single rod withdrawal approach.

PANAC is the GNF core simulator, which is a three-dimensional nuclear BWR core simulator code. In PANAC, the neutronic parameters are obtained from the two-dimensional lattice physics code TGBLA [6] and are parametrically fitted as a function of moderator density, exposure, control state and moderator density history for a given fuel type. TGBLA solves the rod-by-rod thermal spectra by the leakage-dependent integral transport method. The transient TRACG results are compared against PANAC. Both PANAC and TRACG use the TGBLA calculated cross sections as well as the same three-dimensional neutron kinetics model. In PANAC, heat transfer changes are ignored and fuel pin enthalpies are computed by integrating the pin-powers, which result from pin-power reconstruction taking into account the strong flux gradients produced by the moving control rod. In TRACG, a transient solution for the temperature distribution in multiple fuel rods is performed to account for the local peaking and then the calculated temperatures are converted to fuel enthalpies. The same pin-power pin-peaking factors (PPFs) computed by PANAC are used in TRACG to compute pin-enthalpies from nodal enthalpies.

Both codes' CRDA calculation methodologies have been qualified against the same reactivity transient tests which are the Special Power Excursion Reactor Tests (SPERT). From 1964-1970, a series of reactor tests were performed at Idaho National Laboratory (INL). The tests were initiated from cold startup condition at a temperature of 293 K with zero flow. The tests simulated reactivity insertion from $0.68 to $1.21. Comparisons between TRACG rod drop calculations and SPERT experimental tests concluded that the code adequately models the transient event [7].

2. PLANT AND EVENT DESCRIPTION

This study analyzes a CRDA scenario for a generic GE BWR/6. The GE BWR/6 is the most modern GE reactor product line with external recirculation pumps to force flow into the core. This is achieved by two external loop in which the recirculation pumps take water from the downcomer and pump it back in the vessel by providing the driving force for the jet pumps, located in the downcomer annulus all around the vessel, to drive the flow in the lower plenum and eventually in the core. BWR/6 are typically characterized by very large core, with several hundred fuel assemblies. BWR/6 are also characterized by fast control rods SCRAM, with respect to all previous GE BWR product lines. Figure 1 gives a schematic of a typical BWR/6 vessel and internals, and how it is modeled in TRACG.

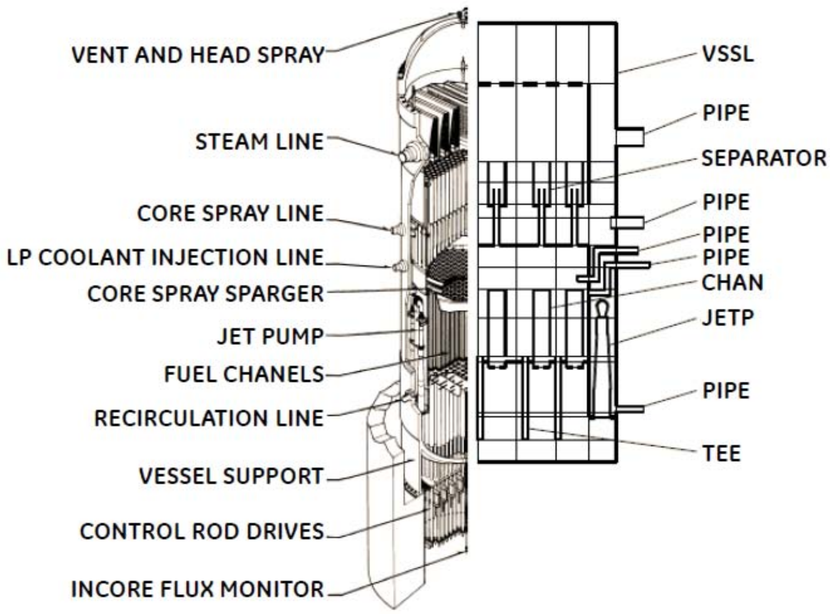

Figure 1. Schematic of a BWR/6 and corresponding TRACG model [8].

In order to model the event, some conservative assumptions were made. These assumptions include the blade falling at the terminal velocity and the blade worth adding reactivity to a core that is already critical. For the CRDA scenario, the control blade drop velocity is 0.7 m/sec. The maximum velocity was applied instantaneously. As the power increases during the event, high neutron flux trip occurs at 120% of rated power. The reactor SCRAM delay is 0.09 seconds. The insertion time table for the SCRAM is 0% at 0.2 sec, 60% at 1.71 seconds, and 100% at 3.7 seconds after SCRAM initiation. For a typical BWR/6 the SCRAM time is faster; however for this application the slower SCRAM speed, which is a conservative assumption, is adopted. The transient time for the simulation is 6.0 seconds. Table 1 provides PANAC initial conditions for the BWR/6 being evaluated.

