Three-dimensional coupled kinetics/thermal-hydraulic benchmark TRIGA experiments

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Abstract

This research project provides separate effects tests in order to benchmark neutron kinetics models coupled with thermal-hydraulic (T/H) models used in best-estimate codes such as the Nuclear Regulatory Commission’s (NRC) RELAP and TRAC code series and industrial codes such as RETRAN. Before this research project was initiated, no adequate experimental data existed for reactivity initiated transients that could be used to assess coupled three-dimensional (3D) kinetics and 3D T/H codes which have been, or are being developed around the world. Using various Test Reactor Isotope General Atomic (TRIGA) reactor core configurations at the Penn State Breazeale Reactor (PSBR), it is possible to determine the level of neutronics modeling required to describe kinetics and T/H feedback interactions. This research demonstrates that the small compact PSBR TRIGA core does not necessarily behave as a point kinetics reactor, but that this TRIGA can provide actual test results for 3D kinetics code benchmark efforts. This research focused on developing in-reactor tests that exhibited 3D neutronics effects coupled with 3D T/H feedback. A variety of pulses were used to evaluate the level of kinetics modeling needed for prompt temperature feedback in the fuel. Ramps and square waves were used to evaluate the detail of modeling needed for the delayed T/H feedback of the coolant. A stepped ramp was performed to evaluate and verify the derived thermal constants for the specific PSBR TRIGA core loading pattern. As part of the analytical benchmark research, the STAR 3D kinetics code (Weader, 1992, STAR: Space and time analysis of reactors, Version 5, Level 3, Users Guide, Yankee Atomic Electric Company, YEAC 1758, Bolton, MA) was used to model the transient experiments. The STAR models were coupled with the one-dimensional (1D) WIGL and LRA and 3D COBRA (Rowe, 1973, COBRA IIIC: A digital computer program for steady-state and transient thermal-hydraulic analysis of rod bundle nuclear fuel elements, Battelle Institute, Richland, WA). T/H models to determine the level of...
T/H modeling required to accurately describe the behaviour of the PSBR TRIGA core during these transient conditions. STAR’s 1D T/H models (WIGL and LRA) were adequate for the rapid pulse events, when accurate temperature-dependent fuel thermal constants were used and the reactor coolant feedback mechanism was small. However, the longer transients (i.e. ramps, square waves) necessitated the use of the COBRA 3D fluid flow analysis coupled with the 3D STAR kinetics model.

1. Introduction and historical background

Reactivity initiated accidents, such as rod ejections and boron dilution events, and postulated events with large positive moderator and T/H feedback [e.g. main steam line breaks (Feltus 1994)] are important neutronic transients that need to be evaluated to determine the overall safety and risk of operating nuclear power plants. These types of transients, as well as others, have been simulated using large T/H system and kinetics computer codes that have been developed by international regulatory bodies as well as the nuclear industry. Before this research project was initiated, no adequate experimental data existed for reactivity-initiated transients that could be used to access the coupled 3D T/H kinetics codes which have been, or are being developed around the world. Rather than an assessment against experimental data, benchmarks have been limited to comparisons between the various 3D kinetics computer code methodologies. While this inter-comparison of dynamics codes is necessary, it is inherently insufficient. The ultimate test of any computer code, like the proof of any scientific theory, is comparisons with actual physical measurements, not comparisons only with other computer models.

A series of pulses, square waves, and ramps were performed at the PSBR TRIGA reactor, for various core loading patterns. For the extensive data collection and benchmark effort, three pulses (1.5, 1.75 and 2.0), three square waves (200, 300 and 500 kW) and three power ramps to 500 kW (with 30, 40 and 50 s periods) were performed. All transients initiated from a power level of 50 W and were used to evaluate the level of detail of T/H modeling needed with 3D kinetics analysis. The pulses were conducted to evaluate the level of neutronics (1D or 3D) necessary for the modeling the prompt temperature feedback mechanism in the TRIGA fuel elements. The ramps and square waves were conducted to evaluate the level (i.e. 1D or 3D) of T/H modeling needed with the coupled 3D neutrons for modeling the delayed T/H feedback mechanism of the complex fluid flow inside the TRIGA core. The square waves and ramps were allowed to reach a steady-state power level and equilibrium core conditions before being scrammed. A stepped ramp (50 W to 750 kW) transient was performed to verify the fuel element thermal constants and related the steady state fuel temperature distributions to the specific core power level. This approach demonstrated that the thermal constants (i.e. heat capacity, specific heat) of the PSBR TRIGA fuel elements as a function of temperature were determined consistently.

