Testing of an Integrated Reactor Core Simulator and Power Conversion System With Simulated Reactivity Feedback

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Abstract

A Direct Drive Gas-Cooled (DDG) reactor core simulator has been coupled to a Brayton Power Conversion Unit (BPCU) for integrated system testing at NASA Glenn Research Center (GRC) in Cleveland, Ohio. This is a closed-cycle system that incorporates an electrically heated reactor core module, turboalternator, recuperator, and gas cooler. Nuclear fuel elements in the gas-cooled reactor design are replaced with electric resistance heaters to simulate the heat from nuclear fuel in the corresponding fast spectrum nuclear reactor. The thermodynamic transient behavior of the integrated system was the focus of this test series. In order to better mimic the integrated response of the nuclear-fueled system, a simulated reactivity feedback control loop was implemented. Core power was controlled by a point kinetics model in which the reactivity feedback was based on core temperature measurements; the neutron generation time and the temperature feedback coefficient are provided as model inputs. These dynamic system response tests demonstrate the overall capability of a nonnuclear test facility in assessing system integration issues and characterizing integrated system response times and response characteristics.

Nomenclature

C_i	delayed-neutron concentration, i^{th} precursor group
P	reactor fission power (W)
t	time (sec)
T	temperature (K)
α_T	temperature coefficient of reactivity (K^{-1})
β	total delayed-neutron fraction (sum of all β_i)
β_i	delayed-neutron fraction, i^{th} precursor group
λ_i	decay constant for the ith precursor group (sec^{-1})
Λ	prompt neutron generation time (sec)
ρ	reactivity (unitless)

I. Introduction

Non-nuclear testing can be used to evaluate the operation of an integrated nuclear system within a reasonable cost and schedule to provide valuable input to the overall system design. Various operational regimes can be studied in non-nuclear testing to validate thermal and thermal hydraulic codes, to assess thermal hydraulic behavior, to characterize stress/strain in the system during operation, and to verify system integration processes (Refs. 1 to 7). However, because the electrically heated core lacks neutrons, the dynamic neutronic response of the system cannot be fully simulated without the incorporation of system models to the simulator control system. Non-nuclear testing with simulated neutronic feedback is expected to provide reasonable approximation of reactor transient behavior because reactivity feedback is very simple in a compact fast reactor (simple, negative, and relatively monotonic temperature feedback, caused mostly by thermal expansion) and can be simulated using measured data from non-nuclear system tests.

In 2004 and 2005 dynamic testing was performed on the electrically heated SAFE-100 and 100a test articles at the NASA Marshall Space Flight Center (MSFC) (Refs. 8 and 9). The Safe Affordable Fission Engine (SAFE) heat pipe cooled reactor concept is a compact, fast-spectrum reactor designed to use highly enriched UN fuel. In this design, passive heat extraction from the reactor core would be accomplished by in-core liquid metal heat pipes. The SAFE-100a, a partial array of the SAFE-100 design, was constructed from 19 modules (each containing three fuel tubes surrounding a sodium-filled stainless steel heat pipe) and was coupled to a prototypic gas-flow heat exchanger (HX). Heat removal was provided by a gas-to-water heat exchanger on the hot side of the intermediate HX gas flow loop; no power conversion system (PCS) was included in the test configuration. Integrated reactor simulator and HX response to system transients were studied through application of a temperature-based feedback model.

The average core temperature, determined from several thermocouples placed throughout the core block, provided state estimation as an input to the reactor simulator control system. The time scale associated with system transients is primarily dictated by the thermal inertia of the core; for the SAFE-100a, the peaks and/or valleys of the power and temperature occurred within 3 to 5 min after transient initiation, followed by modest oscillations until the system stabilized at a new steady state condition approximately 20 to 30 min after the transient was initiated (Ref. 9).

Similar dynamic test techniques were applied to the direct drive gas-cooled reactor system (DDG) during initial reactor core simulator testing at NASA MSFC in 2006 (Refs. 10 and 11). Although both the SAFE-100a and DDG system designs utilize a fast spectrum reactor, the method of cooling the reactors differs significantly and, while the SAFE reactor is a modular design, the DDG utilizes a monolithic block. These differences lead to a variable system response that can be demonstrated and assessed in a nonnuclear test facility. Unlike the SAFE system, no intermediate HX is required for the DDG to PCS integration. Preliminary DDG testing with simulated neutronic response was performed using the neutronic model previously implemented in the SAFE test series and making use of simple instrumentation initially installed in the DDG core for basic check-out tests. Heat removal was accomplished by a gas-to-water heat exchanger located on the hot side of the gas flow loop; no PCS was included in the initial test configuration.

The test series discussed in the current paper integrates the DDG core simulator with a Brayton power conversion unit (BPCU) at NASA Glenn Research Center (GRC) to demonstrate integrated system performance with simulated neutronic feedback implemented in the reactor simulator control system. The companion paper in these proceedings (Ref. 12) provides additional detail on the BPCU design and the series of initial checkout tests conducted on the integrated DDG-BPCU.