Table I. Initial Conditions

Power (MWt)	5.0
Core Flow (% Rated)	30.0
Pressure (Pa)	7.17×10^6 (1000 psia)
Bypass Flow (% Rated)	5.0

3. PANAC ANALYSIS

The limiting control blade is identified using PANAC and a single rod withdrawal approach. PANAC is used for detailed three-dimensional design and operational calculations of BWR neutron flux and power distributions. Thermal performance as a function of control blade position, refueling pattern, coolant flow, reactor pressure, and other operational and design variables are also calculated.

3.1. PANAC Control Blade Movement

A control blade static worth calculation is performed using the three-dimensional core simulator. The calculation matrix considered a number of temperatures within the range of approximately 300-500 K at the beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC) exposures. The control blades are organized into 10 groups.

A predetermined control rod withdrawal sequence was established for the BWR/6 to control the power distribution in the core and minimize the control blade worths. There are some restrictions on control blade maneuvering implemented in this analysis because of BPWS application modeling. Groups 1 through 4 must be fully withdrawn before groups 5 through 10 are moved. For each withdrawal step the rods are withdrawn to their most withdrawn notch for that given step to produce the largest possible step reactivity. Each of the groups 1 through 4 must be fully withdrawn before another group is moved. These first four group withdrawals would not be permissible per the BPWS licensing bases because they do not follow banked withdrawal. Groups 9 and 10 are incrementally withdrawn between the full withdrawal of groups 5 and 6. In the PANAC analysis, every control blade that has been previously moved is assessed to determine its in-sequence static blade worth for every step in the withdrawal sequence.

3.2. PANAC Results

For each of the temperature and exposure combinations the GEH blade worth screening process is used to identify the limiting cases based on the maximum control blade worth. Control blade worth is defined as the reactivity difference between the current position of a blade in the withdrawal sequence and the fully inserted position.

Table II provides the maximum blade worth for the combinations of exposures of the two most limiting temperatures cases with the corresponding blade location, eigenvalues, and hydrogen content. The maximum blade worth was determined by choosing from the cases where the eigenvalues are within a specific criterion based on the core's critical state. Any rod worths calculated in the subcritical or beyond prompt critical are not considered. For an actual plant analysis, the core's critical state may be determined as a function of the exposure-dependent design basis cold eigenvalues. However for this analysis, the design basis eigenvalue is assumed to be 1.0, which is exposure independent.

Two cases were identified for further analysis based on the stated criterion. Table II shows that the hydrogen content for the two limiting temperature cases are within the first portion of the regulatory guidelines corresponding to 150 cal/g enthalpy criteria. The cases chosen are those that demonstrated the highest worth and are evaluated by additional screening criteria that have been developed by GEH. The blade of interest is located at PANAC coordinates (8, 26) and is part of control blade group 3. Therefore, groups 1 and 2 are fully withdrawn as well as the control blades ahead in sequence in group 3. Note the case names are combinations of the cycle exposures (BOC, MOC, and EOC) and the two limiting fluid temperatures (T1, T2) that were identified.

It's important to observe that the analyzed withdrawal sequence does not comply with BPWS rules [1] and constitutes a very limiting case. This is mostly because the first four blade groups do not follow banked withdrawal procedure. In fact, the resulting worths for these cases are approximately 1.5%, which are considered very large.

Table II. PANAC Maximum Rod Worth Results

Case	Worth (%Δk)	Blade (i,j) Location	Eigenvalue	H_2 Concentration (ppm)
T1_BOC	1.29	(12,12)	1.005	53.50
T1_MOC	1.23	(08,26)	1.009	47.66
T1_EOC	**1.47**	**(08,26)**	**1.006**	**45.62**
T2_BOC	1.27	(12,12)	0.997	53.50
T2_MOC	1.09	(08,26)	1.009	47.66
T2_EOC	**1.49**	**(08,26)**	**1.007**	**45.62**

3.3. PANAC CRDA Transient Results

Using PANAC, the fuel enthalpy increases are determined for the limiting cases based on the withdrawal sequence position, exposure, and fluid temperature of the points of interest. The reactivity insertion as a function of time starting from steady-state conditions is computed for the CRDA event. PANAC is a conservative approach to modeling a prompt critical reactivity insertion event due to the adiabatic method and the lack of negative void reactivity feedback.