This research effort, supported by the US NRC, has provided both experimental and analytical benchmark information for coupled T/H and 3D kinetics feedback for reactor safety analysis efforts. The results presented in this paper can be used to
qualify the point, 1D and 3D kinetics models in the NRC’s RELAP and TRAC series of codes as well as industry codes such as RETRAN (Feltus, 1994). The experimental data and analytical results can be used to formulate a standardized benchmark problem for verification of coupled neutrons and T/H codes, which includes the complex T/H and kinetics phenomena exhibited in the TRIGA reactor.

Since the configuration, composition, thermal and nuclear data of the PSBR TRIGA is well-documented (Miller, 1998), other investigators can easily test the capabilities of their 3D coupled neutronics and T/H methodologies against these experimental results. Furthermore, since the actual TRIGA test procedures and instrumentation details are delineated in the research thesis (Miller, 1998), the experiments can be replicated at other TRIGA facilities and used for future experimental benchmarks.

2. Analytical methodology (STAR, WIMS and COBRA)

The neutronic and T/H coupling was accomplished by implementing three T/H codes contained internally within the STAR code (Weader, 1992): COBRA IIC, WIGL, and LRA. The WIGLE and LRA 1D T/H models (Weader, 1992) are utilized by including an additional data cards in the STAR input deck. To implement the COBRA 3D T/H model, both a STAR model and a COBRA IIC model of the PSBR TRIGA were created, and executed using the full neutronics and T/H coupling option in STAR.

The STAR and COBRA models were configured so that each STAR neutronic node has a one-to-one correspondence between each COBRA channel modeled (Miller, 1998). The STAR model includes detailed axial modeling of each of the fuel elements, shim, regulating, transient and safety rods, and the reflector region based on both initial and transient conditions. Two radial nodes per fuel element are used for the radial core layout. The COBRA model uses one channel per neutronic node and includes the pertinent mechanical flow restrictions of the fuel elements. Miller, (1998) provides extensive details about the core geometry and STAR and COBRA models for core loading pattern 48.

The analytical methodology for performing the coupled 3D kinetics and T/H analysis is provided in this section, as well as general descriptions of the STAR (Weader, 1992), COBRA IIIC (Rowe, 1973), and WIMS-D4 cross section generation code (Oak Ridge, 1991). The calculational and numerical methods used by each code and the advantages of each method are also described.

2.1. STAR code description

STAR (space-time reactor analysis) is a 3D reactor kinetics code (Weader, 1992) that can calculate the time-dependent coupled T/H response to reactor transients. The STAR code is based on the QUANDRY analytic nodal method (ANM) and uses a spatial coarse nodal mesh, finite difference technique to solve the time-dependent transient nodal kinetics and diffusion equations. STAR uses standard
flux, fission source method and fully-implicit temporal iterations. A non-linear iteration is used outside of the standard ANM techniques to correct the coarse-mesh finite difference (CMFD) solution to match the higher order QUANDRY ANM solution. The diffusion equations are integrated over the transverse directions, and the spatial integrations solve for the node-averaged fluxes and the face-averaged net leakages. CMFD coupling coefficients based on updated group constants are then calculated to couple the nodal averaged fluxes and the surface flux and current.

STAR has many advantages (Feltus, 1994): (1) in the limit of a fine mesh the CMFD discontinuity factors approach unity and STAR converges to the exact solution of the diffusion equation; (2) the STAR global power level is calculated directly from the 3D nodal power distribution without any quasi-static approximations; (3) STAR uses a true two group solution, instead of $1^{1/2}$ group approximation; (4) no adjustments (i.e. albedoes) are needed at the core/reflector boundary; (5) the transient kinetics equations are solved by a full-implicit time differencing scheme; and (6) as the spatial mesh sizes become small, STAR matches the finite difference solution exactly. Although STAR is limited to rectangular geometries, the hexagonal layout of the TRIGA core can be accurately modeled using 2 nodes per fuel element (Miller, 1998).