II. Neutronics Model

Reactor dynamics can be modeled using the point kinetics equations (PKE), which can be derived from transport and diffusion theory (Hetrick, 1971 (Ref. 13)). The PKE have been used to describe reactor dynamics in previous space reactor system models, including that developed for a nuclear reactor coupled to a closed-loop Brayton cycle (Ref. 14) and for SP-100 system analysis (Ref. 15) In the absence of an external source and written in terms of reactor fission (thermal) power, the PKE are given by (see Nomenclature):

$$\frac{dP(t)}{dt} = \frac{(\rho(t) - \beta)}{\Lambda} P(t) + \lambda_i C_i(t), \qquad (1a)$$

and

$$\frac{dC_i(t)}{dt} = \frac{\beta_i}{\Lambda} P(t) - \lambda_i C_i(t). \qquad (1b)$$

The current value of the reactivity, $\rho(t)$, is affected by the position of the reactor control mechanisms (i.e., drums, sliders, etc.) and by the reactor temperature, which affects both the thermal deformation of the reactor core (and, hence, neutron leakage) and the neutron cross sections in the nuclear fuel (via Doppler broadening). The reactivity is given by:

$$\rho(t) = \rho_o + \alpha_T \left(T(t) - T(t_o) \right), \qquad (2)$$

where ρ_o is the initial steady-state reactivity at time t_o before a transient is applied to the system and α_T is the temperature coefficient of reactivity,

$$\alpha_T = \frac{d\rho}{dT} \qquad (3)$$

In Equation (1), the reactivity is unitless. However, it is often expressed in units of dollars and cents, where 100 cents is equal to $1 and $1 of reactivity is equal to ρ/β. The PKE are solved in a point-wise fashion in the applied dynamic testing. This allows the reactivity to be assumed constant at each iteration of the control loop, solving for the new $P(t)$ based on the current reactivity; the current value of reactivity is then updated at each iteration of the dynamic control loop. Increasing the iteration frequency of the control loop to the maximum data update frequency in the instrumentation (accounting for data communication delays) can improve the accuracy of the applied solution.

Studies of reactor dynamics generally recognize six distinct groups of delayed neutrons. Approximate solutions to the PKE can be found using a smaller number of groups to represent the delayed neutron population. The dynamic models applied to date (both for the SAFE and DDG) include a simple dynamic model having one group of delayed neutrons, applying a weighted average decay constant and the total delayed neutron fraction. The individual decay constants and delayed neutron fractions for a fast spectrum reactor are summarized by Hetrick (Ref. 13), where the total decay constant is $\lambda = 0.0767 \text{ sec}^{-1}$ and the total delayed neutron fraction is $\beta = 0.00642$. Initial application of the model assumes that these values are approximately accurate for both the SAFE and DDG fast spectrum reactors. However, the appropriate parameters should be specifically determined for each reactor design via detailed neutronic analysis as the dynamic test methodology is improved. A one-group representation simplifies the computational model, speeding the real-time implementation of simulated reactivity feedback during test but introducing potential error in the solution for the reactor fission power as a function of time. To improve the fidelity of the neutronic model, a more detailed six-group PKE representation will be applied in future tests. To maintain real-time implementation of the neutronics model, high speed computing must be employed. The DDG-BPCU test series adopted the simplified one-group model to minimize the number of variables introduced from one test series to the next.

The PKE provide a good estimate of the reactor dynamics in a fast spectrum system. Once the relevant feedback mechanisms are understood for a particular reactor design appropriate instrumentation and measurement points can be selected for the nonnuclear test hardware. The importance of the hardware instrumentation will be discussed in the results presented.

A simple model of point reactor temperature feedback has been applied to date. In each test configuration a single bulk temperature reactivity feedback coefficient was applied to the core due to the limited instrumentation in both the SAFE and DDG systems. The test and control methodology discussed in this paper was developed after followed the associated hardware was designed and fabricated. Hence, there was minimal opportunity to introduce instrumentation selected specifically to support dynamic testing with simulated neutronic feedback. Future enhancements to the control methodology with simulated neutronic feedback can be achieved through early instrumentation and control system design as a part of the broader system design tasks.

Specific application of the dynamic test methodology to the DDG-BPCU system will be discussed.

III. Test Article Description

A Direct Drive Gas-Cooled (DDG) reactor core simulator, initially tested at NASA MSFC (Refs. 9 and 16) has been coupled to a Brayton Power Conversion Unit (BPCU) for integrated system testing in a thermal vacuum facility at NASA GRC. The configuration of both the DDG and the BPCU and the integration of these components are described.