In this PANAC analysis, the three-dimensional transient neutron diffusion equations are solved using one neutron energy group and up to six delayed neutron precursor groups. Moderator temperature, exposure, and xenon concentration distributions are assumed to be constant during the transient. Core pressure and inlet conditions are not varied during the calculation. The Doppler reactivity feedback is accounted for by using an adiabatic fuel temperature model. This is a valid assumption because the rapid power response to a CRDA occurs so quickly that it does not allow time for significant heat transfer in the fuel pellet or for large changes in the moderator density. By limiting the changes in core reactivity to the control blade drop, Doppler feedback and SCRAM, PANAC presents a representative analysis of the prompt enthalpy rise in the fuel.

Table III gives the peak power and enthalpy rise as calculated by PANAC. The table shows that the 'Worth 1' case does not initiate an automatic SCRAM because the power does not exceed 120%; however, the power in the 'Worth 2' case does reach the SCRAM set point. The change in peak fuel enthalpy is important to determining the severity of the CRDA transient event, such that compliance with regulatory criteria can be maintained. The peak enthalpy values stay below the 150 cal/g limit that is correlated to the hydrogen content imposed for a CRDA event.

Consistent with the guidance in Standard Review Plan 4.2, the peak pin enthalpy rise is reported at a time corresponding to one pulse width after the peak power. Specifically, this is one full pulse width at half the maximum (FWHM) pulse height of the prompt power pulse. This occurs at a time of 1.20 and 1.16 seconds for the 'Worth 1' and 'Worth 2' cases respectively, as shown in Table III.

Table III. PANAC Results

Case	Time of Peak Power (sec)	Power (% Rated)	Time of Peak Enthalpies (sec)	Peak Pin Enthalpy Rise (cal/g)	Nodal Enthalpy Rise (cal/g)
EOC_T1_Worth1	1.1	96%	1.20	46.4	41.4
EOC_T2_Worth2	1.1	166%	1.16	60.2	51.4

Figures 2 through 4 show the reactivity, enthalpy and power behavior during the CRDA transient event. PANAC implements adiabatic fuel model. This ensures that the energy deposited in the fuel is maximized by not permitting the fuel energy to transfer. This results in an increasing fuel enthalpy since there is no heat transfer to the coolant. In addition, it can be seen that the 'Worth 2' case has a greater prompt power peak and enthalpy rise than the 'Worth 1' case. As the SCRAM took effect on the reactivity, the slope in the rise of enthalpy decreases in 'Worth 2' in comparison to 'Worth 1'.

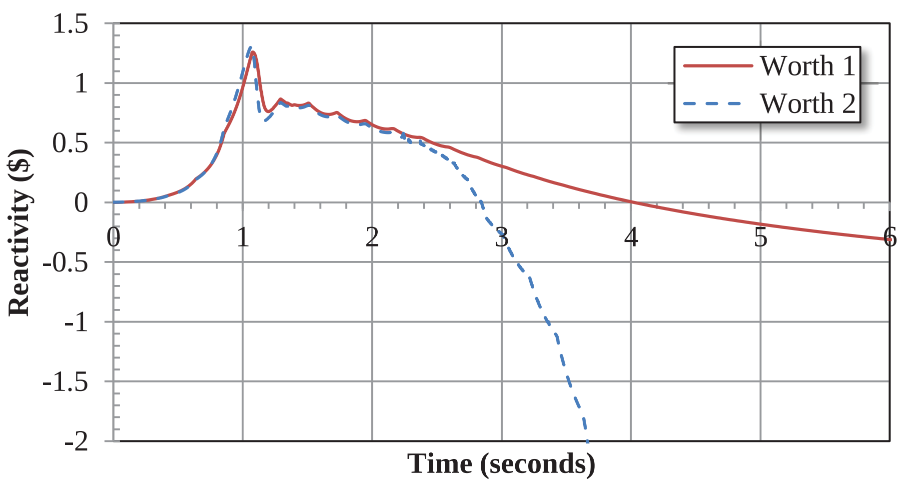

Figure 2. PANAC Reactivity vs. Time Results.