STAR contains four T/H models internally (Weader, 1992): WIGL, LRA, COBRA and VIPRE. These codes calculate the in-core temperatures and fluid properties, and use those properties to determine the T/H feedback for the core neutronics. Each T/H model can be either run as a “stand alone” code, or executed with STAR implicitly. While running transients, STAR makes two calls to COBRA. First STAR executes a static COBRA run to initialize core conditions. The STAR executes a transient COBRA run during each STAR transient time step. STAR inputs the inlet distribution of the coolant temperatures, coolant mass flow rate, the ratio of the local mass flux to the core average mass flux, and the pressure to COBRA. COBRA completes its run and outputs the fuel temperature distribution, coolant density, and temperature back to STAR.

2.2. COBRA IIIC code

The principle code used to model the T/H feedback for the benchmark was COBRA IIIC which is contained internally within STAR. COBRA IIIC (Rowe, 1973) is a steady-state and transient subchannel analysis code that uses a semi-explicit finite difference scheme where the time-dependent boundary conditions are: inlet mass velocity, enthalpy, and exit pressure. COBRA IIIC computes the flow and enthalpy of the coolant in the core subchannels for either sub-cooled or boiling conditions. Transverse flow can be accounted for by considering turbulent and diversion cross-flow mixing between adjacent channels. Each subchannel assumed to contain 1D predominant flow, two-phase, separated slip flow. The void fraction is defined as a function of enthalpy, flow rate, heat flux, pressure, position, and time.

The most important features of COBRA IIIC used for this research are: (a) COBRA can accurately handle transients of fast to intermediate speed; (b) the numerical solution has no stability limitation on space or time steps; (c) a more
complete transverse momentum equation now includes temporal and spatial acceleration of the diversion cross-flow; (d) a fuel pin model option allows calculation of the fuel and cladding temperatures during transients by specifying power density, and (e) improved numerical procedures allow more complete analysis of bundles with partial flow blockages. The complex flow paths and mechanical configuration of the PSBR TRIGA fuel could be modeled with COBRA IIIC for this effort. Since STAR has the COBRA IIIC code already incorporated internally (Weader, 1992), it was necessary to compare the COBRA IIIC model generated for this research with a less-detailed COBRA IV TRIGA model developed by Gougar (1997) for his TRIGA flow control experiments. After the COBRA IIIC model was completed, it was executed with the STAR stand-alone option and compared with Gougar’s COBRA IV results for steady-state conditions (Miller, 1998).

2.3. WIMS-D4 code description

The group constants (i.e. cross-sections) for the STAR models were generated with the WIMS-D4 code (Oak Ridge, 1991). WIMS-D4 is a general lattice program that uses transport theory to calculate flux as a function of energy and position in the cell. WIMS-D4 utilizes a two step process to perform cross-section calculations. The first step calculates the spectra for a few spatial regions of the cell with the full number of energy groups in its library. These spectra are used to condense the basic cross sections into a few energy groups. Second, WIMS-D4 carries out a few group calculation using a detailed spatial representation. The resulting fluxes are then expanded using the previous calculation so that the reaction rates in each spatial point can be calculated using the library group structure.

The WIMS-D4 internal library can provide basic microscopic cross section data for up to 69 energy groups. WIMS-D4 then calculates the macroscopic cross-sections for each material in the cell, and automatically calculates the resonance shielding. The preliminary, many group, few region, spectrum calculation is then carried out using collision probability techniques. The primary, many region, condensed group transport calculation is carried out with either collision techniques or WDSN methods (Oak Ridge, 1991). WIMS-D4 can utilize a variety of different geometric configurations. Calculations can be performed for homogeneous cells, slab arrays, regular rod arrays, rod clusters in cylindrical geometry, and finite cylinders in \((r, z)\) geometry. Point burnup calculations and multicell calculations can also be performed.