A. DDG Core and Simulator Design

The DDG reference design is based on a gas-cooled, UN-fueled, pin-type fast reactor using HeXe gas flowing directly into a recuperated Brayton system to produce electricity for nuclear electric propulsion (Ref. 17). The DDG, which consists of 37 fuel pins in a block matrix, is designed to operate at 32 kWt; this is a reduced power version of a full scale design. Each fuel pin is surrounded by an annular flow channel, which provides direct heat removal by flowing gas. The DDG design was initially developed by Sandia National Laboratory but was transferred to Los Alamos National Laboratories for additional design and testing support. The 37-pin DDG is constructed entirely of stainless steel. The core block is a 53.3 cm (21 in.) long, 16.5 cm (6.5 in.) (flat-to-flat) hexagonal, solid stainless steel block with a pattern of 37, 1.9 cm (¾ in.) diameter holes gun-drilled through the length.

The DDG core is designed to operate using high pressure, 2.4 MPa (350 psia), He/Xe coolant that is directly coupled to the power conversion system. Initial DDG testing at NASA MSFC used He/Ar gas coolant (20% He by mass) due to the significant cost of He/Xe gas. Checkout tests of the DDG-BPCU integrated system at NASA GRC used Krypton gas; all

system response testing discussed here used a He/Xe gas mixture.

He/Xe gas flows along the outside surface of the DDG core (this is the "downcomer" region), flows through the annular coolant flow channels along the heater elements, and reaches a second plenum (shown at the top of Fig. 1) where it exits through a common flow channel to be recirculated by the BPCU. The DDG was designed to operate with gas inlet and outlet temperatures of 650 K and 850 K, respectively. The upper temperature limit in the current test loop was established by limitations of materials employed in the loop construction that would not be typical for a flight unit.

The graphite heater elements used to simulate the heat from nuclear fission were custom-designed by engineers at NASA MSFC (Ref. 18). The elements installed in the DDG have a constant outer diameter and are electrically isolated from the DDG core block through the use of alumina rings at three axial positions on each heater element. The constant diameter provides a flat axial power profile along the reactor simulator. The elements are grouped into four control zones, as shown in Figure 2. Previous application of electric heaters in the SAFE
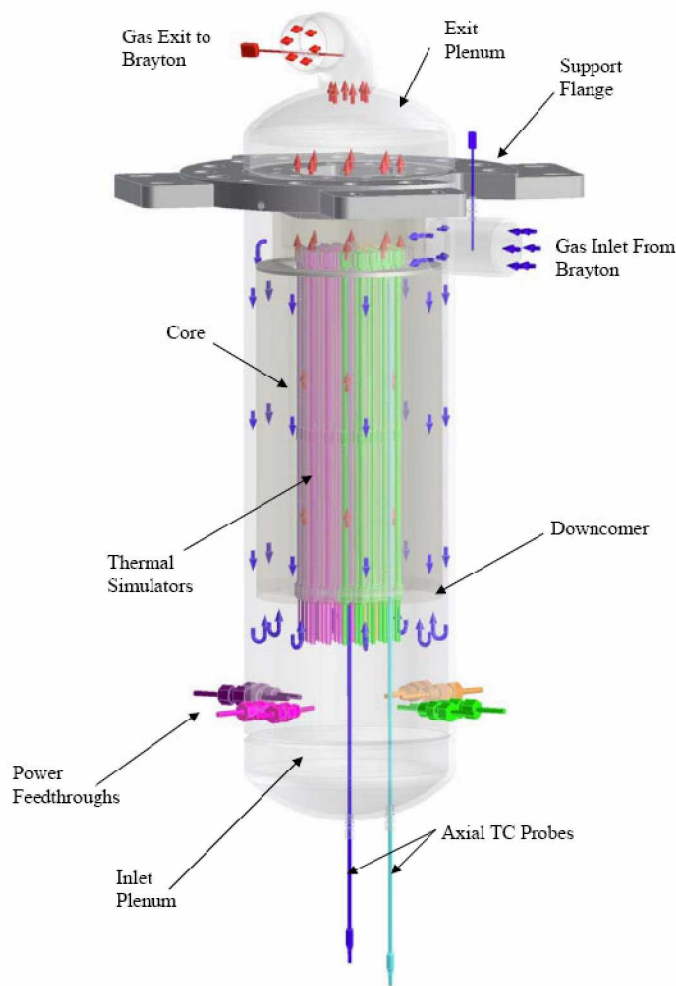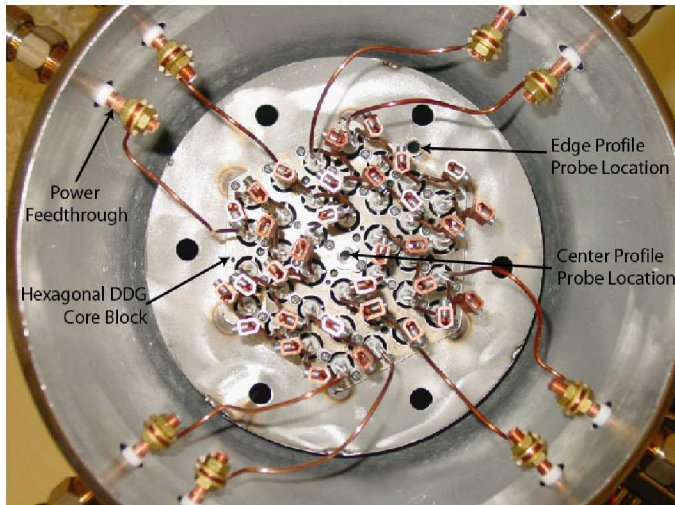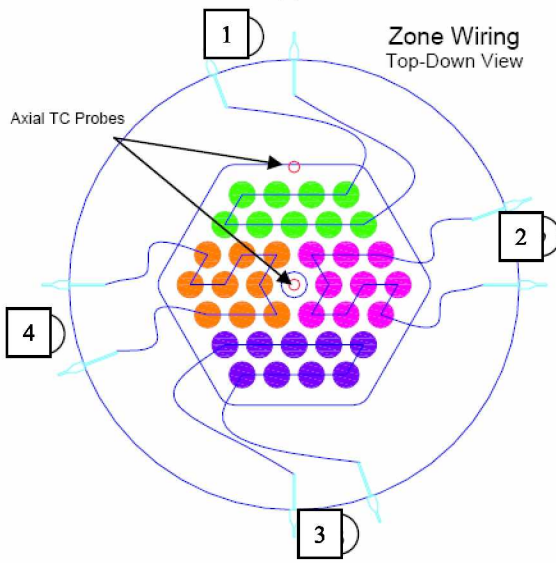