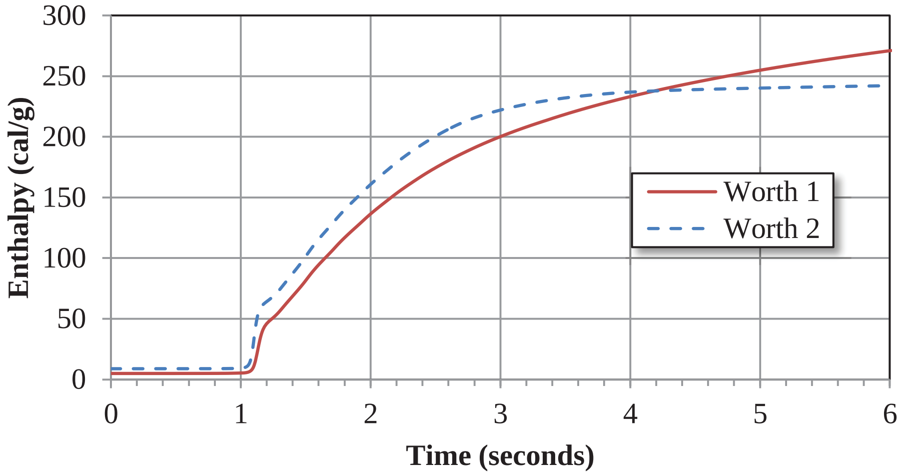

Figure 3. PANAC Enthalpy vs. Time Results.

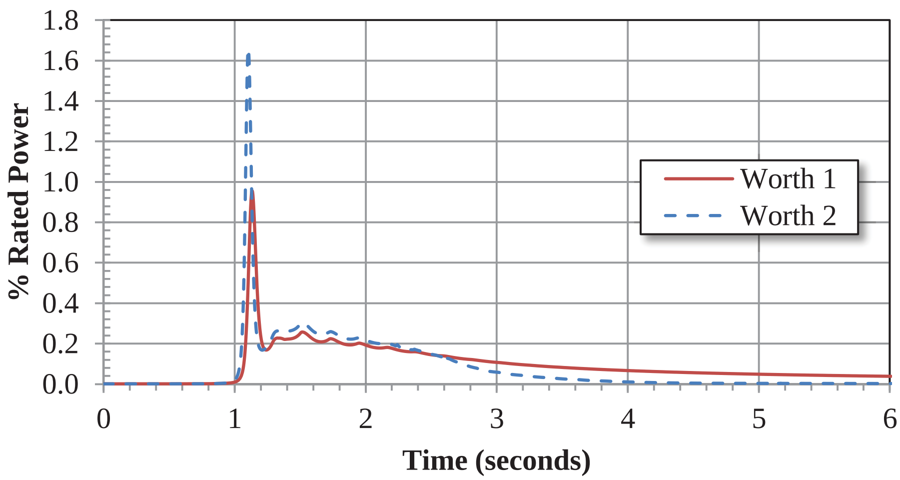

Figure 4. PANAC % Rated Power vs. Time Results.

4. TRACG ANALYSIS

The fuel enthalpy increase resulting from the CRDA transient is also determined using TRACG. The main difference in the transient analysis is TRACG calculation accounts for the heat transfer during the CRDA event. This includes a gap conductance model which uses PRIME [9] generated values. The gap conductance is retrieved as a function of Linear Heat Generation Rate (LHGR) history, instantaneous LHGR as well as exposure. In addition, the gap conductance model treats the changing gap conductance as a function of gap size, which is affected by thermal expansion calculated from the transient power response.

4.1. TRACG Model

The TRACG model was comprised of multiple components to describe the BWR/6 plant. An air level is specified above the separators, such that near the top of the vessel the pressure boundary is specified. Initial fluid and structural temperatures in the vessel were set to each of the cases' temperatures.

The thermal-hydraulic system code modeling of the BWR/6 and channel components was coupled with the three-dimensional neutron kinetics input from PANAC. TRACG first runs a steady-state case using the PANAC calculated power distribution in order to determine the corresponding thermal-hydraulic conditions. TRACG then restarts from this case to run the transient using a fully integrated thermal-hydraulic and three-dimensional neutron kinetics model.

The BWR/6 core is filled with GNF2 fuel. The channel geometry is generated based on fuel dimensions, spacer grids, water rod locations and dimensions, and loss coefficient specifications. The channel groupings are determined based on the location of the accident blade. Each of the channels in a 6x6 area immediately surrounding the accident blade are modeled individually while all the remaining channels are grouped together resulting in 37 channel groups as shown in Fig. 5.