2.4. Modeling TRIGA fuel elements with WIMS-D4, and STAR

In order to model and benchmark these experimental results in the 3D STAR neutronics code, group constants (i.e. cross-sections) must be found for each region of the core. The WIMS/PC-D4 code (Oak Ridge, 1991) was used to generate the cross-sections for this effort because of the unique features of the TRIGA fuel element. TRIGA fuel consists of two types: 8.5 w/o and 12 w/o uranium with 20 w/o enriched U-235. The TRIGA fuel elements each have a unique burnup history that
needed to be included in the cross section generation. Furthermore, the TRIGA fuel has a zirconium center pin and zirconium hydride inside the fuel region that provides the lattice effect (General Atomics, 1967) for prompt feedback as well as the fuel temperature Doppler effect. The zirconium hydride fuel matrix acts as the primary moderator in TRIGA fuels more than the coolant acts as a moderator (General Atomics, 1976). Also, the TRIGA core has graphite regions in the fuel element, control rods and transient rods with air and fuel follower regions (Feltus, 1998; Miller 1998). Previous analysis showed that standard LWR fuel lattice codes such as LEOPARD and CASMO cannot accurately model the TRIGA fuel (Feltus, et al., 1996). Miller (1998) provides the axial geometry information of the fuel elements, transient, regulator, scram and control rods for the steady state and transient conditions in detail.

The WIMS-D4 pin cell model was used to model the TRIGA fuel elements and the SN methods was used for the main 50 group (n.b., 50 groups is the maximum for the WIMS-D4 pin cell model) transport calculations, which were then collapsed into 2 energy group cross sections for the STAR code input. The TRIGA unit cell is modeled as two annuli and a polygon surrounding the annuli. The first annulus consists of the fuel pin and the region between the first and second annuli consists of the stainless steel clad. The region between the second annulus and the polygon is water. Two group cross sections, diffusion constants, effective and infinite multiplication factors, etc., are obtained from the WIMS-D4 output. Different fuel and moderator temperatures were used in the WIMS-D4 cases to accurately model the T/H feedback effects needed for the STAR group constant information.

Miller (1998) provides a detailed description of the cross-section development for each of the experiments and core configurations and the temperature feedback information needed for the STAR model. Cross-section derivatives with respect to temperature were calculated as a linear function between the peak temperature of the individual transient and the initial ambient temperature of the fuel. The cross sections are thus specifically derived for the fuel characteristics during the individual transient taking into account the historical burnup information of each fuel element modeled. Kinetics parameters were carefully determined and confirmed with data for other TRIGA reactors (General Atomics, 1967).

3. Description of the breazeale TRIGA experiments

Selected steady-state power levels were used to find static T/H and neutronic coupling information in terms of flux and temperature distributions. Time-dependent tests using ramps, square waves, and neutron pulses were performed that simulate time-dependent transients with kinetic and T/H feedback. Symmetric and asymmetric core configurations were used to develop spatially dependent kinetics and T/H conditions for the benchmarks. Static and time-dependent power levels, including pulses were used along with the temperature distributions and flux measurements to determine consistent fuel properties and temperature constants. Data for the ramps and square waves are obtained from both the operator console and
the reactor computer. The data collected are flux, power, fuel temperatures, and control rod insertion positions. Data for the rapid pulses are obtained from the console and from a new data acquisition system computer that was developed and tested as part of this research project (Miller, 1998).

Three transient types were chosen to test the neutronics and T/H coupling for the benchmark: pulses, square waves, and ramps. Three pulses (1.50, 1.75 and 2.00) were performed to test the coupling for the nearly instantaneous reactivity insertions which cause immediate Doppler fuel temperature and lattice effect (i.e. zirconium hydride) feedback. Three square waves (200, 350 and 500 kW) were performed to test the coupling of the step reactivity insertions with the delayed T/H feedback from the reactor coolant flow. Three ramps to 500 kW (with 30, 40 and 50 s periods) were performed to evaluate the level of neutronics coupled with T/H needed for modeling the delayed T/H feedback mechanism of the coolant inside the TRIGA core. A stepped ramp transient was performed to relate the steady-state fuel temperature distribution to the core power level, allowing the experimenters to determine and verify the fuel thermal conductivity, heat capacities, and specific heats of the TRIGA fuel elements. Each T/H kinetics transient performed on the PSBR TRIGA was then compared with the STAR/WIMS/COBRA model. All modeling assumptions and parameters were documented for the construction of the benchmark (Miller, 1998) so that the actual tests could be reproduced independently.

For the pulse, square wave, and ramp transient events performed on the TRIGA, two types of data were needed: core power and fuel temperature distributions. The core power for the pulses were measured with an uncompensated ion chamber (UIC) located one foot from the north face of the core at the centerline between core locations C8 and C9, as shown in Fig. 1. For the square waves and ramps, a compensated ion chamber (CIC) on the core centerline was used in the same location. The rationale behind placing the CIC at the core centerline was to obtain a symmetric reading of the core power level. Miller (1998) provides details about the TRIGA core at the start of the tests and the detailed experimental procedures used to conduct these tests.