Figure 1.—DDG flow path; cold gas and hot gas flows are indicated by blue and red arrows, respectively.

(a)

(b)

Figure 2.—DDG heater element control: (a) DDG heater element installation showing electrical connections and (b) schematic showing corresponding power supply control zones.
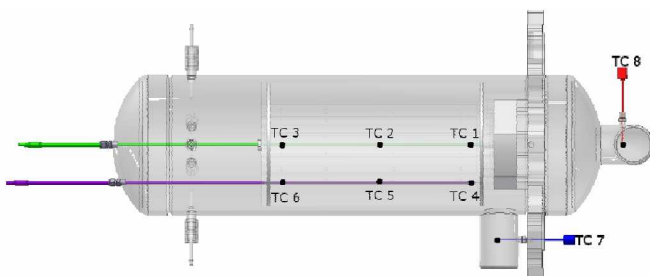

Figure 3.—DDG rendering with thermocouple locations identified.

and initial DDG simulator testing wired the heater elements in radial zones to allow radial distribution of the power (i.e., to simulate the cosine radial power distribution in an operating reactor core). The current power zone configuration, selected to simplify wiring, allows only a flat radial power profile.

To provide increased performance and ease of integration with the BPCU, the upper bonnet flange in the original DDG construction was replaced to provide a more stream-lined, compact design than in previous tests. The modified all-welded DDG design is capable of heating a He/Xe (or other inert gas) mixture to 1000 K, delivering up to 15 kW thermal power; operational pressure for the modified system is 0.7 MPa (100 psia) and the maximum gas flow rate is 0.2 kg/s.

Each of the two axial profile probes, manufactured by Omega, has a 0.6 cm (¼ in.) stainless steel sheath contains three thermocouples (TCs) located approximately at the start (TC3, TC6), axial midline (TC2, TC5), and end (TC1, TC2) of the heated region of the core. The central fuel pin/heater location has been plugged on one end to provide a stagnant gas pocket through the center of the core for installation of the central axial probe. The probe on the core block periphery is inserted in a tube that is seam-welded along the length of the core block surface, providing good thermal contact with the block and allowing for easy removal/replacement of the probe. See probe locations shown in Figure 2(a); additional TCs are located at the gas inlet (TC7) and gas outlet (TC8), as shown in Figure 3.

B. BPCU Description

The BPCU is a 2 kWe unit originally developed for solar dynamic system flight experiments planned for the Mir space station in 1997. The flight experiment was canceled but the unit was tested at GRC as a part of the Solar Dynamic Ground Test Demonstration System (Ref. 19). The convertor was later modified for use in the nuclear electric propulsion test bed at GRC in ~2003.