37	37	37	37	37	37	37	37
37	17	23	25	27	29	31	37
37	18	5	9	11	13	32	37
37	19	6	1	2	14	33	37
37	20	7	4	3	15	34	37
37	21	8	10	12	16	35	37
	22	24	26	28	30	36	37
				37	37	37	37

Figure 5. TRACG Channel Groupings.

4.2. TRACG Analysis and Results

The enthalpy rise as a result of the CRDA event is also determined with TRACG at the time determined using the same FWHM methodology described previously. Table IV summarizes the impact on the core from the transient event. Note the TRACG enthalpy results presented in Table IV include the PANAC based PPF. Depending on the temporal point in the transient, the pin-peaking ranged from 1.00 to 1.24.

The TRACG peak power is greater than that of PANAC. This is because the adiabatic conditions in PANAC provide more negative Doppler reactivity feedback. Due to the incorporation of heat transfer in the TRACG model as well as negative void reactivity feedback there is an actual peak in the TRACG enthalpy distribution whereas in PANAC the adiabatic assumption ensures that the enthalpy can never decrease. These results show that TRACG and PANAC essentially predict, as expected, similar prompt enthalpy rises.

Table IV. TRACG Results

Case	Time of Peak Power (sec)	Power (% Rated)	Time of Peak Enthalpies (sec)	Peak Pin Enthalpy Rise (cal/g)	Nodal Enthalpy Rise (cal/g)
EOC_T1_Worth1	1.1	135%	1.20	52.8	47.3
EOC_T2_Worth2	1.1	219%	1.16	65.2	56.2

5. MODEL COMPARISON

In this section PANAC and TRACG CRDA event modelling is compared. Figure 6 displays how PANAC and TRACG performed against the regulatory criteria. From Fig. 6 it can be seen that there is a significant margin in the enthalpy rise relative to the acceptance criteria. The TRACG values in comparison to the PANAC enthalpy rise values are similar in magnitude.

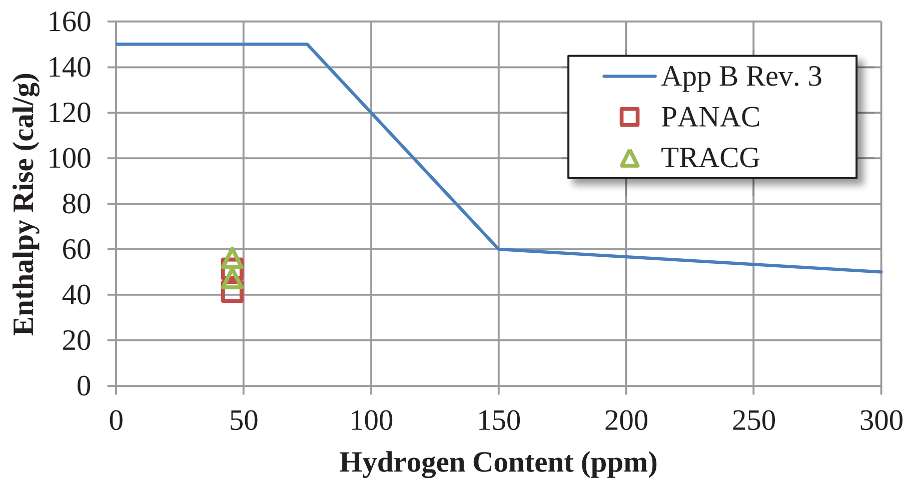

Figure 6. Enthalpy Rise vs. Hydrogen Content.

Figures 7 through 9 present the results for the 'Worth 1' case and Figures 10 through 12 show the results for the 'Worth 2' case. The results show good agreement between the PANAC and TRACG peak reactivity insertion. PANAC under-predicts the power spike relative to TRACG. However, it is clear the reactivity is very consistent in the two codes, as is expected because they both implement the same three-dimensional neutron kinetics model. In addition, the contrasting fuel enthalpy calculation method is evident. In TRACG the enthalpies steadily decrease after reaching a maximum whereas PANAC continues to increase indefinitely. Therefore, at the end of the simulation PANAC's overall enthalpy results are more conservative than TRACG's overall enthalpy results.