Fuel temperatures were measured in various locations throughout the core. The temperature data was obtained through the use of instrumented fuel elements with thermocouples embedded in the zirconium centerline pin. Temperature measurements were taken in instrumented element I-15 at location G8 and instrumented element I-14 at D9 as shown in Fig. 1. The positions of I-15 and I-14 relative to each other allowed for a determination of the core radial temperature distribution for each power level achieved. Three thermocouples in I-14 allowed for taking an axial core temperature distribution. As expected the temperatures were highest in the bottom of the core and lower in the top portion of the fuel. A large temperature gradient exists because of the significant amount of cross-flow enters from the side of the core at the top portion of the reactor (Gougar, 1997) as shown in Fig. 2. Gougar’s experiments (1997) determined that more than 90% of the coolant flow at the top of the core is a result of cross flow coming in from the sides of the reactor.

Three pulses were conducted using $1.50, 1.75 and 2.00$ reactivity insertions. From the UIC data collected, the amount of instantaneous heat generation in the core is
known. From the thermocouple data, the magnitude and response time of the temperature increase is known. With the three different power levels associated with each of the three pulses, this information yielded the prompt fuel temperature feedback as a function of the amount of energy generated. Figs. 3–5 show the experimental results for the $1.50, 1.75$ and $2.00$ pulses.

Three square waves starting at $50$ W and ending at $200, 350$ and $500$ kW were performed using the final power level to set the control rod movement, using the least amount of power overshoot possible with the transient and regulator rod movement (Miller, 1998). Fig. 6 presents the power and temperature data, and shows the rapid, but slight overshoot of the control rod motion during the set-power option. With the initiation of the square waves, the effect of an instantaneous power increase on fuel temperature can be observed as a function of energy. After the desired global power level was achieved, the TRIGA was maintained at that power level until steady-state conditions were reached in the core and coolant. Upon reaching steady-state conditions, the reactor was then scrammed, and the time it took for the fuel to return to ambient pool temperature was observed from the thermocouple data.

Three ramps to $500$ kW were conducted on the PSBR TRICA using $30, 40$ and $50$ s periods starting with an initial power level of $50$ W. With these ramps, i.e. linearly
increasing reactivity insertions, the effect of an exponential power increase on the fuel temperature can be observed. After the desired global power level of 500 kW was achieved the TRIGA core was maintained at 500 kW until steady state conditions were reached, then the reactor was scrammed. The temperature responses were measured as the fuel returned to the ambient pool temperature. Fig. 7 shows the results for each of the three ramp tests.

After the completion of the pulse, square wave and ramp transients, it became evident that one more experiment was needed to complete the calculations of the time constants and temperature properties of the TRIGA fuel elements. A stepped ramp transient was initiated at power level of 50 W and then the power was ramped up to 50 kW using a 40 s period. At 50 kW the TRIGA was allowed to reach steady-state conditions before being ramped again up to 100 kW. This process was continued with increments of 50 kW until a power level of 750 kW was reached, as shown in Fig. 8. After the core reached steady-state conditions at 750 kW, the reactor was scrammed. After the completion of this experiment, the steady-state thermocouple readings were available at four locations in the core as a function of global power for 16 steady-state conditions. The average fuel temperature as a
function of core global power level could then be readily determined. This relationship demonstrated the amount of heat retention in the TRIGA core as a function of total amount of energy produced in the core. This information permitted the calculation of the heat capacity and thermal properties of the TRIGA fuel elements, and is given in detail by Miller (1998).

4. Determination of the TRIGA fuel thermal constants

This research project allowed temperature dependent thermal constants to be found for various transient conditions for the PSBR TRIGA. The method used in this research (Miller, 1998) could be used to find the information for other core loading patterns at the PSU TRIGA and for other research reactors. Each core
loading would produce slightly different values since each core and pattern would produce different temperature distributions for the weighting constants.

The thermocouple data had to be properly averaged to find representative fuel centerline temperatures. The PSBR TRIGA core has a parabolic temperature distribution because of the large amount of cross flow entering from the side of the reactor (Gougar, 1997). The experimental fuel centerline temperatures in I-14 and I-15 (Fig. 1) were compared with previously measured temperature gradients and found consistent with the results reported by Gougar (1997). A representative centerline temperature for the entire TRIGA core loading 48 was calculated using a weighting factor of 0.93347 and measured centerline temperature with the power level:

\[ T_{\text{ave}} = (0.93347)[T_{\text{cf,measured}} - (1.1604)P/K] \]
For future TRIGA experiments, this equation could be modified to use the centerline temperatures directly.