The BPCU is a fully integrated power conversion system including a common shaft turbine-alternator-compressor, recuperator, and gas cooler connected by gas ducts. The gas loop is designed to use a working fluid of 62.7 mole % He and 37.3 mole % Xe gas mixture with an average molecular weight of 83.8 g/mol. Previous testing, conducted with a simple gas heater rather than a reactor simulator, fully characterized the BPCU performance at 48000 and 52000 rpm at a variety of input heater power levels (Refs. 20 and 21).
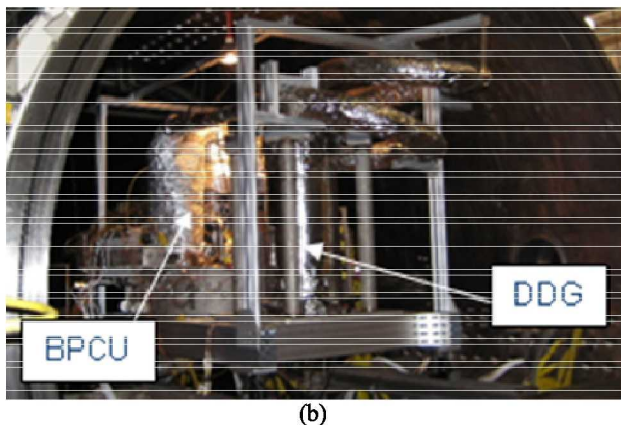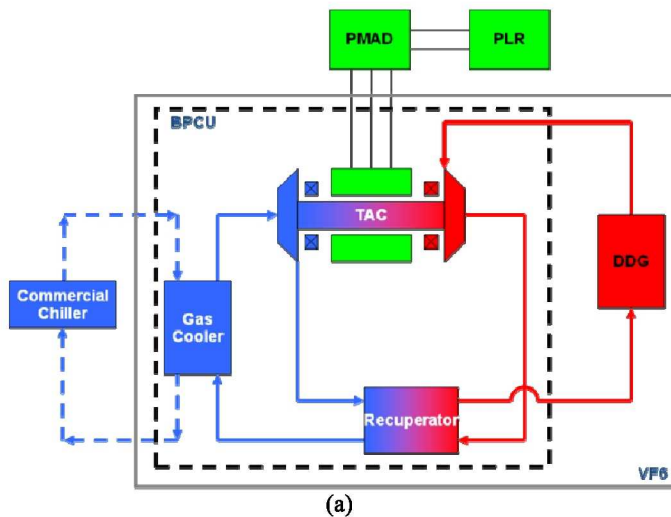
C. DDG-BPCU Assembly and Test Procedure

The DDG was fully assembled with resistive heaters, profile probes, and modified vacuum pressure vessel (all-welded design) prior to shipment to GRC. The DDG hardware was integrated with the BPCU in Vacuum Facility 6 (VF6) housed in NASA GRC Building 301 (Ref. 12). The integrated system schematic and corresponding hardware are shown in Figure 4.

The reactivity feedback control system described in Section II was implemented via a custom-designed LabVIEW (National Instruments Corporation) control program developed at NASA MSFC. The control program allowed for the feedback to be based on any of the eight TCs identified in Figure 3. The reactivity feedback test matrix is summarized in Table I. The initial power levels were selected based on prior DDG-BPCU testing conducted using temperature control; these power levels were refined on the day of the test to achieve a desired outlet gas temperature. The power levels following the indicated transient (e.g., variation of 37 to 46 krpm shaft speed in test series A) are determined by the controller and, hence, are not reported in the test matrix.

The maximum heater element temperature was initially selected as the desired input to the reactivity feedback control loop. Optimally, the fuel simulator (heater element) temperature should be directly measured at various points in the reactor core simulator to estimate the state of the corresponding fueled reactor. However, because this capability was not available in the fuel element simulators selected for DDG testing, one of the eight TCs identified in Figure 3 had to be employed.

(a)

(b)

Figures 4.—(a) DDG-BPCU system schematic and (b) hardware configuration as installed at GRC Vacuum Facility 6.

The heater temperature was initially estimated from TC1 (center, outlet end of core); the temperature at TC1 was closely followed by TC4, such that the change in the temperature at either thermocouple was approximately equivalent. Initial application of the feedback control indicated that the temperatures measured at TCs 1 and 4 suffered from significant time lag relative to changes made in the heater element power levels and, therefore, were not representative of the simulated fuel temperature. This lag resulted from the significant thermal pathway between the heater element and the TC measurement point that included a gas gap (heater element to core block), stainless steel core block structure, a second gas gap (block to TC probe), and probe packing materials. As a result of this thermal lag between the actual heater temperature and that measured at TCs 1 and 4, an alternate measurement point was necessary to estimate the element temperature. The outlet gas temperature, measured by TC8, was selected for feedback control because it was more closely coupled with the heater element (fuel simulator) temperature than measurements in either TC profile probe. It should be noted that the lack of imbedded temperature measurement in the fuel simulators establishes a significant limitation in the instrumentation system that results in a measured power response to flow changes that is not prototypic of a fueled reactor; this limitation can be rectified via instrumented fuel simulators.

TABLE I.—REACTIVITY FEEDBACK TEST MATRIX

Test identifier	Shaft speed (krpm)	Initial DDG electrical power (W)	Initial temperature (TC 8, °C)
A	Variation of BPCU shaft speed @ 555 °C outlet.		
	37 46 37	~3700	555
B	Positive reactivity insertion.		
	37	~3900	555
C	Variation of BPCU shaft speed @ 580 °C outlet.		
	37 46 52 46 37	~4300	580
D	Negative reactivity insertion.		
	37	~4230	580

After first establishing steady state operation, the test matrix described in Table I includes transition between several shaft speeds at each initial gas outlet temperature (series A and C); the 52 krpm shaft speed could not be sustained at the low input power level in test series A due to limitations in the BPCU. System response to both increasing shaft speed (e.g., 37 to 46 rpm) and decreasing shaft speed (e.g., 46 to 37 krpm) was assessed to characterize potential system hysteresis effects. These transients simulate a load change that would result from variation in the electric power extracted from the power conversion system.