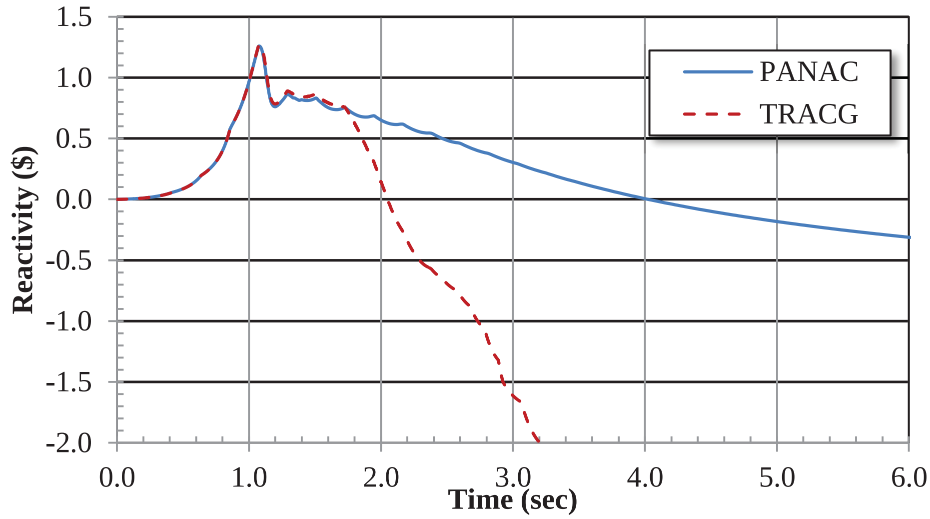

Figure 7. Worth 1 Reactivity Code Comparison.

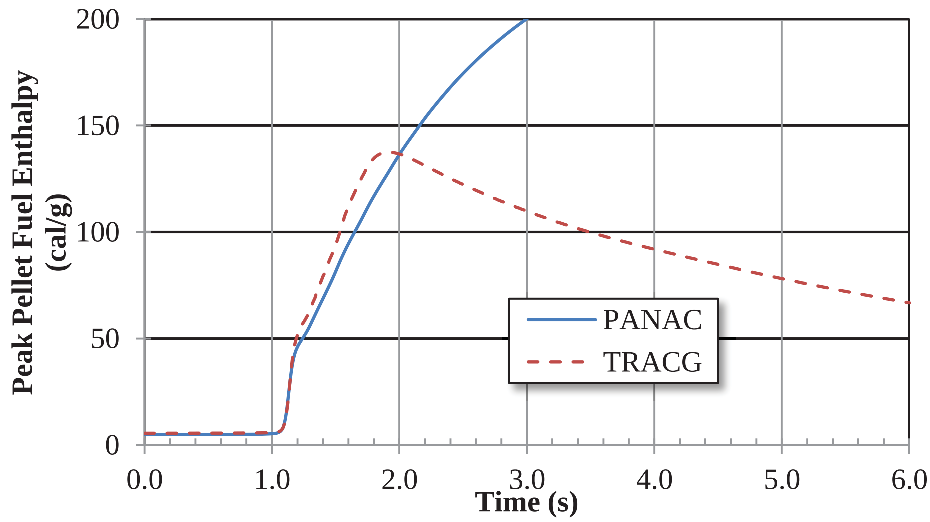

Figure 8. Worth 1 Enthalpy Code Comparison.

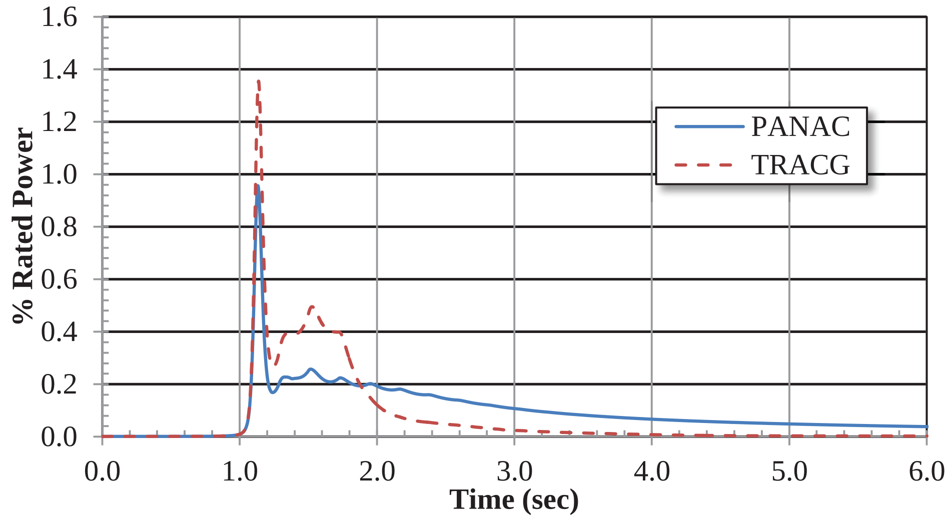

Figure 9. Worth 1 % Rated Power Code Comparison.