Next Miller (1998) determined the fuel rod heat capacity as a function of global power level using transient heat conduction theory and the dimensionless Biot number. The thermal conductivity of the rods was found as a function of the global reactor power level. Data from these and previous tests (Gougar, 1997) measured coolant surrounding rods, fuel centerline temperatures and derived the heat transfer coefficient as a function of the global power level. The thermal conductivity can be found as a function of temperature based on radial geometry. The thermal conductivity proved to be linear with global power level \( P \) in kW (Miller, 1998):

\[
K [\text{W/mK}] = 1.10 + (0.006)P
\]
Fig. 6. Power and temperature responses of 200, 350 and 500 kW square waves.
Fig. 7. Power and temperature responses of ramps to 500 kW with 30, 40 and 50 s periods.
General Atomics (1967) experiments for other TRIGA reactors show a linear relationship between the fuel specific heat and the fuel temperature (integrated temperature for the entire core):

\[ C_p\left[\text{J/cm}^3\text{C}\right] = 2.04 + 0.00417T[\text{C}] \]

Table 1 shows the calculated (i.e. derived) results as well as the experimentally measured values for the fuel specific heat, conductivity, heat capacity for each of the transients performed in this research (Miller, 1998).

Table 1
PSBR TRIGA fuel element thermal constants

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<tr>
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</table>
5. Comparison of the experimental benchmark and analytical results

These transient experiments were modeled with the STAR 3D kinetics code model using the WIMS-D4 two group cross sections. The STAR models of the pulses were coupled with a 1-D WIGL T/H model. After a careful evaluation of earlier results (Feltus et al., 1996), the core thermal constants were evaluated consistently, as described above. When the final temperature dependent thermal constants (i.e. heat transfer coefficients, conductivities, and heat capacities for the fuel elements) were used, as shown in Table 1, it was determined that the power responses calculated by STAR for the pulse events very closely predicted the power responses obtained experimentally.

Figs. 9 and 10 compare the measured test results with the STAR/WIGL model for the $1.50$ and $1.75$ pulses and show that the 1D WIGL T/H model in STAR is capable of handling core coolant conditions that involve simple 1D flow. During these rapid pulses, the fuel heats up essentially adiabatically, and the Doppler and lattice effect feedback quickly reduces the power as the energy remains in the fuel during the short pulse duration until the heat is slowly dissipated out of the fuel, as shown in Figs. 3–5. The STAR/WIGL model was also used to predict the $2.00$ pulse results and it predicted the test results well within the measurement error bands.
even though there is a known miscalibration problem for the PSBR TRIGA for higher power levels (Miller, 1998).

The ramps and square wave events were modeled with STAR using the a 1D LRA adiabatic T/H model (Weader, 1992) and a 3D COBRA IIIC T/H model (Rowe, 1973; Weader, 1992) based on steady state operation test results. One-dimensional T/H models (e.g. WIGL, LRA in STAR) were found to be incapable of handling the complex 3D flow patterns observed in the PSBR TRIGA. The Penn State TRIGA core resides in a pool that is cooled by natural circulation. Complex 3D flow patterns exist throughout the core (Gougar, 1997) which prevent a 1D T/H model from accurately describing the T/H conditions present, especially in a transient which exhibits delayed feedback effects from the coolant.

Miller (1998) demonstrated how poorly the LRA and WIGL models predicted the ramp and square wave test results because these models underpredict the actual flow through the core which is enhanced by the cross flow coming in from the sides of the reactor. The results produced by the 1D LRA model clearly illustrates that the transients with a duration long enough to involve coolant T/H feedback require a three-dimensional model of the core and the coolant flow. The LRA model underpredicted
the power because the reduced flow caused the fuel temperatures to be predicted higher, causing more Doppler feedback, than actually measured seen during the experiments.

The coupled COBRA T/H and STAR kinetics results, as coupled for steady state conditions, and for the time-dependent regimes of the ramps and square waves, were shown to predict the 3D TRIGA power responses and coolant flow exhibited during the actual tests well. Details about the performance of the COBRA T/H model coupled with the STAR kinetics model are given in the research thesis (Miller, 1998) and the project documentation (Feltus, 1998).