With the simulated reactivity feedback control loop implemented, a change in the BPCU shaft speed, which increases or decreases coolant mass flow, results in a variation in the reactor simulator power level such that the average reactor temperature is the same prior to and following the transient, as described by the PKE (see Eqs. (1) to (3)). BPCU shaft speed is adjusted manually on the BPCU control panel and is not integrated into the computer control system. Without simulated neutronic feedback implemented to adjust the core power level the core and gas outlet temperatures would increase/decrease as the gas flow is adjusted via modification to the BPCU shaft speed.

Transients included in tests B and D demonstrate the effect of increasing or decreasing reactor power level, as would be observed in a reactor control maneuver (e.g., due to control drum rotation). This is expressed here as an insertion of positive or negative reactivity. These transients are inserted between test A and C and following test C as a controlled means to increase/decrease core power to the next desired test point (i.e., gas outlet temperature of 555 °C (828 K) to 580 °C (853 K)). In each case the desired reactivity insertion was determined using Equation (2). Assuming initially steady state operation (ρ_o = 0) and α_T = –0.2 cents/K, each of these transients corresponds to a reactivity insertion of ± \$0.05.

Note that in the present application the reactivity insertion transients are implemented as step insertions; adjustments to the reactor power level during flight operation would apply slow, controlled ramp insertions. Hence, the data presented represent a highly conservative estimate of the system response time and oscillatory behavior. Also note that because the BPCU control is performed manually it is unable to adjust in response to increased/decreased core power (i.e., the BPCU does not adjust shaft speed based on changes to the input DDG power).

IV. Results and Discussion

The integrated DDG-BPCU system takes several hours to prepare for testing, including pump-down of the vacuum chamber and heat-up of the DDG hardware using the installed resistive heater elements. After attaining the desired steady-state system temperature at a BPCU shaft speed of 37 krpm, check-out testing of the simulated neutronic feedback control loop was performed to ensure proper setting of the system feedback coefficients. During this pretest stage several feedback coefficients were considered in the control system, ranging from –0.33 to –0.10 cents/K. Because the DDG and installed heater elements are not prototypic of specific reactor core and fuel element design, the feedback coefficient for the test series was selected from a range of values that could be applicable to a fast spectrum reactor core. Using TC1 as a state estimator for the feedback control, large oscillations were observed following a modest transient given a feedback coefficient at –0.33 cents/K. These oscillations tended to grow rather than damp with time, demonstrating the issues associated with improper instrumentation for application of

feedback control. A feedback coefficient of –0.2 cents/K demonstrated a damped oscillatory response of the system to an applied transient. This value corresponds to that applied in previous SAFE-100a and DDG dynamic testing at NASA MSFC.

As noted in Section III.C, TC1, located at the center of the DDG core, was initially selected for feedback control maneuvers. However, due to the significant time lag between control maneuvers and TC response at this position, undesirable oscillations in the core power level were observed following the application of a modest system transient. As a result, the outlet gas temperature measured at TC8 was selected for the current series of integrated system testing. This temperature measurement was much noisier than the core block temperatures (TCs 1 to 6), resulting in a highly sensitive control system that was difficult to drive to a true steady state condition (even temperature variations within the uncertainly of the TC instrumentation could drive a power level adjustment).

Data sampling and control was applied at 10 Hz. To reduce the magnitude of data recorded for each test series, data was recorded to the associated data files only once every 10 sec. Hence, the data used for post-processing and analysis of system performance (e.g., to produce plots included in this paper) indicates smoother output data than was observed on the LabVIEW control interface during test.

Results of each test are summarized in Figures 5 to 8. The system took approximately 1 hr to reach a new steady state for each transient applied. In test series A and C the integrated system responded as predicted in that the measured core temperature was the same as that prior to the transient. In some cases, however, one will note that the system did not reach a true steady state condition with regard to the core power level due to the highly sensitive response that results from use of noisy temperature data.

As noted previously, use of TC8 as the state estimator for the feedback control system makes the assumption that the outlet gas temperature follows the average fuel temperature. This assumption does not account for the time lag associated with increased/decreased heater (fuel simulator) temperature due to a change in the power level and the corresponding increase/decrease in coolant temperature. Although the time lag in TC8 is not as large as that for TCs 1 to 6, the outlet gas temperature still does not provide an exact representation of the average fuel temperature response to changes in the input power level, resulting in nonprototypic response of the DDG relative to a fueled reactor. This instrumentation deficiency could be addressed, in part, by using average gas temperature (average of the inlet, TC7, and outlet, TC8, gas temperatures), but the ability to select multiple TCs for feedback control was not included in the current LabVIEW control program.