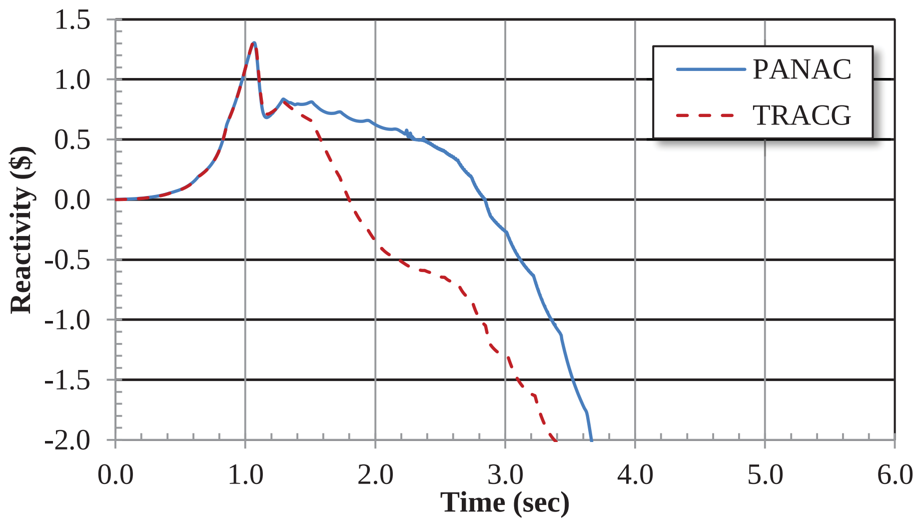

Figure 10. Worth 2 Reactivity Code Comparison.

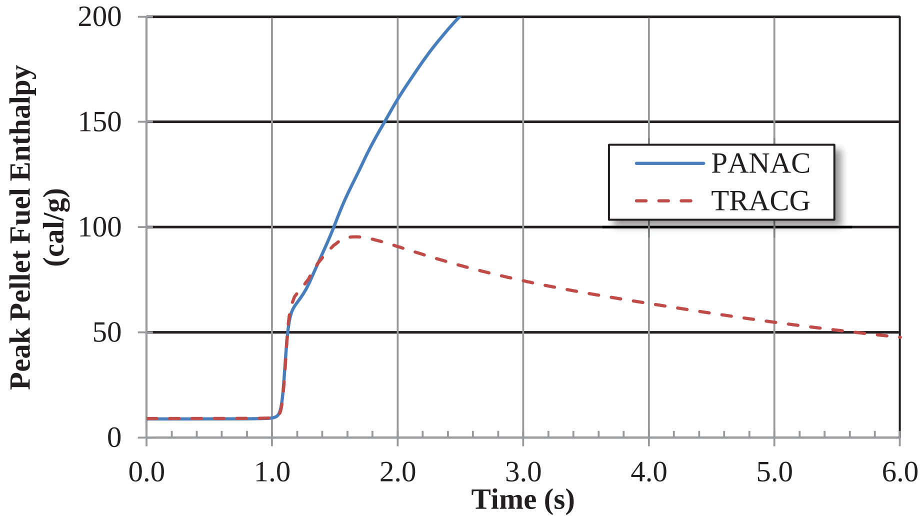

Figure 11. Worth 2 Enthalpy Code Comparison.