6. Significance of this research

As the nuclear industry strives to quantify operating margins and safety limits, it has become necessary to couple T/H codes with 3D kinetics methods. Point kinetics models cannot adequately describe the behavior of large power reactors, especially during asymmetrical events, such as reactivity insertion events, and main steam line breaks (Feltus, 1990, 1994). Many reactor transients depend on local T/H conditions in the core. Coupling T/H methods with full 3D dynamics (i.e. kinetics with feedback) analysis allows for a fully integrated solution between the power production and the moderator and fuel temperature feedback effects.

This research was directly motivated by the need to provide integral effects tests (i.e. actual experiments) in order to benchmark neutron kinetics methods coupled with T/H codes. The PSBR TRIGA has been carefully instrumented and analyzed through previous research efforts, which involved essentially steady state conditions. However, the flexibility of the TRIGA core and its inherent feedback mechanisms allow for challenging transient conditions. Although the PSBR TRIGA core is compact, 3D effects can be measured during various transients (e.g. pulses, ramps, square waves) in terms of the T/H conditions and the core power distribution.

This research effort has provided both experimental and analytical benchmark information for coupled T/H and kinetics feedback for reactor safety analysis efforts. The results can be used to qualify the point, one- and three-dimensional kinetics models in the NRC’s, vendor’s, and industry’s T/H code methodologies. The details gathered during the actual experiments and information determined during the analyses (Miller, 1998) are sufficiently extensive so that a formal benchmark problem effort can be used to validate any coupled 3D kinetics and T/H code package.

7. Summary and conclusions

A series of transients including pulses, square waves, and ramps were performed on the PSBR TRIGA to provide coupled kinetic and T/H benchmark experiments. The pulses were conducted to evaluate the level of neutrons needed for modeling the prompt temperature feedback mechanism of the TRIGA fuel elements. Square
waves and ramps were conducted in order to evaluate the level of neutronics coupled with T/H phenomena necessary for accurate modeling of the delayed T/H feedback mechanism of the coolant within the TRIGA core. This research validated the assumption that the PSU TRIGA provides sufficiently complex fluid flow conditions and kinetics detail for benchmark experiments that could be used to validate coupled kinetics T/H models.

These transient events were modeled with the STAR 3D kinetics code. The STAR models of the pulses were coupled with a 1-D WIGL T/H model. After a review of earlier results, the core thermal constants were evaluated consistently. When the final temperature dependent thermal constants (i.e. heat transfer coefficients, conductivities, and heat capacities for the fuel elements) were used, it was determined that the power responses calculated by STAR for the pulse events very closely predicted the power responses obtained experimentally.

The ramps and square waves were coupled with a 1D LRA adiabatic T/H model and a 3D COBRA IIIC T/H model based on steady state results. Predicted results from STAR and the LRA model for the transient portions of the ramps and square waves are described in the benchmark documentation (Miller, 1998). The results produced by the 1D LRA model clearly illustrates that the transients with a duration long enough to involve coolant T/H feedback require a three-dimensional model of the core and the coolant flow.

The Penn State TRIGA core resides in a pool that is cooled by natural circulation. Complex 3D flow patterns exist throughout the core (Gougar, 1997) which prevent a 1D T/H model from accurately describing the T/H conditions present. The LRA and WIGL 1D T/H models in STAR are capable of handing core coolant conditions that involve simple 1D flow; however, they are not capable of handing complex 3D flow patterns. COBRA and STAR results, as coupled for steady state conditions, and for the time-dependent regimes of the ramps and square waves, can predict the 3D TRIGA coolant flow exhibited during the actual tests.

While inter-comparison of coupled T/H and kinetics codes are needed, it is not adequate or sufficient. This research effort has provided both experimental and analytical benchmark information the coupled T/H and 3D kinetics feedback for reactor safety analysis efforts. These results can be used to qualify the point, one and three dimensional kinetics models in the NRC’s RELAP and TRAC series of codes, as well as industry T/H codes such as RETRAN. Other investigators can test their 3D coupled neutronics and T/H methods against these experimental results to see whether their codes can accurately describe the complex T/H and kinetics effects exhibited in the PSU TRIGA for pulses, ramps, and square waves.

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