Alternately, future core simulator instrumentation should include direct measurement of heater temperature at multiple points across the core. The corresponding fueled reactor will respond to changes in the coolant mass flow by maintaining constant average core temperature; the axial temperature rise

across the core (dT) adjusts to match the core power level to the coolant flow rate. The resulting core power is then a function of the coolant mass flow rate ($\dot{m}$), the specific heat of the coolant (C_p) and the axial temperature rise across the core: $P = \dot{m}C_p dT$. The present analysis assumes that the mass flow rate is directly proportional to the BPCU shaft speed, although this may not be strictly true in some operational regimes; C_p is a constant for a given coolant gas.

Transients implemented in tests B and D also performed as expected. The positive/negative reactivity insertion of ± $0.05 was selected to attain a desired 25 °C (25 K) change in the outlet gas temperature. Following large oscillations in the core power level, as seen in Figures 6 and 8, the desired outlet gas temperature was attained. The positive or negative spike in the core power level could be reduced by application of a smart controller in conjunction with the implemented neutronic feedback control system.

V. Discussion

The test configuration discussed in this report applied a simplistic model of simulated reactivity feedback to the integrated DDG-BPCU system. However, this test methodology was adopted after initial design and fabrication of the DDG core and other system components and minimal opportunity was available to introduce temperature measurement designed specifically for application of feedback control. Axial profile probes installed following earlier DDG testing (Refs. 8, 9, and 16) conducted prior to the current test series provided additional valuable information relative to previous testing, but the complete instrumentation plan introduced in Bragg-Sitton and Webster (Ref. 8) was not adopted due to time and cost constraints. None of the available TC measurements in the DDG core provided a good estimation of "fuel" temperature in a corresponding reactorcore design. Introduction of instrumented heater
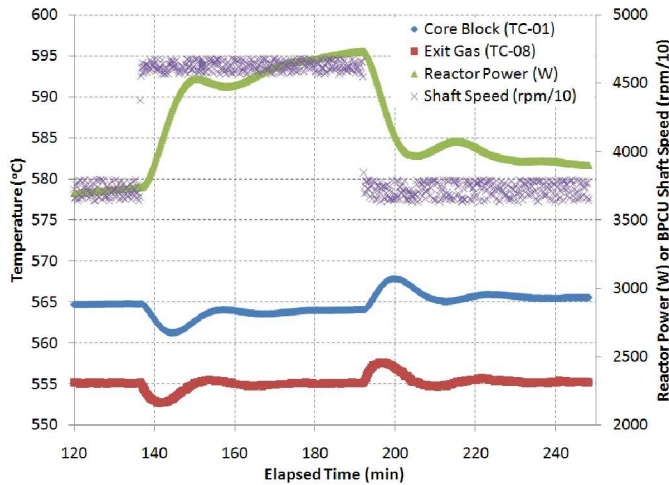

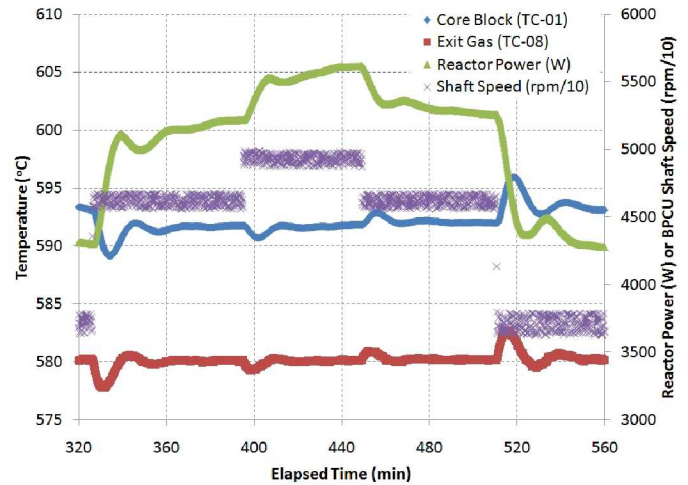Figure 5.—Transient Series A: 555 °C gas outlet temperature.

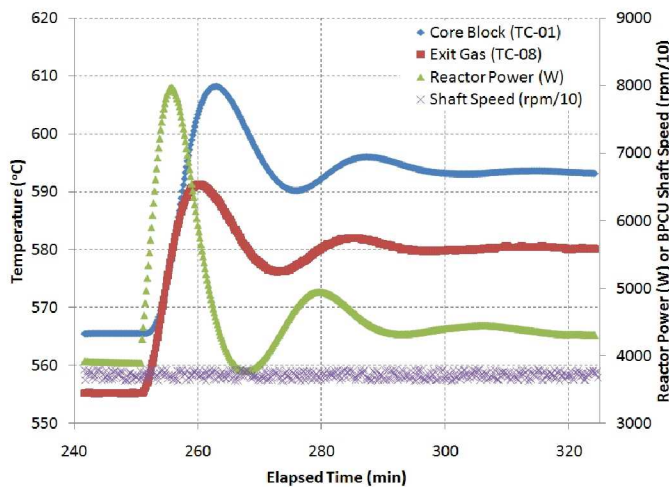Figure 7.—Transient Series C: 580 °C gas outlet temperature.