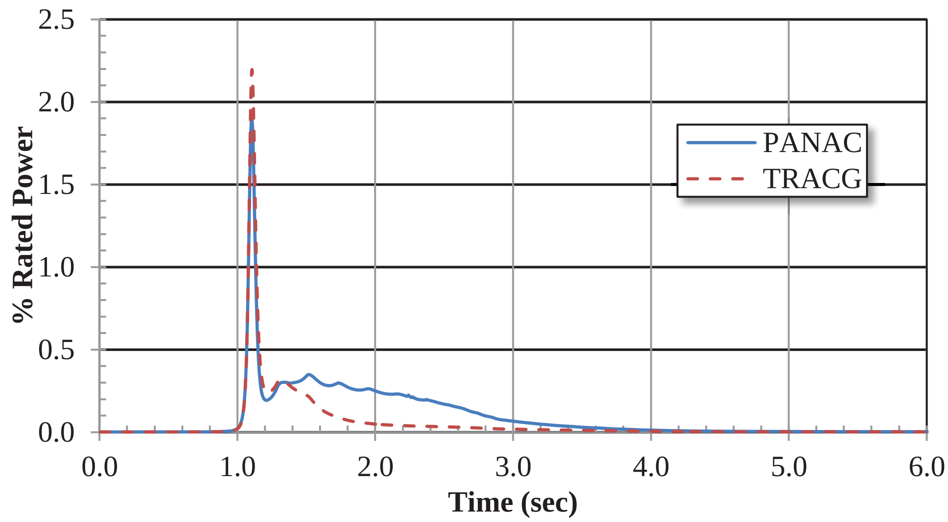

Figure 12. Worth 2 % Rated Power Code Comparison.

Figure 13 presents the reactivity components as predicted by TRACG during the CRDA event for the 'Worth 1' case. During the transient the fuel reactivity is negative due to Doppler reactivity feedback. Due to the control blade drop, the control reactivity increases at the start of the event. The total reactivity peak occurs at around 1.1 seconds and steadily decreases as the negative reactivity effects take control of the event, which includes the SCRAM.

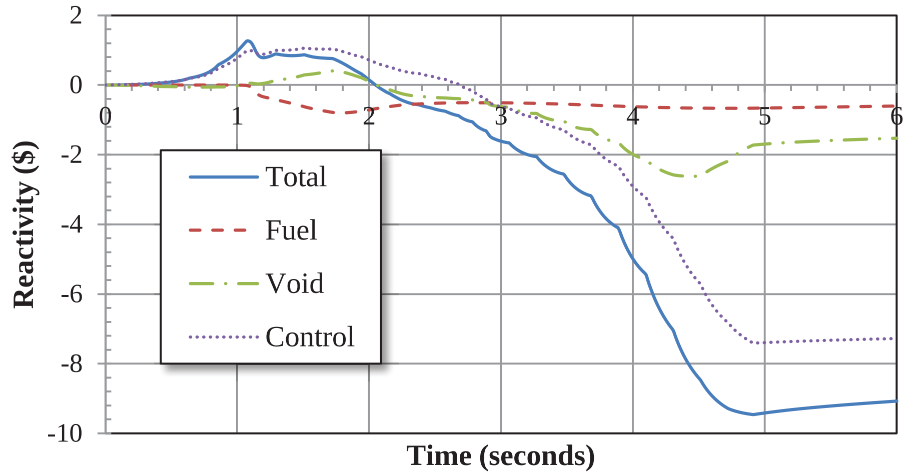

Figure 13. TRACG Worth 1 Reactivity.

6. CONCLUSIONS

The power, reactivity, and enthalpy results are presented for a postulated CRDA event in a representative GE BWR/6 plant with a full core of GNF2 fuel. The transient event was modeled with both the best estimate model code TRACG, the GE Hitachi Nuclear Energy proprietary version of TRAC, and with the GNF core simulator PANAC. The codes model the CRDA event to ensure that the plant's response is compliant with the Standard Review Plan 4.2 enthalpy criteria. The results demonstrate two main outcomes: 1) PANAC and TRACG predict essentially the same prompt enthalpy rise, and 2), even for very limiting CRDA with high rod worths a large margin to the enthalpy criteria. Future work will involve the analysis of additional plant types including BWR/2 and BWR/4.

NOMENCLATURE

BOC	Beginning of Cycle
BPWS	Banked Position Withdrawal Sequence
BWR	Boiling Water Reactor
CRDA	Control Rod Drop Accident
EOC	End of Cycle
FWHM	Full Width Half Max
GE	General Electric
GEH	GE Hitachi
GNF	Global Nuclear Fuels
INL	Idaho National Laboratory
LHGR	Linear Heat Generation Rate
LTR	License Topical Report
MOC	Middle of Cycle
PPF	Pin-Peaking Factor
ppm	Parts per Million
RIA	Reactivity Insertion Accident
TRACG	Transient Reactor Analysis Code GE
T-H	Thermal-Hydraulic

ACKNOWLEDGMENTS

The author acknowledges the support from the General Electric’s Edison Engineering and Development Program and Advanced Courses in Engineering. The author would also like to thank Charles Heck (GE Hitachi Nuclear Energy) and co-authors for their help in writing this paper.

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