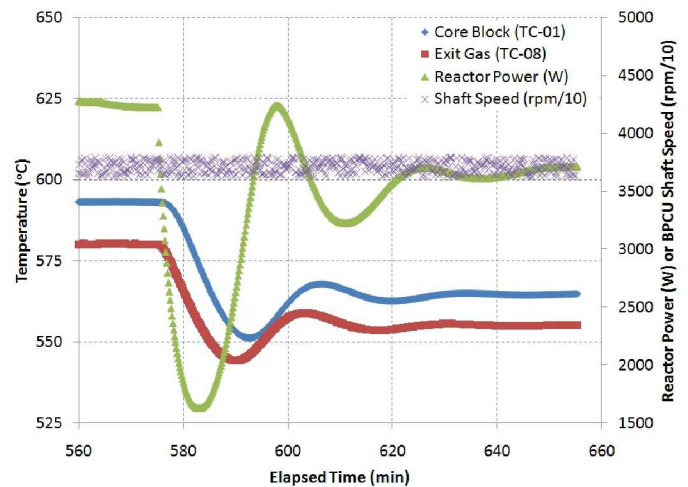Figure 6.—Transient Test B: Positive reactivity insertion.

Figure 8.—Transient Test D: Negative reactivity insertion.

elements that incorporate a temperature measurement system internal to their design would provide improved state estimation in future applications.

Additionally, no mechanism (i.e., temporal or spatial data averaging) was included in the LabVIEW control program to reduce noise in the data signal applied in feedback control. Future testing should adopt noise reduction techniques in the data acquisition system. These techniques could include (1) selection of a feedback control signal from all available TC data or an average of those data points, as applied in the SAFE-100a test series(Ref. 10) in which state estimation was based on an average of all core TCs; (2) application of a moving time average to the measured temperature data (i.e., measured temperature input into the control could be the average of the previous 5 or 10 data points, as applied in SAFE-100 testing (Ref. 11)); or (3) application of a computational filter, such as a Kalman filter, in the control system.

Future nonnuclear testing of fission power systems for space application should adopt a highly realistic approach to system control (with neutronic feedback) early in the design phase to allow appropriate selection of core instrumentation. Improved instrumentation and enhancement of the feedback model made possible by more highly distributed instrumentation (e.g., introduction of multiple feedback components, such as fuel, core block, and reflector components) could provide significant improvement in the accuracy of nonnuclear testing to characterize integrated system performance. Additional improvement in the feedback model will be accomplished through the use of an improved reactor dynamic model, i.e., introduction of six delayed neutron groups in the reactor point kinetics model as has been adopted in other dynamic models of reactor systems (Ref. 14).

The tested control system simulated the inherent transient response of an integrated reactor core and power conversion system by modeling the reactor core neutronic response that results from system temperature variation. Because no constraints were applied to the magnitude of adjustments in the core power level step insertions of reactivity resulted in large oscillations in the core power level. An additional controller can be modeled and introduced to the control system to limit the maximum power level or maximum change rate in the power level following a control maneuver to reduce the potential for system damage (or shutdown if safety settings in the LabVIEW controller are tripped). In this manner, an autonomous reactor control system could be tested in a nonnuclear test environment that integrates a reactor core simulator and power conversion system.

VI. Conclusions

The presented test series demonstrated the integrated operation of an electrically-heated, gas cooled reactor core simulator with a Brayton power conversion unit. Results indicate expected system performance in response to transients

initiated at the BPCU via changes in the shaft rotation speed (to emulate system load changes) or simulated reactor control maneuvers to vary the input power level. As expected for a monolithic core block design that has significant thermal inertia, the observed system time constant was significantly longer (~1 hr) than that observed for earlier testing of a simulated modular core design (~20 min time constant observed in the SAFE-100a test series (Ref. 10)).

The current and previous dynamic system response tests demonstrate the overall capability of a nonnuclear test facility in assessing system integration issues and characterizing integrated system response times and response characteristics.

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13. SUPPLEMENTARY NOTES

14. ABSTRACT
A Direct Drive Gas-Cooled (DDG) reactor core simulator has been coupled to a Brayton Power Conversion Unit (BPCU) for integrated system testing at NASA Glenn Research Center (GRC) in Cleveland, Ohio. This is a closed-cycle system that incorporates an electrically heated reactor core module, turboalternator, recuperator, and gas cooler. Nuclear fuel elements in the gas-cooled reactor design are replaced with electric resistance heaters to simulate the heat from nuclear fuel in the corresponding fast spectrum nuclear reactor. The thermodynamic transient behavior of the integrated system was the focus of this test series. In order to better mimic the integrated response of the nuclear-fueled system, a simulated reactivity feedback control loop was implemented. Core power was controlled by a point kinetics model in which the reactivity feedback was based on core temperature measurements; the neutron generation time and the temperature feedback coefficient are provided as model inputs. These dynamic system response tests demonstrate the overall capability of a nonnuclear test facility in assessing system integration issues and characterizing integrated system response times and response characteristics.

15. SUBJECT TERMS
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